APPLICATION FOR RENEWED OPERATING LICENSE

R. E. GINNA NUCLEAR POWER PLANT

TECHNICAL AND ADMINISTRATIVE INFORMATION CONTENTS

TECHNICAL AND ADMINISTRATIVE INFORMATION

Contents

TECHNICAL AND ADMINISTRATIVE INFORMATION

Contents

TECHNICAL AND ADMINISTRATIVE INFORMATION

Contents

TECHNICAL AND ADMINISTRATIVE INFORMATION

Contents

TECHNICAL AND ADMINISTRATIVE INFORMATION

Contents

TECHNICAL AND ADMINISTRATIVE INFORMATION

Contents

TECHNICAL AND ADMINISTRATIVE INFORMATION

Contents

TECHNICAL AND ADMINISTRATIVE INFORMATION

Contents

TECHNICAL AND ADMINISTRATIVE INFORMATION

Contents

TECHNICAL AND ADMINISTRATIVE INFORMATION

Contents

TECHNICAL AND ADMINISTRATIVE INFORMATION

Contents

TECHNICAL AND ADMINISTRATIVE INFORMATION
Tables

TECHNICAL AND ADMINISTRATIVE INFORMATION
Tables

TECHNICAL AND ADMINISTRATIVE INFORMATION
Tables

TECHNICAL AND ADMINISTRATIVE INFORMATION
Figures

1.0 ADMINISTRATIVE INFORMATION

This application has been prepared to provide the administrative, technical and environmental information required by 10 CFR Part 54 (Reference 1) and 10 CFR Part 51 (Reference 2) to support the renewal of the Operating License for R. E. Ginna Nuclear Power Plant (Ginna Station).

Consistent with the expectations in RIS 2001-05 (Reference 3), one hardcopy of this application is being provided for the public document room, with remaining copies on CD-ROM. Another hard copy is being provided to the NRC License Renewal Project Manager. For the reviewer's convenience, the CD-ROMs also contain copies of the Updated Final Safety Analysis Report (UFSAR) and the license renewal (LR) drawings prepared in support of the license renewal effort. Hyperlinks to those documents are provided in the application where appropriate. Five sets of hardcopy drawings are also being provided to the NRC under a separate cover letter.

The UFSAR, LR drawings, and other references cited within the application are for information only, and are not incorporated by reference into the LRA.

This section of the application provides the following information:

1. Information on the organization of the application (Section 1.1).
2. A general plant description (Section 1.2).
3. The administrative information required by 10 CFR 54.17 and 54.19 (Section 1.3).
4. Summary of abbreviations and passive function code definitions (Section 1.5).
5. A distribution list for written communications related to the application (Section 1.6).

1.1 Application Format and Content

The following discussion describes the content of the Ginna Station License Renewal Application.

Section 1 provides the administrative information required by Part 54 of Title 10 of the Code of Federal Regulations, Sections 17 and 19 (10 CFR 54.17 and 10 CFR 54.19).

Section 2 provides the scoping and screening methodology. Section 2 also describes and justifies the methodology used to determine the systems, structures, and components within the scope of license renewal and the structures and components subject to an aging management review. The system groupings in Sections 2 and 3 are organized to be consistent with NUREG-1800. Table 2.2-1, Plant Level Scoping Results, provides listings of the plant mechanical systems, structures, and electrical/instrumentation and controls (I&C) systems, and identifies those plant systems and structures that are and are not within the scope of license renewal. Sections 2.3, 2.4 and 2.5 provide a description of systems, their intended functions, and for information only, cross references to UFSAR sections and license renewal drawings. Each system subsection has a table listing component groups subject to an Aging Management Review (AMR), their passive intended function, and one or more hyperlinked cross references to the Section 3 table line items providing AMR information. The drawings and UFSAR are provided as a separate attachment for use as review tools.

Section 3 describes the results of the aging management reviews for the components and structures requiring aging management reviews. Section 3 identifies the components and structures subject to aging management review including a comparison to the structures and components identified in the U. S. Nuclear Regulatory Commission's (NRC) "Generic Aging Lessons Learned (GALL) Report," NUREG-1801, which are combined into Standard Review Plan System groups. Those Ginna Station component groups with aging effects, and aging management programs selected to manage those effects which are consistent with the assumptions made in the GALL Report, are contained in a set of tables that are identical to those in GALL Volume I and the Standard Review Plan. A second set of tables in each Section 3 system grouping provides aging management information including Component group, material, environment, aging effect and selected aging management programs for those component groups where one or more of the column details were not consistent with that assumed in the GALL Report. All the Section 3 tables have a final column that contains additional explanatory information specific to that line item. In addition these tables have hyperlinked cross references to the aging management details in Appendix B.

Section 4 includes a list of time-limited aging analyses (TLAAs), as defined by 10 CFR 54.3. It includes the identification of the component or subject, and an explanation of the time dependent aspects of the calculation or analysis. Section 4 demonstrates that the analyses remain valid for the period of extended operation, the analyses have been projected to the end of the period of extended operation, or the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. Section 4 also states that no 10 CFR 50.12 exemption involving a time-limited aging analysis as defined in 10 CFR 54.3 is required during the period of extended operation. There are some issues discussed in Section 4 that are not considered TLAAs. These are included to provide information on some issues that have been routinely discussed as part of Section 4 in past License Renewal Applications.

Appendix A, Updated Final Safety Analysis Report Supplements, contains a summary description of the programs for managing the effects of aging for the period of extended operation. A summary description of the evaluation of time-limited aging analyses for the period of extended operation is also included.

Appendix B, Aging Management Programs, describes the aging management programs and activities and demonstrates that the aging effects on the components and structures within the scope of the License Renewal Rule will be managed such that they will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation. Where the Ginna Station programs are consistent with corresponding programs in the GALL Report, the appropriate GALL program is referenced.

Appendix C is not used for this application.

Appendix D, Technical Specification Changes, concludes that no technical specification changes are necessary to manage the effects of aging during the period of extended operation.

Appendix E, Environmental Report, contains an environmental report analyzing the potential environmental impacts of license renewal, as provided for in NRC regulations 10 CFR 51.53(c) and 10 CFR 54.23. The NRC requires license renewal applicants to provide the NRC with input, in the form of an environmental report, that the NRC will use to meet NEPA requirements as they apply to license renewal [10 CFR 51.53(c)].

The NRC has determined that nuclear power plant license renewal decisions are major federal actions requiring preparation of an environmental impact statement [10 CFR 51.20(a)(2) and 51.95(c)]. In an effort to streamline the license renewal environmental review, the NRC conducted a generic analysis and published the results in NUREG-1437, Generic Environmental Impact Statement for the License Renewal of Nuclear Power Plants (GEIS). To fulfill NEPA requirements, the NRC is required to publish site-specific analyses in the form of a supplemental environmental impact statement to the GEIS.

The information in Section 2, Section 3, and Appendix B fulfills the requirements in 10 CFR 54.21(a). Section 1.4 discusses how the requirements of 10 CFR 54.21(b) will be met. The information in Section 4 fulfills the requirements in 10 CFR 54.21(c). The information in

Appendix A and Appendix D fulfills the requirements in 10 CFR 54.21(d) and 10 CFR 54.22, respectively. The Environmental Report, as required by 10 CFR 54.23, is provided with the Ginna Station License Renewal Application as Appendix E.

1.2 Plant Description

Ginna Station is a two-loop pressurized water reactor, rated at 490 MWe, located in Wayne County, near Rochester, New York. Ginna Station is located on the south shore of Lake Ontario, which is the source of circulating water and is the ultimate heat sink. The site, including the switchyard, contains 488 acres. The turbine and condenser system as well as the nuclear steam supply system were designed and supplied by Westinghouse. The remainder of the plant was designed by either Rochester Gas and Electric Corporation or Gilbert Associates, Incorporated. The replacement steam generators were designed and supplied by Babcock and Wilcox International.

1.3 Information Required by 10 CFR 54.17 and 10 CFR 54.19

1.3.1 Name of Applicant

Rochester Gas and Electric Corporation (RG&E)

RG&E is currently the sole owner and operator of Ginna Station. RGS Energy Group, Inc. (RGS), the parent company of RG&E, is a wholly-owned subsidiary of Energy East Corporation (Energy East).

1.3.2 Address of Applicant

Rochester Gas and Electric Corporation
89 East Avenue
Rochester, NY 14649

RGS Energy Group, Inc.
89 East Avenue
Rochester, NY 14649

Energy East Corporation
Commerce Plaza
Albany, NY 12260

1.3.3 Description of Business or Occupation of Applicant

RG&E is a New York corporation engaged principally in the generation of electricity and the purchase, transmission, distribution and sale of electric power and natural gas in Western New York State. RG&E operates under the general regulatory supervision of the New York State Public Service Commission (NYPSC) and, for its wholesale electricity sales and related interstate activities, RG&E is subject to regulation by the Federal Energy Regulatory Commission (FERC). With respect to its interests in Ginna Station, RG&E recovers its cost of generating electricity through rates subject to the regulatory authority of the NYPSC and FERC.

RG&E is a wholly-owned indirect subsidiary of Energy East, a registered public utility holding company under the Public Utility Holding Company Act of 1935 (PUHCA). Energy East has a number of wholly-owned public utility subsidiaries in the northeastern United States, including New York State Electric & Gas Corporation (NYSEG).

Energy East acquired 100% of the common stock of RGS, which is currently a direct wholly-owned subsidiary of Energy East. RG&E is a direct wholly-owned subsidiary of RGS. Energy East has transferred all of NYSEG's common stock to RGS, so that RG&E and NYSEG can be operated under a combined management structure.

NRC approval of the indirect transfer, as a result of the acquisition of RGS by Energy East, of RG&E's NRC-issued licenses was required. Accordingly, RG&E submitted an application pursuant to Section 50.80. This application was approved on December 10, 2001. The application addressed the issues central to the NRC's review of an indirect license transfer request, including the purpose and nature of the proposed transaction, as well as the impact of the proposed transaction on RG&E's existing financial and technical qualifications, its ability to provide decommissioning funding assurance, and the absence of foreign control or domination over the parent company.

As a result of the recent acquisition of RGS by Energy East, RG&E is now an indirect wholly-owned subsidiary of Energy East. However, this recent change in its ownership has not altered RG&E's status as a distinct corporate entity, nor its status as a regulated utility. RG&E continues to own the assets that it owned prior to the RGS acquisition and continues to engage in the generation of electricity and the purchase, transmission, distribution and sale of electric power and natural gas in its existing service territories. RG&E continues to maintain sole ownership of Ginna Station, and operates the station as an NRC licensee. RG&E continues to recover its costs associated with generation electricity at this facility through rates subject to the regulatory authority of the NYPSC and the FERC.

1.3.4 Organization and Management of Applicant

Rochester Gas and Electric Corporation

Rochester Gas and Electric Corporation is not owned, controlled or dominated by an alien, a foreign corporation, or a foreign government. All officers and directors are citizens of the

United States of America. The names and addresses of the directors and principal officers are provided below:

Directors

Name	Address
Wesley W. von Schack	Commerce Plaza Albany, NY 12260
Kenneth M. Jasinski	Commerce Plaza Albany, NY 12260
Paul C. Wilkens	89 East Avenue Rochester, NY 14649

Principal Officers

Name	Address
Paul C. Wilkens President	89 East Avenue Rochester, NY 14649
Louis L. Bellina Vice-President, Customer Relations	89 East Avenue Rochester, NY 14649
David J. Irish Vice-President, Fossil/Hydro Operations	89 East Avenue Rochester, NY 14649
Mark Keogh Vice-President, Treasurer and Secretary	89 East Avenue Rochester, NY 14649
Robert C. Mecredy Vice-President, Nuclear Operations	89 East Avenue Rochester, NY 14649
Clifton B. Olson Vice-President, Energy Supply	89 East Avenue Rochester, NY 14649
Jessica S. Raines Vice-President, Support Services	89 East Avenue Rochester, NY 14649
Paul G. Ruganis Vice-President, Information Services	89 East Avenue Rochester, NY 14649

William L. Thomas Vice-President, Human Resource Services	89 East Avenue Rochester, NY 14649
Michael B. Whitcraft Vice-President, Energy Delivery	89 East Avenue Rochester, NY 14649
Joseph A. Widay Vice-President and Plant Manager, Ginna Station	89 East Avenue Rochester, NY 14649
Joseph J. Syta Controller	89 East Avenue Rochester, NY 14649
Kathleen C. Spellane Assistant Treasurer	89 East Avenue Rochester, NY 14649

RGS Energy Group, Inc.

RGS Energy Group, Inc. is not owned, controlled or dominated by an alien, a foreign corporation, or a foreign government. All officers and directors are citizens of the United States of America. The names and addresses of the directors and principal officers are provided below:

Directors

Name	Address
Wesley W. von Schack	Commerce Plaza Albany, NY 12260
Kenneth M. Jasinski	Commerce Plaza Albany, NY 12260
Paul C. Wilkens	89 East Avenue Rochester, NY 14649

Principal Officers

Name	Address
Wesley W. von Schack Chairman, President, and Chief Executive Officer	Commerce Plaza Albany, NY 12260

Kenneth M. Jasinski Executive Vice President and Chief Financial Officer	Commerce Plaza Albany, NY 12260
Robert D. Kump Vice President and Secretary	Commerce Plaza Albany, NY 12260

Energy East Corporation

Energy East Corporation is not owned, controlled or dominated by an alien, a foreign corporation, or a foreign government. All officers and directors are citizens of the United States of America. The names and addresses of the directors and principal officers are provided below:

Directors

Name	Address
Richard Aurelio	Commerce Plaza Albany, NY 12260
James A. Carrigg	Commerce Plaza Albany, NY 12260
Joseph J. Castiglia	Commerce Plaza Albany, NY 12260
Lois B. DeFleur	Commerce Plaza Albany, NY 12260
G. Jean Howard	Commerce Plaza Albany, NY 12260
David M. Jagger	Commerce Plaza Albany, NY 12260
John M. Keeler	Commerce Plaza Albany, NY 12260
Ben E. Lynch	Commerce Plaza Albany, NY 12260

Peter J. Moynihan	Commerce Plaza Albany, NY 12260
Walter G. Rich	Commerce Plaza Albany, NY 12260
Wesley W. von Schack	Commerce Plaza Albany, NY 12260

Principal Officers

Name	Address
Wesley W. von Schack Chairman, President and Chief Executive Officer	Commerce Plaza Albany, NY 12260
Kenneth M. Jasinski Executive Vice-President and Chief Financial Officer	Commerce Plaza Albany, NY 12260
Robert D. Kump Vice-President, Treasurer and Secretary	Commerce Plaza Albany, NY 12260
Robert E. Rude Vice-President and Controller	Commerce Plaza Albany, NY 12260

1.3.5 Class of License, Use of Facility, and Period of Time for which the License is Sought

RG&E requests renewal of the Class 103 operating license for Ginna Station (License No. DPR-18) for a period of 20 years beyond the expiration date of the current license on September 18, 2009.

RG&E also requests renewal of the source, special nuclear material, and by-product license that is included within the operating license and that was issued pursuant to 10 CFR Parts 30, 40, and 70.

1.3.6 Earliest and Latest Dates for Alterations, if Proposed

RG&E does not propose to alter the station in connection with this application. The current licensing basis (CLB) will be continued and maintained throughout the period of extended operation.

1.3.7 Listing of Regulatory Agencies Having Jurisdiction and News Publications

The Federal Energy Regulatory Commission (FERC) and the New York State Public Service Commission (NYPSC) are the principal regulators of the company's electric operations.

The Honorable David P. Boergers
Secretary
Federal Energy Regulatory Commission
888 First Street, NE, Room 1A
Washington, DC 20426

Janet H. Deixler
Secretary to the Commission
New York State Public Service Commission
New York State Department of Public Service
3 Empire State Plaza
Albany, NY 12223-1350

The area news publication and its associated address is provided below

Democrat and Chronicle
55 Exchange Blvd
Rochester, NY 14614

1.3.8 Conforming Changes to Standard Indemnity Agreement

10 CFR 54.19(b) requires that license renewal applications include, "…conforming changes to the standard indemnity agreement, 10 CFR 140.92, Appendix B, to account for the expiration term of the proposed renewed license." The current indemnity agreement for the unit does not contain a specific expiration term for the operating license. Therefore, conforming changes to account for the expiration term of the proposed renewed license are not necessary, unless the license number is changed upon issuance of the renewed license.

1.3.9 Restricted Data Agreement

This application does not contain restricted data or other national defense information, nor is it expected that subsequent amendments to the license application will contain such information. However, pursuant to 10 CFR 54.17(g) and 10 CFR 50.37, RG&E, as a part of the application for a renewed operating license, hereby agrees that it will not permit any individual to have access to or any facility to possess Restricted Data or classified National

Security Information until the individual and/or facility has been approved for such access under the provisions of 10 CFR Parts 25 and/or 95.

1.4 Current Licensing Basis Changes During NRC Review

Each year, following the submittal of the Ginna Station License Renewal Application and at least three months before the scheduled completion of the NRC review, Ginna Station will submit amendments to the application pursuant to 10 CFR 54.21(b). These revisions will identify any changes to the current licensing basis that materially affect the contents of the License Renewal Application, including the UFSAR supplements and any other aspects of the application.

1.5 Abbreviations and Passive Function Code Definitions

1.5.1 Abbreviations

This section contains the abbreviations that pertain to the administrative and technical information within the license renewal application. The abbreviations that pertain to the environmental information are included in the front of Appendix E (Environmental Report).

Abbreviation	Definition
AC	Alternating Current
ABVS	Auxiliary Building Ventilation System
ACI	American Concrete Institute
AEC	Atomic Energy Commission
AFW	Auxiliary Feedwater
AISC	American Institute of Steel Construction
AMP	Aging Management Program
AMR	Aging Management Review
AMSAC	ATWS Mitigation System Actuation Circuitry
ANSI	American National Standards Institute
AR	ACTION Report
ARV	Atmospheric Relief Valve
ASCO	Automatic Switch Company
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transients Without a SCRAM
AVT	All-Volatile-Treatment
BADGER	Boron Areal Density Gauge for Evaluating Racks
BAST	Boric Acid Storage Tank
BF3	Boron Trifluoride
BSS	Borated Stainless Steel
BTP	Branch Technical Position
B&W	Babcock and Wilcox

Abbreviation	Definition
BWR	Boiling Water Reactor
CAR	Corrective Action Report
CASS	Cast Austenitic Stainless Steel
CCW	Component Cooling Water
CD-ROM	Compact Disk-Read only Memory
CFR	Code of Federal Regulations
CLB	Current Licensing Basis
CLOC	Closed Loop Outside Containment
CMIS	Configuration Management Information System
CREATS	Control Room Emergency Air Treatment System
CRDM	Control Rod Drive Mechanism
CRFC	Containment Recirculation Fan Cooling
CRT	Cathode Ray Tube
CS	Containment Spray or Carbon Steel
CUF	Cumulative Usage Factor
CVCS	Chemical and Volume Control System
CW	Circulating Water
DAM	Data Acquisition Modules
DBA	Design Basis Accident
DBD	Design Basis Document
DBE	Design Basis Event
DC	Direct Current
DMIMS	Digital Metal Impact Monitoring System
DOJ	U.S. Department of Justice
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EFPY	Effective Full Power Year

Abbreviation	**Definition**
EMPA	Swiss Federal Testing Station
Energy East	Energy East Corporation
EPM-FPPR	Fire Protection Program Report
EPR	Ethylene Propylene Rubber
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
ESF	Engineered Safety Features
ESFAS	Engineered Safety Features Actuation System
FAC	Flow-Accelerated Corrosion
FCC	Federal Communications Commission
F_{en}	Environmental Fatigue Multiplier
FERC	Federal Energy Regulatory Commission
FME	Foreign Material Exclusion
FP	Fire Protection
FRP	Fiberglass Reinforced Plastic
FSER	Final Safety Evaluation Report
GAI	Gilbert Associates, Inc.
GALL	Generic Aging Lessons Learned
GDC	General Design Criterion
GEIS	Generic Environmental Impact Statement
GL	Generic Letter
GSI	Generic Safety Issue
GTR	Generic Technical Report
HAZ	Heat-Affected Zone
HELB	High Energy Line Break
HEPA	High Efficiency Particulate Air
HIC	High Integrity Container

Abbreviation	Definition
HSLAS	High Strength Low Alloy Steel
HTK	High Temperature Kerite
HUT	Holdup Tank
HVAC	Heating, Ventilation, and Air Conditioning
I&C	Instrumentation and Controls
IASCC	Irradiation-Assisted Stress Corrosion Cracking
ID	Inner Diameter
IDR	Identified Deficiency Report
IE	Inspection and Enforcement
IGA	Intergranular Attack
IGSCC	Intergranular Stress Corrosion Cracking
IN	Information Notice
INEL	Idaho National Engineering Laboratories
INPO	Institute of Nuclear Power Operations
IPA	Integrated Plant Assessment
IR	Insulation Resistance
ISI	Inservice Inspection
ITG	Issues Task Group
LBB	Leak-Before-Break
LER	Abnormal Occurrence and Licensee Event Report
LOCA	Loss-of-Coolant Accident
LR	License Renewal
LRA	License Renewal Application
LRE	License Renewal Engineer
LTOPS	Low Temperature Overpressure Protection System
LVDT	Linear Variable Differential Transformer
MDAFW	Motor-Driven Auxiliary Feedwater

Abbreviation	Definition
MIC	Microbiologically Induced Corrosion
MOV	Motor Operated Valve
MRP	Materials Reliability Program
MRPI	Microprocessor Rod Position Indication
MRV	Minimum Required Value
MSIV	Main Steam Isolation Valve
MSSV	Main Steam Safety Valve
MT	Magnetic Particle Test
MWe	Megawatt-electric
NADP	National Atmospheric Deposition Program
NaOH	Sodium Hydroxide
NCR	Non-Conformance Report
NDE	Non-destructive Examination
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NNS	Non-Nuclear Safety
NPS	Nominal Pipe Size
NRC	Nuclear Regulatory Commission
NSR	Non-Safety Related
NSAC	Nuclear Safety Analysis Center
NSSS	Nuclear Steam Supply System
NTN	National Trends Network
NYPSC	New York State Public Service Commission
NYSEG	New York State Electric and Gas
ODSCC	Outside Diameter Stress Corrosion Cracking
PE	Polyethylene
PLL	Predicted Lower Limit

Abbreviation	**Definition**
PORV	Power Operated Relief Valve
PPCS	Plant Process Computer System
PSPM	Periodic Surveillance and Preventive Maintenance Program
PSS	Plant Sampling System
PSSL	Plant Systems and Structures List
PTLR	Pressure-Temperature Limit Report
PTS	Pressurized Thermal Shock
P-T	Pressure-Temperature
PUHCA	Public Utility Holding Company Act of 1935
PVC	Polyvinyl Chloride
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
QA	Quality Assurance
RAI	Request for Additional Information
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RG	Regulatory Guide
RGS	RGS Energy Group, Inc.
RG&E	Rochester Gas and Electric Corporation
RHR	Residual Heat Removal
RIO	Refueling, Inspection and Overhaul Report
RMW	Reactor Makeup Water
RPV	Reactor Pressure Vessel
RTD	Resistance Temperature Detector

Abbreviation	Definition
RT_{PTS}	Reference temperature for pressurized thermal shock
RT_{NDT}	Reference nil ductility transition temperature
ΔRT_{NDT}	Radiation-induced shift in the reference nil ductility transition temperature
RTS	Reactor Trip System
RV	Reactor Vessel
RVI	Reactor Vessel Internals
RWST	Refueling Water Storage Tank
SAFW	Standby Auxiliary Feedwater
SAS	Safety Assessment System
SBO	Station Blackout
SC-1	Safety Class 1
SC-2	Safety Class 2
SC-3	Safety Class 3
SCBA	Self-Contained Breathing Apparatus
SCC	Stress Corrosion Cracking
SEC	Securities and Exchange Commission
SEP	Systematic Evaluation Program
SER	Safety Evaluation Report
SFP	Spent Fuel Pool
SFR	System Function Report
SG	Steam Generator
SI	Safety Injection
SOER	INPO Significant Operating Event Report
SOV	Solenoid Operated Valve
SPING	System-Level Particulate, Iodine, and Nobel Gas Monitors

Abbreviation	Definition
SQUG	Seismic Qualification Utility Group
SR	Safety Related
SRP	Standard Review Plan
SS	Safety Significant or Stainless Steel
SSC	Structure System and Component
SW	Service Water
SwRI	Southwest Research Institute
SWSROP	Service Water System Reliability Optimization Program
TDAFW	Turbine-Driven Auxiliary Feedwater
TLAA	Time-Limited Aging Analyses
TRM	Technical Requirements Manual
TSC	Technical Support Center
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
USE	Upper Shelf Energy
USI	Unresolved Safety Issue
UT	Ultrasonic Testing
UV	Ultraviolet
VCT	Volume Control Tank
VT	Visual Test
WOG	Westinghouse Owner’s Group

1.5.2 Passive Function Code Definitions

This section contains the meanings for the abbreviations used in the Screening results tables to represent the passive functions for components, subcomponents, and structural members. Passive functions are the specific intended functions performed by in-scope passive components in support of system or structure intended functions. Passive components are components that perform an intended function without moving parts or without a change in configuration or properties.

CODE	**DEFINITION**
CORE SUPPORT	CORE SUPPORT
DELIVER VOLTAGE, CURRENT, OR SIGNALS	TO PROVIDE ELECTRICAL CONNECTIONS TO SPECIFIED SECTIONS OF AN ELECTRICAL CIRCUIT TO DELIVER VOLTAGE, CURRENT, OR SIGNALS
DIRECT FLOW	STR-PROVIDE SPRAY SHIELD OR CURBS FOR DIRECTING FLOW (E.G. SAFETY INJECTION FLOW TO CONTAINMENT SUMP)
DIRECT GASEOUS DISCHARGE	STR-PROVIDE PATH FOR RELEASE OF FILTERED AND UNFILTERED GASEOUS DISCHARGE
FASTENING	PROVIDE CONNECTION FASTENING
FILTER	STR-PROVIDE FILTRATION
FIRE BARRIER	STR-PROVIDE RATED FIRE BARRIER TO CONFINE OR RETARD A FIRE FROM SPREADING TO OR FROM ADJACENT AREAS OF THE PLANT
FLAME SUPPRESSION	FLAME SUPPRESSION
FLOOD BARRIER	STR-PROVIDE FLOOD PROTECTION BARRIER (INTERNAL AND EXTERNAL FLOODING EVENT)
FLOW CONTROL	FLOW CONTROL
FLOW DISTRIBUTION	FLOW DISTRIBUTION
GUIDE AND SUPPORT INSTRUMENTATION	GUIDE AND SUPPORT INSTRUMENTATION
GUIDE AND SUPPORT RCCA'S	GUIDE AND SUPPORT RCCA'S
GUIDE AND SUPPORT THERMOCOUPLES	GUIDE AND SUPPORT THERMOCOUPLES
HEAT SINK	STR-PROVIDE HEAT SINK DURING SBO OR DESIGN BASIS ACCIDENTS

CODE	DEFINITION
HEAT TRANSFER	HEAT TRANSFER
HELB SHIELDING	STR-PROVIDE SHIELDING AGAINST HIGH ENERGY LINE BREAKS
HOUSE, PROTECT EQUIPMENT	HOUSE, PROTECT EQUIPMENT
INSULATE AND SUPPORT	INSULATE AND SUPPORT ELECTRICAL CONDUCTOR
JOINT INTEGRITY	MECHANICAL CLOSURE INTEGRITY
LIFTING EQUIPMENT	LIFTING EQUIPMENT
MAINTAIN ELEC SYSTEM INTEGRITY	MAINTAIN ELEC SYSTEM INTEGRITY
MISSILE BARRIER	STR-PROVIDE MISSILE BARRIER (INTERNALLY OR EXTERNALLY GENERATED)
MOUNT & SUPPORT ELEC COMP & INST	MOUNT & SUPPORT ELEC COMP & INST
NO NEI 95-10 PASSIVE FUNCTION	NO NEI 95-10 PASSIVE FUNCTION
NOT EVALUATED	INTENDED FUNCTION NOT REQUIRED TO BE EVALUATED
PIPE WHIP RESTRAINT	STR-PROVIDE PIPE WHIP RESTRAINT
PRESSURE BOUNDARY	STR-PROVIDE PRESSURE BOUNDARY OR ESSENTIALLY LEAK TIGHT BARRIER TO PROTECT PUBLIC HEALTH AND SAFETY IN THE EVENT OF ANY POSTULATED DESIGN BASIS EVENT
PRESSURE BOUNDARY	PRESSURE BOUNDARY
PROVIDE ELECTRICAL ISOLATION	PROVIDE ELECTRICAL ISOLATION
PROVIDE FILTRATION	PROVIDE FILTRATION
PROVIDE FIRE BARRIER	PROVIDE FIRE BARRIER
PROVIDE FLOW	PROVIDE FLOW

CODE	**DEFINITION**
PROVIDE ISOLATION BARRIER	PROVIDE ISOLATION BARRIER
PROVIDE MIXED FLOW	PROVIDE MIXED FLOW
PROVIDE RADIATION SHIELD	PROVIDE RADIATION SHIELD
PROVIDE STRUCTURAL SUPPORT	PROVIDE STRUCTURAL SUPPORT
RADIATION SHIELDING	STR-PROVIDE SHIELDING AGAINST RADIATION AND/OR HEAT
RESERVOIR	RESERVOIR
RESTRICTS FLOW	RESTRICTS FLOW
RX CLNT PRESS BOUNDARY	RX CLNT PRESS BOUNDARY
SHELTER SR	STR-PROVIDE SHELTER/PROTECTION TO SAFETY-RELATED COMPONENTS
SHIELD VESSEL	SHIELD VESSEL
STRUCTURAL SUPPORT	STRUCTURAL SUPPORT
SUPPORT IN-CORE INSTRUMENTATION	SUPPORT IN-CORE INSTRUMENTATION
SUPPORT NSR	STR-PROVIDE STRUCTURAL SUPPORT TO NONSAFETY-RELATED COMPONENTS WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF ANY OF THE REQUIRED SAFETY RELATED FUNCTIONS
SUPPORT PIPE, CABLE, DUCT	SUPPORT PIPE, CABLE, DUCT
SUPPORT RV INTERNALS	SUPPORT RV INTERNALS
SUPPORT SR	STR-PROVIDE STRUCTURAL AND/OR FUNCTIONAL SUPPORT TO SAFETY RELATED EQUIPMENT

CODE	DEFINITION
SUPPORT THIMBLE TUBES	SUPPORT THIMBLE TUBES
THROTTLING	THROTTLING
WATER SOURCE	STR-PROVIDE SOURCE OF COOLING WATER FOR PLANT SHUTDOWN

1.6 Communications

Written communications on this application should be directed to:

Mr. Robert C. Mecredy
Vice-President, Nuclear Operations
R. E. Ginna Nuclear Power Plant
1503 Lake Road
Ontario, NY 14519

With copies to:

Mr. George J. Wrobel
Manager, Nuclear Safety and Licensing
R. E. Ginna Nuclear Power Plant
1503 Lake Road
Ontario, NY 14519

And

Mr. David F. Wilson
R. E. Ginna Nuclear Power Plant
1503 Lake Road
Ontario, NY 14519

Section 1.0 References

1. 10 CFR 54, Requirements for Renewal of Operating Licenses for Nuclear Power Plants, U.S. Nuclear Regulatory Commission.

2. 10 CFR 51, Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions, U.S. Nuclear Regulatory Commission.

3. RIS 2001-05, NRC Regulatory Issue Summary 2001-05, Guidance on Submitting Documents to the NRC By Electronic Information Exchange or on CD-ROM

4. NEI 95-10, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule, Rev. 3, Nuclear Energy Institute, March 2001.

5. Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, NUREG-1800, U.S. Nuclear Regulatory Commission, July 2001.

2.0 SCOPING AND SCREENING METHODOLOGY FOR IDENTIFYING STRUCTURES AND COMPONENTS SUBJECT TO AGING MANAGEMENT REVIEW, AND IMPLEMENTATION RESULTS

2.1 Scoping and Screening Methodology

2.1.1 Introduction

The initial step in scoping is defining the entire plant in terms of major systems and structures. All of these systems and structures are evaluated against the scoping criteria in 10 CFR 54.4 (a)(1), (2), and (3), to determine if they perform, support or could adversely impact a critical safety function for responding to a design basis accident event, or perform or support a specific requirement of one of five regulated events.

This step is accomplished using the UFSAR, Technical Specifications, licensing correspondence files, Design Basis Documents (DBDs), controlled drawings, the Q-list and the Configuration Management Information System (CMIS), a controlled data base which stores equipment and Licensing Basis information.

During the scoping process, even if only a portion of a system or structure meets the scoping criteria of 10 CFR 54.4, the system or structure is identified as in the scope of License Renewal for subsequent screening with exceptions as indicated in Section 2.1.3. The scoping and screening process provides a system description and identifies specific system/structure intended functions. System scoping establishes all the components and structural parts of which the system or structure is composed, defines component internal and external environments, and then specifies which components and structural parts support the specific system/structure intended functions (i.e., are within the License Renewal evaluation boundary). As a result, not all of the components or structural parts that make up in-scope systems and structures may be within the scope of License Renewal, since those components do not support an intended function. The screening process identifies those components subject to aging management review. By identifying the active or passive nature of the SSC and whether the SSC is long-lived, only the appropriate components are forwarded to aging management review. The screening process also defines the License Renewal component passive intended functions (e.g. pressure boundary, restrict flow, etc.).

2.1.2 Plant Level Scoping

10 CFR 54 provides specific criteria for determining which systems, structures, and components should be reviewed and evaluated for inclusion in the scope of License Renewal. Specifically, Section 54.4 of the rule states that:

(a) Plant systems, structures, and components within the scope of this part are:

(1) Safety related systems, structures, and components which are those relied upon to remain functional during and following design basis events (as defined in 10 CFR 50.49(b)(1)) to ensure the following functions:

(i) The integrity of the reactor coolant pressure boundary;

(ii) The capability to shut down the reactor and maintain it in a safe shutdown condition; or

(iii) The capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposures comparable to those referred to in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2), 10 CFR 100.11, as applicable.

(2) All non-safety-related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs (a)(1)(I), (ii), or (iii) above.

(3) All systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the NRC's regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63).

(b) The intended functions that these systems, structures, and components must be shown to fulfill in 54.21 are those functions that are the bases for including them within the scope of License Renewal as specified in paragraphs (a)(1) - (3) of this section.

All plant systems and structures that comprise Ginna Station are reviewed and evaluated against the criteria outlined above to determine which ones meet the requirements for inclusion in the scope of License Renewal. The scoping methodology used by RG&E is consistent with the guidance provided by the NRC in NUREG-1800 and NEI 95-10. Existing plant documentation is used for this review including the UFSAR, Technical Specifications, and the licensing correspondence files that collectively form the Ginna Current Licensing Basis, controlled drawings, and the Configuration Management Information System (CMIS) database. Once the systems and structures have been identified for inclusion in the scope, they will be further evaluated with the next step in the IPA process - screening.

2.1.3 System and Structure Function Determination

The plant Q-list documents Structure System and Component (SSC) Safety Functions in a very structured and comprehensive way. The License Renewal Scoping and Screening

Process initially uses the rule based classification information and process as a direct input. Approved plant procedures govern the process for safety classifications while the CMIS database stores the resultant process output.

Plant structures, systems and components are sorted and tracked within CMIS using a system identifier known as the Plant Systems and Structures List (PSSL) number. This detailed numbering scheme supports plant needs with respect to the maintenance rule and maintenance work and offers a good starting point with respect to identifying license renewal system functional boundaries. License Renewal (LR) systems are based on the PSSL and account for and contain all of the PSSL systems, but do so in a manner that is much more consistent with the broader system descriptions in the UFSAR.

SSC safety classifications are functionally based and are valid no matter what system naming or sorting scheme is used. Therefore, describing unique system boundaries for License Renewal has no impact on the ability to determine if an SSC performs a License Renewal intended function.

The SSC functions detailed in the systems scoping reports are derived from the plant's licensing basis. System (and Structure) Function Codes have been assigned and can be related to the criterion in the American National Standard Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants (ANSI/ANS-51.1-1983) should further clarification be required. Table 2.1-1 details the system function codes used. Systems associated with function codes A through R are systems that perform nuclear safety functions (10 CFR 54.4(a)(1) Criterion 1 functions). Systems associated with function code Y contain components whose failure can affect a nuclear safety function (10 CFR 54.4(a)(2) Criterion 2 functions). Systems associated with function codes Z1 through Z5 contain components that are associated with the License Renewal regulated event set (10 CFR 54.4(a)(3) Criterion 3 functions).

The safety classification process also evaluates how a system component supports the system's safety function. This evaluation accounts for both active and passive design functions (e.g. pressure boundary, a passive function and allow flow, an active function). The classification evaluation is documented in CMIS in the form of numeric codes applied to show how each component function supports every system level function associated with that component. A detailed listing of the codes is contained in approved plant procedures.

An understanding of the relationship between the classification system functions and component safety classification safety rules are important. This is because a system may contain components that are within the scope of the License Renewal Rule even though the host system does not perform a system level design function that would appear to be subject to the rule. It may also be because components within the system have been credited within the CLB to do more functions than was the host system's original design functions.

Example: The Safety Injection (SI) system has as one of its intended functions, Function B, Introduce Emergency Negative Reactivity, to make the reactor subcritical. This is an obvious design function of the system as described in the UFSAR. The SI system also has selected components that are used to achieve post fire safe shutdown. That the system contains such components is not intuitive from simply reviewing the design function of the system.

Thus it is necessary to thoroughly understand and document all the uses of a system, and components within that system, in order to achieve a competent system level scoping list.

In order to differentiate between a system's design function and any other functions associated with components tracked within a system boundary, the individual License Renewal system scoping results are annotated with comments. When a system function comment describes a "primary design system function," this indicates that this is one of the designated nuclear safety functions of that system. When the comments describe "associated design system functions," this indicates a component within the system performs that function, possibly independently of what the system was originally designed to do. Primary system functions are generally discussed in the UFSAR in the chapter devoted to the system. Associated functions are generally discussed either as topical areas in the UFSAR or elsewhere in the CLB. Associated functions may or may not be nuclear safety functions.

System scoping must identify all License Renewal functions associated with components contained within a system. Generally, within the License Renewal System boundary, if the system under review contains any components that meet the License Renewal scoping criteria detailed in 10 CFR 54.4(a), the entire system is considered in-scope and that system moves forward to the License Renewal screening process.

There are two specific exceptions to this dictate:

1) When the only in-scope portion of the system is comprised of components that will receive a commodity group evaluation (e.g. fire barriers, equipment supports, etc.). In this case it is appropriate to identify the system or structure as not being within the scope of License Renewal, however the basis for that determination must be clearly identified.

   Example: The Non-Essential Ventilation Systems contain components that act as fire barriers (fire dampers). Within the system evaluation boundary, no other functions performed by the system are License Renewal intended functions. Therefore, this method of evaluation of the system components that perform the fire barrier function within the Fire Barrier commodity group results in designation of the Non-Essential Ventilation Systems as not being within the scope of License Renewal.

2) When the only in-scope portion of the system is comprised of components that act as containment isolation boundaries. In that case it is appropriate to identify the system as not being within the scope of License Renewal so long as the components that perform the isolation boundary function are evaluated within the Containment Isolation Boundary System.

Example: The Plant Sampling System contains components that act as containment isolation boundaries (valves, pipe). Within the system evaluation boundary no components, other than those that perform the isolation function, perform any additional License Renewal intended functions. Therefore, this method of evaluation of the system components that perform the containment isolation boundary function within the Containment Isolation System results in the designation of Plant Sampling as not being within the scope of License Renewal.

The critical element of system scoping, no matter if the general results case is applied or one of the specific exceptions is invoked, is to ensure that all SSCs that perform License Renewal intended functions are identified and the criteria that made them in-scope to License Renewal is documented.

2.1.4 Design Codes, Standards, and SSC Safety Classifications

The plant was designed and constructed prior to issuance of what are now the standards for Quality Group Classifications and ASME code boundaries. UFSAR Section 3.2, Classification of Structures, Components and Systems, provides a comparison of original construction codes and Regulatory Guide 1.26 and 10 CFR 50.55a. UFSAR Table 3.2-1, Classification of Structures, Systems and Components, summarizes the results of the Systematic Evaluation Program (SEP) review of selected SSCs important to safety.

The Safety Classification process provides a further comprehensive review of plant SSCs and issues Quality Classifications utilizing ANSI/ANS-51.1-1983 as guidance. Consequently, plant SSCs have been designated as Safety Class 1 (SC-1), Safety Class 2 (SC-2), Safety Class 3 (SC-3), Safety Significant (SS) typically described at other facilities as Non-Nuclear Safety related with Augmented Quality, and Non-Nuclear Safety (NNS).

Components designated as SC-1, SC-2, or SC-3 are classified as safety related because they meet the safety related structures, system and components definition in 10 CFR 50.2. Consequently, these components are also subject to full quality assurance requirements.

Quality group flags commensurate with safety classifications are depicted on the plant P&IDs. Components designated as SC-1, 2 or 3 are necessarily within the scope of License Renewal while other system components may or may not be.

2.1.5 Application of License Renewal Scoping Criterion

Figure 2.1-1 provides a basic diagram depicting how the scoping and screening process is executed.

2.1.5.1 Safety-Related Criteria Pursuant to 10 CFR 54.4(a)(1) (Criterion 1)

10 CFR 54.4(a)(1) states that SSCs within the scope of License Renewal include safety related SSCs that are relied upon to remain functional during and following design basis events (as defined in 10 CFR 50.49(b)(1)) to ensure the following functions:

- the integrity of the reactor coolant pressure boundary;
- the capability to shut down the reactor and maintain it in a safe shutdown condition; or
- the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to those referred to in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2), or 10 CFR 100.11 of this chapter as applicable.

As discussed in Section 2.1.4, the plant has an established safety classification process for SSCs. The safety related criterion used in that process encompasses the definition of safety related specified in 10 CFR 54.4(a)(1).

Implementation of the License Renewal Scoping and Screening procedure ensured that the UFSAR, Technical Specifications, design documents, design drawings and SSC safety classifications were reviewed as applicable to ensure all system functions were identified.

Based on this review, the License Renewal intended functions relative to the criteria of 10 CFR 54.4(a)(1) were identified and documented (as discussed in Section 2.1.3).

Thus, the scoping process used to identify safety related systems and structures is consistent with and satisfies the criteria of 10 CFR 54.4(a)(1).

2.1.5.2 Non-Safety Related Criteria Pursuant to 10 CFR 54.4(a)(2) (Criterion 2)

10 CFR 54.4(a)(2) states that SSCs within the scope of License Renewal include non-safety related SSCs whose failure could prevent satisfactory accomplishment of any of the functions identified for safety related SSCs.

The analytical process used to review SSCs for 10 CFR 54.4(a)(2) applicability ensured that the UFSAR, Technical Specifications, design documents, design drawings and the SSC safety classifications were reviewed as applicable to ensure all non-safety SSC functional interactions were identified where a non-safety SSC could fail and prevent the satisfactory accomplishment of a safety function.

However, reviewing design documents a system at a time does not provide information relative to system spatial interactions. Thus, it is possible that a functionally non-safety component whose failure could prevent a safety function could escape being designated as in-scope solely using the analytical review method.

To guard against such instances, the identification of SSCs meeting the criterion of 10 CFR 54.4(a)(2) was performed in a two-phase, synergistic fashion.

- Phase one, the analytical process, evaluates SSCs meeting Criterion 2 that are explicitly identified in the Current Licensing Basis (CLB).
- Phase two, the plant spaces physical review, evaluates SSCs for possible interactions not functionally described in the CLB.

2.1.5.3 Phase 1: Incorporating the Non-Safety SSCs Meeting the Criterion that are Explicitly Identified in the Current Licensing Basis

Scoping and Screening procedures direct the review of the CLB to identify the non-safety SSCs subject to inclusion within the rule. As previously discussed, some non-safety related equipment whose failure could affect a safety related function carry a classification designator of safety significant and are included within the plant Quality Assurance Program. This quality classification grouping may also include components that fulfill 10 CFR 54.4(a)(3) criteria. As discussed in Section 2.1.3, it is possible to segregate classifications by quality criteria rule application. The quality criteria rules in approved plant procedures were reviewed to determine which SSCs met 10 CFR 54.4(a)(2). These rules allow identification of systems which host components that are:

- Credited for HELB (pipe whip, jet impingement)
- Credited for internal flooding (barriers, drains)

- Credited for external flooding
- Credited for internal missiles
- Load handling equipment credited for NUREG-0612
- Alternate/backup systems or equipment credited in mitigating Licensing Basis Events

Three items warrant further clarification and discussion: SR/NNS piping interfaces, seismic II/I supports, and alternate/backup systems and equipment credited in mitigating Design Basis Events.

SR/NNS Piping Interface

The scoping and screening process for mechanical systems utilizes P&IDs to graphically represent License Renewal boundaries. The P&IDs show component classification boundaries at valves. Actual safety boundaries extend to the weld after the first seismic support beyond the P&ID depicted class change. This piping is subject to aging management review (AMR) and is included within the characterization of the piping assets considered to be within the scope of License Renewal.

Seismic II/I Supports

Non-seismic SSCs, which are positioned above or in close proximity to safety related SSCs and whose failure during a seismic event could cause the failure of a safety related SSC, are commonly referred to as Seismic II/I.

This includes:

- Supports for non-safety components whose hangers, supports and mounting hardware must be seismically designed to prevent non-safety related components from damaging adjacent safety related components.
- Non-safety related components and associated supports that can be considered a potential source of jeopardy (e.g. missiles) for nearby safety related components.

NOTE: The pressure boundary aspects of non-safety piping whose failure could affect a safety function are discussed in Section 2.1.5.4.

As discussed in UFSAR Section 3.2, Classification of Structures, Components and Systems, and Section 3.7, Seismic Design, significant analysis and modification efforts have been made to ensure the station is seismically capable. Seismic robustness was further enhanced by implementation of the Seismic Qualification Utility Group (SQUG) processes as noted in UFSAR Section

3.1.2.1.2, General Design Criterion 2, Design Basis for Protection Against Natural Phenomena. These facts notwithstanding, there is no plant labeling scheme useful for differentiating between non-safety supports that are within the scope of License Renewal and those that are not. Consequently, non-safety equipment supports in areas containing safety related equipment will be considered within the scope of License Renewal as a commodity group in accordance with procedure, and will receive subsequent aging management reviews.

Alternate/Backup Systems Credited in Mitigating Licensing Basis Events

The facility was constructed prior to the issuance of the 10 CFR 50 General Design Criteria and the Standard Review Plan. As a result of the NRCs Systematic Evaluation Program (SEP) as well as closure of Unresolved Safety Issues (USIs) and Generic Safety Issues (GSIs), numerous enhancements were made to bring the facility into conformance with the intent of the more modern design criteria. One of the fundamental concepts used by the Nuclear Regulatory Commission (NRC) in assessing the acceptability of the plant design included a review of the ability to achieve safe shutdown given a postulated SSC failure and an initiating event or transient. While those events are not included in the plant's accident analysis set, they are critical elements relied upon by the NRC to judge the safety adequacy of the facility and are essential elements of the CLB. The UFSAR and NUREG-0821 describe the results of the SEP review and any modifications that were required to achieve compliance with the NRCs Safety Evaluation Reports (SERs).

Intrinsic to the CLB are the modifications or strategies used to cope with the loss of safety related SSCs from a variety of events including: Fires, Internal and External Floods, Tornadoes, High Energy Line Breaks, Internally Generated Missiles, Masonry Block Wall Failures, and Seismic Events. Modifications or coping strategies were necessary because, in some areas, the facility was not constructed with sufficient physical separation to prevent the consequences of a single initiating event from damaging both trains of equipment in a mitigating system used to satisfy a safety function. The modifications are well documented and are included within the characterization of the structural assets evaluated in the Civil or Mechanical scoping reviews as appropriate.

Where modifications alone were not an appropriate solution, other equipment sets were selected, and that equipment proven acceptable to achieve safe shutdown given the consequences of the postulated failure. The basic methodology used included ensuring that the postulated failure could not introduce any safety consequences that could not be mitigated by surviving equipment, and identifying which mitigating equipment may be used as a surrogate or backup to the systems explicitly credited in mitigating Design Basis Events.

Example: The screen house was modified to add a dike bisecting the operating floor of the structure. The dike serves to protect vital equipment on the east side from the effects of a flood caused by a failure in the non-safety circulating water pumps on the west side. The dike is within the scope of License Renewal. The screen house can not be protected against the effects of a tornado. The plant had to modify other systems (fire water) to provide a backup cooling source for use of the emergency diesel generators to achieve safe shutdown in the event the screen house is lost. The backup cooling water is within the scope of License Renewal.

Example: Some block wall panels between the intermediate building and the turbine building are susceptible to failure in the event of high-energy line breaks or seismic events. This failure could render the preferred auxiliary feedwater system inoperable. An additional set of standby auxiliary feedwater pumps, not subject to the same failure mechanism, was installed for use in this scenario. In this example, the susceptible bock wall panels are not within the scope of License Renewal as Criterion 2 components even though the safety related preferred auxiliary feedwater pumps would be impaired by falling blocks, because the standby auxiliary feedwater pumps perform the required safety function. However, these very same wall panels are in scope as fire barriers (Criterion 3) and require aging management review appropriate to that function. In other areas of the structure, restraining/deflection devices were installed to prevent blocks from falling on safety equipment. The restraining/deflection devices are within the scope of License Renewal. The resolution of USI A-46 required the use of documented analysis and walkdowns to ensure that the systems needed to shut down the plant and maintain it in a safe condition for 72 hours can withstand a design-basis seismic event. The SSCs selected for achieving safe shutdown following a design basis seismic event are within the scope of License Renewal regardless of their safety classification.

2.1.5.4 Phase 2: Evaluation of Non-Safety SSCs for Failure Modes and Effects or Spatial Interactions not Explicitly Functionally Described in the CLB

After completion of the analytical scoping effort described in Section 2.1.5.3, further evaluations are performed. In plant areas containing safety related equipment, field verifications are performed to ascertain if any systems or system piping segments not already included within the scope of License Renewal from the analytical review are present. If a newly identified system or piping segment has a failure mode or effect that meets Criterion 2, that system or segment is included within the scope of License Renewal.

Example:The evaluation of Diesel Room A shows that the room contains: fuel oil, service water, jacket water, fire water, treated water, and house heating steam. House heating steam is a high energy system and the piping and equipment segments in the diesel room could fail such that the diesel is disabled. The other fluid services in the room are included within the scope of License Renewal based on the analytical review of their functions in supporting the Diesel Generator. House heating steam must now also be included based on the Spaces Review.

Based on these reviews, the License Renewal intended functions relative to the criteria of 10 CFR 54.4(a)(2) were identified and documented (as discussed in Section 2.1.3).

Thus, the scoping process used to identify non-safety SSCs whose failure could affect a safety function is consistent with and satisfies the criteria of 10 CFR 54.4(a)(2).

2.1.5.4.1 Use of Mitigative Features or Preventive Aging Management Techniques for Non-Safety Equipment Whose Failure Could Affect a Safety Function

During the review of SEP Topic III-5.B, “Pipe Breaks Outside Containment,” a comprehensive review of postulated high and moderate energy pipe breaks and cracks in accordance with BTP AJB 3-1 was performed. Postulated breaks and cracks were assumed to occur in the following systems, or portions of systems, not in the scope of license renewal per 10 CFR 54.4(a)(1) or (a)(3).

- Circulating Water in the Screen House and Turbine Building
- Heating Steam in the Screen House, Control Building battery rooms and relay room, Auxiliary Building and Diesel Building
- Feedwater, Condensate, and Main Steam in the Turbine Building
- Chilled Water in the Intermediate Building

Where protective measures were needed to assure safe shutdown capability, mitigative measures were used to cope with the postulated pipe failure in all except one situation. That case is the heating steam lines in the Diesel Building. In the former cases, the mitigative features have been placed within the scope of license renewal. In the latter case, as described in Section 2.1.5.4, all the steam heating piping and components in the Diesel Building have been placed in the scope of license renewal in accordance with the criteria specified in 10 CFR 54.4(a)(2).

Where mitigative features have been employed, those features are within the scope of license renewal and are characterized within the building that houses them. All other fluid system components designated as being within the scope of license renewal for 10 CFR 54.4(a)(2) employ preventive measures to manage aging. These system components are characterized in the system that contains them.

2.1.5.5 Other Scoping Pursuant to 10 CFR 54.4(a)(3) (Criterion 3)

10 CFR 54.4(a)(3) states that SSCs within the scope of License Renewal include all systems and structures relied on in safety analyses or plant evaluations to demonstrate compliance with the Commission’s regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63).

The analytical process used to review SSCs for 10 CFR 54.4(a)(3) applicability ensured that the UFSAR, Technical Specifications, design documents, design drawings and plant safety classifications were reviewed as applicable to ensure all SSCs credited for compliance with the regulated event set are identified. Specific scoping information based on each of these regulations is described in the following sections.

2.1.5.6 Fire Protection (FP)

UFSAR Section 9.5.1, Fire Protection Systems, describes the station fire protection and post fire safe shutdown equipment. All fire protection, detection, mitigation, confinement and safe shutdown equipment used at the station is subject to Criterion 3 scoping review.

Evaluations have been made of equipment needed to meet the fire protection requirements of Appendix A to Branch Technical Position ASB 9.5-1, as well as those needed to meet 10 CFR 50, Appendix R and 10 CFR 50.48. These evaluations are used as fire protection scoping basis documents. All structures and systems that contain components used for fire protection of the SSCs important to safety are within the scope of License Renewal.

NOTE: For commercial insurance purposes, many of the site structures not important to safety also have fire detection and mitigation. Where a fire protection system is not credited in the CLB as important to safety, that system and the SSCs it protects are not within the License Renewal Scope.

2.1.5.7 Environmental Qualification (EQ)

UFSAR Section 3.11, Environmental Design of Mechanical and Electrical Equipment, describes the station's process for Environmental Qualification (EQ). The master list of EQ components is detailed in site-specific procedures. All systems that contain components detailed on the EQ master equipment list are within the scope of License Renewal.

2.1.5.8 Pressurized Thermal Shock (PTS)

UFSAR Section 5.3.3.5, Pressurized Thermal Shock, describes the stations conformance to the 10 CFR 50.61 criterion.

RG&E has made two submittals to the NRC regarding Pressurized Thermal Shock (PTS). The first was made on January 23, 1986 in response to the original version of 10 CFR 50.61. The NRC SE on this submittal was issued on March 11, 1987 and concluded that the calculated reference temperature for pressurized thermal shock (RT_{PTS}) met 10 CFR 50.61 requirements. When 10 CFR 50.61 was amended on May 15, 1991, RG&E made a second submittal on February 13, 1992. The NRC dispositioned this submittal in their SER on the amendment to recapture the construction period for Ginna dated April 20, 1994. In this SER, the NRC found that the reference temperature found for pressurized thermal shock for the Ginna reactor vessel will be well below the screening criteria at the expiration of its current license. The RG&E submittals and NRC SERs did not identify the need for specific plant hardware modifications or reliance on other plant systems. Consequently only the reactor vessel is credited for PTS. Therefore, the only system relied upon for PTS is the Reactor Coolant System.

2.1.5.9 Anticipated Transients Without a Scram (ATWS)

UFSAR Section 7.2.6, Anticipated Transients Without a Scram Mitigation System Actuation Circuitry, describes the system installed to provide conformance with 10 CFR 50.62. All equipment in the system from the sensor output to the final actuation device that is credited for compliance with 10 CFR 50.62 is included in the scope of License Renewal. All systems that host those components are within scope of License Renewal.

2.1.5.10 Station Blackout (SBO)

UFSAR Section 8.1.4.5, Station Blackout Program, describes the station's methodology for coping with a station blackout. The coping strategy basis reference documents include both primary and alternative SSCs available to manage the event. Certain plant areas may heat up from SBO-induced loss of ventilation. Where heatup calculations take credit for structural elements as a heat sink, that structure is also considered a primary SBO mitigating component. The primary mitigating SSCs are within the scope of License Renewal. License Renewal regulatory guidance also mandates the inclusion of the plant system portion of the offsite power SSCs used for SBO recovery beyond those identified in the regulatory commitments made to satisfy 10 CFR 50.63 criteria. Systems and structures that provide a function for SBO coping and systems or structures that provide a function for recovery from an SBO condition in accordance with the current License Renewal regulatory interpretation are within the scope of License Renewal.

Based on these reviews, the License Renewal intended functions relative to the criteria of 10 CFR 54.4(a)(3) were identified and documented (as discussed in Section 2.1.3).

Thus, the scoping process used to identify systems and structures relied upon to mitigate the regulated events of concern is consistent with and satisfies the criteria of 10 CFR 54.4(a)(3).

2.1.6 Interim Staff Guidance Discussion

Recently during license renewal application reviews the NRC staff identified six issues for which additional staff and industry guidance clarification was necessary. They are:

1. Housing of Active Components
2. Concrete Aging Requirements
3. Interpretation of 10 CFR 54(a)2
4. Fire Protection Component Aging Management
5. Treatment of Electrical Fuse Holders
6. Scoping of Station Blackout Components

Following is a discussion of the general process used during the License Renewal Integrated Plant Assessment at Ginna Station for each of these areas:

Housing of Active Components

The Statements of Consideration for 10 CFR 54 provides the License Renewal Rule philosophy that, during the extended period of operation, safety-related functions should be maintained in the same manner and to the same extent as during the current licensing term. Examples of structures and components that perform passive functions are listed in 10 CFR 54.21(a)(1)(ii), which states, “These structures and components include, but are not limited to, pump casings, valve bodies . . .”

Pumps and valves were just an example here, meant to focus the AMR process on the passive function of an SSC. That passive function is not limited to the pressure boundary of the reactor coolant system. The exclusion of an SSC due to its active nature only applies to that portion of the SSC with an active function and not to those portions of the SSC with a passive function. Therefore, at Ginna Station, fan housings and fire damper housings are considered to be within scope and subject to an AMR.

Concrete Aging Requirements

As a result of the performance of AMRs for in-scope concrete components, Ginna Station has concluded that many of these components do not require aging management for the period of extended operation. This conclusion is based on a review of the material of construction, the environment, and industry and plant-specific operating experience for these components. However, for accessible concrete portions of the containment, Ginna Station has implemented the examination requirements and inspection intervals of ASME Section XI, Subsection IWL as an aging management program (AMP) for the period of extended operation. Other structures subject to an aging management review will receive similar inspections as part of the Structures Monitoring Program. The GALL report does not recommend further evaluation of concrete components in inaccessible areas for which the applicant can demonstrate a non-aggressive environment. The environment evident for the inaccessible concrete at Ginna Station is not aggressive; therefore, further evaluation of normally inaccessible structures will only be considered when excavations allow access or when aging effects on accessible concrete structures indicate that potential detrimental aging effects could also be occurring in inaccessible areas.

Interpretation of 10 CFR 54(a)2

10 CFR 54.4(a)(2) states that SSCs within the scope of License Renewal include non-safety related SSCs whose failure could prevent satisfactory accomplishment of any of the functions identified for safety related SSCs.

The analytical process that was used to review the Ginna Station SSCs for 10 CFR 54.4(a)(2) applicability ensured that the UFSAR, Technical Specifications, design documents, design drawings and the SSC safety classifications were reviewed as applicable to identify all non-safety SSC functional interactions where a non-safety SSC could fail and prevent the satisfactory accomplishment of a safety function.

However, reviewing design documents a system at a time did not provide information relative to system spatial interactions. Thus, it was possible that a functionally non-safety component whose failure could prevent a safety function could escape being designated as in-scope solely using the analytical review method.

To guard against such instances, the identification of SSCs meeting the criterion of 10 CFR 54.4(a)(2) was performed using a two-phase approach. Phase one was an analytical process, which evaluated SSCs meeting Criterion 2 that were explicitly identified in the Current Licensing Basis (CLB). Phase two was a plant spaces physical review, which evaluated SSCs for possible interactions that were not explicitly described in the CLB.

The result was synergistic in that the 54.4(a)(2) criterion was applied such that non-safety related SSCs were identified as being within the scope of license renewal when there was a potential either physically or spatially for interaction with the intended function of safety related equipment. A more detailed discussion of this phased methodology is available in Section 2.1.5.3 and Section 2.1.5.4.

Fire Protection Component Aging Management

In a January 28, 2002, letter from the NRC to NEI entitled, “Proposed Staff Guidance On Aging Management Of Fire Protection Systems For License Renewal,” the NRC provided changes to their previous guidance to the industry on aging management for passive SSCs comprising fire protection. As outlined in the program description in Section 2.2.10, Fire Protection Program, and Section 2.2.11, Fire Service Water System Program, Ginna Station intends to provide for aging management in a manner consistent with the proposed guidance.

This includes such guidance as to the performance of volumetric inspections and/or wall thickness evaluations, and visual as well as other techniques. For the Fire Water System Program, Ginna Station will continue to flow test those portions of the sprinkler system that are now routinely tested.

Treatment of Electrical Fuse Holders

Consistent with the requirements specified in 10 CFR 54.4(a), fuse holders (including fuse clips and fuse blocks) are considered to be passive electrical components. Fuse holders are scoped, screened, and included in the aging management review (AMR) in the same manner as terminal blocks and other types of electrical connections. However, fuse holders inside the enclosure of an active component, such as switchgear, power supplies, power inverters, battery chargers, and circuit boards, are considered to be piece parts of the larger assembly. Since piece parts and subcomponents in such an enclosure are inspected regularly and maintained as part of the Ginna Station normal maintenance and surveillance activities, they are considered not subject to an AMR. Fuse holders perform a primary function similar to electrical connections by providing an electrical circuit to deliver rated voltage, current, or signals. These intended functions meet the criteria of 10 CFR 54.4(a). Additionally, these intended functions are performed without moving parts or without a change in configuration or properties as described in 10 CFR 54.21 (a)(1)(i). Fuse holders are therefore passive, long-lived electrical components within the scope of license renewal and subject to an AMR. Therefore, aging management of the fuse holders is required for those cases where fuse holders are not considered piece parts of a larger assembly.

Scoping of Station Blackout Components

NRC guidance on this issue is as follows: “Consistent with the requirements specified in 10 CFR 54.4(a)(3) and 10 CFR 50.63 (a)(1), the plant system portion of the offsite power system should be included within the scope of license renewal.” Further clarification was provided which stated that, “the staff has determined that the plant system portion of the offsite power system that is used to connect the plant to the offsite power source should be included within the scope of the rule. This path typically includes the switchyard circuit breakers that connect to the offsite power system transformers (Startup transformers), the transformers themselves, the intervening overhead or underground circuits between circuit breaker and transformer and transformer and onsite electrical distribution system, and the associated control circuits and structures.”

UFSAR Section 8.1.4.5, Station Blackout Program, describes the station’s methodology for coping with a station blackout. The SSCs assumed to be necessary for the coping strategy including both primary and alternative SSCs available to manage the event are included within the scope of License Renewal.

As stated above the License Renewal regulatory guidance also mandates the inclusion of selected offsite power SSCs used for SBO recovery beyond those identified in the regulatory commitments made to satisfy 10 CFR 50.63 criteria. Therefore, systems and structures that provide a function for SBO coping and systems or structures that provide a function for recovery from an SBO condition in accordance with the current License Renewal regulatory interpretation are also considered within the scope of License Renewal. Additional specific information on this methodology is included in Section 2.1.5.10, Station Blackout.

2.1.7 Component Level Screening (Identification of Components Subject to Aging Management Review)

The requirement to identify SCs subject to an aging management review is specified in 10 CFR 54.21(a)(1) that states:

“Each application must contain the following information:

(a) An integrated plant assessment (IPA). The IPA must:

(1) For those systems, structures, and components within the scope of this part, as delineated in 10 CFR 54.4, identify and list those structures and components subject to an aging management review. Structures and components subject to an aging management review shall encompass those structures and components:

(i) That perform an intended function, as described in 10 CFR 54.4 without moving parts or without a change in configuration or properties. These structures and components include, but are not limited to, the reactor vessel, the reactor coolant system pressure boundary, steam generators, the pressurizer, piping, pump casings, valve bodies, the core shroud, component supports, pressure retaining boundaries, heat exchangers, ventilation ducts, the containment, the containment liner, electrical and mechanical penetrations, equipment hatches, seismic Category 1 structures, electrical cables and connections, cable trays, and electrical cabinets, excluding, but not limited to, pumps (except casing), valves (except body), motors, diesel generators, air compressors, snubbers, the control rod drive, ventilation dampers, pressure transmitters, pressure indicators, water level indicators, switchgears, cooling fans, transistors, batteries, breakers, relays, switches, power inverters, circuit boards, battery chargers, and power supplies; and

(ii) That are not subject to replacement based on a qualified life or specified time period.”

The screening portion of the IPA methodology is divided into three engineering disciplines; mechanical, civil/structural, and electrical/I&C. The relevant aspects of the component/structural component scoping and screening process for mechanical systems, civil structures, and electrical and I&C systems are described below.

For mechanical systems and civil structures, this process establishes evaluation boundaries, determines the SCs that compose the system or structure, determines which of those SCs support system/structure intended functions, and identifies specific SC intended functions. Consequently, not all of the SCs for in-scope systems or structures are in the scope of license renewal. Once the in-scope SCs are identified, the process determines which SCs are subject to an aging management review per the criteria of 10 CFR 54.21(a)(1). Note that the screening process is consistent with the NRC Staff's guidance on consumables provided in Table 2.1-3 of NUREG-1800 (Reference 6).

For electric/I&C systems, a bounding "Plant Spaces" approach as described in NEI 95-10 is taken. This approach establishes evaluation boundaries, determines the electrical and I&C component commodity groups within in-scope systems, identifies specific component and commodity-intended functions and then determines which component commodity groups are subject to an aging management review per the criteria of 10 CFR 54.21(a)(1). This approach calls for component scoping after screening has been performed.

2.1.7.1 Mechanical Systems

For mechanical systems, the component/structural component scoping and screening process is performed on each system identified to be within the scope of license renewal. This process evaluates the individual SCs included within in-scope mechanical systems to identify specific SCs or SC groups that require an aging management review. Electrical interface components associated with mechanical systems that are determined to be in scope are evaluated as described in Section 2.1.7.4, Electrical and I&C Systems.

Mechanical system evaluation boundaries were established for each system within the scope of license renewal. These boundaries were determined by mapping the pressure boundary associated with license renewal system intended functions onto the system flow diagrams. License renewal system intended functions are the functions a system must perform relative to the scoping criteria of 10 CFR 54.4(a)(1), 10 CFR 54.4(a)(2), and 10 CFR 54.4(a)(3).

The sequence of steps performed on each mechanical system determined to be within the scope of license renewal is as follows:

- Based on a review of design drawings and the system component list from the CMIS database, SCs that are included within the system are identified.
- Based on the plant level scoping results, the pressure boundary associated with license renewal system intended functions is mapped onto the system's flow diagrams.

NOTE: The LR evaluation markups for a system have typically been extended to the first normally closed manual valve, check valve or automatic valve that gets a signal to go closed. A normally open manual valve has also been used as a boundary in a few instances where a failure downstream of the valve has no short term effects, can be quickly detected, and the valve can be easily closed by operators to establish the pressure boundary prior to any adverse consequences. However, for SBO, Appendix R, high energy line break (HELB), and flooding events, the LR boundaries for a system have been defined consistent with the boundaries established in the CLB evaluations. Those boundaries do not always coincide with an isolation device. As described in Section 2.1.5.3, actual aging management evaluation boundaries may extend beyond the graphical depiction of the screening boundary.

- The system SCs that are within the scope of license renewal (i.e., required to perform a license renewal system intended function) are identified.
- Component intended functions for in-scope SCs are identified. The component intended functions identified are based on the guidance of NEI 95-10.
- The in-scope SCs that perform an intended function without moving parts or without a change in configuration or properties (screening criterion of 10 CFR 54.21(a)(1)(i)) are identified. Active/passive screening determinations are based on the guidance in Appendix B to NEI 95-10.
- The passive, in-scope SCs that are not subject to replacement based on a qualified life or specified time period (screening criterion of 10 CFR 54.21(a)(1)(ii)) are identified as requiring an aging management review. The determination of whether passive, in-scope SC has a qualified life or specified replacement time period was based on a review of plant-specific information, including the component database, maintenance programs, and procedures.

  Example: Some of the ventilation systems in the scope of license renewal include system filters such as fiberglass prefilter elements, HEPA filters and charcoal filters. These system filters are also in the scope of license renewal, but are periodically replaced and are not subject to aging management review. Periodic testing and inspection programs are in place to monitor filter performance such that system intended functions are maintained.

2.1.7.2 Civil Structures

For structures, the component/structural component scoping and screening process is performed on each structure identified to be within the scope of license renewal. This method evaluates the individual SCs included within in-scope structures to identify specific SCs or SC groups that require an aging management review.

The sequence of steps performed on each structure determined to be within the scope of license renewal is as follows:

- Based on a review of design drawings, the structure component list from the CMIS database and plant walkdowns, SCs that are included within the structure are identified. These SCs include items such as walls, pipe and equipment supports, conduit, cable trays, electrical enclosures, instrument panels, and related supports.
- The CLB is reviewed and compared to the walkdown results. Appurtenances such as flood barriers, missile shields, jet impingement shields, etc., relied upon in the licensing basis are verified as accounted for within a structure.
- The SCs that are within the scope of license renewal (i.e., required to perform license renewal system intended functions) are identified.
- Component intended functions for in-scope SCs are identified. The component intended functions identified are based on the guidance of NEI 95-10.
- The in-scope SCs that perform an intended function without moving parts or without a change in configuration or properties (screening criterion of 10 CFR 54.21(a)(1)(I)) are identified. Active/passive screening determinations are based on the guidance in Appendix B to NEI 95-10.
- The passive, in-scope SCs that are not subject to replacement based on a qualified life or specified time period (screening criterion of 10 CFR 54.21(a)(1)(ii)) are identified as requiring an aging management review. The determination of whether a passive, in-scope SC has a qualified life or specified replacement time period was based on a review of plant-specific information, including the component database, maintenance programs and procedures, vendor manuals, and plant experience.

NOTE: The structural component function(s) may support the intended function(s) of the structure or may have a unique function that does not support the intended function of the structure. A case in point is the spent fuel storage racks that are located in the Auxiliary Building. A unique function of the spent fuel storage racks is to maintain separation of the fuel assemblies to prevent criticality, which is not considered to be an intended function of the Auxiliary Building itself. The spent fuel racks are evaluated in the Spent Fuel Cooling and Fuel Storage System.

Structural steel, anchor bolts, base plates, etc. that are required to support non-safety related components to prevent physical interactions with safety related equipment are subject to aging management reviews. These components must remain in place such that they do not impact equipment that is required to perform a safety function in such a way as to prevent the equipment from performing its safety function.

Materials such as caulking and waterstops are identified generically. Limited situations may exist where these materials are important to maintaining the integrity of the components to which they are connected. The license renewal structure or component intended functions supported by these materials are restricted to two functions. These functions are:

1) Providing a rated fire barrier.

2) Providing a flood barrier.

Sealants and caulking that support the fire barrier function are addressed as part of the fire barrier penetration seals. Waterstops that support the flood barrier function are addressed with the wall or floor within which the sealant/waterstop is contained. Flood barriers are addressed in the buildings that contain them.

2.1.7.3 Structural Commodity Groups

Civil structures within the scope of License Renewal also house and support functionally unique features that may be included within the scope of the rule. These structural elements are best described and evaluated within structural commodity groupings.

Example: The Auxiliary Building contains hundreds of fire barriers and seals. Most of these barriers perform a license renewal intended function. Rather than listing all of the barrier numbers as a subset of the Auxiliary Building components, the barriers are binned together in a commodity group and structural screening is performed on the group.

The structural commodity screening evaluation groups are:

- Fire doors, barrier penetration seals and wraps
- Racks, panels and electrical enclosures, pipe and equipment supports.

2.1.7.4 Electrical and I&C Systems

Screening for electrical/ I&C components is performed on a generic component commodity group basis for all electrical/I&C systems, as well as the electrical/I&C component commodity groups associated with in-scope mechanical systems and civil structures. The methodology employed is consistent with the guidance in NEI 95-10 and NUREG-1800. Components within the scope of 10 CFR 50.49 (Environmental Qualification) are not subject to an aging management review

(AMR) based on the screening criteria of 10 CFR 54.21(a)(1)(ii). This is supported by NUREG-1800, Section 2.5.3. Additionally, each electrical and I&C system received a system level function review as described in Section 2.1.3. Although not necessary for determining the population of the passive long lived electrical components subject to aging management review, the system level review provides useful documentation of a systems role within the current licensing basis.

Screening for electrical and I&C system commodity groups used the plant spaces approach and the bounding review technique. Using this methodology, initially all passive long lived electrical and I&C commodity groups are considered subject to an AMR. The plant is segregated into areas where common, bounding environmental parameters can be assigned. Electrical commodity groups that have been installed in the area are identified and, among those, the commodity that represents the limiting aging characteristics is identified. The selected commodity is compared to the plant space service conditions and an assessment is made as to whether this limiting material will be able to maintain its function for the period of extended operation (i.e. receives an AMR). The results and conclusions of the AMR may indicate that aging management activities are required for components with specific material/environment combinations. Using these results, component specific scoping may be performed to limit the number of components for which aging management activities are required, or eliminate aging management activities altogether if nothing remains in the material/environment group population. Component scoping is performed by determining if a device served by the representative material commodity group has a License Renewal intended function and determining if credible failure modes would result in a loss of intended function. The intended function of the device being served can usually be determined by reviewing the intended functions of the devices being interconnected by the commodity group.

During electrical screening, passive, long-lived electrical/I&C sub-components that are part of larger components are evaluated as part of the larger component. For example, wires and connections internal to relays, relay racks, motors, motor control centers, and control boards are not considered as subject to aging management review. The sub-components are part of the active component and therefore are not evaluated separately. The boundary components for the electrical/I&C component review are the incoming 34.5 kV switchyard bus breakers and the generator step up transformer. These reference points represent the transition from site controlled power systems to the power systems maintained as part of the local distribution grid. All other components associated with offsite substations are not included in the evaluation boundary. This does not immediately exclude the on-site transformer yard and associated component commodity groups.

Based on a review of previous license renewal applications, NEI 95-10, and standard industry guidance for aging evaluation of electrical commodities, the following list represents the passive electrical/I&C component commodity groups:

- Insulated Cables and Connections (including splices, connectors, terminal blocks and fuse holders)
- Electrical Penetration Assemblies
- Phase Bus
- Switchyard Bus
- Transmission Conductors
- Uninsulated Ground Conductors
- High Voltage Insulators

A review of the UFSAR, the plant's database, and design basis documents was performed to validate the commodity group applicability to Ginna Station. The review did not identify a need to add any additional groups. For these commodities, the intended function is to electrically connect a specified section of an electrical circuit to deliver voltage, current or signal or, in the case of high voltage insulators, to insulate and support an electrical conductor.

Because the electrical screening and scoping is iterative in nature, the electrical and I&C commodities above are initially considered subject to aging management review.

2.1.7.5 Screening of Stored Equipment

In response to the NRC letter from Chris Grimes to Doug Walters (NEI) dated February 11, 1999, Subject: Screening of Equipment Kept in Storage, a review has been performed to identify equipment that (1) is maintained in storage, (2) is reserved for installation in the plant in response to a design basis event (DBE), and (3) requires an AMR. In addition to passive components, the review has also considered stored active components that are not routinely inspected, tested, and maintained.

The Appendix R stored equipment is used to restore power to pre-selected plant components and to provide cooling to certain areas after a fire in order to attain cold shutdown. The stored equipment identified as requiring an aging management review is cable and connectors.

2.1.7.6 Screening of Thermal Insulation

In response to NRC staff requests for additional information (RAI) on other license renewal applications, a screening review has been performed of thermal insulation. The review has concluded that only the thermal insulation used on the primary containment liner is included within the scope of License Renewal.

2.1.7.7 Identification of Short-lived Components and Consumables

The screening process has identified the passive components and structural members within the scope of license renewal. That process, as described in Section 2.1.7, has not attempted to identify those components that can be treated as short-lived and, therefore, do not require an AMR. These determinations have been made during the AMR process. It was during this part of the process that the plant procedures being credited for managing the effects of aging have been reviewed. If a procedure was found to provide for the periodic replacement of the component, or the component was found to have an established qualified life (e.g., for EQ purposes), the component has been identified as short-lived and an aging management review has not been required for that component. Consumables are a special class of short-lived items that can include packing, gaskets, component seals, O-rings, oil, grease, component filters, system filters, fire extinguishers, fire hoses, and air packs. Many types of consumables are part of a component such as a valve or a pump and, therefore, have not been identified during screening. Items potentially treatable as consumables have been evaluated consistent with the information presented in Table 2.1-3 of Reference 6. The results of that evaluation are presented below.

2.1.7.7.1 Packing, Gaskets, Component Seals, and O-Rings

Packing, gaskets, component mechanical seals, and O-rings are typically used to provide a leak-proof seal when components are mechanically joined together. These items are commonly found in components such as valves, pumps, heat exchangers, ventilation units/ducts, and piping segments. These types of consumables are considered subcomponents of the identified components and, therefore, are not subject to their own condition or performance monitoring. Therefore, the AMR for the component has included an evaluation of the sealing materials in those instances where it could not be demonstrated that one of the following conditions exist:

1. The sealing materials are short-lived because they are replaced on a fixed frequency or have a qualified life established (e.g., for EQ purposes), or
2. The sealing materials are not relied on in the CLB to maintain any of the following:

- Leakage below established limits
- System pressure high enough to deliver specified flow rates
- A pressure envelope for a space

Note: Sealants used to provide flood and fire barriers are addressed in Section 2.1.7.2.

2.1.7.7.2 **Oil, Grease, and Filters**

Oil, grease, and filters (both system and component filters) have been treated as consumables because either:

- A program for periodic replacement exists, or
- A monitoring program (e.g., predictive analysis activities, condition monitoring) exists that replaces these consumables, based on established performance criteria, when their condition begins to degrade but before there is a loss of intended function.

2.1.7.7.3 **Fire Extinguishers, Fire Hoses, and Air Packs**

Components such as fire hoses, fire extinguishers, self-contained breathing apparatus (SCBA), and SCBA cylinders are considered to be consumables and are routinely tested or inspected. The Fire Protection Program complies with the applicable safety standards (NFPA-10 for fire extinguishers; NFPA-1962 for fire hoses; 42 CFR 84, 29 CFR 19.10, 29 CFR 19.26, NUREG-41, and ANSI-Z88.2 for air packs), which specify performance and condition monitoring programs for these specific components. Fire hoses and fire extinguishers are inspected and hydrostatically tested periodically and must be replaced if they do not pass the test or inspection. SCBA and SCBA cylinders are inspected and periodically tested and must be replaced if they do not pass the test or inspection. The Fire Protection Program determines the replacement criterion of these components that are routinely checked by tests or inspections to assure operability. Therefore, while these consumables are in the scope of license renewal, they do not require an AMR.

Section 2.1 References

1. Regulatory Guide 1.97, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.
2. NEI 95-10, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 -The License Renewal Rule, Rev. 3, Nuclear Energy Institute, March 2001.
3. Branch Technical Position (BTP) APCSB 9.5-1, Appendix A, Guidelines for Fire Protection for Nuclear Power Plants August 23, 1976.
4. Letter of August 5, 1999 from Christopher I. Grimes of the NRC to Douglas J. Walters of NEI, Subject: License Renewal Issue No. 98-0082, Scoping Guidance.
5. Letter of February 11, 1999 from Christopher I. Grimes of the NRC to Doug Walters of NEI, Subject: Request for Additional Information Regarding Generic License Renewal Issue No. 98-0102, Screening of Equipment that is Kept in Storage.
6. Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, NUREG-1800, U.S. Nuclear Regulatory Commission, July 2001.

Table 2.1-1 System Function Codes

System Function Code	Function Description	ANSI 51.1 Discussion Paragraph	Notes	10 CFR 54.4(a) Criterion
A	MAINTAIN REACTOR CORE ASSEMBLY GEOMETRY	4.1.1, 4.4.1		1
B	INTRODUCE EMERGENCY NEGATIVE REACTIVITY TO MAKE THE REACTOR SUBCRITICAL	4.2.1, 4.8.1	Functions include limiting the introduction of positive reactivity.	1
C	INTRODUCE NEGATIVE REACTIVITY TO ACHIEVE OR MAINTAIN SUBCRITICAL REACTOR CONDITION	4.2.1		1
D	SENSE OR PROVIDE PROCESS CONDITIONS AND GENERATE SIGNALS FOR REACTOR TRIP AND ENGINEERED SAFETY FEATURES ACTUATION	4.3.1		1
E	PROVIDE REACTOR COOLANT PRESSURE BOUNDARY	4.4.1		1
F	REMOVE RESIDUAL HEAT FROM THE RCS	4.5.1	Residual heat removal by direct recirculation of reactor coolant. This system function does not address emergency core cooling via Engineered Safety Features Actuation.	1
G	PROVIDE EMERGENCY CORE COOLANT WHERE THE ECCS PROVIDES COOLANT DIRECTLY TO THE CORE	4.8.1	This function includes coolant that is provided to the core via RCS piping. This function also addresses coolant inventory that is maintained for use by the ECCS to provide emergency core cooling and to introduce negative reactivity.	1

Table 2.1-1 System Function Codes

System Function Code	Function Description	ANSI 51.1 Discussion Paragraph	Notes	10 CFR 54.4(a) Criterion
H	PROVIDE EMERGENCY HEAT REMOVAL FROM THE REACTOR COOLANT SYSTEM USING SECONDARY HEAT REMOVAL CAPABILITY	4.10.1	Secondary heat removal capability addresses the secondary side of the steam generators, and steam generator secondary cooling paths, for example: Auxiliary Feedwater, Relief Valves/Lines.	1
J	PROVIDE HEAT REMOVAL FROM SAFETY RELATED HEAT EXCHANGERS	4.7.1	This function addresses heat removal as necessary to provide a nuclear safety function. This function does not include emergency heat removal from the secondary side of the Steam Generators (System Function H).	1
K	PROVIDE PRIMARY CONTAINMENT BOUNDARY	4.9.1	This function addresses any primary containment fission product barrier or primary containment radioactive material holdup or isolation.	1
L	PROVIDE EMERGENCY HEAT REMOVAL FROM PRIMARY CONTAINMENT AND PROVIDE CONTAINMENT PRESSURE CONTROL	4.11.1		1
M	PROVIDE EMERGENCY REMOVAL OF RADIOACTIVE MATERIAL FROM THE PRIMARY CONTAINMENT ATMOSPHERE	4.11.1		1
N	CONTROL COMBUSTIBLE GAS MIXTURES IN THE PRIMARY CONTAINMENT ATMOSPHERE	4.11.1		1

Table 2.1-1 System Function Codes

System Function Code	Function Description	ANSI 51.1 Discussion Paragraph	Notes	10 CFR 54.4(a) Criterion
O	MAINTAIN EMERGENCY TEMPERATURES WITHIN AREAS CONTAINING SAFETY CLASS 1,2,3 COMPONENTS	4.12.1		1
P	ENSURE ADEQUATE COOLING IN THE SPENT FUEL POOL	4.13.1	Cooling to maintain stored fuel within acceptable temperature limits.	1
Q	PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1,2,3 COMPONENTS	4.14.1		1
R	STRUCTURALLY SUPPORT OR HOUSE SAFETY CLASS 1,2,3 COMPONENTS	4.18.1		1
S	SPECIAL CAPABILITY CLASS FUNCTIONS	N/A	Components within the system are Safety Significant (augmented quality). For the purposes of License Renewal, components which are special capability class are treated under the Criterion 3 Codes Z1 through Z5.	N/A
T	NON-NUCLEAR SAFETY CLASS FUNCTIONS	N/A		N/A
X	SFR FUNCTION NOT APPLICABLE AT COMPONENT LEVEL	N/A	Not a system level function. Component performs a safety related function beyond the boundaries of the respective system specific design, such as accident monitoring.	1

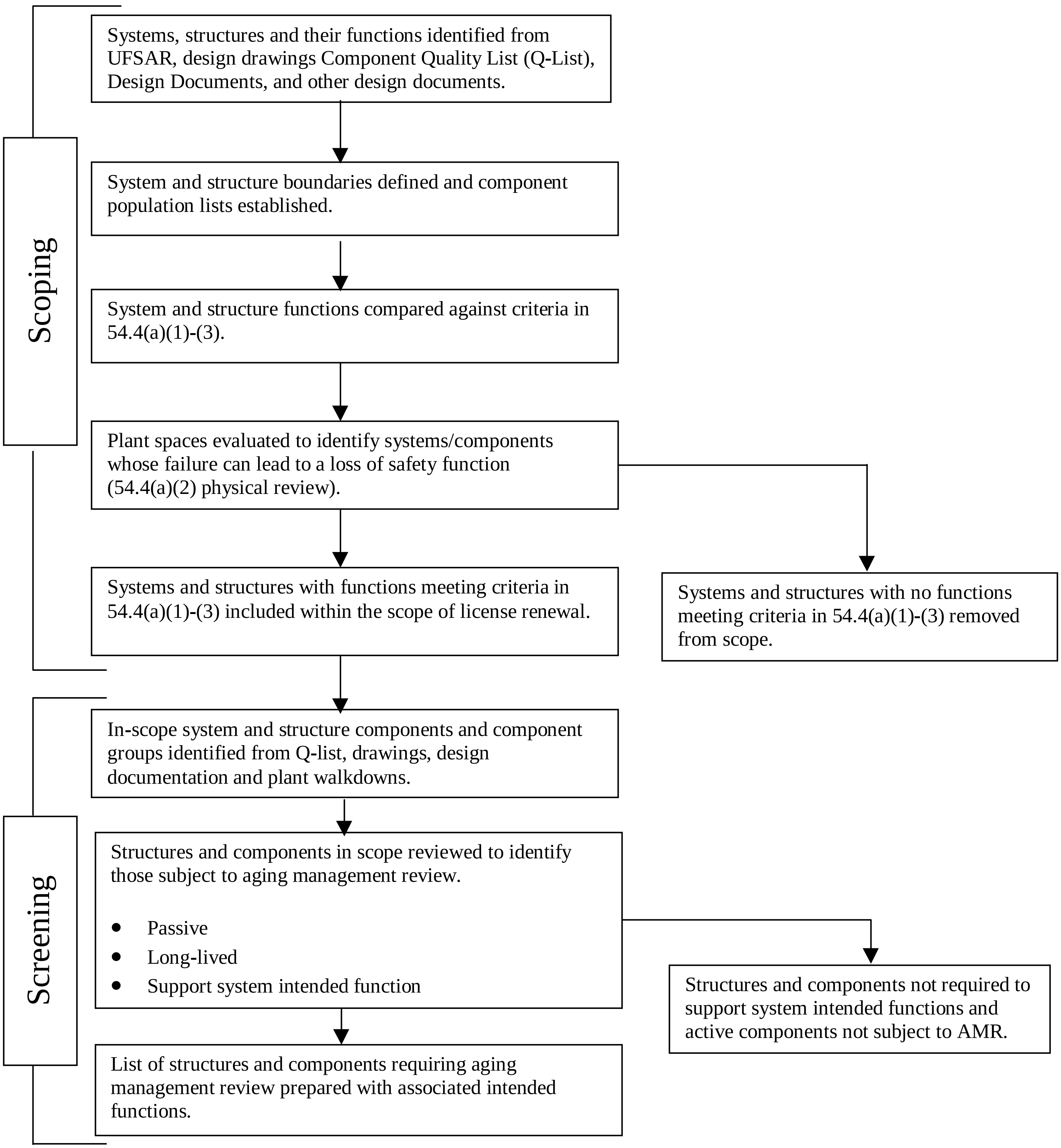
Table 2.1-1 System Function Codes

System Function Code	Function Description	ANSI 51.1 Discussion Paragraph	Notes	10 CFR 54.4(a) Criterion
Y	LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSCs WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	N/A		2
Z1	LICENSE RENEWAL CRITERION 3 - SSCs RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	N/A		3
Z2	LICENSE RENEWAL CRITERION 3 - SSCs RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ENVIRONMENTAL QUALIFICATION (10 CFR 50.49)	N/A		3
Z3	LICENSE RENEWAL CRITERION 3 - SSCs RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR PRESSURIZED THERMAL SHOCK (10 CFR 50.61)	N/A		3

Table 2.1-1 System Function Codes

System Function Code	Function Description	ANSI 51.1 Discussion Paragraph	Notes	10 CFR 54.4(a) Criterion
Z4	LICENSE RENEWAL CRITERION 3 - SSCs RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ANTICIPATED TRANSIENTS WITHOUT SCRAM (10 CFR 50.62)	N/A		3
Z5	LICENSE RENEWAL CRITERION 3 - SSCs RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR STATION BLACKOUT (10 CFR 50.63)	N/A		3

Figure 2.1-1 Scoping and Screening Process Overview



2.2 Plant Level Scoping Results

The systems, structures, and commodities at Ginna Station were evaluated as to whether they were within the scope of license renewal, using the methodology described in Section 2.1. The results are shown below in Table 2.2-1.

Table 2.2-1 Plant Level Scoping Results

Description	Within Scope of License Renewal?	Comments
SRP Evaluation Group: Reactor Vessel, Internals, and Reactor Coolant System		
Reactor Coolant, Core, and Internals (Section 2.3.1.1)	Yes	The Regenerative and Letdown Heat Exchangers are included in this system.
Non-Class 1 RCS Components (Section 2.3.1.6)	Yes	
SRP Evaluation Group: Engineered Safety Features		
Containment Hydrogen Detectors and Recombiners (Section 2.3.2.4)	Yes	
Containment Isolation Components (Section 2.3.2.5)	Yes	Piping and components in the Heating Steam System, Plant Air Systems, Plant Sampling Systems, and Containment Structure that are associated with the primary containment boundary are included in CICS.
Containment Spray (Section 2.3.2.2)	Yes	
Residual Heat Removal (Section 2.3.2.3)	Yes	
Safety Injection (Section 2.3.2.1)	Yes	
SRP Evaluation Group: Auxiliary Systems		
Chemical and Volume Control (Section 2.3.3.1)	Yes	The Regenerative and Letdown Heat Exchangers are included in the Reactor Coolant, Core, and Internals system.
Chilled Water (Section 2.3.3.15)	No	

Table 2.2-1 Plant Level Scoping Results

Description	Within Scope of License Renewal?	Comments
Circulating Water (Section 2.3.3.14)	No	Those portions of the CW system that support the delivery of lake water sufficient for the use of service water and fire water pumps are evaluated within the Service Water System. The interface with the Circulating Water System that provides circulating water flood detection is evaluated within the Reactor Protection System.
Component Cooling Water (Section 2.3.3.2)	Yes	
Containment Ventilation Systems (Section 2.3.3.9)	Yes	
Cranes, Hoists, and Lifting Devices (Section 2.3.3.11)	Yes	NUREG-0612 cranes are included in this group.
Essential Ventilation Systems (Section 2.3.3.10)	Yes	
Emergency Power (Section 2.3.3.8)	Yes	
Fire Protection (Section 2.3.3.6)	Yes	Fire protection commodity group items are included within this system.
Fuel Handling (Section 2.3.3.16)	No	Cranes, new and spent fuel storage racks, the spent fuel pool and cavity liners are evaluated separately.
Heating Steam (Section 2.3.3.7)	Yes	Piping and components associated with the primary containment boundary are included with CICS.
Non-Essential Ventilation Systems (Section 2.3.3.19)	No	Components that act as fire barriers (fire dampers) are included in the Fire Barrier commodity group.
Plant Air Systems (Section 2.3.3.18)	No	Piping and components associated with the primary containment boundary are included with CICS.

Table 2.2-1 Plant Level Scoping Results

Description	Within Scope of License Renewal?	Comments
Plant Sampling Systems (Section 2.3.3.17)	No	Piping and components associated with the primary containment boundary are included with CICS.
Radiation Monitoring (Section 2.3.3.13)	Yes	
Service Water (Section 2.3.3.5)	Yes	Those portions of the CW system that support the delivery of lake water sufficient for the use of service water and fire water pumps are evaluated within the Service Water System.
Site Service and Facility Support (Section 2.3.3.20)	No	
Spent Fuel Cooling and Fuel Storage (Section 2.3.3.3)	Yes	New and spent fuel storage racks, the spent fuel pool, transfer tube, and cavity liners are included in this group.
Treated Water (Section 2.3.3.12)	Yes	
Waste Disposal (Section 2.3.3.4)	Yes	
SRP Evaluation Group: Steam and Power Conversion System		
Auxiliary Feedwater (Section 2.3.4.3)	Yes	
Feedwater and Condensate (Section 2.3.4.2)	Yes	
Main and Auxiliary Steam (Section 2.3.4.1)	Yes	
Turbine-Generator and Supporting Systems (Section 2.3.4.4)	Yes	
SRP Evaluation Group: Containments, Structures and Component Supports		
All Volatile Water Treatment Building (Section 2.4.2.6)	Yes	

Table 2.2-1 Plant Level Scoping Results

Description	Within Scope of License Renewal?	Comments
Auxiliary Building (Section 2.4.2.1)	Yes	
Cable Tunnel (Section 2.4.2.10)	Yes	
Component Supports Commodity Group (Section 2.4.2.12)	Yes	Commodity grouping associated with Essential Buildings and Yard Structures.
Containment Structures (Section 2.4.1)	Yes	
Control Building (Section 2.4.2.5)	Yes	
Diesel Building (Section 2.4.2.4)	Yes	
Essential Buildings and Yard Structures (Section 2.4.2)	Yes	
Essential Yard Structures (Section 2.4.2.11)	Yes	
Intermediate Building (Section 2.4.2.2)	Yes	
Non-Essential Buildings and Yard Structures (Section 2.4.3)	No	
Screen House Building (Section 2.4.2.7)	Yes	
Service Building (Section 2.4.2.9)	Yes	
Standby Auxiliary Feedwater Building (Section 2.4.2.8)	Yes	
Turbine Building (Section 2.4.2.3)	Yes	
SRP Evaluation Group: Electrical Components		
120 VAC Vital Instrument Buses (Section 2.5.2)	Yes	
125 VDC Power (Section 2.5.3)	Yes	

Table 2.2-1 Plant Level Scoping Results

Description	Within Scope of License Renewal?	Comments
4160 VAC Power (Section 2.5.4)	Yes	
480 VAC Power (Section 2.5.5)	Yes	
Control Rod Drive and Nuclear Process Instruments (Section 2.5.6)	Yes	
Engineered Safety Features Actuation (Section 2.5.10)	Yes	
Misc. AC Power and Lighting (Section 2.5.7)	Yes	
Offsite Power (Section 2.5.8)	Yes	Those portions of the Offsite Power System physically located on site are included within this group.
Plant Communications (Section 2.5.11)	Yes	
Plant Process Computers (Section 2.5.12)	No	
Plant Security (Section 2.5.13)	No	
Reactor Protection (Section 2.5.9)	Yes	Those portions of the Circulating Water System that provide circulating water flood detection are evaluated within the Reactor Protection System.
Seismic and Meteorological Instrumentation (Section 2.5.14)	No	

2.3 Scoping and Screening Results: Mechanical Systems

2.3.1 Reactor Coolant System

2.3.1.1 Reactor Coolant (Class 1)

System Description

NOTE: Reactor Coolant System (RCS) Class 1 components, Steam Generators, the Pressurizer, and the Reactor Vessel are reviewed and evaluated as unique specific topical areas. For clarity purposes the system drawings for Class 1 RCS components include showing the above RCS equipment, with the Class 1 portion clearly denoted with flags.

The Reactor Coolant System transports the heat generated in the reactor core to secondary heat removal systems. The RCS system also acts in conjunction with the fuel and the primary containment system to provide defense in-depth with respect to preventing fission products from escaping to the environment. Consequently the RCS is associated with mitigating virtually all accidents, transients and events.

The principal components of the Reactor Coolant System include the reactor vessel, pressurizer, steam generators, reactor coolant pumps, and the essential class 1 piping and valves (including the regenerative and letdown heat exchangers). The Reactor Coolant System consists of two identical heat transfer loops connected in parallel to the reactor vessel. Each loop contains a circulating pump and a steam generator.

Aging management reviews for the following system components were performed using the Westinghouse aging management review WCAPs and the corresponding applicant action item requirements detailed in the appropriate NRC Safety Evaluation Report:

- Reactor Vessel
- Reactor Vessel Internals
- Reactor Coolant System Class 1
- Reactor Coolant System Non-Class 1
- Pressurizer
- Steam Generator

The following fluid systems interface with the Reactor Coolant System:

Plant Sampling
Residual Heat Removal
Chemical and Volume Control
Waste Disposal
Safety Injection
Component Cooling Water

The following RCS subsystems system descriptions are provided below for further detail:

- Reactor Vessel - Section 2.3.1.2
- Reactor Vessel Internals - Section 2.3.1.3
- Pressurizer - Section 2.3.1.4
- Steam Generators - Section 2.3.1.5

System Function Listing

In addition to the System Functions described above, the Reactor Coolant System also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code A MAINTAIN REACTOR CORE ASSEMBLY GEOMETRY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Reactor Coolant, Core, and Internals system perform this primary design system function. (Reactor Vessel internals)

Code B INTRODUCE EMERGENCY NEGATIVE REACTIVITY TO MAKE THE REACTOR SUBCRITICAL	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Reactor Coolant, Core, and Internals system perform this primary design system function. (Control Rods)

Code D SENSE OR PROVIDE PROCESS CONDITIONS AND GENERATE SIGNALS FOR REACTOR TRIP AND ENGINEERED SAFETY FEATURES ACTUATION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Reactor Coolant, Core, and Internals system perform this primary design system function. Reactor Coolant temperature detectors are included as Class 1 boundaries.

Code E PROVIDE REACTOR COOLANT PRESSURE BOUNDARY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Reactor Coolant, Core, and Internals system perform this primary design system function.

Code F REMOVE RESIDUAL HEAT FROM THE RCS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Reactor Coolant, Core, and Internals system perform this associated design system function.

Code G PROVIDE EMERGENCY CORE COOLANT WHERE THE ECCS PROVIDES COOLANT DIRECTLY TO THE CORE	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Reactor Coolant, Core, and Internals system perform this associated design system function.

Code H PROVIDE EMERGENCY HEAT REMOVAL FROM THE REACTOR COOLANT SYSTEM USING SECONDARY HEAT REMOVAL CAPABILITY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Reactor Coolant, Core, and Internals system perform this primary design system function. (Steam Generators)

Code J PROVIDE HEAT REMOVAL FROM SAFETY RELATED HEAT EXCHANGERS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Reactor Coolant, Core, and Internals system perform this associated design system function. (Excess Letdown Heat Exchanger and RCP thermal barrier heat exchangers)

Code K PROVIDE PRIMARY CONTAINMENT BOUNDARY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Reactor Coolant, Core, and Internals system perform this associated design system function.

Code L PROVIDE EMERGENCY HEAT REMOVAL FROM PRIMARY CONTAINMENT AND PROVIDE CONTAINMENT PRESSURE CONTROL	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Reactor Coolant, Core, and Internals system perform this associated design system function

Code R	Cri 1	Cri 2	Cri 3				
STRUCTURALLY SUPPORT OR HOUSE SAFETY CLASS 1, 2, 3 COMPONENTS			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Reactor Coolant, Core, and Internals system perform this primary design system function. (Reactor Vessel)

Code S	Cri 1	Cri 2	Cri 3				
SPECIAL CAPABILITY CLASS FUNCTIONS			FP	EQ	PTS	AT	SB

Comment: Components within the Reactor Coolant, Core, and Internals system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Reactor Coolant, Core, and Internals system that perform special capability class functions are tracked under the Criterion 3 codes (Z1 through Z5). Reg Guide 1.97 Category 3 post accident monitoring variables and mid-loop level detection

Code X	Cri 1	Cri 2	Cri 3				
SFR FUNCTION NOT APPLICABLE AT COMPONENT LEVEL			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Reactor Coolant, Core, and Internals system perform specific safety related functions different from and in addition to the system level functions (e.g. Reg Guide 1.97 Category 1).

Code Z1	Cri 1	Cri 2	Cri 3				
LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Reactor Coolant, Core, and Internals system perform this function.

Code Z2	Cri 1	Cri 2	Cri 3				
LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ENVIRONMENTAL QUALIFICATION (10 CFR 50.49)			FP	EQ	PTS	AT	SB
				X			

Comment: Components within the Reactor Coolant, Core, and Internals system perform this function.

Code Z3 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR PRESSURIZED THERMAL SHOCK (10 CFR 50.61)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
					X		

Comment: Components within the Reactor Coolant, Core, and Internals system perform this function

Code Z5 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR STATION BLACKOUT (10 CFR 50.63)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: Components within the Reactor Coolant, Core, and Internals system perform this function.

UFSAR Reference

Additional Reactor Coolant System details are provided in Section 5.1, Section 5.2, and Section 5.4 of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Reactor Coolant System are listed below:

33013-1246,1
33013-1247
33013-1258
33013-1260
33013-1262,2
33013-1263
33013-1264
33013-1265,1
33013-1278,1
33013-2248
33013-2278

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.1-1 along with each Component Group’s passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.1-1 Reactor Coolant (Class 1)

Component Group	Passive Function	Aging Management Reference
Class 1		
VALVES ≥ 4 IN. NPS	PRESSURE BOUNDARY	Table 3.2-1 Line Number (1) Table 3.2-1 Line Number (19) Table 3.2-1 Line Number (24) Table 3.2-2 Line Number (13) Table 3.2-2 Line Number (14) Table 3.2-2 Line Number (34)
VALVES < 4 IN. NPS	PRESSURE BOUNDARY	Table 3.2-1 Line Number (1) Table 3.2-1 Line Number (19) Table 3.2-1 Line Number (24) Table 3.2-2 Line Number (13) Table 3.2-2 Line Number (14) Table 3.2-2 Line Number (34)
REACTOR COOLANT PUMPS (CASING AND MAIN FLANGE)	PRESSURE BOUNDARY	Table 3.2-1 Line Number (1) Table 3.2-1 Line Number (19) Table 3.2-1 Line Number (24) Table 3.2-2 Line Number (34)
REACTOR COOLANT PUMPS (THERMAL BARRIER FLANGE)	PRESSURE BOUNDARY	Table 3.2-2 Line Number (15) Table 3.2-2 Line Number (34)
THERMAL BARRIER HEAT EXCHANGER TUBING	PRESSURE BOUNDARY HEAT TRANSFER	Table 3.2-2 Line Number (15) This applies to both passive functions.
ORIFICES AND REDUCERS	PRESSURE BOUNDARY THROTTLING	Table 3.2-2 Line Number (15) This applies to both passive functions. Table 3.2-2 Line Number (34) This applies only to the pressure boundary passive function.
PIPING AND FITTINGS ≥ 4 IN. NPS	PRESSURE BOUNDARY	Table 3.2-1 Line Number (1) Table 3.2-1 Line Number (24) Table 3.2-2 Line Number (34)

Table 2.3.1-1 Reactor Coolant (Class 1)

Component Group	Passive Function	Aging Management Reference
PRIMARY LOOP ELBOWS	PRESSURE BOUNDARY	Table 3.2-1 Line Number (1) Table 3.2-1 Line Number (10) Table 3.2-1 Line Number (20) Table 3.2-2 Line Number (13) Table 3.2-2 Line Number (34)
PIPING AND FITTINGS < 4 IN. NPS	PRESSURE BOUNDARY	Table 3.2-1 Line Number (1) Table 3.2-1 Line Number (6) Table 3.2-1 Line Number (24) Table 3.2-2 Line Number (34)
BOLTING FOR FLANGED PIPING JOINTS, RCP'S, AND VALVE CLOSURES	MECHANICAL CLOSURE INTEGRITY	Table 3.2-1 Line Number (22) Table 3.2-1 Line Number (26)
REACTOR COOLANT PUMP LUGS	PRESSURE BOUNDARY	Table 3.2-2 Line Number (16) Table 3.2-2 Line Number (34)

2.3.1.2 Reactor Vessel

Component Description

The Ginna Station RPV, as the principal component of the RCS, contains the heat-generating core and associated supports, controls and instrumentation, and coolant circulating channels. Primary outlet and inlet nozzles provide for the exit of heated coolant and its return to the RPV for recirculation through the core.

The Ginna Station RPV consists of a cylindrical shell with a hemispherical bottom head and a flanged and gasketed removable upper head. The RPV shell is fabricated from integral ring forgings joined by circumferential welds. The RPV contains the core, core support structures, rod control clusters, thermal shield or neutron shield panels, and other parts directly associated with the core. Inlet and outlet nozzles are located at an elevation between the head flange and the core. The body of the RPV is low-alloy carbon steel, and the inside surfaces in contact with coolant are clad with austenitic stainless steel to minimize corrosion. The RPV is supported by steel pads integral with the coolant nozzles. The pads rest on steel base plates atop a support structure attached to the concrete foundation.

UFSAR Reference

Additional Reactor Vessel details are provided in Section 5.3 of the UFSAR.

Subcomponents Subject to an Aging Management Review

The subcomponents of the Reactor Vessel that require aging management review are indicated in Table 2.3.1-2 along with each subcomponent's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.1-2 Reactor Vessel

Subcomponent	Passive Function	Aging Management Reference
CRDM ROD TRAVEL HOUSINGS	PRESSURE BOUNDARY	Table 3.2-1 Line Number (24) Table 3.2-2 Line Number (34)
CRDM LATCH HOUSINGS	PRESSURE BOUNDARY	Table 3.2-1 Line Number (24) Table 3.2-2 Line Number (34)
CRDM HOUSING TUBES (HEAD ADAPTERS)	PRESSURE BOUNDARY	Table 3.2-1 Line Number (23) Table 3.2-2 Line Number (2) Table 3.2-2 Line Number (34)
VENT PIPE	PRESSURE BOUNDARY	Table 3.2-1 Line Number (23) Table 3.2-2 Line Number (2) Table 3.2-2 Line Number (34)
CLOSURE HEAD DOME	PRESSURE BOUNDARY	Table 3.2-1 Line Number (26) Table 3.2-2 Line Number (1) Table 3.2-2 Line Number (2) Table 3.2-2 Line Number (33)
CLOSURE HEAD FLANGE	PRESSURE BOUNDARY	Table 3.2-1 Line Number (26) Table 3.2-2 Line Number (1) Table 3.2-2 Line Number (2) Table 3.2-2 Line Number (33)
VESSEL FLANGE	PRESSURE BOUNDARY SUPPORT RV INTERNALS	Table 3.2-1 Line Number (26) Table 3.2-1 Line Number (28) Table 3.2-2 Line Number (1) Table 3.2-2 Line Number (2) These apply to both passive functions.

Table 2.3.1-2 Reactor Vessel

Subcomponent	Passive Function	Aging Management Reference
O-RING LEAK MONITOR TUBES	PRESSURE BOUNDARY	Table 3.2-1 Line Number (9) Table 3.2-2 Line Number (34)
UPPER SHELL	PRESSURE BOUNDARY	Table 3.2-1 Line Number (4) Table 3.2-2 Line Number (1) Table 3.2-2 Line Number (2) Table 3.2-2 Line Number (4) Table 3.2-2 Line Number (5) Table 3.2-2 Line Number (33)
PRIMARY INLET NOZZLES	PRESSURE BOUNDARY	Table 3.2-1 Line Number (4) Table 3.2-2 Line Number (1) Table 3.2-2 Line Number (2) Table 3.2-2 Line Number (4) Table 3.2-2 Line Number (5) Table 3.2-2 Line Number (33)
PRIMARY OUTLET NOZZLES	PRESSURE BOUNDARY	Table 3.2-1 Line Number (4) Table 3.2-2 Line Number (1) Table 3.2-2 Line Number (2) Table 3.2-2 Line Number (4) Table 3.2-2 Line Number (5) Table 3.2-2 Line Number (33)
PRIMARY NOZZLE SAFE ENDS	PRESSURE BOUNDARY	Table 3.2-1 Line Number (10) Table 3.2-1 Line Number (24) Table 3.2-2 Line Number (2) Table 3.2-2 Line Number (34)
INTERMEDIATE SHELL (INCLUDING CIRCUMFERENTIAL WELD)	PRESSURE BOUNDARY	Table 3.2-1 Line Number (3) Table 3.2-1 Line Number (4) Table 3.2-2 Line Number (1) Table 3.2-2 Line Number (2) Table 3.2-2 Line Number (5) Table 3.2-2 Line Number (33)
LOWER SHELL	PRESSURE BOUNDARY	Table 3.2-1 Line Number (4) Table 3.2-2 Line Number (1) Table 3.2-2 Line Number (2) Table 3.2-2 Line Number (5) Table 3.2-2 Line Number (33)

Table 2.3.1-2 Reactor Vessel

Subcomponent	Passive Function	Aging Management Reference
CORE SUPPORT LUGS	SUPPORT RV INTERNALS	Table 3.2-1 Line Number (9) Table 3.2-2 Line Number (2) Table 3.2-2 Line Number (3)
BOTTOM HEAD TORUS	PRESSURE BOUNDARY SUPPORT RV INTERNALS	Table 3.2-2 Line Number (1) Table 3.2-2 Line Number (2) Table 3.2-2 Line Number (5) Table 3.2-2 Line Number (33) These apply to both passive functions.
BOTTOM HEAD DOME	PRESSURE BOUNDARY SUPPORT RV INTERNALS	Table 3.2-2 Line Number (1) Table 3.2-2 Line Number (2) Table 3.2-2 Line Number (5) Table 3.2-2 Line Number (33) These apply to both passive functions.
INSTRUMENTATION TUBES AND SAFE ENDS	PRESSURE BOUNDARY SUPPORT THIMBLE TUBES	Table 3.2-1 Line Number (9) Table 3.2-2 Line Number (2) Table 3.2-2 Line Number (34) These apply to both passive functions.
BMI GUIDE TUBES	PRESSURE BOUNDARY SUPPORT THIMBLE TUBES	Table 3.2-1 Line Number (8) Table 3.2-2 Line Number (1) Table 3.2-2 Line Number (2) Table 3.2-2 Line Number (34) These apply to both passive functions.
SEAL TABLE FITTINGS	PRESSURE BOUNDARY SUPPORT THIMBLE TUBES	Table 3.2-2 Line Number (1) Table 3.2-2 Line Number (34) These apply to both passive functions.
VENTILATION SHROUD SUPPORT RING	STRUCTURAL SUPPORT	Table 3.2-2 Line Number (5) Table 3.2-2 Line Number (33)
CLOSURE STUDS, NUTS, AND WASHERS	MECHANICAL CLOSURE INTEGRITY	Table 3.2-1 Line Number (18) Table 3.2-1 Line Number (26) Table 3.2-1 Line Number (35) Table 3.2-2 Line Number (6)

Table 2.3.1-2 Reactor Vessel

Subcomponent	Passive Function	Aging Management Reference
REFUELING SEAL LEDGE	STRUCTURAL SUPPORT	Table 3.2-2 Line Number (5) Table 3.2-2 Line Number (33)
NOZZLE SUPPORT PADS	STRUCTURAL SUPPORT	Table 3.2-1 Line Number (26) Table 3.2-2 Line Number (33)

2.3.1.3 Reactor Vessel Internals

Component Description

The Ginna Station RVIs consist of two basic assemblies, i.e.,

- Upper internals assembly that is removed during each refueling operation to obtain access to the reactor core. The top of this assembly is clamped to a ledge below the vessel-head mating surface by the reactor vessel head. The core barrel fuel alignment pins of the lower internals assembly guides the bottom of the upper internals assembly.
- Lower internals assembly that can be removed, if desired following a complete core unload. This assembly is clamped at the same ledge below the vessel-head mating surface and closely guided at the bottom by radial/clevis assemblies.

UFSAR Reference

Additional Reactor Vessel Internals details are provided in Section 3.9.5 and Section 4.2.1 of the UFSAR.

Subcomponents Subject to an Aging Management Review

The subcomponents of the Reactor Vessel Internals that require aging management review are indicated in Table 2.3.1-3 along with each subcomponent's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.1-3 Reactor Vessel Internals

Subcomponent	Passive Function	Aging Management Reference
LOWER CORE PLATE AND FUEL PINS	CORE SUPPORT FLOW DISTRIBUTION	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (31) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (10) These apply to both passive functions.
LOWER SUPPORT FORGING	CORE SUPPORT FLOW DISTRIBUTION	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (31) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (8) Table 3.2-2 Line Number (9) These apply to both passive functions.
LOWER SUPPORT COLUMNS	CORE SUPPORT	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (31) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7)
CORE BARREL AND FLANGE	CORE SUPPORT FLOW DISTRIBUTION	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (31) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (10) These apply to both passive functions.
RADIAL KEYS AND CLEVIS INSERTS	CORE SUPPORT	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (28) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (9)
BAFFLE AND FORMER ASSEMBLY	CORE SUPPORT FLOW DISTRIBUTION	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (31) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) These apply to both passive functions.

Table 2.3.1-3 Reactor Vessel Internals

Subcomponent	Passive Function	Aging Management Reference
CORE BARREL OUTLET NOZZLE	FLOW DISTRIBUTION	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (31) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (8) Table 3.2-2 Line Number (9)
SECONDARY CORE SUPPORT	CORE SUPPORT FLOW DISTRIBUTION	Table 3.2-2 Line Number (11) This applies to both passive functions.
DIFFUSER PLATES	FLOW DISTRIBUTION	Table 3.2-2 Line Number (11)
UPPER SUPPORT PLATE ASSEMBLY	GUIDE AND SUPPORT RCCA'S	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (9)
UPPER CORE PLATE AND FUEL ALIGNMENT PINS	CORE SUPPORT FLOW DISTRIBUTION	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (9) Table 3.2-2 Line Number (10) These apply to both passive functions.
UPPER SUPPORT COLUMNS	GUIDE AND SUPPORT RCCA'S	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (9)
RCCA GUIDE TUBES AND FLOW DOWNCOMERS	GUIDE AND SUPPORT RCCAS	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (9)
GUIDE TUBE SUPPORT PINS	GUIDE AND SUPPORT RCCAS	Table 3.2-1 Line Number (8) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (10) Table 3.2-2 Line Number (11)

Table 2.3.1-3 Reactor Vessel Internals

Subcomponent	Passive Function	Aging Management Reference
UPPER CORE PLATE ALIGNMENT PINS	GUIDE AND SUPPORT RCCAS	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (28) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (9)
HOLD-DOWN SPRING	CORE SUPPORT	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (30) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (9) Table 3.2-2 Line Number (10) Table 3.2-2 Line Number (12)
HEAD/VESSEL ALIGNMENT PINS	CORE SUPPORT	Table 3.2-2 Line Number (11)
THERMAL SHIELD AND NEUTRON PANELS	SHIELD VESSEL	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (31) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7)
BMI COLUMNS AND FLUX THIMBLES	GUIDE AND SUPPORT INSTRUMENTATION	Table 3.2-1 Line Number (28) Table 3.2-2 Line Number (11)
HEAD COOLING SPRAY NOZZLES	FLOW DISTRIBUTION	Table 3.2-2 Line Number (11)
UPPER INSTRUMENTATION COLUMN, CONDUIT AND SUPPORTS	GUIDE AND SUPPORT THERMOCOUPLES	Table 3.2-2 Line Number (11)
BOLTING (UPPER SUPPORT COLUMN, GUIDE TUBE, CLEVIS INSERT)	CORE SUPPORT	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (30) Table 3.2-1 Line Number (31) Table 3.2-1 Line Number (33) Table 3.2-1 Line Number (36) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (9) Table 3.2-2 Line Number (12)

Table 2.3.1-3 Reactor Vessel Internals

Subcomponent	Passive Function	Aging Management Reference
BOLTING (LOWER SUPPORT COLUMN, BAFFLE/FORMER, BARREL/FORMER)	CORE SUPPORT	Table 3.2-1 Line Number (5) Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (12) Table 3.2-1 Line Number (13) Table 3.2-1 Line Number (31) Table 3.2-1 Line Number (33) Table 3.2-1 Line Number (36) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (12)

2.3.1.4 Pressurizer

Component Description

The Ginna pressurizer is part of the reactor coolant system (RCS) and is located inside containment. The RCS pressure control consists of the pressurizer vessel equipped with electric heaters, safety valves, relief valves, pressurizer spray, interconnecting piping, and instrumentation. During operation, the pressurizer contains saturated water and steam maintained at the desired saturation temperature and pressure by the electric heaters and pressurizer spray. The chemical and volume control system (CVCS) maintains the desired water level in the pressurizer during steady-state operation by a pressurizer level control instrumentation system.

During normal operation, the external electrical network imposes load changes on the plant turbine generator. These load changes cause temperature changes in the RCS. Since the reactor rod control system which controls the reactor coolant temperature, does not respond instantaneously during a load transient, the pressurizer pressure control system is designed to absorb the reactor coolant volume surges and limit pressure variations during the initial transient period prior to an effective response by the reactor rod control system. The pressurizer performs the following functions:

- Maintains the required reactor coolant pressure (pressure boundary function) during steady-state operation and normal heatup and cooldown.
- Limits pressure changes, to an allowable range, that are caused by reactor coolant thermal expansion and contraction during normal plant load changes and transients.

UFSAR Reference

Additional Pressurizer details are provided in Section 5.4.7 of the UFSAR.

Subcomponents Subject to an Aging Management Review

The subcomponents of the Pressurizer that require aging management review are indicated in Table 2.3.1-4 along with each subcomponent's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.1-4 Pressurizer

Subcomponent	Passive Function	Aging Management Reference
LOWER HEAD	PRESSURE BOUNDARY	Table 3.2-1 Line Number (24) Table 3.2-1 Line Number (26) Table 3.2-2 Line Number (33)
SURGE NOZZLE	PRESSURE BOUNDARY	Table 3.2-1 Line Number (24) Table 3.2-2 Line Number (19) Table 3.2-2 Line Number (33)
SURGE NOZZLE SAFE END	PRESSURE BOUNDARY	Table 3.2-1 Line Number (24) Table 3.2-2 Line Number (34)
HEATER WELL AND HEATER SHEATH	PRESSURE BOUNDARY	Table 3.2-1 Line Number (24) Table 3.2-2 Line Number (34)
SHELL	PRESSURE BOUNDARY	Table 3.2-1 Line Number (24) Table 3.2-1 Line Number (26) Table 3.2-2 Line Number (33)
INSTRUMENT NOZZLES THERMOWELLS	PRESSURE BOUNDARY	Table 3.2-1 Line Number (24) Table 3.2-2 Line Number (34)
UPPER HEAD	PRESSURE BOUNDARY	Table 3.2-1 Line Number (24) Table 3.2-1 Line Number (26) Table 3.2-2 Line Number (33)
SPRAY NOZZLE	PRESSURE BOUNDARY	Table 3.2-1 Line Number (24) Table 3.2-2 Line Number (19) Table 3.2-2 Line Number (33)
SPRAY NOZZLE SAFE END	PRESSURE BOUNDARY	Table 3.2-1 Line Number (24) Table 3.2-2 Line Number (34)

Table 2.3.1-4 Pressurizer

Subcomponent	Passive Function	Aging Management Reference
SAFETY NOZZLE	PRESSURE BOUNDARY	Table 3.2-2 Line Number (17) Table 3.2-2 Line Number (19) Table 3.2-2 Line Number (33)
SAFETY NOZZLE SAFE END	PRESSURE BOUNDARY	Table 3.2-1 Line Number (24) Table 3.2-2 Line Number (34)
RELIEF NOZZLE	PRESSURE BOUNDARY	Table 3.2-2 Line Number (17) Table 3.2-2 Line Number (19) Table 3.2-2 Line Number (33)
RELIEF NOZZLE SAFE END	PRESSURE BOUNDARY	Table 3.2-1 Line Number (24) Table 3.2-2 Line Number (34)
MANWAY COVER	PRESSURE BOUNDARY	Table 3.2-2 Line Number (18) Table 3.2-2 Line Number (19) Table 3.2-2 Line Number (33)
SUPPORT SKIRT AND FLANGE	STRUCTURAL SUPPORT	Table 3.2-1 Line Number (26) Table 3.2-1 Line Number (29) Table 3.2-2 Line Number (33)
MANWAY COVER BOLTS	MECHANICAL CLOSURE INTEGRITY	Table 3.2-1 Line Number (22) Table 3.2-1 Line Number (26) Table 3.2-2 Line Number (33)

2.3.1.5 Steam Generators

Component Description

The Steam Generators (SGs) form the boundary between the radioactive primary (Class 1 piping) and the non-radioactive secondary systems. There are two identical steam generators installed in containment, one in each reactor coolant system (RCS) loop. The SG is a vertical shell and tube heat exchanger, where heat transferred from a single-phase fluid at high temperature and pressure (RCS) on the tube side is used to generate a two-phase (steam-water) mixture at a lower temperature and pressure on the shell side. The reactor coolant flows through the primary side, or inverted U-tubes, entering and leaving through the nozzles located in the hemispherical bottom head of the steam generator. The primary head is divided into inlet and outlet chambers by a vertical partition plate extending from the head to the tube sheet.

The steam-water mixture is generated on the secondary, or shell side. Feedwater entering the steam generators through a feed ring, mixes with recirculated fluid and flows downward around the tube bundle inner shroud, then enters the tube bundle area where heat is transferred from the RCS. A small portion of the tube bundle located near the tubesheet functions as a preheater to raise the temperature of the fluid to the saturation point. The remaining area of the tube bundle secondary side operates in the heat transfer nucleate boiling region. The wet vapor rises and is dried to a near moisture-free condition as it exits the steam generator at the outlet nozzle at the top of the shell.

At steady-state conditions, the fluid inventory and heat content on both the primary and secondary sides of the steam generator is constant, requiring a virtually constant mass flow on the primary side and a makeup (feedwater) mass flow rate that matches the combined steam flow and blowdown mass flow rates.

UFSAR Reference

Additional Steam Generator details are provided in Section 5.4.2 of the UFSAR.

Subcomponents Subject to an Aging Management Review

The subcomponents of the Steam Generators that require aging management review are indicated in Table 2.3.1-5 along with each subcomponents' passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.1-5 Steam Generators

Subcomponent	Passive Function	Aging Management Reference
PRIMARY INLET AND OUTLET NOZZLES	PRESSURE BOUNDARY	Table 3.2-1 Line Number (32) Table 3.2-2 Line Number (20) Table 3.2-2 Line Number (26) Table 3.2-2 Line Number (33)
PRIMARY INLET AND OUTLET NOZZLE SAFE ENDS	PRESSURE BOUNDARY	Table 3.2-1 Line Number (32) Table 3.2-2 Line Number (20) Table 3.2-2 Line Number (34)

Table 2.3.1-5 Steam Generators

Subcomponent	Passive Function	Aging Management Reference
TUBESHEET	PRESSURE BOUNDARY	Table 3.2-2 Line Number (21) Table 3.2-2 Line Number (22)
DIVIDER PLATE	FLOW DISTRIBUTION	Table 3.2-2 Line Number (23)
U-TUBES	PRESSURE BOUNDARY HEAT TRANSFER	Table 3.2-1 Line Number (15) This applies to both passive functions.
PRIMARY MANWAYS	PRESSURE BOUNDARY	Table 3.2-1 Line Number (26) Table 3.2-2 Line Number (33)
STEAM GENERATOR SHELL AND TRANSITION CONE	PRESSURE BOUNDARY	Table 3.2-1 Line Number (2) Table 3.2-2 Line Number (20) Table 3.2-2 Line Number (33)
FEEDWATER NOZZLE	PRESSURE BOUNDARY	Table 3.2-1 Line Number (21) Table 3.2-2 Line Number (20) Table 3.2-2 Line Number (24) Table 3.2-2 Line Number (33)
STEAM OUTLET NOZZLE	PRESSURE BOUNDARY	Table 3.2-1 Line Number (21) Table 3.2-2 Line Number (20) Table 3.2-2 Line Number (24) Table 3.2-2 Line Number (33)
STEAM FLOW RESTRICTOR	RESTRICTS FLOW	Table 3.2-2 Line Number (24)
BLOWDOWN PIPING NOZZLES AND SECONDARY SIDE SHELL PENETRATIONS	PRESSURE BOUNDARY	Table 3.2-1 Line Number (9) Table 3.2-2 Line Number (20) Table 3.2-2 Line Number (24) Table 3.2-2 Line Number (33)
SECONDARY CLOSURES	PRESSURE BOUNDARY	Table 3.2-2 Line Number (24) Table 3.2-2 Line Number (33)
INTERNAL SHROUD, PRIMARY AND SECONDARY DECKS	STRUCTURAL SUPPORT	Table 3.2-2 Line Number (24)

Table 2.3.1-5 Steam Generators

Subcomponent	Passive Function	Aging Management Reference
LATTICE GRID TUBE SUPPORTS	STRUCTURAL SUPPORT	Table 3.2-2 Line Number (25)
U-BEND RESTRAINTS	STRUCTURAL SUPPORT	Table 3.2-2 Line Number (25)
PRIMARY CHANNEL HEAD	PRESSURE BOUNDARY	Table 3.2-1 Line Number (26) Table 3.2-1 Line Number (32) Table 3.2-2 Line Number (20) Table 3.2-2 Line Number (33)
PRIMARY MANWAY BOLTS	MECHANICAL CLOSURE INTEGRITY	Table 3.2-1 Line Number (22) Table 3.2-1 Line Number (26) Table 3.2-2 Line Number (33)
SECONDARY SIDE CLOSURE BOLTS	MECHANICAL CLOSURE INTEGRITY	Table 3.2-1 Line Number (22) Table 3.2-2 Line Number (33)
SUPPORT PADS	STRUCTURAL SUPPORT	Table 3.2-2 Line Number (26) Table 3.2-2 Line Number (27) Table 3.2-2 Line Number (33)
SEISMIC LUGS	STRUCTURAL SUPPORT	Table 3.2-2 Line Number (27) Table 3.2-2 Line Number (33)

2.3.1.6 Reactor Coolant (Non-Class 1)

System Description (Non-Class1)

NOTE: Reactor Coolant System (RCS) Class 1 components, Steam Generators, the Pressurizer, and Reactor Vessel are reviewed and evaluated as unique specific topical areas. For clarity purposes the system drawings for Non-Class 1 RCS components include showing the above RCS equipment, but the Class 1 portion is clearly denoted with flags.

The Non-Class 1 Reactor Coolant System (RCS) components system includes all of the safety class 2, 3 and non-nuclear safety grade equipment used to functionally support the Reactor Coolant System. Non-Class 1 RCS equipment is used to sense and provide signals for reactor trip and Engineered Safety Features Actuation. Equipment included within the system boundary is also used for safe shutdown following fires and Station Blackout Events. The Non-Class 1 RCS components system also contains equipment that is Environmentally Qualified.

The principal components of the Non-Class 1 RCS Components system include all RCS interconnected non-class 1 piping instruments and instrument lines, reactor coolant pump motor coolers and heat exchangers, and the pressurizer power operated relief valve (PORV) nitrogen actuation system. Also included within the evaluation boundary are the PORV and safety valve downstream tail piping up to and including the pressurizer relief tank, the reactor vessel level monitoring system, the low RCS loop level instrumentation, in-core nuclear detector drive detector isolation and the essential piping valves and ancillary equipment necessary to support the function of the reactor coolant system.

System Function Listing

In addition to the System Functions described above, the Non-Class 1 RCS Components System also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code D SENSE OR PROVIDE PROCESS CONDITIONS AND GENERATE SIGNALS FOR REACTOR TRIP AND ENGINEERED SAFETY FEATURES ACTUATION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Non-Class 1 RCS Components system perform this primary design system function.

Code E PROVIDE REACTOR COOLANT PRESSURE BOUNDARY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Non-Class 1 RCS Components system perform this primary design system function.

Code H PROVIDE EMERGENCY HEAT REMOVAL FROM THE REACTOR COOLANT SYSTEM USING SECONDARY HEAT REMOVAL CAPABILITY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Non-Class 1 RCS Components system perform this associated design system function. This function is associated with instrument loop power supplies.

Code J PROVIDE HEAT REMOVAL FROM SAFETY RELATED HEAT EXCHANGERS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Non-Class 1 RCS Components system perform this associated design system function. This function is associated with the RCP motor upper bearing oil coolers.

Code K PROVIDE PRIMARY CONTAINMENT BOUNDARY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Non-Class 1 RCS Components system perform this associated design system function.

Code Q PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Non-Class 1 RCS Components system perform this associated design system function.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Non-Class 1 RCS Components system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Non-Class 1 RCS Components system that perform special capability class functions are tracked under the Criterion 2 code (Y) and Criterion 3 codes (Z1 through Z5). The “S” function includes the Digital Metal Impact Monitoring System and Reg Guide 1.97 Category 2 and 3 post accident monitoring variables, non-safety reactor trip signals, etc.

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Non-Class 1 RCS Components system perform this associated design system function.

Code X SFR FUNCTION NOT APPLICABLE AT COMPONENT LEVEL	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Non-Class 1 RCS Components system perform specific safety related functions different from and in addition to the system level functions (e.g. Reg Guide 1.97 Category 1).

Code Y	Cri 1	Cri 2	Cri 3				
LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Non-Class 1 RCS Components system perform this associated design system function. The integrity of the downstream tailpieces from the pressurizer safeties is assumed with respect to the analysis of jet thrust forces during a safety valve lift. The analysis does not extend to the interconnecting pipe which also drains to the pressurizer relief tank

Code Z1	Cri 1	Cri 2	Cri 3				
LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Reactor Coolant, Core, and Internals system perform this function.

Code Z2	Cri 1	Cri 2	Cri 3				
LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ENVIRONMENTAL QUALIFICATION (10 CFR 50.49)			FP	EQ	PTS	AT	SB
				X			

Comment: Components within the Reactor Coolant, Core, and Internals system perform this function.

Code Z5	Cri 1	Cri 2	Cri 3				
LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR STATION BLACKOUT (10 CFR 50.63)			FP	EQ	PTS	AT	SB
							X

Comment: Components within the Non-Class 1 RCS Components system perform this function.

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.1-6 along with each Component Group’s passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

UFSAR Reference

Additional Reactor Coolant Non-Class 1 details are provided in Section 5.1, Section 5.2, Section 5.4, and Table 6.2-15a of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Reactor Coolant Non-Class 1 are listed below:

33013-1246,1	33013-1265,1
33013-1247	33013-1278,1
33013-1258	33013-2248
33013-1260	33013-2278
33013-1262,2	33013-1887
33013-1263	33013-1888
33013-1264	33013-1890

Table 2.3.1-6 Reactor Coolant (Non-Class 1)

Component Group	Passive Function	Aging Management Reference
ACCUMULATOR	PRESSURE BOUNDARY	Table 3.2-2 Line Number (28) Table 3.2-2 Line Number (33)
CONDENSING CHAMBER	PRESSURE BOUNDARY	Table 3.2-2 Line Number (28) Table 3.2-2 Line Number (33)
COOLER	PRESSURE BOUNDARY	Table 3.2-2 Line Number (30) Table 3.2-2 Line Number (31)
CS COMPONENTS	PRESSURE BOUNDARY	Table 3.2-2 Line Number (28) Table 3.2-2 Line Number (33)
FASTENERS (BOLTING)	MECHANICAL JOINT INTEGRITY	Table 3.2-1 Line Number (22) Table 3.2-1 Line Number (26) Table 3.2-2 Line Number (33)
HEAT EXCHANGER	PRESSURE BOUNDARY HEAT TRANSFER	Table 3.2-2 Line Number (30) Table 3.2-2 Line Number (31) Table 3.2-2 Line Number (33) These apply to both passive functions.
OPERATOR	PRESSURE BOUNDARY	Table 3.2-2 Line Number (28) Table 3.2-2 Line Number (33)

Table 2.3.1-6 Reactor Coolant (Non-Class 1)

Component Group	Passive Function	Aging Management Reference
PIPE	PRESSURE BOUNDARY	Table 3.2-1 Line Number (6) Table 3.2-2 Line Number (29) Table 3.2-2 Line Number (28) Table 3.2-2 Line Number (29) Table 3.2-2 Line Number (33)
SEAL TABLE	SUPPORT IN-CORE INSTRUMENTATION	Table 3.2-2 Line Number (32) Table 3.2-2 Line Number (34)
STRAINER HOUSING	PRESSURE BOUNDARY	Table 3.2-2 Line Number (28) Table 3.2-2 Line Number (34)
TEMPERATURE ELEMENT[1]	PRESSURE BOUNDARY	Table 3.2-1 Line Number (6) Table 3.2-2 Line Number (34)
VALVE BODY	PRESSURE BOUNDARY	Table 3.2-1 Line Number (6) Table 3.2-2 Line Number (28) Table 3.2-2 Line Number (33) Table 3.2-2 Line Number (34)

1. Selected instruments were conservatively included within the scope of License Renewal. Consideration was given to the consequences of an instrument housing pressure boundary failure. Where an instrument was unisolable from a pressure source and is of sufficient size that a system function would be degraded should the pressure boundary fail, that instrument is included for License Renewal review.

2.3.2 Engineered Safety Features Systems

The following systems are addressed in this section:

- Safety Injection System (Section 2.3.2.1)
- Containment Spray System (Section 2.3.2.2)
- Residual Heat Removal System (Section 2.3.2.3)
- Containment Hydrogen Detectors and Recombiners System (Section 2.3.2.4)
- Containment Isolation Components (Section 2.3.2.5)

2.3.2.1 Safety Injection (SI)

System Description

The Safety Injection System supports RCS inventory and reactivity control during accident and post-accident circumstances by automatically delivering borated water to the reactor vessel for cooling under high and low reactor coolant pressure conditions. Additionally, the system serves to insert negative reactivity into the Reactor core in the form of borated water during an uncontrolled plant cooldown following a steam line break or an inadvertent valve operation. The Safety Injection System is also credited for use in safe shutdown following some fires and contains components that are part of the Environmental Qualification Program.

Adequate core cooling following a loss-of-coolant accident is provided by the Safety Injection System (SI), which operates as follows:

1. Injection of borated water by the passive accumulators.
2. Injection by the high-pressure safety injection pumps drawing borated water from the RWST.
3. Provide capability for injection by the residual heat removal pumps also drawing borated water from the RWST.
4. Recirculation of reactor coolant and injection water from the containment sump to the reactor coolant system by the residual heat removal pumps and the SI pumps, if needed (piggy-back operation).

The principal components of the SI system are two passive accumulators (one for each loop), high-head safety injection pumps, interface with low-head safety injection (Residual Heat Removal pumps), and the essential piping and valves. The accumulators are passive devices that discharge into the cold leg of each loop. During MODES 1 and 2 the refueling water storage tank (RWST) is aligned to the suction of the high-head safety injection pumps and residual heat removal pumps. The Containment Spray System shares the RWST liquid capacity with the SI system. After the injection phase, coolant spilled from the break and water injected by the safety injection system and the containment spray is cooled and recirculated from the sump to the reactor coolant system by the low-pressure safety injection system or, if needed, by the high-pressure safety injection system.

The following fluid systems interface with the Safety Injection System:

Reactor Coolant	Waste Disposal
Residual Heat Removal	Plant Air
Containment Spray	Spent Fuel Cooling and Fuel Storage
Chemical and Volume Control	Component Cooling Water
Service Water	

System Function Listing

In addition to the System Functions described above, the Safety Injection System also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code B INTRODUCE EMERGENCY NEGATIVE REACTIVITY TO MAKE THE REACTOR SUBCRITICAL	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Safety Injection system perform this primary design system function. The Safety Injection system increases the boron concentration in the Reactor Coolant system during the injection phase of Safety Injection to ensure adequate reactor shutdown margin in the event of a secondary pipe break. The Safety Injection system provides sufficient boron to maintain an adequate post-LOCA sump boron concentration to ensure shutdown of the core. The Safety Injection system delivers borated water to the Reactor Coolant system, as necessary, to compensate for Xenon decay to maintain hot shutdown margin.

Code G PROVIDE EMERGENCY CORE COOLANT WHERE THE ECCS PROVIDES COOLANT DIRECTLY TO THE CORE	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Safety Injection system perform this primary design system function. The Safety Injection system delivers borated cooling water to the Reactor Coolant system during the injection phase to support core cooling.

Code J PROVIDE HEAT REMOVAL FROM SAFETY RELATED HEAT EXCHANGERS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Safety Injection system perform this associated design system function (e.g. seal water heat exchangers and/or heat exchanger interfaces with other fluid systems).

Code K PROVIDE PRIMARY CONTAINMENT BOUNDARY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Safety Injection system perform this associated design system function. A portion of the Safety Injection system is a Closed Loop Outside Containment (CLOC) pressure boundary.

Code L PROVIDE EMERGENCY HEAT REMOVAL FROM PRIMARY CONTAINMENT AND PROVIDE CONTAINMENT PRESSURE CONTROL	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Safety Injection system perform this primary design system function. The Safety Injection system provides the liquid capacity of the Refueling Water Storage Tank for the Containment Spray system to provide emergency heat removal from primary containment and provide containment pressure control.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Safety Injection system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Safety Injection system that perform special capability class functions are tracked under the Criterion 3 codes (Z1 through Z5). The Safety Injection systems Refueling Water Storage Tank provides a source of borated water during refueling shutdown to flood the refueling cavity.

Additionally, the Safety Injection system provides a backup source of water during normal plant operations for the Chemical and Volume Control system charging pumps in the event the normal source (VCT) is lost. The Refueling Water Storage Tank also provides one of two required boric acid sources sufficient to provide the required shutdown margin at cold shutdown, xenon-free conditions from any expected operating condition.

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Safety Injection system perform this associated design system function. The Safety Injection system contains non-nuclear safety class components such as normally isolated sample points and test connections.

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Safety Injection system perform this associated design system function. The Safety Injection system contains non-safety piping segments and components whose failure could impact a safety related function.

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Safety Injection system perform this associated design system function. The safety injection system is capable of isolating RCS boundaries, preventing loss of inventory of the RWST and supplying RWST inventory to the charging system to support reactor coolant make-up capability and reactivity control. Additionally, a Safety Injection pump may be used following some fire events to provide inventory and reactivity control for safe shutdown.

Code Z2 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ENVIRONMENTAL QUALIFICATION (10 CFR 50.49)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: Components within the Safety Injection system are designated as Environmentally Qualified.

UFSAR Reference

Additional Safety Injection System details are provided in Section 6.3 and Table 6.2-15a of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Safety Injection System are listed below:

33013-1246,2
33013-1250,1
33013-1261
33013-1262,1
33013-1262,2
33013-1887
33013-1888

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.2-1 along with each Component Group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.2-1 Safety Injection (SI)

Component Group	Passive Function	Aging Management Reference
ACCUMULATOR	PRESSURE BOUNDARY	Table 3.3-1 Line Number (9) Table 3.3-2 Line Number (1)
CS COMPONENTS	PRESSURE BOUNDARY	Table 3.3-1 Line Number (10)
FASTENERS (BOLTING)	JOINT INTEGRITY	Table 3.3-1 Line Number (10) Table 3.3-1 Line Number (11) Table 3.3-2 Line Number (9) Table 3.3-2 Line Number (10) Table 3.3-2 Line Number (11)
FLOW ELEMENT	PRESSURE BOUNDARY	Table 3.3-1 Line Number (9) Table 3.3-2 Line Number (18) Table 3.3-2 Line Number (19)

Table 2.3.2-1 Safety Injection (SI)

Component Group	Passive Function	Aging Management Reference
HEAT EXCHANGER	HEAT TRANSFER PRESSURE BOUNDARY	Table 3.3-1 Line Number (8) Table 3.3-2 Line Number (23) Table 3.3-2 Line Number (24) Table 3.3-2 Line Number (25) Table 3.3-2 Line Number (30) Table 3.3-2 Line Number (31) These apply to the pressure boundary passive function. Table 3.3-2 Line Number (26) Table 3.3-2 Line Number (27) Table 3.3-2 Line Number (28) Table 3.3-2 Line Number (29) These apply to the heat transfer passive function.
INDICATOR[1]	PRESSURE BOUNDARY	Table 3.3-1 Line Number (9) Table 3.3-2 Line Number (34)
ORIFICE	PRESSURE BOUNDARY RESTRICTS FLOW	Table 3.3-1 Line Number (9) This applies to both passive functions. Table 3.3-2 Line Number (38) Table 3.3-2 Line Number (39) These apply to the pressure boundary passive function.
PIPE	PRESSURE BOUNDARY	Table 3.3-1 Line Number (9) Table 3.3-2 Line Number (46) Table 3.3-2 Line Number (49) Table 3.3-2 Line Number (50)
PUMP CASING	PRESSURE BOUNDARY	Table 3.3-2 Line Number (54) Table 3.3-2 Line Number (55) Table 3.3-2 Line Number (56)
TANK	PRESSURE BOUNDARY	Table 3.3-1 Line Number (3) Table 3.3-1 Line Number (9) Table 3.3-2 Line Number (62)

Table 2.3.2-1 Safety Injection (SI)

Component Group	Passive Function	Aging Management Reference
VALVE BODY	PRESSURE BOUNDARY	Table 3.3-1 Line Number (9) Table 3.3-2 Line Number (71) Table 3.3-2 Line Number (73) Table 3.3-2 Line Number (74) Table 3.3-2 Line Number (76) Table 3.3-2 Line Number (77) Table 3.3-2 Line Number (78) Table 3.3-2 Line Number (79) Table 3.3-2 Line Number (80) Table 3.3-2 Line Number (81) Table 3.3-2 Line Number (90) Table 3.3-2 Line Number (92) Table 3.3-2 Line Number (93)

1. Selected instruments were conservatively included within the scope of License Renewal. Consideration was given to the consequences of an instrument housing pressure boundary failure. Where an instrument was unisolable from a pressure source and is of sufficient size that a system function would be degraded should the pressure boundary fail, that instrument is included for License Renewal review.

2.3.2.2 Containment Spray (CS)

System Description

The Containment Spray (CS) system, in conjunction with the Containment Ventilation system is designed to remove sufficient heat from the containment atmosphere following an accident condition to maintain the containment pressure below design limits. The CS system, in conjunction with the sodium hydroxide (NaOH) tank, is also capable of reducing the iodine and particulate fission product inventories in the containment atmosphere such that the offsite radiation exposure resulting from a LOCA is within the guidelines established by 10 CFR 100. The CS System also contains components that are part of the Environmental Qualification Program.

The principal components of the containment spray system include two pumps, one tank, two spray headers, two eductors, spray nozzles, the essential piping and valves. The system initially takes suction from the refueling water storage tank (RWST). When a low level is reached in the RWST, the spray pump suction is fed from the discharge of the residual heat removal pumps if continued spray is required.

During the period of time that the spray pumps draw from the RWST, approximately 20 gpm of spray additive will be added to the refueling water in each train by using a liquid eductor enabled by the spray pump discharge. The fluid passing from the NaOH tank will then mix with the fluid entering the pump suction. The results will be a solution suitable for the removal of iodine. The containment spray system provides a 100% redundant backup to the containment post-accident charcoal system for iodine removal capability following a LOCA. For operation in the recirculation mode, water is supplied through the residual heat removal pumps.

The following fluid systems interface with the Containment Spray System:

Residual Heat Removal	Safety Injection
Component Cooling Water	Instrument Air
Waste Disposal	Containment Ventilation

System Function Listing

In addition to the System Functions listed above, the Containment Spray System also supports additional functions (associated design system functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code J PROVIDE HEAT REMOVAL FROM SAFETY RELATED HEAT EXCHANGERS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Containment Spray system perform this associated design system function (e.g. seal water heat exchangers cooled by CCW).

Code K PROVIDE PRIMARY CONTAINMENT BOUNDARY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Containment Spray system perform this associated design system function. A portion of the Containment Spray system outside containment is a closed loop system.

Code L PROVIDE EMERGENCY HEAT REMOVAL FROM PRIMARY CONTAINMENT AND PROVIDE CONTAINMENT PRESSURE CONTROL	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Containment Spray system perform this primary design system function. The Containment Spray system delivers treated water to the containment spray headers during accident conditions which could over pressurize containment thus ensuring containment pressure does not exceed its design value.

Code M PROVIDE EMERGENCY REMOVAL OF RADIOACTIVE MATERIAL FROM THE PRIMARY CONTAINMENT ATMOSPHERE	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Containment Spray system perform this primary design system function. The Containment Spray system delivers treated water to the containment spray headers to support removal of elemental iodine from the containment atmosphere in the event of a Loss of Coolant Accident.

Code Q PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Containment Spray system perform this associated design system function. (This function denotes protecting the other MCC vital loads from a failure of a load in the Containment Spray system.)

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Containment Spray system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Containment Spray system that perform special capability class functions are tracked under the Criterion 3 codes (Z1 through Z5).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Containment Spray system perform this associated design system function. The Containment Spray system contains non-nuclear safety class components such as normally isolated sample points and test connections.

Code Z2 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ENVIRONMENTAL QUALIFICATION (10 CFR 50.49)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: Components within the Containment Spray system are designated as Environmentally Qualified.

UFSAR Reference

Additional Containment Spray System details are provided in Section 6.5.2, Section 6.2.2, and Table 6.2-15a of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Containment Spray System are listed below:

33013-1246,2
33013-1261
33013-1891

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.2-2 along with each Component Group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.2-2 Containment Spray (CS)

Component Group	Passive Function	Aging Management Reference
CS COMPONENTS	PRESSURE BOUNDARY	Table 3.3-1 Line Number (10)
EDUCTOR	PRESSURE BOUNDARY PROVIDE MIXED FLOW	Table 3.3-1 Line Number (9) This applies to both passive functions. Table 3.3-2 Line Number (8) This applies to the pressure boundary passive function.
FASTENERS (BOLTING)	JOINT INTEGRITY	Table 3.3-1 Line Number (10) Table 3.3-1 Line Number (11) Table 3.3-2 Line Number (9) Table 3.3-2 Line Number (10) Table 3.3-2 Line Number (11)
FLOW NOZZLES	PRESSURE BOUNDARY PROVIDE FLOW	Table 3.3-2 Line Number (20) . Table 3.3-2 Line Number (21) These apply to both passive functions.
HEAT EXCHANGER	HEAT TRANSFER PRESSURE BOUNDARY	Table 3.3-1 Line Number (8) This applies to both passive functions. Table 3.3-2 Line Number (23) This applies to the pressure boundary passive function.
INDICATOR[1]	PRESSURE BOUNDARY	Table 3.3-1 Line Number (9) Table 3.3-2 Line Number (34)

Table 2.3.2-2 Containment Spray (CS)

Component Group	Passive Function	Aging Management Reference
ORIFICE	PRESSURE BOUNDARY RESTRICTS FLOW	Table 3.3-2 Line Number (35) Table 3.3-2 Line Number (36) Table 3.3-2 Line Number (37) These apply to both passive functions.
PIPE	PRESSURE BOUNDARY	Table 3.3-1 Line Number (9) Table 3.3-2 Line Number (46) Table 3.3-2 Line Number (49) Table 3.3-2 Line Number (50) Table 3.3-2 Line Number (52) Table 3.3-2 Line Number (53)
PUMP CASING	PRESSURE BOUNDARY	Table 3.3-2 Line Number (54) Table 3.3-2 Line Number (55) Table 3.3-2 Line Number (56)
TANK	PRESSURE BOUNDARY	Table 3.3-2 Line Number (62) Table 3.3-2 Line Number (63) Table 3.3-2 Line Number (64)
TRANSMITTER[1]	PRESSURE BOUNDARY	Table 3.3-1 Line Number (9) Table 3.3-2 Line Number (68)
VALVE BODY	PRESSURE BOUNDARY	Table 3.3-1 Line Number (9) Table 3.3-2 Line Number (71) Table 3.3-2 Line Number (74) Table 3.3-2 Line Number (76) Table 3.3-2 Line Number (77) Table 3.3-2 Line Number (82) Table 3.3-2 Line Number (83) Table 3.3-2 Line Number (90) Table 3.3-2 Line Number (92) Table 3.3-2 Line Number (93) Table 3.3-2 Line Number (95) Table 3.3-2 Line Number (96)

1. Selected instruments were conservatively included within the scope of License Renewal. Consideration was given to the consequences of an instrument housing pressure boundary failure. Where an instrument was unisolable from a pressure source and is of sufficient size that a system function would be degraded should the pressure boundary fail, that instrument is included for License Renewal review.

2.3.2.3 Residual Heat Removal (RHR)

System Description

The Emergency Core Cooling Systems include the use of the Residual Heat Removal System (RHR). The system automatically delivers borated water to the reactor vessel for cooling under low reactor coolant pressure conditions. The RHR system, in conjunction with the Main and Auxiliary Steam system, is designed to transfer the fission product decay heat and other residual heat from the reactor core to the component cooling water system and the atmosphere at a rate such that design limits of the fuel and the primary system coolant boundary are not exceeded. The RHR system also contains components credited for use in safe shutdown following some fires and components that are part of the Environmental Qualification Program.

Adequate core cooling following a loss-of-coolant accident is provided by the safety injection (emergency core cooling) system, which operates as follows:

1. Injection of borated water by the passive accumulators.
2. Injection by the high-pressure safety injection pumps drawing borated water from the RWST.
3. Injection by the residual heat removal pumps also drawing borated water from the RWST.
4. Recirculation of reactor coolant and injection water from the containment sump to the reactor coolant system by the residual heat removal pumps.

The principal components of the RHR system are two RHR (low head safety injection) pumps, two heat exchangers, and the essential piping and valves. Note: The residual heat removal system discharge line is not used for an Emergency Core Cooling System (ECCS) function that would require MOV-720 or MOV-721 to open; however, a branch of the residual heat removal discharge line provides low-pressure safety injection to the reactor vessel via parallel lines with one normally closed motor-operated valve (MOV-852A or B) and one check valve (CV-853A or B) in each line.

During MODES 1 and 2 the refueling water storage tank (RWST) is aligned to the suction of the high-head safety injection pumps and residual heat removal pumps. After the injection phase, coolant spilled from the break and water injected by the safety injection system and the containment spray is cooled and recirculated to the reactor coolant system by the low-pressure safety injection (residual heat removal) system or, if needed, by the high-pressure safety injection system.

If reactor coolant system depressurization to below the shutoff head of the residual heat removal pumps occurs before the injection mode of the safety injection system is terminated, the residual heat removal pumps will be used in the recirculation mode. The residual heat removal pumps will take suction from the containment sump, circulate the spilled coolant through the residual heat removal heat exchangers, and return the coolant to the reactor via the reactor vessel nozzles. If depressurization of the reactor coolant system proceeds slowly, the high-pressure safety injection pumps are aligned to take suction from the residual heat removal pumps, and inject flow into the reactor coolant system cold legs. The RHR pumps and heat exchangers, in conjunction with the Containment Spray System, may also be used during the recirculation phase to supply water from the containment sump for use in heat and pressure control of the containment atmosphere.

After the steam generators have been used to reduce the reactor coolant temperature to 350°F, decay heat cooling is initiated by aligning the residual heat removal pumps to take suction from the reactor coolant system loop A hot leg and discharge through the residual heat removal heat exchangers to the loop B cold leg.

The following fluid systems interface with Residual Heat Removal:

Reactor Coolant	Safety Injection
Containment Spray	Chemical and Volume Control
Component Cooling Water	

System Function Listing

In addition to the System Functions described above, the Residual Heat Removal System also contains components which support additional functions (associated design system functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code F REMOVE RESIDUAL HEAT FROM THE RCS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Residual Heat Removal system perform this primary design system function. The Residual Heat Removal system, in conjunction with the Safety Injection system, recirculates and cools the water that is collected in the containment sump and returns it to the Reactor Coolant system during the ECCS recirculation phase to support long term cooling.

Code G PROVIDE EMERGENCY CORE COOLANT WHERE THE ECCS PROVIDES COOLANT DIRECTLY TO THE CORE	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Residual Heat Removal system perform this primary design system function. The Residual Heat Removal system, in conjunction with the Safety Injection system, delivers borated cooling water to the Reactor Coolant system during the ECCS injection phase to support core cooling.

Code J PROVIDE HEAT REMOVAL FROM SAFETY RELATED HEAT EXCHANGERS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Residual Heat Removal system perform this associated design system function (e.g. seal water heat exchangers).

Code K PROVIDE PRIMARY CONTAINMENT BOUNDARY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Residual Heat Removal system perform this associated design system function. The Residual Heat Removal system has containment isolation valves. A portion of the Residual Heat Removal system outside containment is a closed loop system.

Code L PROVIDE EMERGENCY HEAT REMOVAL FROM PRIMARY CONTAINMENT AND PROVIDE CONTAINMENT PRESSURE CONTROL	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Residual Heat Removal system perform this primary design system function. The Residual Heat Removal system provides the capability to supply water to the suction of the Containment Spray pumps when in recirculation mode.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Residual Heat Removal system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Residual Heat Removal system that perform special capability class functions are tracked under the Criterion 3 codes (Z1 through Z5). The Residual Heat Removal system removes sensible and decay heat from the Reactor Coolant system (RCS) during cooldown, cold shutdown and refueling shutdown.

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Residual Heat Removal system perform this associated design system function. The Residual Heat Removal system contains non-nuclear safety class components such as normally isolated sample points and test connections.

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Residual Heat Removal system perform this associated design system function. The Residual Heat Removal system may be used following some fire events to provide reactivity control and decay heat removal for safe shutdown.

Code Z2 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ENVIRONMENTAL QUALIFICATION (10 CFR 50.49)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: Components within the Residual Heat Removal system are designated as Environmentally Qualified.

UFSAR Reference

Additional Residual Heat Removal System details are provided in Section 6.3.2.3, Section 5.4.5, and Table 6.2-15a of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Residual Heat Removal System are listed below:

33013-1245
33013-1247
33013-1260
33013-1264
33013-1272,2
33013-1278,2
33013-1890

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.2-3 along with each Component Group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.2-3 Residual Heat Removal (RHR)

Component Group	Passive Function	Aging Management Reference
CS COMPONENTS	PRESSURE BOUNDARY	Table 3.3-1 Line Number (10)
FASTENERS (BOLTING)	JOINT INTEGRITY	Table 3.3-1 Line Number (10) Table 3.3-1 Line Number (11) Table 3.3-2 Line Number (9) Table 3.3-2 Line Number (10) Table 3.3-2 Line Number (11)
FLOW ELEMENT	PRESSURE BOUNDARY	Table 3.3-1 Line Number (9) Table 3.3-2 Line Number (19)
HEAT EXCHANGER	HEAT TRANSFER PRESSURE BOUNDARY	Table 3.3-1 Line Number (8) This applies to both passive functions. Table 3.3-2 Line Number (22) Table 3.3-2 Line Number (23) Table 3.3-2 Line Number (33) These apply to the pressure boundary passive function.
INDICATOR[1]	PRESSURE BOUNDARY	Table 3.3-1 Line Number (9) Table 3.3-2 Line Number (34)

Table 2.3.2-3 Residual Heat Removal (RHR)

Component Group	Passive Function	Aging Management Reference
ORIFICE	PRESSURE BOUNDARY RESTRICTS FLOW	Table 3.3-1 Line Number (9) This applies to both passive functions. Table 3.3-2 Line Number (39) This applies to the pressure boundary passive function.
PIPE	PRESSURE BOUNDARY	Table 3.3-1 Line Number (9) Table 3.3-2 Line Number (46) Table 3.3-2 Line Number (48) Table 3.3-2 Line Number (49) Table 3.3-2 Line Number (50)
PUMP CASING	PRESSURE BOUNDARY	Table 3.3-1 Line Number (9) Table 3.3-2 Line Number (58)
SWITCH[1]	PRESSURE BOUNDARY	Table 3.3-1 Line Number (9) Table 3.3-2 Line Number (61)
TEMPERATURE ELEMENT[1]	PRESSURE BOUNDARY	Table 3.3-1 Line Number (9) Table 3.3-2 Line Number (65)
VALVE BODY	PRESSURE BOUNDARY	Table 3.3-1 Line Number (9) Table 3.3-2 Line Number (71) Table 3.3-2 Line Number (73) Table 3.3-2 Line Number (74) Table 3.3-2 Line Number (76) Table 3.3-2 Line Number (77) Table 3.3-2 Line Number (92) Table 3.3-2 Line Number (93)

1. Selected instruments were conservatively included within the scope of License Renewal. Consideration was given to the consequences of an instrument housing pressure boundary failure. Where an instrument was unisolable from a pressure source and is of sufficient size that a system function would be degraded should the pressure boundary fail, that instrument is included for License Renewal review.

2.3.2.4 Containment Hydrogen Detectors and Recombiners

System Description

Two trains of containment hydrogen detectors and hydrogen recombiner units are available to the plant. The purpose of these units is to detect and control combustible gas mixtures in the primary containment atmosphere. Portions of these trains are Environmentally Qualified. Because containment hydrogen buildup is a relatively slow process, the recombiner equipment located outside of containment is maintained at a lesser degree of prompt readiness than any other engineered safety feature. Those portions of the recombiner system are considered non-safety related components whose failure could prevent the satisfactory accomplishment of a safety related function.

The principal components of the Containment Hydrogen Detection and Recombiner System include:

For the detection portion, two hydrogen concentration monitoring devices, a local analyzer/control panels, remote monitoring/control panels and their corresponding essential piping and valves. The hydrogen monitoring system is capable of operation during post-accident conditions. The monitors are normally maintained in an isolated standby mode. The recombiner portion consists of two blowers and combustion chambers complete with main burner, two igniters (one a spare), pilot burner, and a dilution chamber, two control panels and the corresponding essential piping and valves. Each combustor is fired by an externally supplied fuel gas, employing containment air as the oxidant. The air supply blowers deliver primary combustion air and quench air to reduce the unit exhaust temperature. Hydrogen in the containment air is oxidized in passing through the combustion chamber. Hydrogen gas is also used as the externally supplied fuel so that noncondensible combustion products, which would cause a progressive rise in containment pressure, are avoided. Oxygen gas is made up through a separate containment feed to prevent depletion of containment oxygen below the concentration required for stable operation of the combustor.

The following fluid systems interface with the Containment Hydrogen Detection and Recombiner System:

Waste Disposal	Plant Air

System Function Listing

In addition to the System Functions described above, the Containment Hydrogen Detection and Recombiner System also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code K PROVIDE PRIMARY CONTAINMENT BOUNDARY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Containment Hydrogen Detectors and Recombiners system perform this associated design system function.

Code N CONTROL COMBUSTIBLE GAS MIXTURES IN THE PRIMARY CONTAINMENT ATMOSPHERE	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Containment Hydrogen Detectors and Recombiners system perform this primary design system function.

Code Q PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Containment Hydrogen Detectors and Recombiners system perform this associated design system function.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Containment Hydrogen Detectors and Recombiners system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Containment Hydrogen Detectors and Recombiners system that perform special capability class functions are tracked under the System Function code Y (Criterion 2).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Containment Hydrogen Detectors and Recombiners system perform this associated design system function.

Code X SFR FUNCTION NOT APPLICABLE AT COMPONENT LEVEL	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Containment Hydrogen Detectors and Recombiners system perform specific safety related functions different from and in addition to the system level functions (e.g. Reg Guide 1.97 Category 1).

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Containment Hydrogen Detectors and Recombiners system perform this associated design system function. Some Containment Hydrogen Detectors and Recombiners components located outside containment are maintained in a quality program but are not classified as nuclear safety related. The basis for this determination is that this standby system has ample time for operability testing and repairs prior to being placed in service. Additionally the system requires the delivery of vendor supplied gases for operation. This not-withstanding, those portions of the system which control or transport the gas mixtures outside of containment are conservatively categorized for License Renewal purposes as SSC's whose failure could prevent the satisfactory accomplishment of a safety related function.

Code Z2 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ENVIRONMENTAL QUALIFICATION (10 CFR 50.49)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: Components within the Containment Hydrogen Detectors and Recombiners system are designated as Environmentally Qualified.

UFSAR Reference

Additional Containment Hydrogen Detection and Recombiner System details are provided in Section 6.2.5, Section 1.5.10, and Table 6.2-15a of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Containment Hydrogen Detection and Recombiner System are listed below:

33013-1274	33013-1889
33013-1275,1	33013-1892
33013-1275,2	33013-1899,1
33013-1278,1	

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.2-4 along with each Component Group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.2-4 Containment Hydrogen Detectors and Recombiners

Component Group	Passive Function	Aging Management Reference
BLOWER CASING	PRESSURE BOUNDARY	Table 3.3-2 Line Number (2) Table 3.3-2 Line Number (3)
CONTROLLER[1]	PRESSURE BOUNDARY	Table 3.3-2 Line Number (4) Table 3.3-2 Line Number (5) Table 3.3-2 Line Number (6)
CS COMPONENTS	PRESSURE BOUNDARY	Table 3.3-1 Line Number (10)
FASTENERS (BOLTING)	JOINT INTEGRITY	Table 3.3-1 Line Number (10) Table 3.3-2 Line Number (11) Table 3.3-2 Line Number (9) Table 3.3-2 Line Number (10) Table 3.3-2 Line Number (11)
FILTER HOUSING	PRESSURE BOUNDARY	Table 3.3-2 Line Number (12) Table 3.3-2 Line Number (13)
FLOW ELEMENT	PRESSURE BOUNDARY	Table 3.3-2 Line Number (16) Table 3.3-2 Line Number (17)

Table 2.3.2-4 Containment Hydrogen Detectors and Recombiners

Component Group	Passive Function	Aging Management Reference
PIPE	PRESSURE BOUNDARY	Table 3.3-1 Line Number (3) Table 3.3-2 Line Number (40) Table 3.3-2 Line Number (42) Table 3.3-2 Line Number (43) Table 3.3-2 Line Number (44) Table 3.3-2 Line Number (45) Table 3.3-2 Line Number (46) Table 3.3-2 Line Number (49) Table 3.3-2 Line Number (50) Table 3.3-2 Line Number (51)
PUMP CASING	PRESSURE BOUNDARY	Table 3.3-2 Line Number (57) Table 3.3-2 Line Number (58)
RECOMBINER CASING	PRESSURE BOUNDARY	Table 3.3-2 Line Number (59) Table 3.3-2 Line Number (60)
VALVE BODY	PRESSURE BOUNDARY	Table 3.3-1 Line Number (3) Table 3.3-2 Line Number (69) Table 3.3-2 Line Number (71) Table 3.3-2 Line Number (74) Table 3.3-2 Line Number (84) Table 3.3-2 Line Number (85) Table 3.3-2 Line Number (86) Table 3.3-2 Line Number (88) Table 3.3-2 Line Number (89) Table 3.3-2 Line Number (90) Table 3.3-2 Line Number (91) Table 3.3-2 Line Number (92) Table 3.3-2 Line Number (93) Table 3.3-2 Line Number (94)
VENTILATION DUCTWORK	PRESSURE BOUNDARY	Table 3.3-2 Line Number (97) Table 3.3-2 Line Number (98)

1. Selected instruments were conservatively included within the scope of License Renewal. Consideration was given to the consequences of an instrument housing pressure boundary failure. Where an instrument was unisolable from a pressure source and is of sufficient size that a system function would be degraded should the pressure boundary fail, that instrument is included for License Renewal review.

2.3.2.5 Containment Isolation Components

System Description

Some plant support systems have no license renewal intended functions at the system level as described in the Updated Final Safety Analysis Report (UFSAR) but do have piping segments and valves that act as part of the containment pressure boundary. The Containment structure itself has piping and valves used for containment pressure sensors and for provisions to test the containment access hatches and containment isolation flange o-rings. The Containment Isolation Components System contains the non-structural equipment detailed in the UFSAR as performing a containment isolation boundary function where the system containing that equipment has no other safety related system function. Components evaluated in the Containment Isolation Components system are relied upon to achieve safe shutdown following some fires and contains components that are part of the Environmental Qualification Program.

The principal components of the Containment Isolation Boundary Components System include pipes and valves. A summary of the system lines penetrating Containment and the boundaries employed for containment isolation is presented in UFSAR Table 6.2-15a.

Each system whose piping penetrates the Containment boundary is designed to maintain or establish isolation of the Containment from the outside environment under any accident for which isolation is required, and assuming a coincident independent single failure or malfunction occurring in any active system component within the isolated bounds. Piping penetrating the Containment is designed for pressures at least equal to the containment design pressure. Containment isolation boundaries are provided as necessary in lines penetrating the Containment to ensure that no unrestricted release of radioactivity can occur.

The following Mechanical Systems interface with the Containment Isolation Components System:

Plant Air Systems
Plant Sampling
Heating Steam

System Function Listing

In addition to the System Functions described above, the Containment Isolation Boundary Components System also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code K PROVIDE PRIMARY CONTAINMENT BOUNDARY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Containment Isolation Components system perform this primary design system function. Some plant support systems have no license renewal intended functions at the system level as described in the Updated Final Safety Analysis Report (UFSAR) but do have piping segments and valves that act as part of the containment pressure boundary. The Containment structure itself has piping and valves used for containment pressure sensors and for provisions to test the containment access hatches and containment isolation flange o-rings.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Containment Isolation Components system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Containment Isolation Components system that perform License Renewal Criterion 3 functions are tracked under the Criterion 3 codes (Z1 through Z5). Components within the Containment Isolation Components system support monitoring Reg. Guide 1.97 Cat 3 variables.

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Containment Isolation Components system perform this associated design system function. Primary sample containment isolation valves are credited with maintaining RCS inventory for safe shutdown following fires.

Code Z2 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ENVIRONMENTAL QUALIFICATION (10 CFR 50.49)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: Components within the Containment Isolation Components system are designated as Environmentally Qualified (RCS sample containment isolation solenoid valve 14104S).

UFSAR Reference

Additional Containment Isolation Components System details are provided in Section 6.2.4 and Table 6.2-15a of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Containment Isolation Components System are listed below:

33013-1248	33013-1886,2
33013-1261	33013-1893
33013-1278,1	33013-1915
33013-1279	33013-1887
33013-1882	33013-1888
33013-1884,1	33013-1890
33013-1884,2	33013-1899,1

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.2-5 along with each Component Group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.2-5 Containment Isolation Components

Component Group	Passive Function	Aging Management Reference
CS COMPONENTS	PRESSURE BOUNDARY	Table 3.3-1 Line Number (10)
DELAY COIL	PRESSURE BOUNDARY	Table 3.3-1 Line Number (9) Table 3.3-2 Line Number (7)

Table 2.3.2-5 Containment Isolation Components

Component Group	Passive Function	Aging Management Reference
FASTENERS (BOLTING)	JOINT INTEGRITY	Table 3.3-1 Line Number (10) Table 3.3-2 Line Number (11) Table 3.3-2 Line Number (9) Table 3.3-2 Line Number (10) Table 3.3-2 Line Number (11)
FLANGE	PRESSURE BOUNDARY	Table 3.3-1 Line Number (4) Table 3.3-2 Line Number (14) Table 3.3-2 Line Number (15)
PIPE	PRESSURE BOUNDARY	Table 3.3-1 Line Number (4) Table 3.3-1 Line Number (9) Table 3.3-2 Line Number (40) Table 3.3-2 Line Number (41) Table 3.3-2 Line Number (43) Table 3.3-2 Line Number (44) Table 3.3-2 Line Number (47) Table 3.3-2 Line Number (49) Table 3.3-2 Line Number (50)
THERMOWELL	PRESSURE BOUNDARY	Table 3.3-2 Line Number (66) Table 3.3-2 Line Number (67)
VALVE BODY	PRESSURE BOUNDARY	Table 3.3-1 Line Number (4) Table 3.3-1 Line Number (9) Table 3.3-2 Line Number (70) Table 3.3-2 Line Number (72) Table 3.3-2 Line Number (73) Table 3.3-2 Line Number (74) Table 3.3-2 Line Number (75) Table 3.3-2 Line Number (86) Table 3.3-2 Line Number (87) Table 3.3-2 Line Number (88) Table 3.3-2 Line Number (89) Table 3.3-2 Line Number (91) Table 3.3-2 Line Number (92) Table 3.3-2 Line Number (93)

2.3.3 **Auxiliary Systems**

The following systems are addressed in this section:

- Chemical Volume and Control System (Section 2.3.3.1)
- Component Cooling Water System (Section 2.3.3.2)
- Spent Fuel Cooling and Fuel Storage System (Section 2.3.3.3)
- Waste Disposal System (Section 2.3.3.4)
- Service Water System (Section 2.3.3.5)
- Fire Protection System (Section 2.3.3.6)
- Heating Steam (Section 2.3.3.7)
- Emergency Power System (Section 2.3.3.8)
- Containment Ventilation Systems (Section 2.3.3.9)
- Essential Ventilation Systems (Section 2.3.3.10)
- Cranes, Hoists, and Lifting Devices (Section 2.3.3.11)
- Treated Water System (Section 2.3.3.12)
- Radiation Monitoring (Section 2.3.3.13)
- Circulating Water - Not Within Scope of License Renewal (Section 2.3.3.14)
- Chilled Water - Not Within Scope of License Renewal (Section 2.3.3.15)
- Fuel Handling - Not Within Scope of License Renewal (Section 2.3.3.16)
- Plant Sampling - Not Within Scope of License Renewal (Section 2.3.3.17)
- Plant Air - Not Within Scope of License Renewal (Section 2.3.3.18)
- Non-Essential Ventilation - Not Within Scope of License Renewal (Section 2.3.3.19)
- Site Service and Facility Support - Not Within Scope of License Renewal (Section 2.3.3.20)

2.3.3.1 **Chemical and Volume Control (CVCS)**

System Description

The Chemical and Volume Control System (CVCS) controls and maintains reactor coolant system inventory and purity through the process of makeup and letdown, and provides seal injection flow to the reactor coolant pump seals. In addition to the reactivity control achieved by the control rods, reactivity control is provided by CVCS, which regulates the concentration of boric acid solution neutron absorber in the reactor coolant system. In order to perform the above functions a continuous feed-and-bleed is maintained between the reactor coolant system and the chemical and volume control system. The CVCS is also credited for use in safe shutdown following Station Blackout events and some fire events. Selected large volume CVCS tanks are considered non safety equipment whose failure could affect a safety function due to their potential to cause flooding effects.

The principal components of CVCS are variable speed charging pumps, tanks, heat exchangers, demineralizers, and the essential piping and valves. The chemical and volume control system controls and maintains reactor coolant system inventory and purity through the process of makeup and letdown, and provides seal injection flow to the reactor coolant pump seals. The letdown portion of the system consists of a regenerative heat exchanger and a nonregenerative heat exchanger to cool the reactor coolant letdown and three parallel orifice valves to reduce the pressure. The coolant is passed through purification and deborating demineralizers, as necessary, where corrosion and fission products are removed. The coolant is then routed to the volume control tank. Seal return flow passes from the reactor coolant pump seals, through a containment isolation valve and the seal-water heat exchanger, before returning to the volume control tank. The seal return line is at low pressure and temperature. The charging pumps draw from the volume control tank and inject into the reactor coolant system, both through the normal makeup path and via the reactor coolant pump seals.

The following fluid systems interface with the Chemical and Volume Control System:

Reactor Coolant	Waste Disposal
Residual Heat Removal	Instrument Air
Spent Fuel Cooling and Fuel Storage	Service Water
Component Cooling Water	Treated Water

System Function Listing

In addition to the System Functions described above, the CVCS also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code J PROVIDE HEAT REMOVAL FROM SAFETY RELATED HEAT EXCHANGERS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Chemical and Volume Control system perform this associated design system function.

Code K PROVIDE PRIMARY CONTAINMENT BOUNDARY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Chemical and Volume Control system perform this associated design system function.

Code L PROVIDE EMERGENCY HEAT REMOVAL FROM PRIMARY CONTAINMENT AND PROVIDE CONTAINMENT PRESSURE CONTROL	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Chemical and Volume Control system perform this associated design system function. (Interface with RWST)

Code Q PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Chemical and Volume Control system perform this associated design system function.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Chemical and Volume Control system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Chemical and Volume Control system that perform special capability class functions are tracked under the Criterion 3 codes (Z1 through Z5). The “S” function also includes identifying components which provide or maintain sufficient reactor coolant inventory and reactivity control to achieve and maintain normal shutdown conditions.

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Chemical and Volume Control system perform this associated design system function.

Code X SFR FUNCTION NOT APPLICABLE AT COMPONENT LEVEL	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Chemical and Volume Control system perform specific safety related functions different from and in addition to the system level functions. (AOV-392A performs both a relief and an isolation function.)

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Chemical and Volume Control system perform this associated design system function. This function identifies non-safety tanks whose failure can affect safety related systems due to flooding. (BAST's, CVCS HUT's, RMW Tank, and VCT). In addition, some non-safety pipe is used to provide a relief path protecting containment isolation valves and piping.

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Chemical and Volume Control system perform this associated design system function.

Code Z5 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR STATION BLACKOUT (10 CFR 50.63)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: Components within the Chemical and Volume Control system perform this associated design system function.

UFSAR Reference

Additional Chemical and Volume Control System details are provided in Section 9.3.4 and Table 6.2-15a of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Chemical and Volume Control System are listed below:

33013-1245	33013-1269
33013-1246,2	33013-1274
33013-1247	33013-1278,2
33013-1258	33013-1891
33013-1260	33013-1887
33013-1262,1	33013-1888
33013-1264	33013-1889
33013-1265,1	33013-1890
33013-1265,2	33013-1892
33013-1266	33013-2274
33013-1267	33013-2275,1
33013-1268	33013-2275,2

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.3-1 along with each Component Group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.3-1 Chemical and Volume Control (CVCS)

Component Group	Passive Function	Aging Management Reference
CONDENSER	PRESSURE BOUNDARY	Table 3.4-2 Line Number (11) Table 3.4-2 Line Number (12) Table 3.4-2 Line Number (13) Table 3.4-2 Line Number (14) Table 3.4-2 Line Number (15) Table 3.4-2 Line Number (16)
COOLER	PRESSURE BOUNDARY	Table 3.4-2 Line Number (27) Table 3.4-2 Line Number (28) Table 3.4-2 Line Number (29) Table 3.4-2 Line Number (30) Table 3.4-2 Line Number (31) Table 3.4-2 Line Number (32)

Table 2.3.3-1 Chemical and Volume Control (CVCS)

Component Group	Passive Function	Aging Management Reference
CS COMPONENTS	PRESSURE BOUNDARY	Table 3.4-1 Line Number (13)
FASTENERS (BOLTING)	JOINT INTEGRITY	Table 3.4-1 Line Number (13) Table 3.4-1 Line Number (23) Table 3.4-2 Line Number (79) Table 3.4-2 Line Number (80) Table 3.4-2 Line Number (81)
FILTER HOUSING	PRESSURE BOUNDARY	Table 3.4-2 Line Number (99) Table 3.4-2 Line Number (101) Table 3.4-2 Line Number (102)
FLOW ELEMENT	PRESSURE BOUNDARY	Table 3.4-2 Line Number (107) Table 3.4-2 Line Number (108) Table 3.4-2 Line Number (109) Table 3.4-2 Line Number (110)
HEAT EXCHANGER	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (8) Table 3.4-1 Line Number (14) Table 3.4-2 Line Number (144) Table 3.4-2 Line Number (151)
PIPE	PRESSURE BOUNDARY	Table 3.4-2 Line Number (235) Table 3.4-2 Line Number (236) Table 3.4-2 Line Number (238) Table 3.4-2 Line Number (239) Table 3.4-2 Line Number (240) Table 3.4-2 Line Number (241) Table 3.4-2 Line Number (242) Table 3.4-2 Line Number (243) Table 3.4-2 Line Number (245) Table 3.4-2 Line Number (246) Table 3.4-2 Line Number (247) Table 3.4-2 Line Number (248) Table 3.4-2 Line Number (249) Table 3.4-2 Line Number (250)
PULSATION DAMPER	PRESSURE BOUNDARY	Table 3.4-2 Line Number (252) Table 3.4-2 Line Number (253) Table 3.4-2 Line Number (254)
PUMP CASING	PRESSURE BOUNDARY	Table 3.4-1 Line Number (4) Table 3.4-2 Line Number (270)

Table 2.3.3-1 Chemical and Volume Control (CVCS)

Component Group	Passive Function	Aging Management Reference
TANK	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (336) Table 3.4-2 Line Number (338) Table 3.4-2 Line Number (343) Table 3.4-2 Line Number (344) Table 3.4-2 Line Number (346) Table 3.4-2 Line Number (347) Table 3.4-2 Line Number (348) Table 3.4-2 Line Number (349) Table 3.4-2 Line Number (350) Table 3.4-2 Line Number (351)
TEMPERATURE ELEMENT[1]	PRESSURE BOUNDARY	Table 3.4-2 Line Number (353) Table 3.4-2 Line Number (356) Table 3.4-2 Line Number (357) Table 3.4-2 Line Number (358) Table 3.4-2 Line Number (359) Table 3.4-2 Line Number (360) Table 3.4-2 Line Number (361) Table 3.4-2 Line Number (365) Table 3.4-2 Line Number (366) Table 3.4-2 Line Number (367) Table 3.4-2 Line Number (368) Table 3.4-2 Line Number (369) Table 3.4-2 Line Number (370)
TRANSMITTER[1]	PRESSURE BOUNDARY	Table 3.4-2 Line Number (372) Table 3.4-2 Line Number (373) Table 3.4-2 Line Number (374) Table 3.4-2 Line Number (375) Table 3.4-2 Line Number (376) Table 3.4-2 Line Number (377) Table 3.4-2 Line Number (378) Table 3.4-2 Line Number (379) Table 3.4-2 Line Number (381) Table 3.4-2 Line Number (382)

Table 2.3.3-1 Chemical and Volume Control (CVCS)

Component Group	Passive Function	Aging Management Reference
VALVE BODY	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (384) Table 3.4-2 Line Number (385) Table 3.4-2 Line Number (386) Table 3.4-2 Line Number (393) Table 3.4-2 Line Number (394) Table 3.4-2 Line Number (398) Table 3.4-2 Line Number (399) Table 3.4-2 Line Number (400) Table 3.4-2 Line Number (401) Table 3.4-2 Line Number (402) Table 3.4-2 Line Number (403) Table 3.4-2 Line Number (405) Table 3.4-2 Line Number (406) Table 3.4-2 Line Number (407) Table 3.4-2 Line Number (408) Table 3.4-2 Line Number (409) Table 3.4-2 Line Number (410) Table 3.4-2 Line Number (411) Table 3.4-2 Line Number (412) Table 3.4-2 Line Number (425) Table 3.4-2 Line Number (429) Table 3.4-2 Line Number (432) Table 3.4-2 Line Number (444) Table 3.4-2 Line Number (446) Table 3.4-2 Line Number (447) Table 3.4-2 Line Number (452) Table 3.4-2 Line Number (453) Table 3.4-2 Line Number (454) Table 3.4-2 Line Number (455) Table 3.4-2 Line Number (456) Table 3.4-2 Line Number (457) Table 3.4-2 Line Number (459) Table 3.4-2 Line Number (461) Table 3.4-2 Line Number (462) Table 3.4-2 Line Number (463) Table 3.4-2 Line Number (464) Table 3.4-2 Line Number (465) Table 3.4-2 Line Number (466) Table 3.4-2 Line Number (467)

1. Selected instruments were conservatively included within the scope of License Renewal. Consideration was given to the consequences of an instrument housing pressure boundary failure. Where an instrument was unisolable from a pressure source and is of sufficient size that a system function would be degraded should the pressure boundary fail, that instrument is included for License Renewal review.

2.3.3.2 Component Cooling Water (CCW)

System Description

The Component Cooling Water (CCW) systems' function is to remove heat from safety related heat exchangers during plant operation, plant cooldown, and postaccident conditions. Components within the CCW system are also credited for use in safe shutdown following some fire events.

The principal components of the Component Cooling Water system are pumps, heat exchanges, the surge tank and the essential piping and valves. A single CCW pump circulates chromated water through parallel flow paths into various components where it picks up heat from other systems and transfers the heat to the service water (SW) system via the component cooling water (CCW) heat exchangers. The surge tank accommodates expansion, contraction, and inleakage of water, and ensures a continuous component cooling water (CCW) supply until a leaking cooling line can be isolated. The component cooling loop serves as an intermediate system between the radioactive fluid systems and the service water (SW) system. Since the component cooling water (CCW) system loop is used as an engineered safety feature, containment isolation valves are not automatically closed. That portion of the loop located outside the containment is not required to be a closed system.

The following fluid systems interface with the Component Cooling Water system:

Reactor Coolant	Safety Injection
Containment Spray	Plant Sampling
Waste Disposal	Service Water

System Function Listing

In addition to the System Functions listed above, the Component Cooling Water System also supports additional functions (associated design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

<table>
<tr><td rowspan="3">Code J
PROVIDE HEAT REMOVAL FROM SAFETY RELATED HEAT EXCHANGERS</td><td rowspan="2">Cri 1</td><td rowspan="2">Cri 2</td><td colspan="5">Cri 3</td></tr>
<tr><td>FP</td><td>EQ</td><td>PTS</td><td>AT</td><td>SB</td></tr>
<tr><td>X</td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
</table>

Comment: Components within the Component Cooling Water system perform this primary design system function.

Code K PROVIDE PRIMARY CONTAINMENT BOUNDARY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Component Cooling Water system perform this associated design system function.

Code Q PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Component Cooling Water system perform this associated design system function.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Component Cooling Water system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Component Cooling Water system that perform special capability class functions are tracked under the Criterion 3 codes (Z1 through Z5).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Component Cooling Water system perform this associated design system function.

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Component Cooling Water system perform this associated design system function.

UFSAR Reference

Additional Component Cooling Water System details are provided in Section 9.2.2 and Table 6.2-15a of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Component Cooling Water System are listed below:

33013-1245	33013-1887
33013-1246,1	33013-1888
33013-1246,2	33013-1889
33013-1250,2	33013-1890
33013-1273,1	33013-1892
33013-1891	33013-1899,1
33013-1251,2	

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.3-2 along with each Component Group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.3-2 Component Cooling Water (CCW)

Component Group	Passive Function	Aging Management Reference
COOLER	PRESSURE BOUNDARY	Table 3.4-2 Line Number (26) Table 3.4-2 Line Number (32)
CS COMPONENTS	PRESSURE BOUNDARY	Table 3.4-1 Line Number (13)
FASTENERS (BOLTING)	JOINT INTEGRITY	Table 3.4-1 Line Number (13) Table 3.4-1 Line Number (23) Table 3.4-2 Line Number (79) Table 3.4-2 Line Number (80) Table 3.4-2 Line Number (81)
FLOW ELEMENT	PRESSURE BOUNDARY	Table 3.4-2 Line Number (107) Table 3.4-2 Line Number (108) Table 3.4-2 Line Number (111)

Table 2.3.3-2 Component Cooling Water (CCW)

Component Group	Passive Function	Aging Management Reference
HEAT EXCHANGER	PRESSURE BOUNDARY HEAT TRANSFER	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (14) Table 3.4-2 Line Number (120) Table 3.4-2 Line Number (130) Table 3.4-2 Line Number (132) Table 3.4-2 Line Number (133) Table 3.4-2 Line Number (151) Table 3.4-2 Line Number (152) Table 3.4-2 Line Number (153) Table 3.4-2 Line Number (154) These apply to the pressure boundary passive function. Table 3.4-1 Line Number (16) Table 3.4-2 Line Number (137) Table 3.4-2 Line Number (138) Table 3.4-2 Line Number (140) Table 3.4-2 Line Number (141) Table 3.4-2 Line Number (142) These apply to the heat transfer passive function.
INDICATOR[1]	PRESSURE BOUNDARY	Table 3.4-2 Line Number (175) Table 3.4-2 Line Number (176) Table 3.4-2 Line Number (179)
ORIFICE	PRESSURE BOUNDARY	Table 3.4-2 Line Number (196) Table 3.4-2 Line Number (200)
PIPE	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (14) Table 3.4-2 Line Number (221) Table 3.4-2 Line Number (228) Table 3.4-2 Line Number (234) Table 3.4-2 Line Number (235) Table 3.4-2 Line Number (244)
PUMP CASING	PRESSURE BOUNDARY	Table 3.4-1 Line Number (14) Table 3.4-2 Line Number (262)
SWITCH[1]	PRESSURE BOUNDARY	Table 3.4-2 Line Number (332) Table 3.4-2 Line Number (333)

Table 2.3.3-2 Component Cooling Water (CCW)

Component Group	Passive Function	Aging Management Reference
TANK	PRESSURE BOUNDARY	Table 3.4-2 Line Number (343) Table 3.4-2 Line Number (352)
TEMPERATURE ELEMENT[1]	PRESSURE BOUNDARY	Table 3.4-2 Line Number (354) Table 3.4-2 Line Number (362)
TRANSMITTER[1]	PRESSURE BOUNDARY	Table 3.4-2 Line Number (373) Table 3.4-2 Line Number (380)
VALVE BODY	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (14) Table 3.4-2 Line Number (386) Table 3.4-2 Line Number (394) Table 3.4-2 Line Number (404) Table 3.4-2 Line Number (425) Table 3.4-2 Line Number (429) Table 3.4-2 Line Number (436) Table 3.4-2 Line Number (444) Table 3.4-2 Line Number (446) Table 3.4-2 Line Number (447)

1. Selected instruments were conservatively included within the scope of License Renewal. Consideration was given to the consequences of an instrument housing pressure boundary failure. Where an instrument was unisolable from a pressure source and is of sufficient size that a system function would be degraded should the pressure boundary fail, that instrument is included for License Renewal review.

2.3.3.3 Spent Fuel Cooling and Fuel Storage

System Description

The spent fuel pool (SFP) cooling system is designed to remove heat from the SFP, which is generated by stored spent fuel. The heat from the SFP is rejected to the Service Water System. The spent fuel pool is a Seismic Category I design, reinforced-concrete structure totally clad with stainless steel. The SFP provides structural support to the spent fuel racks. The spacing and materials of construction of the spent fuel racks work in conjunction with the spent fuel pool water chemistry to provide reactivity control. The SFP concrete elements are evaluated within the Auxiliary Building structure.

The principal components of the Spent Fuel Cooling and Fuel Storage system include pumps, tanks, heat exchangers and the essential piping and valves. Hoses are used to connect the skid-mounted equipment into the system. The new and spent fuel storage racks, and the pool and transfer canal liner are included as components within the Spent Fuel Cooling and Fuel Storage system.

The spent fuel cooling system was originally designed as a single train, non-safety system. The system has been modified to add additional cooling flow paths and equipment. The SFP cooling system now consists of three cooling loops. The primary cooling path is loop B. This loop is safety related, seismically qualified, and functions as the preferred system for ensuring adequate cooling in the SFP. The backup loops include permanently installed loop A and a skid-mounted loop. Together these loops act as a 100% back up to the B loop in that they are capable of removing the decay heat from stored spent fuel and a full core off load.

SFP cooling piping is so arranged that failure of any pipeline does not drain the SFP. To protect against the possibility of complete loss of water in the SFP, the upper suction line penetrates the SFP near the top of the pool. The lower suction line penetrates SFP approximately 5 ft.-4 in. above the top of the fuel racks to preclude the possibility of draining the pool and to ensure a minimum water level of 5 ft.-4 in. above the top of the fuel. The SFP cooling water return line, which terminates at the bottom of the SFP, contains an 0.25-in. vent hole near the normal SFP water level so that the pool water cannot be siphoned. The clarity and purity of the spent fuel pool water is maintained by passing approximately 60 gpm of the loop flow through a filter and demineralizer.

The original spent fuel storage racks provided capacity for the storage of 210 fuel assemblies. In 1976, the NRC approved the replacement of the original racks with higher density flux trap type. This expanded the storage capability from 210 to 595 fuel assemblies. In 1984, the NRC approved the conversion of six flux trap type racks to high-density fixed poison type racks. This further expanded the storage capacity from 595 to 1016 fuel assemblies. At this point, the spent fuel pool was divided into two regions. Region 1 comprised three flux trap type racks to accommodate a full core off-load. Region 2 consisted of six high-density fixed poison (Boraflex) type racks for the storage of 840 fuel assemblies that satisfied minimum burnup criteria and had cooled for a minimum of 60 days.

In 1998, the NRC approved re-racking the spent fuel pool. This re-rack effort will be done in two phases, reconfigured the pool to accommodate a net increase of 353 locations. This is accomplished by retaining the six existing high-density region 2 racks (840 minus 12 for attachment of new racks = 828 locations) and installing new borated stainless steel (BSS) racks with up to 541 additional storage locations for a total of 1369 storage locations after completion of both phases.

After completion of phase 1 of the re-rack, the pool has three types of racks in two regions. Region 1 contains new high-density flux-trap design BSS racks designated as type 3 for fresh and spent fuel. Region 2 contains the existing Boraflex racks designated as type 1 and new high-density BSS racks designated as type 2. With the completion of phase 1, the pool contains 1321 storage locations. Phase 2 has not yet been performed.

As noted above, Boraflex fixed absorber material is provided in the region 2 type 1 racks of the spent fuel pool. The absorber assemblies are welded in place in each storage cell, thus precluding inadvertent mechanical removal. To address concerns with Boraflex degradation as presented in Generic Letter 96-04, RG&E performed tests in February 1998 of the B-10 areal density of 24 representative Boraflex panels in region 2 of the spent fuel pool using the Boron Areal Density Gauge for Evaluating Racks (BADGER). During the testing, degradation beyond the four inch gap assumption of the criticality analysis was noted on selected Boraflex panels. This data indicated that some panels had undergone dissolution beyond expected levels and placed the spent fuel pool in an unanalyzed condition. This event and the results of the associated assessment that was performed were reported to the NRC. In addition, the Technical Specifications were changed to ensure that controls are in place to verify at least 2300 ppm of soluble boron is maintained in the spent fuel pool. Consequently, Boraflex is not relied upon for reactivity control of the stored spent fuel.

New fuel is delivered by truck to the site in approved containers. The assemblies are removed, inspected, and transferred to the new fuel storage racks using the auxiliary building crane. The storage location on the operating level of the auxiliary building facilitates the unloading of trucks and the transfer of the fuel assemblies. The Seismic Category I storage vault contains specially constructed racks which ensure a minimum 20-in. center-to-center spacing of the new fuel assemblies. This spacing ensures a KEFF less than 0.95 for the accidental full water density flooding scenario and less than 0.98 for the accidental low water density (optimum moderation) flooding scenario. The storage area is located above grade, on the Auxiliary Building operating floor, to help prevent this from occurring. The new fuel storage area is configured to store 12 fuel assemblies.

Seismic II/I and Heavy Loads (NUREG-0612) interfacing issues with respect to the Spent Fuel Pool and Fuel Storage system are addressed in Auxiliary Building and Fuel Handling equipment evaluations.

The following mechanical systems interface with the Spent Fuel Cooling and Fuel Storage system:

Safety Injection System
Chemical and Volume Control
Service Water
Waste Disposal
Treated Water

System Function Listing

In addition to the System Intended Functions listed above, the Spent Fuel Cooling and Fuel Storage System also contains components which support additional functions (associated design system functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code J	Cri 1	Cri 2	Cri 3				
PROVIDE HEAT REMOVAL FROM SAFETY RELATED HEAT EXCHANGERS			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Spent Fuel Cooling and Fuel Storage system perform this associated design system function.

Code K	Cri 1	Cri 2	Cri 3				
PROVIDE PRIMARY CONTAINMENT BOUNDARY			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Spent Fuel Cooling and Fuel Storage system perform this associated design system function.

Code P	Cri 1	Cri 2	Cri 3				
ENSURE ADEQUATE COOLING IN THE SPENT FUEL POOL			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Spent Fuel Cooling and Fuel Storage system perform this primary design system function.

Code Q PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Spent Fuel Cooling and Fuel Storage system perform this associated design system function.

Code R STRUCTURALLY SUPPORT OR HOUSE SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Spent Fuel Cooling and Fuel Storage system perform this primary design system function. In addition to pool cooling, the Spent Fuel Cooling system boundary includes the new and spent fuel storage racks and the spent fuel pool liner.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Spent Fuel Cooling and Fuel Storage system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Spent Fuel Cooling and Fuel Storage system that perform special capability class functions are tracked under the System Function code Y (Criterion 2). Components within the Spent Fuel Cooling and Fuel Storage system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Spent Fuel Cooling and Fuel Storage system that perform special capability class functions are tracked under the System Function code Y (Criterion 2).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Spent Fuel Cooling and Fuel Storage system perform this associated design system function.

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Spent Fuel Cooling and Fuel Storage system perform this associated design system function. Ginna licensing basis includes one safety related Spent Fuel Cooling and Fuel Storage Loop. Due to the importance of providing backup cooling, equipment which supports the backup cooling function has been conservatively included in the scope of License Renewal and assigned this associated design system function. The Spent Fuel Cooling and Fuel Storage system contains numerous occurrences of safety class changes at locations containing open valves. These locations are where process line instrument connections transition from pipe to tube at instruments which do not provide safety related information. For purposes of License Renewal, these areas are considered in scope and have been assigned this associated system design function.

UFSAR Reference

Additional Spent Fuel Cooling and Fuel Storage System details are provided in Section 9.1.1, Section 9.1.2, Section 9.1.3, and Table 6.2-15a of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Spent Fuel Cooling and Fuel Storage System are listed below:

33013-1248 33013-1250,2

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.3-3 along with each Component Group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.3-3 Spent Fuel Cooling and Fuel Storage

Component Group	Passive Function	Aging Management Reference
CS COMPONENTS	PRESSURE BOUNDARY	Table 3.4-1 Line Number (13)
DEMINERALIZER	PRESSURE BOUNDARY	Table 3.4-2 Line Number (52) Table 3.4-2 Line Number (53) Table 3.4-2 Line Number (54)

Table 2.3.3-3 Spent Fuel Cooling and Fuel Storage

Component Group	Passive Function	Aging Management Reference
DIAPHRAGM SEAL	PRESSURE BOUNDARY	Table 3.4-2 Line Number (55) Table 3.4-2 Line Number (56)
FASTENERS (BOLTING)	JOINT INTEGRITY	Table 3.4-1 Line Number (13) Table 3.4-1 Line Number (23) Table 3.4-2 Line Number (79) Table 3.4-2 Line Number (80) Table 3.4-2 Line Number (81)
FILTER HOUSING	PRESSURE BOUNDARY	Table 3.4-2 Line Number (99) Table 3.4-2 Line Number (101) Table 3.4-2 Line Number (102)
FLOW ELEMENT	PRESSURE BOUNDARY	Table 3.4-2 Line Number (108) Table 3.4-2 Line Number (109) Table 3.4-2 Line Number (110)
HEAT EXCHANGER	PRESSURE BOUNDARY HEAT TRANSFER	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (119) Table 3.4-2 Line Number (144) Table 3.4-2 Line Number (146) Table 3.4-2 Line Number (148) Table 3.4-2 Line Number (149) Table 3.4-2 Line Number (150) These apply to the pressure boundary passive function. Table 3.4-2 Line Number (145) Table 3.4-2 Line Number (147) These apply to the heat transfer passive function.
INDICATOR[1]	PRESSURE BOUNDARY	Table 3.4-2 Line Number (176) Table 3.4-2 Line Number (177) Table 3.4-2 Line Number (178)
PIPE	PRESSURE BOUNDARY	Table 3.4-2 Line Number (236) Table 3.4-2 Line Number (238) Table 3.4-2 Line Number (239)
PULSATION DAMPER	PRESSURE BOUNDARY	Table 3.4-2 Line Number (252) Table 3.4-2 Line Number (253) Table 3.4-2 Line Number (254)

Table 2.3.3-3 Spent Fuel Cooling and Fuel Storage

Component Group	Passive Function	Aging Management Reference
PUMP CASING	PRESSURE BOUNDARY	Table 3.4-2 Line Number (270) Table 3.4-2 Line Number (274) Table 3.4-2 Line Number (275)
SPECTACLE FLANGE	PRESSURE BOUNDARY	Table 3.4-2 Line Number (295) Table 3.4-2 Line Number (296) Table 3.4-2 Line Number (297)
STRAINER HOUSING	PROVIDE FILTRATION	Table 3.4-2 Line Number (306) Table 3.4-2 Line Number (307)
STRUCTURE	PROVIDE RADIATION SHIELD STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.4-1 Line Number (9) Table 3.4-1 Line Number (11) These apply to both passive functions. Table 3.4-1 Line Number (10) This applies to the structural support SR equipment passive function.
TANK	PRESSURE BOUNDARY	Table 3.4-2 Line Number (343) Table 3.4-2 Line Number (346) Table 3.4-2 Line Number (347)
TEMPERATURE ELEMENT[1]	PRESSURE BOUNDARY	Table 3.4-2 Line Number (354) Table 3.4-2 Line Number (356) Table 3.4-2 Line Number (357)
VALVE BODY	PRESSURE BOUNDARY	Table 3.4-2 Line Number (394) Table 3.4-2 Line Number (398) Table 3.4-2 Line Number (399) Table 3.4-2 Line Number (430) Table 3.4-2 Line Number (435) Table 3.4-2 Line Number (447) Table 3.4-2 Line Number (452) Table 3.4-2 Line Number (453)

1. Selected instruments were conservatively included within the scope of License Renewal. Consideration was given to the consequences of an instrument housing pressure boundary failure. Where an instrument was unisolable from a pressure source and is of sufficient size that a system function would be degraded should the pressure boundary fail, that instrument is included for License Renewal review.

2.3.3.4 Waste Disposal

System Description

The Waste Disposal System provides equipment necessary to collect, process, and prepare for disposal of potentially radioactive liquid, gaseous, and solid wastes produced as a result of reactor operation. Radioactive fluids entering the waste disposal system are collected in sumps and tanks until determination of subsequent treatment methods can be made. The consequences of a Radioactive release from a subsystem or component are evaluated in UFSAR Section 15.7 which concludes that accidental gaseous and liquid radioactive releases from the Waste Disposal system will not pose a safety hazard to the public relative to 10 CFR 100 releases. The Waste Disposal system contains two environmentally qualified sump pumps, which discharge to the waste holdup tank. The waste holdup tank provides a holdup capacity reserved to abate RHR pump seal failure spillage. Other system tanks contain volumes of liquid, which if spilled, could prevent the satisfactory accomplishment of a safety-related function. Additionally components within the system act in concert with structural features to prevent internal floods from propagating.

The principal components of the Waste Disposal system are the demineralizing systems, the waste gas compressors, tanks and the essential piping, pumps and valves. Liquid wastes requiring cleanup before release are collected and processed by a vendor supplied demineralization system. Gaseous waste is re-used as tank cover gas or stored for decay and subsequent release.

The following fluid systems interface with the Waste Disposal System:

Reactor Coolant	Residual Heat Removal
Instrument Air	Containment Spray
Spent Fuel Cooling and Fuel Storage	Chemical and Volume Control
Component Cooling Water	Service Water
Safety Injection	

System Function Listing

In addition to the System Functions described above, the Waste Disposal System also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code J	Cri 1	Cri 2	Cri 3				
PROVIDE HEAT REMOVAL FROM SAFETY RELATED HEAT EXCHANGERS			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Waste Disposal system perform this associated design system function (Component Cooling and Service Water interface with heat exchangers).

Code K	Cri 1	Cri 2	Cri 3				
PROVIDE PRIMARY CONTAINMENT BOUNDARY			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Waste Disposal system perform this associated design system function.

Code P	Cri 1	Cri 2	Cri 3				
ENSURE ADEQUATE COOLING IN THE SPENT FUEL POOL			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Waste Disposal system perform this associated design system function (Waste Disposal interfaces with transfer slot drain).

Code Q	Cri 1	Cri 2	Cri 3				
PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Waste Disposal system perform this associated design system function.

Code S	Cri 1	Cri 2	Cri 3				
SPECIAL CAPABILITY CLASS FUNCTIONS			FP	EQ	PTS	AT	SB

Comment: Components within the Waste Disposal system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Waste Disposal system that perform License Renewal Criterion 3 functions are tracked under the Criterion 3 codes (Z1 through Z5).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Waste Disposal system perform this associated design system function.

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Waste Disposal system perform this associated design system function. (Flood protection, backflow of oil through floor drains)

Code Z2 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ENVIRONMENTAL QUALIFICATION (10 CFR 50.49)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: Components within the Waste Disposal system are designated as Environmentally Qualified (Residual Heat Removal pit sump pumps).

UFSAR Reference

Additional Waste Disposal System details are provided in Section 11.2, Section 11.3, Section 11.4, Section 3.4.2, Section 9.3.3, and Table 6.2-15a of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Waste Disposal System are listed below:

33013-1246,2
33013-1247
33013-1258
33013-1261
33013-1262,1
33013-1262,2
33013-1265,2
33013-1267
33013-1268
33013-1270,1
33013-1272,2
33013-1273,1
33013-1274
33013-1278,2
33013-1891
33013-2287
33013-1259
33013-1270,2
33013-1271
33013-1272,1
33013-1273,2
33013-1276
33013-1277,2
33013-1887
33013-1888
33013-1889
33013-1890
33013-1892
33013-1895
33013-1899,1
33013-1900,1
33013-2279,2
33013-2280
33013-2289
33013-2742

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.3-4 along with each Component Group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.3-4 Waste Disposal

Component Group	Passive Function	Aging Management Reference
CS COMPONENTS	PRESSURE BOUNDARY	Table 3.4-1 Line Number (13)
FASTENERS (BOLTING)	JOINT INTEGRITY	Table 3.4-1 Line Number (13) Table 3.4-1 Line Number (23) Table 3.4-2 Line Number (79) Table 3.4-2 Line Number (80) Table 3.4-2 Line Number (81)
HEAT EXCHANGER	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (14) Table 3.4-2 Line Number (132)

Table 2.3.3-4 Waste Disposal

Component Group	Passive Function	Aging Management Reference
ORIFICE	PRESSURE BOUNDARY RESTRICTS FLOW	Table 3.4-2 Line Number (197) This applies to the pressure boundary passive function. Table 3.4-2 Line Number (199) This applies to both passive functions.
PIPE	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (202) Table 3.4-2 Line Number (229) Table 3.4-2 Line Number (230) Table 3.4-2 Line Number (231) Table 3.4-2 Line Number (236) Table 3.4-2 Line Number (237)
PUMP CASING	PRESSURE BOUNDARY	Table 3.4-2 Line Number (256) Table 3.4-2 Line Number (258) Table 3.4-2 Line Number (270) Table 3.4-2 Line Number (272) Table 3.4-2 Line Number (273) Table 3.4-2 Line Number (274) Table 3.4-2 Line Number (275)
TANK	PRESSURE BOUNDARY	Table 3.4-2 Line Number (343) Table 3.4-2 Line Number (345)
VALVE BODY	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (386) Table 3.4-2 Line Number (391) Table 3.4-2 Line Number (393) Table 3.4-2 Line Number (394) Table 3.4-2 Line Number (397) Table 3.4-2 Line Number (398) Table 3.4-2 Line Number (399) Table 3.4-2 Line Number (428) Table 3.4-2 Line Number (429) Table 3.4-2 Line Number (444) Table 3.4-2 Line Number (446) Table 3.4-2 Line Number (447) Table 3.4-2 Line Number (451) Table 3.4-2 Line Number (452) Table 3.4-2 Line Number (453)

2.3.3.5 Service Water (SW)

System Description

The Service Water (SW) system takes suction from the ultimate heat sink and supplies the cooling water used to provide heat removal from safety related heat exchangers. SW is also the normal suction supply to the standby auxiliary feedwater system and an alternate supply to the preferred auxiliary feedwater system where it is used to provide emergency heat removal from the reactor coolant system using secondary heat removal capability. The SW system is also credited for use in safe shutdown following some fires. The SW system provides multiple water source flow paths to ensure the availability of the ultimate heat sink. These flow paths include non-safety related equipment whose failure could prevent the satisfactory accomplishment of a safety related function. Portions of the SW distribution system serving safeguards equipment are designed as Seismic Category I. Other portions of the service water (SW) system serving non-safety loads are designated as nonseismic and are capable of being isolated from the Seismic Category I portion.

The principal components of the SW system are four service water pumps, a single loop supply header, essential isolation valves, and other essential piping including the normal and standby discharge header and the intake piping systems that transports water from the lake to the SW pump suction bay.

The SW system consists of a single loop header supplied by two separate, 100% capacity, safety related pump trains. The loop header supplies the cooling water to safety related and non-safety related components and system heat exchangers inside the containment, auxiliary, intermediate, turbine, and diesel generator buildings. The non-safety related and long-term safety functions (e.g., component cooling water heat exchangers) can be isolated from the loop header through use of redundant motor operated isolation valves. In addition to supplying cooling water to heat exchangers, the system supplies seal water to the circulating water pumps and the vacuum pumps, flushing water to the traveling screens and makeup water to the fire water storage tank via the fire booster pump.

The following fluid systems interface with the Service Water System:

Safety Injection	Waste Disposal
Residual Heat Removal	Plant Air
Containment Spray	Spent Fuel Cooling and Fuel Storage
Chemical and Volume Control	Component Cooling Water

Auxiliary Feedwater	Fire Protection
Circulating Water	Treated Water
Plant Sampling	

System Function Listing

In addition to the System Functions described above, the Service Water System also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code H	Cri 1	Cri 2	Cri 3				
PROVIDE EMERGENCY HEAT REMOVAL FROM THE REACTOR COOLANT SYSTEM USING SECONDARY HEAT REMOVAL CAPABILITY			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Service Water system perform this primary design system function (emergency source of feedwater).

Code J	Cri 1	Cri 2	Cri 3				
PROVIDE HEAT REMOVAL FROM SAFETY RELATED HEAT EXCHANGERS			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Service Water system perform this primary design system function.

Code K	Cri 1	Cri 2	Cri 3				
PROVIDE PRIMARY CONTAINMENT BOUNDARY			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Service Water system perform this associated design system function.

Code Q	Cri 1	Cri 2	Cri 3				
PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Service Water system perform this associated design system function.

Code S	Cri 1	Cri 2	Cri 3				
SPECIAL CAPABILITY CLASS FUNCTIONS			FP	EQ	PTS	AT	SB

Comment: Components within the Service Water system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Service Water system that perform special capability class functions are tracked under the Criterion 2 code (Y) and Criterion 3 codes (Z1 through Z5). Control Room cooling, Reg Guide 1.97, Backup Spent Fuel Pool Cooling.

Code T	Cri 1	Cri 2	Cri 3				
NON-NUCLEAR SAFETY CLASS FUNCTIONS			FP	EQ	PTS	AT	SB

Comment: Components within the Service Water system perform this associated design system function. Cooling to non-safety related loads, e.g., Hydrogen cooler, Bus Duct coolers, etc.

Code Y	Cri 1	Cri 2	Cri 3				
LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Service Water system perform this associated design system function. Non-safety components within the Service Water system provide an alternate flow paths to ensure the availability of the ultimate heat sink in the event of loss of circulating water. Service Water also supplies cooling to the skid mounted SFP Heat Exchanger via temporary connections.

Code Z1	Cri 1	Cri 2	Cri 3				
LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Service Water system perform this associated design system function.

UFSAR Reference

Additional Service Water System details are provided in Section 3.3.3.3.7, Section 9.2.1, and Table 6.2-15a of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Service Water System are listed below:

33013-1237
33013-1238
33013-1250,1
33013-1250,2
33013-1250,3
33013-1893
33013-1908,3
33013-1251,1
33013-1251,2
33013-1885,1
33013-1885,2
33013-1892
33013-1894,2
33013-1895
33013-1900,1
33013-1921
33013-1925

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.3-5 along with each Component Group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.3-5 Service Water (SW)

Component Group	Passive Function	Aging Management Reference
CS COMPONENTS	PRESSURE BOUNDARY	Table 3.4-1 Line Number (13)
EXPANSION JOINT	PRESSURE BOUNDARY	Table 3.4-2 Line Number (65)
FASTENERS (BOLTING)	JOINT INTEGRITY	Table 3.4-1 Line Number (13) Table 3.4-1 Line Number (23) Table 3.4-2 Line Number (79) Table 3.4-2 Line Number (80) Table 3.4-2 Line Number (81)
FLOW ELEMENT	PRESSURE BOUNDARY	Table 3.4-1 Line Number (16) Table 3.4-2 Line Number (108)

Table 2.3.3-5 Service Water (SW)

Component Group	Passive Function	Aging Management Reference
INDICATOR[1]	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (16) Table 3.4-2 Line Number (167) Table 3.4-2 Line Number (169) Table 3.4-2 Line Number (176)
PIPE	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (16) Table 3.4-2 Line Number (207) Table 3.4-2 Line Number (208) Table 3.4-2 Line Number (210) Table 3.4-2 Line Number (211) Table 3.4-2 Line Number (212) Table 3.4-2 Line Number (220) Table 3.4-2 Line Number (224) Table 3.4-2 Line Number (226) Table 3.4-2 Line Number (235) Table 3.4-2 Line Number (236)
PUMP CASING	PRESSURE BOUNDARY	Table 3.4-1 Line Number (16)
STRAINER HOUSING	PRESSURE BOUNDARY PROVIDE FILTRATION	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (303) Table 3.4-2 Line Number (304) These apply to the pressure boundary passive function. Table 3.4-1 Line Number (16) This applies to both passive functions.
STRUCTURE	HOUSE, PROTECT EQUIPMENT	Table 3.4-2 Line Number (309) Table 3.4-2 Line Number (310)
SWITCH[1]	PRESSURE BOUNDARY	Table 3.4-1 Line Number (16) Table 3.4-2 Line Number (331) Table 3.4-2 Line Number (332)
TEMPERATURE ELEMENT[1]	PRESSURE BOUNDARY	Table 3.4-1 Line Number (16) Table 3.4-2 Line Number (353) Table 3.4-2 Line Number (354)

Table 2.3.3-5 Service Water (SW)

Component Group	Passive Function	Aging Management Reference
VALVE BODY	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (16) Table 3.4-2 Line Number (393) Table 3.4-2 Line Number (396) Table 3.4-2 Line Number (417) Table 3.4-2 Line Number (421) Table 3.4-2 Line Number (422) Table 3.4-2 Line Number (423) Table 3.4-2 Line Number (429) Table 3.4-2 Line Number (446) Table 3.4-2 Line Number (447) Table 3.4-2 Line Number (449)

1. Selected instruments were conservatively included within the scope of License Renewal. Consideration was given to the consequences of an instrument housing pressure boundary failure. Where an instrument was unisolable from a pressure source and is of sufficient size that a system function would be degraded should the pressure boundary fail, that instrument is included for License Renewal review.

2.3.3.6 Fire Protection (FP)

System Description

The Fire Protection system relies on a strategy that includes combustible materials control, fire detection, fire confinement and fire suppression.

Fixed water spray and sprinkler systems, fixed Halon systems, hose lines, along with portable and wheeled extinguishers, provide fire suppression. A fire header supplies water from Lake Ontario, using a motor driven or diesel driven fire pump, to the suppression systems and hose reel stations. There is also a contingency to use a fire department pumper truck is to provide back-up fire suppression ability.

A city water yard loop supplies water from the Town of Ontario to the plant fire hydrants and to the Screenhouse service water pump area sprinkler system. The city water yard loop is also used as a backup cooling water supply to the emergency diesel generators and the standby auxiliary feedwater pumps in the event that service water is unavailable.

The fire water system can be used as a backup for the service water system supply to spent fuel pool heat exchanger A, the standby spent fuel pool heat exchanger, motor driven auxiliary feedwater pumps, standby auxiliary feedwater pumps, and the diesel generator lube-oil coolers and jacket water heat exchangers via. temporary hoses. Provisions are also available to use firewater to flush debris off the Circulating Water intake travelling screens when a high-pressure wash is required.

The fire system has a protective signaling system that alarms locally in selected parts of the plant and transmits fire alarm, supervisory, and trouble signals to the control room. Fire barriers are located throughout the plant to separate established fire areas from each other and also to separate certain safety areas from the remainder of the plant. These barriers are designed to stop a fire from propagating from one area to the other. Fire areas have been defined, based upon separation of equipment and essential safe shutdown cables, to ensure that at least one train of safe shutdown equipment is kept free from the effects of a fire. Fire prevention and mitigation considerations have been included in the design of ventilation systems, drain systems, lighting systems, communications systems, electrical and instrument cables, oil collection systems, and civil layout and materials selections. Materials selection and transient combustible material controls help ensure that the Fire Protection system is capable of achieving its suppression function.

The principal components of the Fire Protection system are two segregated fire headers, a diesel and a motor driven pump, the fire water tank, hoses and hose stations, hydrants and the essential piping, spray heads, nozzles and valves. The system also includes two fixed halon gas suppression systems with their corresponding tanks, pipes, valves and nozzles. Detection, signaling, valve alignment and temperature monitoring systems provide information and automatic controls. Fire barriers, penetration seals, ventilation dampers and doors are addressed in a separate commodity group evaluation as described below.

Fire Barrier Commodity Group

The fire hazards analysis submitted to the NRC in February 1977 identified the fire barriers in the plant and the requirements for maintaining their integrity. The Fire Protection Program Report (EPM-FPPR) identifies all the barriers currently necessary as well as their requirements. These barrier requirements were determined by the fire loading calculated for each area subject to a potential fire hazard. As a result of this analysis, several design modifications were implemented at the plant including upgrading of the rating of original barriers and installing new barriers. Additional barriers were also installed to achieve compliance with 10 CFR 50 Appendix R requirements. Fire barriers are located throughout the plant to separate major areas from each other and also to separate certain safety-related areas from the remainder of the plant. These are designed to stop a fire from propagating from one area to another or to minimize the effects of a fire. Penetrations in these barriers are sealed with appropriate materials to match the requirements of the barrier and have been evaluated to demonstrate they provide an acceptable level of fire protection.

A fire barrier is a continuous vertical or horizontal membrane, such as a wall or floor/ceiling assembly, that is designed and constructed with a specified fire resistance rating. Fire barriers limit the spread or damage of fire and may restrict the movement of smoke. Some barriers may have protected openings.

The principal types of fire barriers include:

Fire Rated Assembly - A passive fire protection feature that is used to separate redundant fires safe shutdown capabilities. A fire rated assembly includes fire rated walls, floors, ceilings, equipment hatches, stairwells, doors, dampers, and penetration seals.

Fire Rated Penetration Seal - An opening in a fire barrier for the passage of pipe, cable, etc., which has been sealed so as not to reduce the integrity of the fire barrier.

Internal Conduit Seals:

A. Smoke and Hot Gas Seals - Noncombustible seals installed inside conduit openings to prevent the passage of smoke and hot gasses through fire barriers. These seals may be located at the fire barrier or at the nearest conduit entry on both sides of the fire barrier. Smoke and hot gas seals are not required to have a fire resistance rating equal to the fire barrier they are installed in.

B. Heat and Fire Seals - Fire rated seals installed inside conduits at or in close proximity to the fire barrier. Heat and fire seals have the same or greater fire resistance rating as the fire barrier they are installed in.

Penetration Seal - Materials, devices, or assemblies installed in communicating spaces across barriers, which provide effective sealing against defined environmental exposure criteria to achieve the same functional requirement as that originally intended by the structural member or the barrier.

Fire Wall - A wall having adequate fire resistance and structural stability under fire conditions to accomplish the purpose of subdividing buildings to restrict the spread of fire.

Fire Break (Fire Stop) - A passive fire protection feature of construction intended to limit flame propagation along vertical or horizontal cable tray runs.

Fire Damper - A device, installed in the air distribution system, designed to close automatically upon detection of heat, to interrupt migratory air flow, and to restrict the passage of flame.

Fire Door - The door component of a fire door assembly.

Fire Door Assembly - Any combination of a fire door, frame, hardware, and other accessories, that together provide a specific degree of fire protection to the opening.

Fire Wrap - A passive fire and/or heat resistant covering (Hymec Wrap) used to protect or shield safe shutdown circuits.

Fire Proofing - A passive cementitious coating applied to steel to provide fire resistance.

The structural components of fire barriers are evaluated with the civil structure that contains the penetration (e.g. block walls, concrete floors, etc.). That notwithstanding, the Fire Barrier Commodity Group includes fire doors and door frames (including roll up doors) and fire dampers and damper frames, along with any penetration seals that may make up the transition between those elements and a civil structure. Also included in the commodity group is any additional framing and material used to construct fire resistant enclosures within civil structures (e.g. the turbine seal oil unit enclosure).

Fire barriers are constructed, maintained, and inspected in accordance with specific procedures subject to rigorous quality control measures. Penetration and barrier specific details such as materials configuration, depth of sealing materials, and inspection acceptance criteria are documented and controlled in accordance with special procedures and processes.

The Fire Barrier Commodity Group evaluation boundary includes all of the barrier types described above. Although the generic commodity group review uses representative asset numbers to reflect barrier materials, each fire barrier is labeled in the plant with a unique number. Plant procedures and drawings specifically detail the construction, repair and inspection criteria distinctive to the specific application. Plant procedures and drawing also distinguish which barrier is credited in the licensing basis with respect to fire protection and which barrier is installed for commercial property conservation. The aging management review of these barrier configurations is comprised of an aging effect evaluation of the constituent material of construction for each of the barriers that are in-scope to the License Renewal rule.

The following fluid systems interface with the Fire Protection system:

Auxiliary Feedwater	Service Water
Plant Air	Emergency Power

System Function Listing

In addition to the System Functions listed above, the Fire Protection System also supports additional functions (associated design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code K	Cri 1	Cri 2	Cri 3				
PROVIDE PRIMARY CONTAINMENT BOUNDARY			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Fire Protection system perform this associated design system function.

Code Q	Cri 1	Cri 2	Cri 3				
PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Fire Protection system perform this associated design system function.

Code S	Cri 1	Cri 2	Cri 3				
SPECIAL CAPABILITY CLASS FUNCTIONS			FP	EQ	PTS	AT	SB

Comment: Components within the Fire Protection system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Fire Protection system that perform special capability class functions are tracked under the Criterion 2 code (Y) and Criterion 3 codes (Z1 through Z5).

Code T	Cri 1	Cri 2	Cri 3				
NON-NUCLEAR SAFETY CLASS FUNCTIONS			FP	EQ	PTS	AT	SB

Comment: Components within the Fire Protection system perform this associated design system function (e.g. fill and pressurization components for the fire water storage tank, system drains, etc.).

Code X	Cri 1	Cri 2	Cri 3				
SFR FUNCTION NOT APPLICABLE AT COMPONENT LEVEL			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Fire Protection system perform specific safety related functions different from and in addition to the system level functions (e.g. safety related position indication for containment isolation valve 9227).

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Fire Protection system perform this primary design system function.

Code Z5 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR STATION BLACKOUT (10 CFR 50.63)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: Components within the Fire Protection system perform this associated design system function.

UFSAR Reference

Additional Fire Protection System details are provided in Section 9.5.1, Section 9.5.1.2, Section 9.5.1.3, Section 7.4.4, and Table 6.2-15a of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Fire Protection System are listed below:

33013-1242
33013-1607
33013-1866
33013-1867
33013-1870
33013-1885,2
33013-1893
33013-1989
33013-1990,1
33013-1990,2
33013-1991
33013-1992
33013-1993,1
33013-1993,2
33013-2287
33013-2344
33013-2345
33013-2346
33013-2347
33013-2348
33013-2349
33013-2350
33013-2351
33013-2352
33013-2353
33013-2354
33013-2355
33013-2356
33013-2357
33013-2359
33013-1871
33013-1892
33013-1895
33013-1896
33013-2341
33013-2342
33013-2343

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.3-6 along with each Component Group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.3-6 Fire Protection (FP)

Component Group	Passive Function	Aging Management Reference
BELL[1]	PRESSURE BOUNDARY	Table 3.4-1 Line Number (20) Table 3.4-2 Line Number (7)
COMPRESSOR CASING (included for conservatism)	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (10)
CONTROLLER[1]	PRESSURE BOUNDARY	Table 3.4-1 Line Number (20) Table 3.4-2 Line Number (17)

Table 2.3.3-6 Fire Protection (FP)

Component Group	Passive Function	Aging Management Reference
COOLER	PRESSURE BOUNDARY	Table 3.4-1 Line Number (20) Table 3.4-2 Line Number (18) Table 3.4-2 Line Number (21) Table 3.4-2 Line Number (22) Table 3.4-2 Line Number (23)
CS COMPONENTS	PRESSURE BOUNDARY	Table 3.4-1 Line Number (13)
CUTTER ASSEMBLY	PRESSURE BOUNDARY	Table 3.4-2 Line Number (43) Table 3.4-2 Line Number (44)
ENGINE CASING	PRESSURE BOUNDARY	Table 3.4-1 Line Number (20) Table 3.4-2 Line Number (58)
FASTENERS (BOLTING)	JOINT INTEGRITY	Table 3.4-1 Line Number (13) Table 3.4-1 Line Number (23) Table 3.4-2 Line Number (79) Table 3.4-2 Line Number (80) Table 3.4-2 Line Number (81)
FILTER HOUSING	PRESSURE BOUNDARY	Table 3.4-2 Line Number (83) Table 3.4-2 Line Number (85) Table 3.4-2 Line Number (86) Table 3.4-2 Line Number (98) Table 3.4-2 Line Number (100)
FLAME ARRESTOR	FLAME SUPPRESSION	Table 3.4-2 Line Number (103) Table 3.4-2 Line Number (104)
FLOW ELEMENT	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (20)
FLOW NOZZLES	PRESSURE BOUNDARY PROVIDE FLOW	Table 3.4-1 Line Number (20) This applies to both passive functions. Table 3.4-2 Line Number (114) This applies to the pressure boundary passive function.
GAS CYLINDER	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (115)

Table 2.3.3-6 Fire Protection (FP)

Component Group	Passive Function	Aging Management Reference
HAND CONTROL STATION	PRESSURE BOUNDARY	Table 3.4-1 Line Number (20) Table 3.4-2 Line Number (116) Table 3.4-2 Line Number (117) Table 3.4-2 Line Number (118)
HEAT EXCHANGER	PRESSURE BOUNDARY HEAT TRANSFER	Table 3.4-1 Line Number (20) This applies to both passive functions. Table 3.4-2 Line Number (126) This applies to the pressure boundary passive function. Table 3.4-2 Line Number (136) This applies to the heat transfer passive function.
HEATING ELEMENT	PRESSURE BOUNDARY	Table 3.4-2 Line Number (159) Table 3.4-2 Line Number (161)
HOSE REEL	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (20)
INDICATOR[1]	PRESSURE BOUNDARY	Table 3.4-2 Line Number (172) Table 3.4-2 Line Number (173) Table 3.4-2 Line Number (174)
LEVEL GLASS	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (6) Table 3.4-2 Line Number (187) Table 3.4-2 Line Number (188) Table 3.4-2 Line Number (189) Table 3.4-2 Line Number (190)
ORIFICE	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (20) Table 3.4-2 Line Number (195) Table 3.4-2 Line Number (196) Table 3.4-2 Line Number (198)

Table 2.3.3-6 Fire Protection (FP)

Component Group	Passive Function	Aging Management Reference
PIPE	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (6) Table 3.4-1 Line Number (20) Table 3.4-2 Line Number (202) Table 3.4-2 Line Number (203) Table 3.4-2 Line Number (207) Table 3.4-2 Line Number (209) Table 3.4-2 Line Number (212) Table 3.4-2 Line Number (213) Table 3.4-2 Line Number (214) Table 3.4-2 Line Number (215) Table 3.4-2 Line Number (219) Table 3.4-2 Line Number (222) Table 3.4-2 Line Number (224) Table 3.4-2 Line Number (227)
PROTOMATIC	PRESSURE BOUNDARY	Table 3.4-1 Line Number (20) Table 3.4-2 Line Number (251)
PUMP CASING	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (6) Table 3.4-1 Line Number (20) Table 3.4-2 Line Number (262) Table 3.4-2 Line Number (263) Table 3.4-2 Line Number (269) Table 3.4-2 Line Number (271)
RELEASE ASSEMBLY	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (285) Table 3.4-2 Line Number (286) Table 3.4-2 Line Number (287) Table 3.4-2 Line Number (288) Table 3.4-2 Line Number (289)
SCREEN	PRESSURE BOUNDARY	Table 3.4-1 Line Number (20) Table 3.4-2 Line Number (290)
SPECTACLE FLANGE	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (20) Table 3.4-2 Line Number (293) Table 3.4-2 Line Number (294)

Table 2.3.3-6 Fire Protection (FP)

Component Group	Passive Function	Aging Management Reference
SPRINKLER HEAD	PRESSURE BOUNDARY PROVIDE FLOW	Table 3.4-1 Line Number (20) Table 3.4-2 Line Number (298) These apply to both passive functions.
STRAINER HOUSING	PRESSURE BOUNDARY	Table 3.4-1 Line Number (6) Table 3.4-1 Line Number (20) Table 3.4-2 Line Number (300) Table 3.4-2 Line Number (301) Table 3.4-2 Line Number (302) Table 3.4-2 Line Number (303) Table 3.4-2 Line Number (305)
STRUCTURE	FIRE BARRIER	Table 3.4-1 Line Number (19) Table 3.4-1 Line Number (25) Table 3.4-2 Line Number (308) Table 3.4-2 Line Number (311) Table 3.4-2 Line Number (312) Table 3.4-2 Line Number (313) Table 3.4-2 Line Number (314) Table 3.4-2 Line Number (315) Table 3.4-2 Line Number (316) Table 3.4-2 Line Number (317) Table 3.4-2 Line Number (318) Table 3.4-2 Line Number (319) Table 3.4-2 Line Number (320) Table 3.4-2 Line Number (321) Table 3.4-2 Line Number (322) Table 3.4-2 Line Number (323) Table 3.4-2 Line Number (324) Table 3.4-2 Line Number (325) Table 3.4-2 Line Number (326) Table 3.4-2 Line Number (327) Table 3.4-2 Line Number (328) Table 3.4-2 Line Number (329) Table 3.4-2 Line Number (330)

Table 2.3.3-6 Fire Protection (FP)

Component Group	Passive Function	Aging Management Reference
TANK	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (6) Table 3.4-1 Line Number (20) Table 3.4-1 Line Number (21) Table 3.4-2 Line Number (334) Table 3.4-2 Line Number (339) Table 3.4-2 Line Number (341) Table 3.4-2 Line Number (342) Table 3.4-2 Line Number (343) Table 3.4-2 Line Number (344)
TRANSMITTER[1]	PRESSURE BOUNDARY	Table 3.4-1 Line Number (20) Table 3.4-2 Line Number (371)
VALVE BODY	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (6) Table 3.4-1 Line Number (20) Table 3.4-2 Line Number (394) Table 3.4-2 Line Number (395) Table 3.4-2 Line Number (414) Table 3.4-2 Line Number (415) Table 3.4-2 Line Number (416) Table 3.4-2 Line Number (417) Table 3.4-2 Line Number (420) Table 3.4-2 Line Number (427) Table 3.4-2 Line Number (428) Table 3.4-2 Line Number (429) Table 3.4-2 Line Number (431) Table 3.4-2 Line Number (433) Table 3.4-2 Line Number (437) Table 3.4-2 Line Number (444) Table 3.4-2 Line Number (446) Table 3.4-2 Line Number (447) Table 3.4-2 Line Number (450) Table 3.4-2 Line Number (458)

1. Selected instruments were conservatively included within the scope of License Renewal. Consideration was given to the consequences of an instrument housing pressure boundary failure. Where an instrument was unisolable from a pressure source and is of sufficient size that a system function would be degraded should the pressure boundary fail, that instrument is included for License Renewal review.

2.3.3.7 Heating Steam

System Description

The Heating Steam system supports habitability and equipment reliability by maintaining plant area temperatures within acceptable bounds. In addition to supporting ventilation functions, heating system also provides process steam for the waste disposal system evaporator. The system does not perform any nuclear safety function. (Note that at one time the heating steam penetrated containment. Those blanked off sections of abandoned pipe are evaluated within the Containment Isolation Components system.) The Heating Steam system contains pressurized, high temperature fluid and has pipe routing and equipment locations in close proximity to safety related equipment. Accordingly, some localized pipe segment and equipment are considered as non-safety components whose failure could prevent the satisfactory accomplishment of a safety function.

The Heating Steam system is categorized a moderate energy system. Consequently the effects of heating steam pipe breaks have been evaluated. Evaluations were subsequently performed to ensure the plant could achieve and maintain safe shutdown following postulated system failures. As a results of the evaluation, pipe whip and jet impingement protection was provided for the 6-in. heating steam line riser located on the intermediate floor of the auxiliary building to protect safety-related electrical equipment in the vicinity of the riser. Additionally, heating steam lines were removed from the relay room and air handling room in order to maintain a mild environment for the purpose of environmental qualification of electrical equipment in the rooms. The mitigative equipment is evaluated in the appropriate civil/structural assessment. As a result of these analysis and modifications, the only portion of the Heating Steam system considered as non-safety components whose failure could prevent the accomplishment of a safety function are those portions of the system contained in the Diesel Generator rooms.

The principal components of Heating Steam are the boiler, tanks, pumps, condensate collection tanks, unit heaters and essential piping and valves. The heating steam is provided from the house boiler, located in the screen house or from a connection in the main steam system. The systems provided with house steam include: unit heaters in the screen house, intermediate building, auxiliary building, turbine building, diesel generator rooms, auxiliary building air handling units, containment purge supply unit, boric acid batch tank, gas stripper, and the boron recycle evaporator.

The following fluid systems interface with the Heating Steam System:
Main and Auxiliary Steam

System Function Listing

In addition to the System Functions listed above, the Heating Steam System also supports additional functions (associated design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Heating Steam system perform this associated system function (e.g. habitability heating).

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Heating Steam system perform this associated design system function. Components within the Heating Steam system pass through the Emergency Diesel Generator rooms and therefore a failure of these components could affect the ability of safety related components to perform their intended functions.

UFSAR Reference

Additional Heating Steam System details are provided in Section 9.4.10 and Section 3.6 of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Heating Steam System are listed below:

33013-1250,1	33013-1896
33013-1266	33013-1913
33013-1893	33013-1914
33013-1908,3	33013-1916,1
33013-1915	33013-1916,2
33013-1874	33013-1917
33013-1890	33013-2274

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.3-7 along with each Component Group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.3-7 Heating Steam System

Component Group	Passive Function	Aging Management Reference
HEATER	PRESSURE BOUNDARY	Table 3.4-2 Line Number (155) Table 3.4-2 Line Number (156)
PIPE	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (206)
STRAINER HOUSING	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (299)
TRAP HOUSING	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (383)
VALVE BODY	PRESSURE BOUNDARY	Table 3.4-2 Line Number (430) Table 3.4-2 Line Number (440)

2.3.3.8 Emergency Power

System Description

The diesel generating Emergency Power system provides electrical power for safety related components when the preferred power supply is not available. The Emergency Power sources become available automatically following the loss of the preferred power supply within a time consistent with the requirements of the engineered safety features and the shutdown systems under both normal and accident conditions. Components within the Emergency Power system are also credited for use in safe shutdown following some fires. Emergency Power system reliability is a critical element in ensuring that the station demonstrates compliance with regulations for station blackout.

Included in the Emergency Power system are two safety related station Emergency Diesel Generators (EDGs) and the Technical Support Center (TSC) diesel generator. Each EGD is capable of automatically starting and sequentially accepting the power requirements of one complete set of safeguards equipment. Each EDG provides the necessary power to cool the core and maintain the containment pressure within the design value for a loss of coolant accident (coincident with a loss of offsite power). The diesels start automatically when loss of voltage is sensed on the bus they supply. The EDGs also start automatically upon receipt of a safety injection signal. The EDGs are normally operated from the control room but EDG A is equipped with a control station that allows the unit to be electrically divorced from the control room and operated locally. The TSC diesel generator can be used to supply a battery charger in order to support vital DC for long term recovery following some fire scenarios.

The principal components of the EDGs include two diesel engines. Each engine is equipped with its own; turbo charger, air start sub-system, lube oil and cooling water sub-systems, fuel oil sub-system, ventilation system and essential piping and valves. (Ventilation requirements are evaluated separately within the ventilation systems.) The TSC diesel requires its own similar sub-systems to function but uses batteries rather than air as a starting mode of force.

The following fluid systems interface with the Emergency Power System:

Service Water

System Function Listing

In addition to the System Functions described above, the Emergency Power System also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code J	Cri 1	Cri 2	Cri 3				
PROVIDE HEAT REMOVAL FROM SAFETY RELATED HEAT EXCHANGERS			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Emergency Power system perform this associated system function.

Code Q	Cri 1	Cri 2	Cri 3				
PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Emergency Power system perform this primary system function.

Code S	Cri 1	Cri 2	Cri 3				
SPECIAL CAPABILITY CLASS FUNCTIONS			FP	EQ	PTS	AT	SB

Comment: Components within the Emergency Power system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Emergency Power system that perform special capability class functions are tracked under the Criterion 3 codes (Z1 through Z5).

Code T	Cri 1	Cri 2	Cri 3				
NON-NUCLEAR SAFETY CLASS FUNCTIONS			FP	EQ	PTS	AT	SB

Comment: Components within the Emergency Power system perform this associated system function.

Code Z1	Cri 1	Cri 2	Cri 3				
LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Emergency Power system perform this associated system function.

UFSAR Reference

Additional Emergency Power System details are provided in Section 9.5.4, Section 9.5.5, Section 9.5.6, Section 9.5.7, Section 9.5.8, and Section 8.3.1.1.6 of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Emergency Power System are listed below:

33013-1239,1	33013-1250,1
33013-1239,2	33013-2288

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.3-8 along with each Component Group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.3-8 Emergency Power

Component Group	Passive Function	Aging Management Reference
ACCUMULATOR	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (7)
COOLER	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (18) Table 3.4-2 Line Number (19) Table 3.4-2 Line Number (20) Table 3.4-2 Line Number (24) Table 3.4-2 Line Number (25)
ENGINE CASING	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (57)
EXPANSION JOINT	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (61) Table 3.4-2 Line Number (62) Table 3.4-2 Line Number (63) Table 3.4-2 Line Number (68) Table 3.4-2 Line Number (69) Table 3.4-2 Line Number (70)

Table 2.3.3-8 Emergency Power

Component Group	Passive Function	Aging Management Reference
FAN CASING	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (76)
FASTENERS (BOLTING)	JOINT INTEGRITY	Table 3.4-1 Line Number (23) Table 3.4-2 Line Number (79) Table 3.4-2 Line Number (80)
FILTER HOUSING	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (82) Table 3.4-2 Line Number (83) Table 3.4-2 Line Number (84) Table 3.4-2 Line Number (85) Table 3.4-2 Line Number (88) Table 3.4-2 Line Number (89) Table 3.4-2 Line Number (90) Table 3.4-2 Line Number (91) Table 3.4-2 Line Number (97) Table 3.4-2 Line Number (99)
GOVERNOR	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5)
HEAT EXCHANGER	PRESSURE BOUNDARY HEAT TRANSFER	Table 3.4-2 Line Number (121) Table 3.4-2 Line Number (122) Table 3.4-2 Line Number (123) Table 3.4-2 Line Number (128) Table 3.4-2 Line Number (130) Table 3.4-2 Line Number (133) These apply to the pressure boundary passive function. Table 3.4-2 Line Number (127) Table 3.4-2 Line Number (129) Table 3.4-2 Line Number (131) These apply to the heat transfer passive function.
HEATING ELEMENT	PRESSURE BOUNDARY	Table 3.4-2 Line Number (159) Table 3.4-2 Line Number (160) Table 3.4-2 Line Number (161)
INDICATOR[1]	PRESSURE BOUNDARY	Table 3.4-2 Line Number (167) Table 3.4-2 Line Number (168) Table 3.4-2 Line Number (170) Table 3.4-2 Line Number (171)

Table 2.3.3-8 Emergency Power

Component Group	Passive Function	Aging Management Reference
LEVEL GLASS	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (7) Table 3.4-2 Line Number (180) Table 3.4-2 Line Number (181) Table 3.4-2 Line Number (182) Table 3.4-2 Line Number (183) Table 3.4-2 Line Number (184) Table 3.4-2 Line Number (185) Table 3.4-2 Line Number (186)
LUBRICATOR	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5)
MUFFLER	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (193) Table 3.4-2 Line Number (194)
ORIFICE	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (14)
PIPE	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (7) Table 3.4-1 Line Number (14) Table 3.4-2 Line Number (212) Table 3.4-2 Line Number (217) Table 3.4-2 Line Number (218) Table 3.4-2 Line Number (219) Table 3.4-2 Line Number (223) Table 3.4-2 Line Number (225) Table 3.4-2 Line Number (228) Table 3.4-2 Line Number (233)
PUMP CASING	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (7) Table 3.4-1 Line Number (14) Table 3.4-2 Line Number (260) Table 3.4-2 Line Number (262) Table 3.4-2 Line Number (264) Table 3.4-2 Line Number (266) Table 3.4-2 Line Number (267)
STRAINER HOUSING	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (7)

Table 2.3.3-8 Emergency Power

Component Group	Passive Function	Aging Management Reference
TANK	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (7) Table 3.4-2 Line Number (335) Table 3.4-2 Line Number (337) Table 3.4-2 Line Number (340) Table 3.4-2 Line Number (341)
TEMPERATURE ELEMENT[1]	PRESSURE BOUNDARY	Table 3.4-2 Line Number (354) Table 3.4-2 Line Number (355) Table 3.4-2 Line Number (363) Table 3.4-2 Line Number (364)
VALVE BODY	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-1 Line Number (7) Table 3.4-2 Line Number (389) Table 3.4-2 Line Number (417) Table 3.4-2 Line Number (418) Table 3.4-2 Line Number (419) Table 3.4-2 Line Number (424) Table 3.4-2 Line Number (427) Table 3.4-2 Line Number (429) Table 3.4-2 Line Number (431) Table 3.4-2 Line Number (432) Table 3.4-2 Line Number (438) Table 3.4-2 Line Number (439) Table 3.4-2 Line Number (445) Table 3.4-2 Line Number (447) Table 3.4-2 Line Number (448) Table 3.4-2 Line Number (450) Table 3.4-2 Line Number (459) Table 3.4-2 Line Number (460)

1. Selected instruments were conservatively included within the scope of License Renewal. Consideration was given to the consequences of an instrument housing pressure boundary failure. Where an instrument was unisolable from a pressure source and is of sufficient size that a system function would be degraded should the pressure boundary fail, that instrument is included for License Renewal review.

2.3.3.9 Containment Ventilation

System Description

The Containment Ventilation Systems function to provide emergency heat removal from the containment atmosphere, to remove radioactive material from the containment atmosphere, and to provide containment pressure control. Portions of the system function to maintain specific containment concrete temperatures below the threshold where long term aging effects are manifested. Thus the containment ventilation system is considered to contain non-safety related equipment whose failure could prevent the satisfactory accomplishment of a safety function (e.g. penetration cooling). The Containment Ventilation system also contains components used for fire detection and components that are Environmentally Qualified.

Included within the scope of the Containment Ventilation System are the following subsystems:

a. Containment recirculation cooling and filtration system.

b. Control rod drive mechanism cooling system.

c. Reactor compartment cooling system.

d. Refueling water surface and purge system.

e. Containment auxiliary charcoal filter system.

f. Containment post-accident charcoal filter system.

g. Containment shutdown purge system.

h. Containment mini-purge system.

i. Penetration cooling system.

The principal components of the Containment Ventilation System include filters, fans, dampers, valves, heat exchangers and the essential ductwork and piping. Fire dampers contained in the system are evaluated as a separate commodity group.

The containment recirculation fans, control rod drive mechanism fans, and reactor compartment fans are direct-driven units, each with standby units for redundancy. The fans and motors of these units are provided with vibration detecting devices to detect abnormal operating conditions in the early stages of the disturbance. Each of the associated systems is provided with flow switches to verify existence of air flow in the associated duct system. Dampers in the following systems and ducts are provided with air by dual supply air mains: primary compartment ducts, dome ducts, containment auxiliary charcoal filter systems, butterfly valves which isolate the post-accident charcoal filters, and containment purge supply and exhaust ducts. Two of the four fans and coolers plus one containment spray pump (i.e. one train of each system) are required to provide sufficient capacity to maintain the containment pressure within design limits after a loss-of-coolant accident or steam line break accident. The containment recirculation fan cooler electrical connections and other equipment in the containment necessary for operation of the system are capable of operating under the environmental conditions following a loss-of-coolant accident.

The control rod drive cooling system consists of fans and ductwork to draw air through the control rod drive mechanism shroud and eject it to the main containment volume. The reactor compartment cooling system consists of a plenum, cooling coils, fans, and ductwork arranged to supply cool air to the annulus between the reactor vessel and the primary shield and to the nuclear instrumentation external to the reactor. The refueling water surface and purge system supplies air to the surface of the refueling cavity and exhaust from the area above the refueling manipulator crane to protect the operators during refueling operations. The containment auxiliary charcoal filter system’s purpose is to absorb radioactive iodine vapor and radioactive particles that may occur as a result of normal primary system leakage inside the containment. The containment shutdown purge system is independent of the main auxiliary building exhaust system and includes provisions for both supply and exhaust air. The supply system includes an outside air connection to roughing filters, heating coils, fans, duct system, and supply penetration with a butterfly valve outside containment and a blind flange inside containment. The exhaust system includes an exhaust penetration with a butterfly valve and a blind flange identical to those above, a duct system, a filter bank with high efficiency particulate air and charcoal filters, fans, and a building exhaust vent. The shutdown purge supply and exhaust duct blind flanges inside the containment are closed during MODES 1, 2, 3 and 4. The containment mini-purge system is capable of purging containment during MODES 1 and 2 at a relatively low flow rate (approximately 1500 cfm). The exhaust is through a 6-in. line to the auxiliary building charcoal filters. The penetration cooling system is used to cool hot mechanical containment penetrations. The containment penetration cooling system is designed to prevent the bulk concrete temperature surrounding the penetrations from exceeding 150°F.

The following fluid systems interface with the Containment Ventilation System:

Waste Disposal	Plant Air
Service Water	

System Function Listing

In addition to the System Functions described above, the Containment Ventilation System also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code J PROVIDE HEAT REMOVAL FROM SAFETY RELATED HEAT EXCHANGERS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Containment Ventilation system perform this associated design system function (Service Water heat removal from Containment Recirculating Fan Coolers, etc.).

Code K PROVIDE PRIMARY CONTAINMENT BOUNDARY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Containment Ventilation system perform this associated design system function.

Code L PROVIDE EMERGENCY HEAT REMOVAL FROM PRIMARY CONTAINMENT AND PROVIDE CONTAINMENT PRESSURE CONTROL	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Containment Ventilation system perform this primary design system function.

Code M PROVIDE EMERGENCY REMOVAL OF RADIOACTIVE MATERIAL FROM THE PRIMARY CONTAINMENT ATMOSPHERE	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Containment Ventilation system perform this primary design system function.

Code O MAINTAIN EMERGENCY TEMPERATURES WITHIN AREAS CONTAINING SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Containment Ventilation system perform this primary design system function.

Code Q PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Containment Ventilation system perform this associated design system function.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Containment Ventilation system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Containment Ventilation system that perform special capability class functions are tracked under the Criterion 2 code (Y) and Criterion 3 codes (Z1 through Z5). Augmented quality equipment in the ventilation system includes Recirc Fan Condensate Level, Containment Purge, RG-1.97 Category 2 Post Accident monitoring variables, etc.

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Containment Ventilation system perform this associated design system function.

Code X SFR FUNCTION NOT APPLICABLE AT COMPONENT LEVEL	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Containment Ventilation system perform specific safety related functions different from and in addition to the system level functions (e.g. supplementary breakers for fans to electrical protect penetrations).

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Containment Ventilation system perform this associated design system function. Containment penetration and reactor compartment cooling fans maintain containment concrete temperature during normal operation below the temperature at which damage occurs.

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Containment Ventilation system perform this associated design system function (temperature elements).

Code Z2 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ENVIRONMENTAL QUALIFICATION (10 CFR 50.49)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: Components within the Containment Ventilation system are designated as Environmentally Qualified (Fan cooler motors, containment temperature, etc.).

UFSAR Reference

Additional Containment Ventilation System details are provided in Section 6.2.2, Section 6.5.1.2, Section 9.4.1, and Table 6.2-15a of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Containment Ventilation System are listed below:

33013-1250,1
33013-1250,3
33013-1261
33013-1278,1
33013-1863
33013-1864
33013-1865
33013-1866
33013-1870
33013-1884,1
33013-1884,2
33013-1893
33013-1915
33013-1887
33013-1888
33013-1916,1

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.3-9 along with each Component Group’s passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.3-9 Containment Ventilation

Component Group	Passive Function	Aging Management Reference
AIR OPERATED DAMPER HOUSING	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (1) Table 3.4-2 Line Number (2) Table 3.4-2 Line Number (3) Table 3.4-2 Line Number (4) Table 3.4-2 Line Number (6)
COOLING COIL	PRESSURE BOUNDARY	Table 3.4-2 Line Number (34) Table 3.4-2 Line Number (35) Table 3.4-2 Line Number (36) Table 3.4-2 Line Number (37) Table 3.4-2 Line Number (40) Table 3.4-2 Line Number (41)
CS COMPONENTS	PRESSURE BOUNDARY	Table 3.4-1 Line Number (13)
DAMPER HOUSING/FRAME	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (47) Table 3.4-2 Line Number (48) Table 3.4-2 Line Number (50)
EXPANSION JOINT	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (59) Table 3.4-2 Line Number (60) Table 3.4-2 Line Number (66) Table 3.4-2 Line Number (67)
FAN CASING	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5)
FASTENERS (BOLTING)	JOINT INTEGRITY	Table 3.4-1 Line Number (13) Table 3.4-1 Line Number (23) Table 3.4-2 Line Number (79) Table 3.4-2 Line Number (80) Table 3.4-2 Line Number (81)
FILTER HOUSING	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (92) Table 3.4-2 Line Number (93) Table 3.4-2 Line Number (97) Table 3.4-2 Line Number (98)
FLANGE	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (105)

Table 2.3.3-9 Containment Ventilation

Component Group	Passive Function	Aging Management Reference
HEAT EXCHANGER	PRESSURE BOUNDARY HEAT TRANSFER	Table 3.4-1 Line Number (5) This applies to both passive functions. Table 3.4-2 Line Number (124) Table 3.4-2 Line Number (125) Table 3.4-2 Line Number (143) Table 3.4-2 Line Number (146) These apply to the pressure boundary passive function. Table 3.4-2 Line Number (134) Table 3.4-2 Line Number (135) Table 3.4-2 Line Number (139) These apply to the heat transfer passive function.
HVAC EQUIPMENT PACKAGE[1]	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (162) Table 3.4-2 Line Number (163) Table 3.4-2 Line Number (164)
PIPE	PRESSURE BOUNDARY	Table 3.4-1 Line Number (2) Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (232) Table 3.4-2 Line Number (235) Table 3.4-2 Line Number (236)
VALVE BODY	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (386) Table 3.4-2 Line Number (387) Table 3.4-2 Line Number (392) Table 3.4-2 Line Number (394) Table 3.4-2 Line Number (413) Table 3.4-2 Line Number (417) Table 3.4-2 Line Number (420) Table 3.4-2 Line Number (425) Table 3.4-2 Line Number (426) Table 3.4-2 Line Number (428) Table 3.4-2 Line Number (429) Table 3.4-2 Line Number (444) Table 3.4-2 Line Number (445) Table 3.4-2 Line Number (446) Table 3.4-2 Line Number (447)

Table 2.3.3-9 Containment Ventilation

Component Group	Passive Function	Aging Management Reference
VENTILATION DUCTWORK	PRESSURE BOUNDARY	Table 3.4-2 Line Number (468) Table 3.4-2 Line Number (469)

1. HVAC equipment packages include the pressure boundary attributes associated with the package and sub-components such as filter housings, internal damper housings, and fan housings. Both the HVAC package units and their associated sub-components are uniquely identified on plant drawings.

2.3.3.10 **Essential Ventilation Systems**

System Description

The Essential Ventilation Systems functions to maintain temperatures within specified limits in areas containing safety related equipment. Additionally, the control room emergency air-treatment portion of the system is designed to filter the control room atmosphere during periods when the control room is isolated and to maintain radiation levels in the control room at acceptable levels following design-basis accidents. (Note: Radiation detection and toxic gas monitoring are evaluated within the radiation monitoring system.) Ventilation is also required for emergency diesel generator operation, for the technical support center diesel generator and its associated equipment, and the standby auxiliary feedwater pumps, all of which maybe used for safe shutdown following some fire events.

Included within the scope of the Essential Ventilation Systems are the following subsystems:

a. Auxiliary Building Ventilation

b. Intermediate Building Ventilation

c. Standby Auxiliary Feedwater Building Ventilation

d. Diesel Generator Building Ventilation

e. Control Building Ventilation

f. Technical Support Center Ventilation

The principal components of the Essential Ventilation Systems include filters, fans, dampers, valves, heat exchangers, conditioning/chiller packages and the essential ductwork and piping. Fire dampers contained in the system are evaluated as a separate commodity group.

The auxiliary building has a non-safety heating, ventilation, and air conditioning system which provides clean, filtered, and tempered air to the operating floor of the auxiliary building and to the surface of the decontamination pit and spent fuel storage pool. The system exhausts air from the equipment rooms and open areas of the auxiliary building, and from the decontamination pit and spent fuel pool (SFP) through a closed exhaust system. The exhaust system includes a 100%-capacity bank of high efficiency particulate air filters and redundant 100%-capacity fans discharging to the atmosphere via the plant vent. This arrangement ensures the proper direction of air flow for removal of airborne radioactivity from the auxiliary building. Included in the auxiliary building exhaust system is a separate charcoal filter circuit, which exhausts from rooms where fission-product activity may accumulate during MODES 1 and 2 in concentrations exceeding the average levels expected in the rest of the building. Although no credit for this system is assumed in the plant safety analysis, this circuit is capable of providing exhaust ventilation from the areas containing pumps and related piping and valving which are used to recirculate containment sump liquid following a LOCA. A full-flow charcoal filter bank is provided in the circuit, along with two 50%-capacity exhaust fans. The air-operated suction and discharge dampers associated with each fan are interlocked with the fan such that they are fully open when the fan is operating and fully closed when the fan is stopped. These dampers fail to the open position on loss of control signal or control air. The fans discharge to the main auxiliary building exhaust system containing the high efficiency particulate air (HEPA) filter bank. To ensure a path for the charcoal (and HEPA) filtered exhaust to the plant vent if the main exhaust fans are not operating, a fail-open damper is installed in a bypass circuit around the two main exhaust fans. In addition to the main auxiliary building ventilation system (ABVS), the residual heat removal, safety injection, containment spray, and charging pump motors are provided with additional cooling provisions when the pumps are operating. The safety injection and containment spray pump motors are located in an open area in the basement of the auxiliary building and share three service-water-cooled heat exchangers. In 1992, service water to these heat exchangers was blanked off. The charging pumps and residual heat removal pumps are located in individual rooms, each room being provided with two cooling units consisting of redundant fans, water-cooled heat exchangers, and ductwork for circulating the cooled air. The capacity of each charging pump cooling unit is sufficient to maintain acceptable room-ambient temperatures with the minimum number of pumps required for system operation in service. The cooling units in the residual heat removal pump pit are not required for the operation of the residual heat removal pumps, even if both pumps are operating. In the event of a loss of offsite power, the ABVS main supply and exhaust fans would be inoperable. However, all other fans in the ABVS are supplied by emergency diesel power, including the pump cooling circuits for safety-related pump motors, as described above. Analysis has shown that the three levels of the auxiliary building and the residual heat removal pump pit would remain within acceptable limits when the outside air was at its maximum expected temperature and there were no cooling units operating. Since the auxiliary building is a very large volume building, it is not expected that there would be a significant postaccident temperature increase except in some local areas near hot piping and large motors. The spent fuel pool (SFP) area ventilation system is a part of the ABVS. The system serves to control airborne radioactivity in the SFP area during normal operating conditions. This is accomplished by directing air from the auxiliary building supply air unit across both the SFP and the decontamination pit to exhaust air ducts which are connected to the suction of the auxiliary building exhaust fan C. Exhaust air from the SFP water surface is drawn through roughing filters and, depending on system alignment, charcoal filters. Discharge from the auxiliary building exhaust fan C passes through HEPA filters, a main auxiliary building exhaust fan, and then out the plant vent.

The non-safety intermediate building ventilation system includes a supply fan that exhausts air from the intermediate building cleanside to the intermediate building restricted area side. Two additional exhaust fans, which are located in the intermediate building restricted area side, draw ventilation air from various areas of both the clean and restricted area sides of the intermediate building and discharge to the auxiliary building discharge header plant vent duct. Ventilation air is provided to the intermediate building cleanside through louvered outside air intakes, which are located in the east wall of the intermediate building. Additional ventilation air capability is available to be drawn into the intermediate building cleanside from the turbine building through a louvered wall opening, which is installed in front of a rolling fire door installed in the fire barrier wall.

The standby auxiliary feedwater pump (SAFW) room cooling and heating system provides cooling and heating as required to maintain the pump room temperature within the design temperature range. This cooling and heating system is needed to provide an acceptable environment for the equipment in the pump room, which includes the two SAFW and their electric drive motors. The SAFW room cooling system is capable of operation whenever the SAFW are needed for operation. This is a result of the fact that the cooling system provides the air-cooling required for continuous operation of the pump motors.

A given cooling unit is automatically started whenever its corresponding SAFW is started. Due to its safety-related nature, the cooling system must remain functional during all modes of plant operation including the period during and after a safe shutdown earthquake.

The diesel generators are housed in adjacent but separate rooms, each room serviced by a safety-related ventilation system. Each room has two inlet fans supplying outside air. Each fan takes suction from a common header and discharges through separate ductwork, dampers, and discharge diffusers. One fan in each room discharges a supply of air directly on the instrument and control cabinets. Excess air is discharged to the outdoors through automatic, pressure-actuated room vents, backdraft dampers, and wall-mounted louvers. No refrigeration or service water (SW) air-cooling is used.

The control room ventilation system is normally operated using a large percentage of recirculated air. The fresh air intake can be closed to control the intake of airborne activity if monitors indicate that such action is appropriate. The control room emergency air-treatment system is designed to filter the control room atmosphere during periods when the control room is isolated and to maintain radiation levels in the control room at acceptable levels following the design-basis accident. This system circulates air from the control room, control room office and kitchen through return air ductwork to a central air conditioning unit located in the air handling room. The air is drawn into the unit through roughing type filters and either heated or cooled as required by electric heating or chilled water coils. Conditioned air is directed back to the rooms through a supply air ductwork system. The entire control room emergency zone air volume is turned over approximately 12 times every hour. During normal operation, fresh makeup air is admitted to this system through an intake louver located in the outside wall of the turbine building, the amount varying between 0% to 25% of the unit flow rate, depending on outside air temperature. Pneumatically operated dampers can be positioned from the control room to isolate the fresh air intake and to place a separate charcoal filter unit in service. The charcoal filter unit includes both high efficiency particulate air (HEPA) filters and 2-in. deep charcoal adsorbers for removing radioactive particulates and gaseous iodine from the control room atmosphere. Its capacity is approximately 25% of the system flow rate and the unit is installed in a normally isolated bypass circuit. In the event of high radiation levels in the control room, the control room radiation instrumentation will automatically close the redundant dampers in the fresh air intake duct and the dampers in the return air duct to the turbine building, and will open the damper in the charcoal filter unit inlet duct to allow 2000 cfm of the recirculation air to flow through the HEPA filters and charcoal adsorbers. This signal will also start a separate fan to provide flow through the charcoal filter unit. Until radioactivity in the control room atmosphere is reduced to a safe level, system flow will be in a closed cycle from the control room, with approximately 25% bypass flow through the charcoal filter unit, through the air conditioning unit, and back to the control room. The dampers can also be positioned to permit fresh air makeup to the system through the charcoal filter unit. Since all control room penetrations, including doors, are designed to high leak-tightness standards and the control room is maintained at essentially atmospheric pressure, the infiltration of contaminated air into the control room is limited to a very low rate. The control building ventilation system includes within its boundary battery and relay room ventilation. Supplemental heating and cooling to the battery rooms is provided by a non-seismic air conditioning unit, with associated service water piping, ventilation ductwork, electric heating coil, and fire dampers. The electric heating coil is seismically mounted in the heating, ventilation, and air conditioning unit discharge duct. The unit and associated ductwork and piping are designed to function during all plant modes. Although the overall design is nonseismic, the piping and ductwork are designed to maintain structural integrity during a design-basis earthquake. Each battery room has an ac-powered propeller exhaust fan that takes suction from the area to remove hydrogen gas generated by the batteries. Also, there is a separate emergency dc-powered ventilation system that is manually actuated in the event of low air flow in the ductwork of either of these battery room exhaust fans. The relay room contains two self-contained, water-cooled air-cooling units that maintain a normal room temperature.

The technical support center heating, ventilation, and air conditioning system maintains year-round occupancy comfort levels, the heating, ventilation, and air conditioning system provides personnel protection from airborne radiological contaminants, maintains a positive pressure relative to the outside, and provides cooling, heating, and ventilation required by special areas (e.g. computer room).

The following fluid systems interface with the Essential Ventilation Systems:

Waste Disposal	Plant Air
Service Water	Heating Steam
Chilled Water	

System Function Listing

In addition to the System Functions described above, the Essential Ventilation Systems also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code J PROVIDE HEAT REMOVAL FROM SAFETY RELATED HEAT EXCHANGERS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Essential Ventilation Systems perform this associated design system function. Service water interface with Standby Auxiliary Feed Building ventilation.

Code O MAINTAIN EMERGENCY TEMPERATURES WITHIN AREAS CONTAINING SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Essential Ventilation Systems perform this primary design system function. This function includes establishing the Control Room emergency air treatment system boundary.

Code Q PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Essential Ventilation Systems perform this associated design system function.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Essential Ventilation Systems perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Essential Ventilation Systems that perform special capability class functions are tracked under the Criterion 3 codes (Z1 through Z5). Augmented quality equipment in the ventilation systems include Relay Room, Battery Room, and Technical Support Center ventilation. Spent Fuel Pool ventilation and filtration is required during fuel handling operations in the Auxiliary Building.

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Essential Ventilation Systems perform this associated design system function.

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Essential Ventilation Systems perform this associated design system function. Non-safety ductwork that supports control room ventilation.

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Essential Ventilation Systems perform this associated design system function. Selected ventilation components support safe shutdown following fire.

UFSAR Reference

Additional Essential Ventilation Systems details are provided in Section 9.4.2, Section 9.4.3, Section 9.4.4, Section 9.4.8, Section 9.4.9, Section 9.5.1.2.4.3, and Section 6.4.2.2 of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Essential Ventilation Systems are listed below:

33013-1250,1	33013-1872
33013-1250,2	33013-1875
33013-1256	33013-1888
33013-1867	33013-1889
33013-1869	33013-1894,1
33013-1870	33013-1896
33013-1873	33013-1897,1
33013-2350	33013-1914
33013-2352	33013-1916,1
33013-1251,1	33013-1917
33013-1868	33013-1920
33013-1871	

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.3-10 along with each Component Group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.3-10 Essential Ventilation Systems

Component Group	Passive Function	Aging Management Reference
AIR OPERATED DAMPER HOUSING	PRESSURE BOUNDARY	Table 3.4-2 Line Number (3) Table 3.4-2 Line Number (5) Table 3.4-2 Line Number (6)
BLOWER CASING	PRESSURE BOUNDARY	Table 3.4-2 Line Number (8) Table 3.4-2 Line Number (9)
COOLING COIL	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (33) Table 3.4-2 Line Number (38) Table 3.4-2 Line Number (39) Table 3.4-2 Line Number (40) Table 3.4-2 Line Number (42)
CS COMPONENTS	PRESSURE BOUNDARY	Table 3.4-1 Line Number (13)

Table 2.3.3-10 Essential Ventilation Systems

Component Group	Passive Function	Aging Management Reference
DAMPER HOUSING/FRAME	PRESSURE BOUNDARY	Table 3.4-2 Line Number (45) Table 3.4-2 Line Number (46) Table 3.4-2 Line Number (47) Table 3.4-2 Line Number (49) Table 3.4-2 Line Number (51)
EXPANSION JOINT	PRESSURE BOUNDARY	Table 3.4-1 Line Number (2) Table 3.4-2 Line Number (64)
FAN CASING	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (71) Table 3.4-2 Line Number (72) Table 3.4-2 Line Number (73) Table 3.4-2 Line Number (74) Table 3.4-2 Line Number (75) Table 3.4-2 Line Number (77) Table 3.4-2 Line Number (78)
FASTENERS (BOLTING)	JOINT INTEGRITY	Table 3.4-1 Line Number (13) Table 3.4-1 Line Number (23) Table 3.4-2 Line Number (79) Table 3.4-2 Line Number (80) Table 3.3-2 Line Number (81)
FILTER HOUSING	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (87) Table 3.4-2 Line Number (94) Table 3.4-2 Line Number (95) Table 3.4-2 Line Number (96)
HEAT EXCHANGER	PRESSURE BOUNDARY HEAT TRANSFER	Table 3.4-1 Line Number (5) This applies to both passive functions. Table 3.4-2 Line Number (125) This applies to the pressure boundary passive function. Table 3.4-2 Line Number (135) This applies to the heat transfer passive function.
HEATING COIL	PRESSURE BOUNDARY	Table 3.4-2 Line Number (157) Table 3.4-2 Line Number (158)

Table 2.3.3-10 Essential Ventilation Systems

Component Group	Passive Function	Aging Management Reference
HVAC EQUIPMENT PACKAGE[1]	PRESSURE BOUNDARY	Table 3.4-2 Line Number (163) Table 3.4-2 Line Number (165) Table 3.4-2 Line Number (166)
MOTOR OPERATED DAMPER	PRESSURE BOUNDARY	Table 3.4-2 Line Number (191) Table 3.4-2 Line Number (192)
VENTILATION DUCTWORK	PRESSURE BOUNDARY	Table 3.4-2 Line Number (468) Table 3.4-2 Line Number (470) Table 3.4-2 Line Number (471) Table 3.4-2 Line Number (472)

1. HVAC equipment packages include the pressure boundary attributes associated with the package and sub-components such as filter housings, internal damper housings, and fan housings. Both the HVAC package units and their associated sub-components are uniquely identified on plant drawings.

2.3.3.11 Cranes, Hoists, and Lifting Devices

System Description

The Cranes, Hoists and Lifting Devices equipment group consists of the load handling systems that comply with NUREG-0612, Control of Heavy Loads. As such, the equipment group is considered to contain non-safety components whose failure could affect a safety function.

The principal components of the Cranes, Hoists and Lifting Devices equipment group include the Reactor Head Lifting Device, the Reactor Internals Lifting Device, and the load carrying elements of the Containment Main Crane, the Auxiliary Building Main Crane, and the Spent Fuel Pool and Containment Refueling Bridge cranes as well as selected jib and monorail hoists. Included are cables, hooks and the moving load bearing elements. The crane rails and supports that interface with building structural members are evaluated within the building that contains them.

These overhead load-handling systems were identified to have the potential for a heavy load drop that could result in damage to safe shutdown equipment. The majority of the plant crane hoists and lifting devices are excluded from NUREG-0612 because of administrative controls over their use or because of their distance from safety related equipment.

System Function Listing

In addition to the System Functions described above, the Cranes, Hoists and Lifting devices equipment group also contains components which support additional functions (associated design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the group, or specific components contained in the group, is provided in the summary below.

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components in the Cranes, Hoists, and Lifting devices equipment group perform this associated design system function.Components in the Cranes, Hoists, and Lifting devices equipment group perform this associated design system function.

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components in the Cranes, Hoists, and Lifting devices equipment group perform this associated design system function. By definition of NUREG-0612, the safe handling of heavy loads is a non safety affecting safety function or the load handing systems are required to meet single failure proof criteria.

UFSAR Reference

Additional Cranes, Hoists, and Lifting Devices System details are provided in Section 9.1.2.1.7, Section 9.1.4.3.1, Section 9.1.4.3.6, Section 9.1.4.3.7, and Section 9.1.5 of the UFSAR.

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.3-11 along with each Component Group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.3-11 Cranes, Hoists, and Lifting Devices

Component Group	Passive Function	Aging Management Reference
CRANE	STRUCTURAL SUPPORT	Table 3.4-1 Line Number (15)
FASTENERS (BOLTING)	JOINT INTEGRITY	Table 3.4-1 Line Number (23) Table 3.4-2 Line Number (79) Table 3.4-2 Line Number (80)

2.3.3.12 **Treated Water**

System Description

The Treated Water system is comprised of the following secondary plant sub-systems: Demineralized water production, Domestic (potable) Water, Secondary Water Chemical Treatment and non-radioactive liquid waste disposal (floor drains, secondary sample effluents, etc.). The Treated Water sub-systems are non-safety auxiliary systems that support the functionality of other process systems. The Treated Water system contains floor drains and equipment whose failure could prevent the satisfactory accomplishment of a safety function (flood mitigation and backflow of oil through floor drain prevention).

The principal components of the Treated Water system are pumps, tanks, ion exchange vessels and the essential piping, hoses and valves necessary for the sub-systems to function. The primary water treatment system or mobile demineralizer trucks process Domestic Water to provide demineralized water to the reactor makeup water tank, the component cooling water surge tank, and the condensate storage tanks and various local locations throughout the plant via a piping distribution network. The All-Volatile-Treatment (AVT) chemistry system uses chemical addition and ion exchange to treat condensate water in order to reduce the corrosion of equipment in the secondary system and minimizes the fouling of heat transfer surfaces. The AVT regeneration wastes are collected in neutralization tanks and sampled to determine disposition methods. The catalytic oxygen removal system reduces condensate dissolved oxygen by mixing hydrogen with the condensate and reducing the free oxygen to water by exposure of the mixture to a metal catalyst surface. The secondary plant equipment and floor drains serve to route leakage from equipment and compartments in order to provide proper control of leakage, prevent uncontrolled communication between areas as necessary, and to allow monitoring of leakage prior to disposition. Where drains from safety-related areas are tied into drains from areas that contain a large quantity of flammable liquid, backflow protection is provided to prevent possible spread of a liquid fire via the drain system. An underground retention tank is the collection point for the various building floor and equipment drains. It retains these effluents for sampling and treatment prior to discharging into the circulating water discharge.

The following fluid systems interface with the Treated Water System:

Feedwater and Condensate	Component Cooling Water
Circulating Water	Plant Sampling

System Function Listing

In addition to the System Functions listed above, the Treated Water System also supports additional functions (associated design system functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code K PROVIDE PRIMARY CONTAINMENT BOUNDARY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Treated Water system perform this associated design system function.

Code Q PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Treated Water system perform this associated design system function.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Treated Water system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Treated Water system that perform special capability class functions are tracked under the Criterion 2 code (Y).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Treated Water system perform this associated design system function.

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Treated Water system perform this associated design system function. Components within the Treated Water system are used for dewatering vital areas and preventing backflow of oil through floor drains.

UFSAR Reference

Additional Treated Water System details are provided in Section 9.2.3, Section 9.5.1.2.4.5, Section 10.7.7, and Table 6.2-15a of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Treated Water System are listed below:

33013-1237	33013-1908,1
33013-1245	33013-1908,2
33013-1248	33013-1909
33013-1893	33013-1910,1
33013-1908,3	33013-1910,2
33013-2287	33013-1911,1
33013-2681	33013-1911,2
33013-1235	33013-1912
33013-1896	33013-1913
33013-1897,1	33013-1914
33013-1897,2	33013-2250
33013-1898	33013-2276
33013-1899,2	33013-2286
33013-1907	33013-2742

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.3-12 along with each Component Group’s passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.3-12 Treated Water

Component Group	Passive Function	Aging Management Reference
CS COMPONENTS	PRESSURE BOUNDARY	Table 3.4-1 Line Number (13)

Table 2.3.3-12 Treated Water

Component Group	Passive Function	Aging Management Reference
FASTENERS (BOLTING)	JOINT INTEGRITY	Table 3.4-1 Line Number (13) Table 3.4-1 Line Number (23) Table 3.4-2 Line Number (79) Table 3.4-2 Line Number (80) Table 3.4-2 Line Number (81)
PENETRATION SEAL[1]	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5)
PIPE	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (204) Table 3.4-2 Line Number (205) Table 3.4-2 Line Number (212) Table 3.4-2 Line Number (216)
PUMP CASING	PRESSURE BOUNDARY	Table 3.4-2 Line Number (255) Table 3.4-2 Line Number (257) Table 3.4-2 Line Number (261) Table 3.4-2 Line Number (262) Table 3.4-2 Line Number (265)
VALVE BODY	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (388) Table 3.4-2 Line Number (390) Table 3.4-2 Line Number (434) Table 3.4-2 Line Number (441) Table 3.4-2 Line Number (442) Table 3.4-2 Line Number (443) Table 3.4-2 Line Number (446) Table 3.4-2 Line Number (450) Table 3.4-2 Line Number (460) Table 3.4-2 Line Number (461)

1. Floor drain sumps are baffled to prevent cross-communication. These metallic baffles are considered penetration seals.

2.3.3.13 Radiation Monitoring

System Description

The function of the Radiation Monitoring system is to detect any plant problem, which may lead to a radiation hazard and/or release of radioactivity to the environment. The system also warns the operators of this hazard so that appropriate actions may be taken. To accomplish this function, the system utilizes both area and process radiation monitors. Some radiation monitors sense parameters and generate signals that interface with engineered safety features actuation (e.g. containment isolation) or are used to monitor for reactor coolant leakage. Others provide automatic non-safety process system control functions as a result of a high alarm. Radiation monitors also ensure control room habitability by generating an isolation signal used to secure the control room ventilation envelope. The Control Room Emergency Air Treatment System also contains non-nuclear safety toxic gas detection that electrically interfaces with the radiation monitoring systems control room ventilation isolation signal. These toxic gas monitors are included within the evaluation boundary of the radiation monitoring system as non-safety components whose failure could prevent the satisfactory accomplishment of a safety related function. The Radiation Monitoring system also contains post accident monitoring instrumentation that is Environmentally Qualified.

The principal components of the Radiation Monitoring system include area monitors, process monitors, System-Level Particulate, Iodine, and Nobel Gas Monitors (SPING), data acquisition modules (DAM), computer interface and terminal equipment, toxic gas detectors, and the pumps, valves and essential ductwork and piping necessary for their functioning. UFSAR Section 11.5, Process And Effluent Radiation Monitoring And Sampling Systems," provides a detailed description of all the radiation monitors and their functions.

The following fluid systems interface with the Radiation Monitoring System (note that the pressure boundary interfaces are evaluated within the system being monitored):

Reactor Coolant	Waste Disposal
Component Cooling	Service Water
Treated Water	

System Function Listing

In addition to the System Functions described above, the Radiation Monitoring System also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code D SENSE OR PROVIDE PROCESS CONDITIONS AND GENERATE SIGNALS FOR REACTOR TRIP AND ENGINEERED SAFETY FEATURES ACTUATION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Radiation Monitoring system perform this primary design system function.

Code J PROVIDE HEAT REMOVAL FROM SAFETY RELATED HEAT EXCHANGERS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Radiation Monitoring system perform this associated design system function (Service Water interface with RM-20B and RE-20B).

Code K PROVIDE PRIMARY CONTAINMENT BOUNDARY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Radiation Monitoring system perform this primary design system function.

Code O MAINTAIN EMERGENCY TEMPERATURES WITHIN AREAS CONTAINING SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Radiation Monitoring system perform this associated design system function.

Code Q PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Radiation Monitoring system perform this associated design system function.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Radiation Monitoring system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Radiation Monitoring system that perform License Renewal Criterion 3 functions are tracked under the Criterion 3 codes (Z1 through Z5).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Radiation Monitoring system perform this associated design system function.

Code X SFR FUNCTION NOT APPLICABLE AT COMPONENT LEVEL	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Radiation Monitoring system perform specific safety related functions different from and in addition to the system level functions (e.g. Reg Guide 1.97 Category 1).

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Radiation Monitoring system perform this associated design system function. Some components could have pressure boundary failures which, if they occurred outside the safety boundary, could dilute the sample and affect the safety function of the device. Additionally, Control Room Toxic Gas monitoring is included for review within the boundaries of the Radiation Monitoring system.

Code Z2 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ENVIRONMENTAL QUALIFICATION (10 CFR 50.49)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: Components within the Radiation Monitoring system are designated as Environmentally Qualified.

UFSAR Reference

Additional Radiation Monitoring system details are provided in Section 6.4.2.2.3, Section 6.4.5, Section 11.5, Section 12.3.4, and Table 6.2-15a of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Radiation Monitoring system are listed below:

33013-1231
33013-1245
33013-1250,2
33013-1250,3
33013-1278,2
33013-1866
33013-1867
33013-1893
33013-2287
33013-1271
33013-1871
33013-1904
33013-1912

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.3-13 along with each Component Group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.3-13 Radiation Monitoring

Component Group	Passive Function	Aging Management Reference
CS COMPONENTS	PRESSURE BOUNDARY	Table 3.4-1 Line Number (13)
FASTENERS (BOLTING)	JOINT INTEGRITY	Table 3.4-1 Line Number (13) Table 3.4-1 Line Number (23) Table 3.4-2 Line Number (79) Table 3.4-2 Line Number (80) Table 3.4-2 Line Number (81)
FILTER HOUSING	PRESSURE BOUNDARY	Table 3.4-2 Line Number (97) Table 3.4-2 Line Number (99)
FLOW ELEMENT	PRESSURE BOUNDARY	Table 3.4-2 Line Number (106) Table 3.4-2 Line Number (108)
FLOW METER[1]	PRESSURE BOUNDARY	Table 3.4-2 Line Number (112) Table 3.4-2 Line Number (113)

Table 2.3.3-13 Radiation Monitoring

Component Group	Passive Function	Aging Management Reference
PIPE	PRESSURE BOUNDARY	Table 3.4-2 Line Number (232) Table 3.4-2 Line Number (236)
PUMP CASING	PRESSURE BOUNDARY	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (259) Table 3.4-2 Line Number (268) Table 3.4-2 Line Number (270)
RADIATION DETECTOR HOUSING	PRESSURE BOUNDARY	Table 3.4-2 Line Number (276) Table 3.4-2 Line Number (277) Table 3.4-2 Line Number (278) Table 3.4-2 Line Number (279) Table 3.4-2 Line Number (280) Table 3.4-2 Line Number (281)
RADIATION MONITOR SKID	PRESSURE BOUNDARY	Table 3.4-2 Line Number (282) Table 3.4-2 Line Number (283) Table 3.4-2 Line Number (284)
SPECIAL ELEMENT	PRESSURE BOUNDARY	Table 3.4-2 Line Number (291) Table 3.4-2 Line Number (292)
VALVE BODY	PRESSURE BOUNDARY	Table 3.4-2 Line Number (392) Table 3.4-2 Line Number (394) Table 3.4-2 Line Number (426) Table 3.4-2 Line Number (429) Table 3.4-2 Line Number (444) Table 3.4-2 Line Number (445) Table 3.4-2 Line Number (447)

1. Selected instruments were conservatively included within the scope of License Renewal. Consideration was given to the consequences of an instrument housing pressure boundary failure. Where an instrument was unisolable from a pressure source and is of sufficient size that a system function would be degraded should the pressure boundary fail, that instrument is included for License Renewal review.

2.3.3.14 Circulating Water - Not Within Scope of License Renewal

System Description

The Circulating Water (CW) system is designed to provide a reliable supply of water, regardless of weather or lake conditions, to the suction of the screen house pumps. Those portions of the CW system that support the delivery of lake water sufficient for the use of service water and fire water pumps are evaluated within the Service Water system. Those portions of the Circulating Water system that provide circulating water flood detection are evaluated within the Reactor Protection system. Consequently, within the system evaluation boundary there are no components that perform License Renewal Intended Functions.

The principal components of the CW system are: the circulating water pumps, travelling screens, chlorine addition tanks and pumps and the essential piping and valves. The function of the circulating water system is to provide a reliable supply of water to condense the steam exhausted from the low-pressure turbines.

The circulating water system is a nonseismic piping system whose primary function is to remove heat from the steam cycle via the main condensers. To achieve this end, the system consists of an intake structure specially designed to minimize the possibility of clogging, an inlet tunnel, four traveling screens, two circulating water pumps, and a discharge canal. The intake tunnel and the intake tunnel bypass (loss of lake valve and bypass piping) are evaluated within the Service Water system. The intake is designed to withstand, without loss of function, ground accelerations of 0.2g, acting in the vertical and horizontal planes simultaneously. The probability of water stoppage due to plugging of the inlet has been reduced to an extremely low value by incorporating design features in the system. Before the inlet plenum water reaches the pump suctions, the water passes through the four parallel traveling screens. Service water pump discharge is used to periodically flush the debris off the screens into a collecting trough where it is carried away.

The following fluid systems interface with the Circulating Water System:

Service Water	Waste Disposal
Fire Protection	

System Function Listing

A comprehensive listing of all functions associated with the Circulating Water system, or specific components contained in the system, is provided in the summary below.

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Circulating Water system perform this primary design system function.

UFSAR Reference

Additional Circulating Water system details are provided in Section 10.6 of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Circulating Water system are listed below

33013-1250,1
33013-1885,1
33013-1885,2
33013-1896
33013-1921

2.3.3.15 Chilled Water - Not Within Scope of License Renewal

System Description

The Chilled Water system supports normal habitability and equipment reliability by maintaining Control Room and office space temperature within acceptable bounds during normal operating conditions. Accordingly, components within the Chilled Water system do not perform any License Renewal intended functions.

The principal components of the Chilled Water system are the chilled water pumps, the chiller units, a surge tank and the essential piping and valves. Chilled water (which uses service water as a heat sink) supplies chilled water to the control room HVAC unit and various cooling coils within individual service building HVAC units.

The following fluid systems interface with the Chilled Water System:

Service Water
Treated Water

System Function Listing

A comprehensive listing of all functions associated with the Chilled Water system, or specific components contained in the system, is provided in the summary below.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Chilled Water system perform this primary design system function (augmented quality). Components within the Chilled Water system maintain an atmosphere in the Main Control Room conducive to continuous occupancy during any mode of normal operation or event.

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Chilled Water system perform this associated design system function (e.g. chemical addition for water treatment, test and drain valves, etc.)

UFSAR Reference

Additional Chilled Water system details are provided in Section 6.4.2.2 of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Chilled Water system are listed below

33013-1250,3	33013-1899,2
33013-1876	33013-1920
33013-1877	33013-2282
33013-1879	

2.3.3.16 Fuel Handling - Not Within Scope of License Renewal

System Description

The Fuel Handling system provide a safe and effective means for transporting and handling reactor fuel from the time the fuel reaches the plant in an unirradiated condition until it is placed in the spent fuel pool racks to await final long term storage. Note: The Fuel Handling system boundary does not include any cranes, hoists or lifting devices categorized under NUREG-0612, Control of

Heavy Loads. Cranes, new and spent fuel storage racks, the spent fuel pool and cavity liners are evaluated separately. Accordingly, components within the Fuel Handling Systems do not perform any License Renewal intended functions.

The principal components of the Fuel Handling system include the new fuel elevator, the underwater air motor driven fuel conveyor car, the pneumatic control assembly equipment for the fuel manipulator cranes, fuel and reactor internals handling tools, control equipment for safety interlocks, and essential valves and air tubing.

Special precautions are taken in all fuel handling operations to minimize the possibility of damage to fuel assemblies during transport to and from the spent fuel pool (SFP) and during installation in the reactor. All handling operations on irradiated fuel are conducted under water. The handling tools used in the fuel handling operations are conservatively designed and the associated devices are of a fail-safe design.

In the fuel storage area, administrative controls and geometric constraints ensure that the fuel assemblies are spaced in a pattern which prevents any possibility of a criticality accident. Also, crane interlocks and administrative controls prevent carrying heavy objects, such as a spent fuel transfer cask, over the fuel assemblies in the storage racks. In addition, administratively, only one fuel assembly can be handled at a given time over storage racks containing spent fuel. The motions of the cranes which move the fuel assemblies are limited to a relatively low maximum speed. Caution is exercised during fuel handling to prevent the fuel assembly from striking another fuel assembly or structures in the containment or spent fuel pool. The fuel handling equipment suspends the fuel assembly in the vertical position during fuel movements, except when the fuel is moved through the transport tube.

The following fluid systems interface with the Fuel Handling System:

Plant Air

System Function Listing

A comprehensive listing of all functions associated with the Fuel Handling system, or specific components contained in the Fuel Handling system, is provided within the summary below.

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Fuel Handling system perform this associated design system function.

Note: Equipment used to handle heavy loads (NUREG-0612) is evaluated within Cranes, Hoists and Lifting Devices.

UFSAR Reference

Additional Fuel Handling system details are provided in Section 9.1.2.1.7, Section 9.1.4, and Section 15.7.3 of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Fuel Handling system are listed below:

33013-1887 33013-1889

2.3.3.17 Plant Sampling - Not Within Scope of License Renewal

System Description

The Plant Sampling system provide representative nuclear process systems (e.g. primary coolant) and non-nuclear, non-radioactive process systems (e.g. condensate) samples for laboratory analysis. Equipment for sampling secondary and non-radioactive fluids is separated from the equipment provided for reactor coolant samples. No components within the Plant Sampling System evaluation boundary perform license renewal intended functions. Leakage and drainage resulting from the radioactive sampling operations are collected and drained to tanks located in the waste disposal system. Components associated with containment isolation are evaluated in the Containment Isolation system. Safety related interface components (e.g. heat exchangers) are evaluated within the system that is used to remove heat.

The principal components of the Plant Sampling systems include heat exchangers, pumps, tanks, and the essential piping and valves. Two types of samples are obtained by the nuclear process sampling portion of the system: high temperature-high pressure reactor coolant system and steam generator blowdown samples which originate inside the reactor containment, and low temperature-low pressure samples from the chemical and volume control and auxiliary coolant systems. Typical information obtained from the primary coolant analyses includes reactor coolant boron and chloride concentrations; fission product radioactivity level; corrosion product concentration and chemical additive concentrations; and oxygen, hydrogen, and fission gas content.

The nuclear process portion of the sample system also includes a Post Accident Sampling System. The postaccident sampling system is designed to allow the station to obtain and analyze reactor coolant, containment air, and containment sump samples within 3 hours after the decision is made to sample. The postaccident sampling system also permits routine sampling of these process streams. In-line chemical instrumentation is provided in a liquid and gas sample panel that remotely determines important chemical parameters of the reactor coolant, containment air, and containment sump A. In addition, the liquid and gas sample panel enables acquisition of both diluted and undiluted grab samples of the reactor coolant and containment air for isotopic analysis in the counting lab.

The non-nuclear or secondary sampling system is provided with a number of sampling points. Inline analyzers are provided for selected parameters to allow continuous information useful in evaluating secondary conditions and in developing corrective actions when required. Major elements of the non-nuclear process sampling portion of the system include: Steam Generator Blowdown Sampling, Main Condenser Hotwell Sampling, Condensate Sampling, Feedwater Sampling, Main Steam Sampling and Heater Drain Tank Sampling. Typical information obtained from secondary sampling include: pH, Conductivity, Chlorides, Sulfate, Sodium, Ethanolamine and Ammonia.

The following fluid systems interface with the Plant Sampling System:

Reactor Coolant	Residual Heat Removal
Chemical and Volume Control	Service Water
Component Cooling Water	Waste Disposal
Plant Air Systems	Main and Auxiliary Steam
Feed Water and Condensate	

System Function Listing

A comprehensive listing of all functions associated with the Plant Sampling system, or specific components contained in the system, is provided in the summary below

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Primary Sampling portion of the Plant Sampling systems perform this associated design system function (augmented quality). This function supports monitoring Reg Guide 1.97, Category 3 Variables (Post Accident Sampling).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Plant Sampling systems perform this associated design system function.

UFSAR Reference

Additional Plant Sampling systems details are provided in Section 9.3.2 of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Plant Sampling systems are listed below:

33013-1245
33013-1251,2
33013-1278,1
33013-1278,2
33013-1279
33013-1866
33013-1888
33013-1890
33013-1893
33013-1894,1
33013-1894,2
33013-1895
33013-1899,1
33013-1908,3
33013-1911,2
33013-2251,1
33013-2251,2
33013-2251,3
33013-2251,4
33013-2281,1
33013-2281,2
33013-2287
33013-2711,1
33013-2711,2
33013-2711,3
33013-2711,4
33013-2739

2.3.3.18 Plant Air - Not Within Scope of License Renewal

System Description

The Plant Air systems, although supplying valves in safety related systems, is not designed as a safety-related system. Safety related systems using instrument air are designed such that upon loss of air pressure each component will fail in a position of greater safety. Components that require a pneumatic motive of force to achieve a safety function (e.g. pressurizer power operated relief valves) have nitrogen backup that is evaluated with the system containing those components. Those portions of Plant Air that act as containment isolation devices are evaluated in the Containment Isolation system. Consequently, within the system evaluation boundary there are no components that perform License Renewal Intended Functions.

The principal components of the Plant Air systems are compressors, tanks, filters, dryers and the essential piping and valves. The instrument air system supplies clean, dry air for valve operators, and piping penetration pressurization. The service air system supplies air for maintenance and service use and the backup eductor for vapor extraction of the turbine-generator bearing drains. A backup source of air supply to the instrument air header is from the service air system. The instrument air system produces 120 to 125 psig dry, filtered air used chiefly as the motive power for valve actuation. The system consists of three air compressors with an associated aftercooler and air reservoir for each compressor. Air from the receivers is supplied to the instrument air header through filters and an air dryer. The instrument air header delivers air to the various valve actuators, piping penetration pressurization system, and containment air and proof test system. The service air system produces 115 to 125 psig dry, filtered air used in the maintenance air connections throughout the station, for fire water storage tank pressurization, and the turbine lube-oil system. The system consists of one air compressor with an integral aftercooler and associated air receivers. A cross-tie between service air and instrument air allows the service air system to supply the instrument air header if instrument air pressure drops below 90 psig. The cross-tie occurs prior to the instrument air filters. Therefore, air being supplied to the instrument air header will always pass through the filters and dryer. A cross-connect between the service air system and the instrument air system allows both systems to be supplied by a single rotary screw air compressor. A pressure regulator valve will stop air flow to the service air system if pressure on the service air side drops below 100 psig. Administratively, the instrument air system and service air system are cross-connected only when one of the rotary screw air compressors is in operation.

The following fluid systems interface with the Plant Air System:
Service Water

System Function Listing

A comprehensive listing of all functions associated with the Plant Air system, or specific components contained in the system, is provided in the summary below.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Plant Air systems perform this associated design system function (augmented quality).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Plant Air systems perform this associated design system function.

UFSAR Reference

Additional Plant Air systems details are provided in Section 9.3.1 and Section 3.5.1.3.2.5 of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Plant Air systems are listed below:

33013-1248	33013-1892
33013-1251,2	33013-1893
33013-1278,1	33013-1894,1
33013-1866	33013-1894,2
33013-1882	33013-1895
33013-1883	33013-1896
33013-1884,1	33013-1897,1
33013-1884,2	33013-1897,2
33013-1886,1	33013-1898
33013-1886,2	33013-1899,1
33013-1887	33013-1899,2
33013-1888	33013-1900,1
33013-1889	33013-1900,2
33013-1890	33013-1903
33013-1891	33013-1925

2.3.3.19 Non-Essential Ventilation - Not Within Scope of License Renewal

System Description

The Non-Essential Ventilation Systems provide heating, ventilation and air conditioning to non-vital areas and plant equipment. Components within the Non-Essential Ventilation Systems do not perform any License Renewal intended functions.

The principal components of the Non-Essential Ventilation Systems include filters, fans, dampers, valves, heat exchangers, conditioning/chiller packages and the essential ductwork and piping and valves. Fire dampers contained in the system are evaluated as a separate commodity group.

The turbine building, while not requiring a heating, ventilation, and air conditioning system, uses roof vent fans, wall vent fans, windows, and unit heaters for ventilation and temperature control. The fans are not supplied by emergency power, and loss of these fans would not be critical to a safe shutdown. Included in the turbine building is the main feedwater pump room. Main feedwater pump equipment cooling systems use a mixture of outside air and room air to control the room and equipment temperatures. No mechanical means of heating or cooling is used. A temperature control system controls the feedwater pump room return air dampers and equipment outside air dampers that admit air to the equipment air supply fan plenum mixed at a setpoint temperature.

The service building ventilation system consists of air handling units serving the various areas of the service building. Air from uncontaminated areas is exhausted through roof exhaust fans. Air from areas of potential contamination, such as laboratories equipped with hoods, are exhausted through the controlled intermediate building controlled access area exhaust fans. Controlled access area fans 1A and 1B include high efficiency particulate air and charcoal filter banks, a low-flow alarm, dampers, and fans. These fans take suction from the following areas and discharge to the Auxiliary Building HEPA filter vent which is exhausted by the main Auxiliary Building exhaust system to the main vent header:

a. Men's and women's decontamination general areas.

b. Radiation protection and chemistry office general area.

c. Primary sample room general area.

d. Primary sample hood.

e. Primary and secondary sample lab hoods.

f. Hot shop general area.

The All-Volatile-Treatment Building Ventilation system provides ventilation and heating to maintain required temperatures for the all-volatile-treatment (condensate demineralizer) building and the condensate booster pump area of the Turbine Building.

The evaluation boundary for Non-Essential Ventilation Systems also includes baseboard circulating radiant heat in the service building and any Heating Ventilation and Air Conditioning associated with non-safety buildings not used in direct support of power production (e.g. Engineering Building, Butler Building, Records Management (Steam Generator) Building, etc.).

The following fluid systems interface with the Non-Essential Ventilation Systems:

Plant Air	Heating Steam
Site Services and Facility Support	Chilled Water

System Function Listing

A comprehensive listing of all functions associated with the Non-Essential Ventilation Systems, or specific components contained in the system, is provided in the summary below.

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Non-Essential Ventilation systems perform this associated design system function.

UFSAR Reference

Additional Non-Essential Ventilation Systems details are provided in Section 9.4.5, Section 9.4.6, and Section 9.4.7 of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Non-Essential Ventilation Systems are listed below:

33013-1872
33013-1873
33013-1874
33013-1875
33013-1876
33013-1877
33013-1878
33013-1879
33013-1880
33013-1881
33013-1895
33013-1897,1
33013-1899,1
33013-1899,2
33013-1909
33013-1913
33013-2250

2.3.3.20 Site Service and Facility Support - Not Within Scope of License Renewal

Description

Site Service and Facility Support Systems evaluation boundary includes domestic (potable) water, domestic hot water and the site sewage transfer to the municipal treatment system. Components within the Site Service and Facility Support Systems do not perform any License Renewal intended functions.

The principal components of the Site Service and Facility Support Systems include heat exchangers, hot water heaters, pumps and essential piping and valves. Domestic water is used for drinking, showers, eye wash stations and various domestic applications. The sewage transfer system pumps collected sanitary discharges from the site to the municipal sanitary header offsite. The sewage transfer system does not interconnect with any potentially radioactive systems so no radiation monitoring is required.

The following fluid systems interface with Site Service and Facility Support Systems:

Non-Essential Ventilation Systems
Treated Water
Plant Sampling

System Function Listing

A comprehensive listing of all functions associated with the Site Service and Facility Support Systems, or specific components contained in the system, is provided within the summary below.

Code T	Cri 1	Cri 2	Cri 3				
NON-NUCLEAR SAFETY CLASS FUNCTIONS			FP	EQ	PTS	AT	SB

Comment: Components within the Site Service and Facility Support system perform this associated design system function.

UFSAR Reference

Additional Site Service and Facility Support Systems details are provided in Section 1.2.6 and Section 1.2.12 the UFSAR.

License Renewal Drawings

The license renewal drawings for the Site Service and Facility Support Systems are listed below:

33013-1279
33013-1908,3
33013-2250
33013-2252
33013-2681

2.3.4 Steam and Power Conversion Systems

The following systems are addressed in this section:

- Main and Auxiliary Steam System (Section 2.3.4.1)
- Feedwater and Condensate System (Section 2.3.4.2)
- Auxiliary Feedwater System (Section 2.3.4.3)
- Turbine-Generator and Supporting Systems (Section 2.3.4.4)

2.3.4.1 Main and Auxiliary Steam

System Description

The Main and Auxiliary Steam system provides heat removal from the reactor coolant system during normal, accident and post accident conditions. During off-normal conditions the system provides emergency heat removal from the RCS using secondary heat removal capability. Some non-safety related portions of piping in the system have failure modes which could prevent the satisfactory accomplishment of safety related functions (high energy line breaks). The system is also credited for safe shutdown following Station Blackout events, some fire events, and contains components which are part of the Environmental Qualification Program. Selected safety valve discharge vent piping is considered non safety equipment whose failure could affect a safety function due to their importance in directing steam flow out of a safety related area. The conversion of the heat produced in the reactor to electrical energy is evaluated in the Turbine Generator system.

The principal components of the main steam portion of the system include the secondary side of two steam generators, where the main steam lines begin. Each steam line has a flow restrictor, four main steam safety valves, an atmospheric dump valve and a steam admission valve to the turbine-driven auxiliary feedwater pump (TDAFW). The two steam lines join together in the intermediate building prior to entering the turbine building. Each steam line is also equipped with a fast closing Main Steam Isolation Valve (MSIV) and a Main Steam non-return check valve. These valves prevent reverse flow in the steam lines which would result from an upstream steam line break or they isolate any downstream steam line break at the common header. The atmospheric relief valves (ARVs) have two functions. They offer overpressure protection to the steam generator at a setpoint below the main steam safety valves (MSSVs) setpoints and can be used to maintain no-load T_{AVG} or perform a plant cooldown in the event the steam dump to the condenser is not available.

The principal components of the auxiliary steam portion of the system include the piping valves and tanks in the extraction steam and steam generator blowdown sub-systems. In extraction steam, five stages of extraction are provided; two from the high-pressure turbine, one of which is the exhaust, and three stages from the low-pressure turbines. There are also two steam dump lines with four relief valves each to the condenser.

Continuous steam generator blowdown is used to reduce the quantities of solids that accumulate in the steam generators as a result of the boiling process. The blowdown recovery system is designed to recover both the blowdown water and heat. Each steam generator has a blowdown header located at the bottom of the shell side just above the tubesheet. Both steam generators are equipped with independent blowdown piping from the connecting steam generator nozzles to a flash tank. The piping transports the removed fluid and entrapped debris away from the steam generator, through containment penetrations, to a common flash tank in the turbine building basement. Flashed steam is vented from the flash tank to low-pressure feedwater heater 3A for heat recovery. The vented steam condenses in the feedwater heater and returns to the condenser through the feedwater heater drain system. The remaining condensate in the blowdown flash tank is drained directly to condenser 1B through a level control valve.

The following fluid systems interface with Main and Auxiliary Steam:

Reactor Coolant	Auxiliary Feedwater
Plant Sampling	Feedwater and Condensate
Turbine Generator	Plant Air

System Function Listing

In addition to the System Functions listed above, the Main and Auxiliary Steam System also supports additional functions (associated design system functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code H	Cri 1	Cri 2	Cri 3				
PROVIDE EMERGENCY HEAT REMOVAL FROM THE REACTOR COOLANT SYSTEM USING SECONDARY HEAT REMOVAL CAPABILITY			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Main and Auxiliary Steam system perform this primary design system function.

Code K PROVIDE PRIMARY CONTAINMENT BOUNDARY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Main and Auxiliary Steam system perform this primary design system function.

Code Q PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Main and Auxiliary Steam system perform this associated design system function.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Main and Auxiliary Steam system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Main and Auxiliary Steam system that perform special capability class functions are tracked under the Criterion 3 codes (Z1 through Z5).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Main and Auxiliary Steam system perform this associated design system function.

Code X SFR FUNCTION NOT APPLICABLE AT COMPONENT LEVEL	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Main and Auxiliary Steam system perform specific safety related functions different from and in addition to the system level functions. Flow elements in Main Steam pipe used for ESFAS.

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Main and Auxiliary Steam system perform this associated design system function. Portions of the Main Steam system resist HELB. Safety valve discharge vent stack pipe directs steam out of the intermediate building.

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Main and Auxiliary Steam system perform this associated design system function.

Code Z2 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ENVIRONMENTAL QUALIFICATION (10 CFR 50.49)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: Components within the Main and Auxiliary Steam system are designated as environmentally qualified.

Code Z5 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR STATION BLACKOUT (10 CFR 50.63)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: Components within the Main and Auxiliary Steam system perform this associated design system function.

UFSAR Reference

Additional Main and Auxiliary Steam System details are provided in Section 5.4.6, Section 10.1.1, Section 10.3, Section 10.7, and Table 6.2-15a of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Main and Auxiliary Steam System are listed below:

33013-1231	33013-1899,1
33013-1232	33013-1903
33013-1277,1	33013-1905
33013-1893	33013-1916,2
33013-1277,2	33013-1918,1
33013-1885,1	33013-1918,2
33013-1894,1	33013-1919,1
33013-1894,2	33013-1919,2
33013-1895	33013-1924
33013-1896	33013-2251,2

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.4-1 along with each Component Group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.4-1 Main and Auxiliary Steam

Component Group	Passive Function	Aging Management Reference
CONDENSING CHAMBER	PRESSURE BOUNDARY	Table 3.5-1 Line Number (2) Table 3.5-1 Line Number (5) Table 3.5-1 Line Number (6) Table 3.5-1 Line Number (7)
CONVERTER[1]	PRESSURE BOUNDARY	Table 3.5-2 Line Number (4) Table 3.5-2 Line Number (5)
CS COMPONENTS	PRESSURE BOUNDARY	Table 3.5-1 Line Number (13)
FASTENERS (BOLTING)	JOINT INTEGRITY	Table 3.5-1 Line Number (8) Table 3.5-1 Line Number (13) Table 3.5-2 Line Number (7) Table 3.5-2 Line Number (8) Table 3.5-2 Line Number (9)
FLOW ELEMENT	PRESSURE BOUNDARY	Table 3.5-2 Line Number (11) Table 3.5-2 Line Number (12) Table 3.5-2 Line Number (15) Table 3.5-2 Line Number (16)

Table 2.3.4-1 Main and Auxiliary Steam

Component Group	Passive Function	Aging Management Reference
OPERATOR	PRESSURE BOUNDARY	Table 3.5-1 Line Number (5) Table 3.5-2 Line Number (24)
PIPE	PRESSURE BOUNDARY	Table 3.5-1 Line Number (2) Table 3.5-1 Line Number (5) Table 3.5-1 Line Number (6) Table 3.5-1 Line Number (7) Table 3.5-2 Line Number (28) Table 3.5-2 Line Number (29) Table 3.5-2 Line Number (33) Table 3.5-2 Line Number (34) Table 3.5-2 Line Number (35) Table 3.5-2 Line Number (39) Table 3.5-2 Line Number (41)
POSITIONER[1]	PRESSURE BOUNDARY	Table 3.5-2 Line Number (42) Table 3.5-2 Line Number (43)
PRESSURE RELAY[1]	PRESSURE BOUNDARY	Table 3.5-2 Line Number (44) Table 3.5-2 Line Number (45)
SCREEN	PRESSURE BOUNDARY	Table 3.5-2 Line Number (48) Table 3.5-2 Line Number (49)
VALVE BODY	PRESSURE BOUNDARY	Table 3.5-1 Line Number (2) Table 3.5-1 Line Number (5) Table 3.5-1 Line Number (6) Table 3.5-1 Line Number (7) Table 3.5-2 Line Number (56) Table 3.5-2 Line Number (57) Table 3.5-2 Line Number (58) Table 3.5-2 Line Number (62) Table 3.5-2 Line Number (63) Table 3.5-2 Line Number (67) Table 3.5-2 Line Number (68) Table 3.5-2 Line Number (69) Table 3.5-2 Line Number (73) Table 3.5-2 Line Number (75)

1. Selected instruments were conservatively included within the scope of License Renewal. Consideration was given to the consequences of an instrument housing pressure boundary failure. Where an instrument was unisolable from a pressure source and is of sufficient size that a system function would be degraded should the pressure boundary fail, that instrument is included for License Renewal review.

2.3.4.2 Feedwater and Condensate

System Description

The Feedwater and Condensate systems function to condense the steam exhausted from the low-pressure turbines, collect and store this condensate, and then send it back to the steam generator for reuse. Components within the system are used to provide emergency heat removal from the reactor coolant system using secondary heat removal capability. The Engineered Safety Features Actuation System (ESFAS) provides actuation signals for feedwater isolation. Portions of the main feedwater piping systems in the intermediate building the turbine building have failure modes and effects which could prevent the satisfactory accomplishment of a safety related function (high energy line breaks). The feedwater lines are equipped with a non-return check valve and an isolation valve in each line. The non-return valve is the boundary between Seismic Category I and nonseismic feedwater piping and prevents the steam generator from blowing back through the feedwater lines if damage occurs to the nonseismic portion. Components within the Feedwater and Condensate system are also credited for use in safe shutdown following Station Blackout events and some fires. Additionally, components within the system perform functions used to mitigate anticipated transients without a scram (ATWS) and components that are part of the Environmental Qualification Program.

The principal components of the Feedwater and Condensate system are the feedwater and condensate pumps, the feedwater regulating and bypass valves, feedwater heaters and the essential piping and valves. The steam that leaves the exhaust of the low-pressure turbines enters the main condenser as saturated steam with low moisture content. This steam is condensed by the circulating water, which passes through the tubes of the condenser. The condensed steam collects in the condenser hotel from which the condensate pumps take suction. The condensate pumps increase the pressure of the water and provide suction head for the condensate booster pumps. The condensate booster pumps in turn provide sufficient suction head for the main feedwater pumps. Between the condensate pumps and the condensate booster pumps is the condensate demineralizer system, which maintains condensate water purity. The condensate booster pumps flow condensate through the condensate cooler, hydrogen coolers, air ejector condensers, gland steam condenser, and low-pressure heaters to the suction of the feedwater pumps. The feedwater pumps send feedwater through the high-pressure heaters to the steam generators via the feedwater regulating valves.

The following fluid systems interface with the Feedwater and Condensate system:

Reactor Coolant	Auxiliary Feedwater
Plant Air	Plant Sampling
Turbine Generator	Main and Auxiliary Steam

System Function Listing

In addition to the System Functions listed above, the Feedwater and Condensate System also supports additional functions (associated design system functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code H PROVIDE EMERGENCY HEAT REMOVAL FROM THE REACTOR COOLANT SYSTEM USING SECONDARY HEAT REMOVAL CAPABILITY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Feedwater and Condensate system perform this associated design system function.

Code K PROVIDE PRIMARY CONTAINMENT BOUNDARY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Feedwater and Condensate system perform this associated design system function.

Code Q PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Feedwater and Condensate system perform this associated design system function.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Feedwater and Condensate system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Feedwater and Condensate system that perform special capability class functions are tracked under the Criterion 3 codes (Z1 through Z5).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Feedwater and Condensate system perform this associated design system function.

Code X SFR FUNCTION NOT APPLICABLE AT COMPONENT LEVEL	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Feedwater and Condensate system perform specific safety related functions different from and in addition to the system level functions. Components within the Feedwater and Condensate system monitor Reg Guide 1.97 Category 1 variables.

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Feedwater and Condensate system perform this associated design system function. Portions of the Feedwater system resist HELB.

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Feedwater and Condensate system perform this associated design system function.

Code Z2 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ENVIRONMENTAL QUALIFICATION (10 CFR 50.49)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: Components within the Feedwater and Condensate system are designated as environmentally qualified.

Code Z4 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ANTICIPATED TRANSIENTS WITHOUT SCRAM (10 CFR 50.62)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
						X	

Comment: Components within the Feedwater and Condensate system perform this associated design system function.

Code Z5 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR STATION BLACKOUT (10 CFR 50.63)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: Components within the Feedwater and Condensate system perform this associated design system function.

UFSAR Reference

Additional Feedwater and Condensate System details are provided in Section 5.4.6, Section 10.1.1, Section 10.4, and Table 6.2-15a of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Feedwater and Condensate System are listed below:

33013-1232
33013-1236,1
33013-1236,2
33013-1237
33013-1238
33013-1915
33013-1233
33013-1235
33013-1251,1
33013-1251,2
33013-1252
33013-1277,2
33013-1885,1
33013-1894,1
33013-1894,2
33013-1895
33013-1896
33013-1897,1
33013-1897,2
33013-1899,2
33013-1902
33013-1903
33013-1904
33013-1905
33013-1909
33013-1918,1
33013-1918,2
33013-1919,1
33013-1919,2
33013-1921
33013-1922
33013-1923
33013-1924
33013-2276
33013-2277
33013-2711,1
33013-2739

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.4-2 along with each Component Group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.4-2 Feedwater and Condensate

Component Group	Passive Function	Aging Management Reference
CONDENSING CHAMBER	PRESSURE BOUNDARY	Table 3.5-1 Line Number (2) Table 3.5-1 Line Number (5) Table 3.5-1 Line Number (6)
CS COMPONENTS	PRESSURE BOUNDARY	Table 3.5-1 Line Number (13)
FASTENERS (BOLTING)	JOINT INTEGRITY	Table 3.5-1 Line Number (8) Table 3.5-1 Line Number (13) Table 3.5-2 Line Number (7) Table 3.5-2 Line Number (8) Table 3.5-2 Line Number (9)

Table 2.3.4-2 Feedwater and Condensate

Component Group	Passive Function	Aging Management Reference
FLOW ELEMENT	PRESSURE BOUNDARY	Table 3.5-1 Line Number (2) Table 3.5-1 Line Number (5) Table 3.5-1 Line Number (6)
PIPE	PRESSURE BOUNDARY	Table 3.5-1 Line Number (2) Table 3.5-1 Line Number (5) Table 3.5-1 Line Number (6) Table 3.5-2 Line Number (34) Table 3.5-2 Line Number (35) Table 3.5-2 Line Number (38) Table 3.5-2 Line Number (39) Table 3.5-2 Line Number (40) Table 3.5-2 Line Number (41)
TEMPERATURE ELEMENT[1]	PRESSURE BOUNDARY	Table 3.5-2 Line Number (51) Table 3.5-2 Line Number (52) Table 3.5-2 Line Number (53) Table 3.5-2 Line Number (54) Table 3.5-2 Line Number (55)
VALVE BODY	PRESSURE BOUNDARY	Table 3.5-1 Line Number (2) Table 3.5-1 Line Number (5) Table 3.5-1 Line Number (6) Table 3.5-2 Line Number (68) Table 3.5-2 Line Number (69) Table 3.5-2 Line Number (72) Table 3.5-2 Line Number (73) Table 3.5-2 Line Number (74) Table 3.5-2 Line Number (75)

1. Selected instruments were conservatively included within the scope of License Renewal. Consideration was given to the consequences of an instrument housing pressure boundary failure. Where an instrument was unisolable from a pressure source and is of sufficient size that a system function would be degraded should the pressure boundary fail, that instrument is included for License Renewal review.

2.3.4.3 Auxiliary Feedwater (AFW)

System Description

The Auxiliary Feedwater (AFW) system is designed to maintain the steam generator water inventory when the normal feedwater system is not available. During accident and post accident conditions the AFW system supplies feedwater to the steam generators in order provide emergency heat removal from the reactor coolant system using secondary heat removal capability (atmosphere or main condenser). The AFW system is also credited for use in mitigating Anticipated Transients Without a Scram (ATWS), and safe shutdown following Station Blackout Events (SBO) and some fires.

The principal components of the AFW system are electric motor-driven and steam turbine-driven pumps, the turbine-driven feedwater pump (TDAFW) oil system, and the essential piping and valves. The preferred auxiliary feedwater system is divided into two independent trains. There are two motor-driven pumps powered from separate redundant 480-V safeguards emergency buses which can receive power from either onsite or offsite sources. Each motor-driven pump can provide 100% of the preferred auxiliary feedwater system flow required for decay heat removal and can be cross-connected to provide flow to either steam generator. There is also a turbine-driven pump which can receive motive steam from each steam line and provide flow to either or both steam generators. The turbine-driven pump provides 200% of the flow required for decay heat removal.

A standby auxiliary feedwater system (SAFW) provides flow in case the preferred auxiliary feedwater system pumps are inoperable (e.g. a high-energy line break event could render inoperable the three preferred auxiliary feedwater pumps). The standby auxiliary feedwater system (SAFW) uses two motor-driven pumps which can be aligned to separate service water (SW) system loops. The standby auxiliary feedwater system (SAFW) has the same features as the preferred auxiliary feedwater system pumps with regard to functional capability and power supply separation. The system is manually actuated from the control room.

The condensate storage tanks are the normal (preferred) suction source for delivery of cooling water to the steam generators. The safety-related supply is from the plant service water system with the fire water system as a back-up source.

The following fluid systems interface with the Auxiliary Feedwater System:

Main and Auxiliary Steam	Service Water
Plant Air Systems	Treated Water
Feedwater and Condensate	Fire Protection

System Function Listing

In addition to the System Functions described above, the Auxiliary Feedwater System also contains components which support additional functions (associated design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code H PROVIDE EMERGENCY HEAT REMOVAL FROM THE REACTOR COOLANT SYSTEM USING SECONDARY HEAT REMOVAL CAPABILITY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Auxiliary Feedwater system perform this primary design system function.

Code J PROVIDE HEAT REMOVAL FROM SAFETY RELATED HEAT EXCHANGERS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Auxiliary Feedwater system perform this associated design system function.

Code K PROVIDE PRIMARY CONTAINMENT BOUNDARY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Auxiliary Feedwater system perform this associated design system function.

Code Q PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Auxiliary Feedwater system perform this associated design system function.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Auxiliary Feedwater system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Auxiliary Feedwater system that perform special capability class functions are tracked under the Criterion 2 code (Y) and Criterion 3 codes (Z1 through Z5).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Auxiliary Feedwater system perform this associated design system function.

Code X SFR FUNCTION NOT APPLICABLE AT COMPONENT LEVEL	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Auxiliary Feedwater system perform specific safety related functions different from and in addition to the system level functions (some electrical components within this system do not perform nuclear safety functions, but are not electrically isolated from safety related circuits).

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Auxiliary Feedwater system perform this associated design system function (Condensate storage tanks and Turbine Driven Aux Feed Pump steam trap drain components).

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Auxiliary Feedwater system perform this associated design system function.

Code Z4	Cri 1	Cri 2	Cri 3				
LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ANTICIPATED TRANSIENTS WITHOUT SCRAM (10 CFR 50.62)			FP	EQ	PTS	AT	SB
						X	

Comment: Components within the Auxiliary Feedwater system perform this associated design system function.

Code Z5	Cri 1	Cri 2	Cri 3				
LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR STATION BLACKOUT (10 CFR 50.63)			FP	EQ	PTS	AT	SB
							X

Comment: Components within the Auxiliary Feedwater system perform this associated design system function.

UFSAR Reference

Additional Auxiliary Feedwater System details are provided in Section 7.2.6, Section 10.5, Section 10.7.4, and Table 6.2-15a of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Auxiliary Feedwater System are listed below:

33013-1231	33013-1893
33013-1234	33013-2285
33013-1237	33013-1892
33013-1238	33013-2739

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.3.4-3 along with each Component Group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.4-3 Auxiliary Feedwater (AFW)

Component Group	Passive Function	Aging Management Reference
ACCUMULATOR	PRESSURE BOUNDARY	Table 3.5-1 Line Number (4) Table 3.5-1 Line Number (5)
CONTROLLER[1]	PRESSURE BOUNDARY	Table 3.5-2 Line Number (1) Table 3.5-2 Line Number (2) Table 3.5-2 Line Number (3)
COOLER	PRESSURE BOUNDARY	Table 3.5-1 Line Number (4) Table 3.5-1 Line Number (9) Table 3.5-2 Line Number (6)
CS COMPONENTS	PRESSURE BOUNDARY	Table 3.5-1 Line Number (13)
FASTENERS (BOLTING)	JOINT INTEGRITY	Table 3.5-1 Line Number (8) Table 3.5-1 Line Number (13) Table 3.5-2 Line Number (7) Table 3.5-2 Line Number (8) Table 3.5-2 Line Number (9)
FILTER HOUSING	PRESSURE BOUNDARY	Table 3.5-1 Line Number (4) Table 3.5-2 Line Number (10)
FLOW ELEMENT	PRESSURE BOUNDARY	Table 3.5-2 Line Number (12) Table 3.5-2 Line Number (13) Table 3.5-2 Line Number (14)
GOVERNOR	PRESSURE BOUNDARY	Table 3.5-1 Line Number (2) Table 3.5-1 Line Number (5)
HEAT EXCHANGER	HEAT TRANSFER PRESSURE BOUNDARY	Table 3.5-1 Line Number (4) This applies to both passive functions. Table 3.5-1 Line Number (9) Table 3.5-2 Line Number (18) Table 3.5-2 Line Number (20) These apply to the heat transfer passive function. Table 3.5-2 Line Number (17) Table 3.5-2 Line Number (19) Table 3.5-2 Line Number (21) These apply to the pressure boundary passive function.

Table 2.3.4-3 Auxiliary Feedwater (AFW)

Component Group	Passive Function	Aging Management Reference
LEVEL GLASS	PRESSURE BOUNDARY	Table 3.5-2 Line Number (22) Table 3.5-2 Line Number (23)
ORIFICE	PRESSURE BOUNDARY RESTRICTS FLOW	Table 3.5-1 Line Number (4) Table 3.5-2 Line Number (25) Table 3.5-2 Line Number (26) Table 3.5-2 Line Number (27) These apply to both passive functions.
PIPE	PRESSURE BOUNDARY	Table 3.5-1 Line Number (2) Table 3.5-1 Line Number (4) Table 3.5-1 Line Number (5) Table 3.5-1 Line Number (6) Table 3.5-2 Line Number (30) Table 3.5-2 Line Number (31) Table 3.5-2 Line Number (32) Table 3.5-2 Line Number (34) Table 3.5-2 Line Number (35) Table 3.5-2 Line Number (36) Table 3.5-2 Line Number (37)
PUMP CASING	PRESSURE BOUNDARY	Table 3.5-1 Line Number (2) Table 3.5-1 Line Number (5) Table 3.5-2 Line Number (46) Table 3.5-2 Line Number (47)
SPEED INCREASER	PRESSURE BOUNDARY	Table 3.5-1 Line Number (4) Table 3.5-1 Line Number (5)
TANK	PRESSURE BOUNDARY	Table 3.5-1 Line Number (2) Table 3.5-1 Line Number (4) Table 3.5-1 Line Number (5) Table 3.5-2 Line Number (50)
TRAP HOUSING	PRESSURE BOUNDARY	Table 3.5-1 Line Number (2) Table 3.5-1 Line Number (5)

Table 2.3.4-3 Auxiliary Feedwater (AFW)

Component Group	Passive Function	Aging Management Reference
VALVE BODY	PRESSURE BOUNDARY	Table 3.5-1 Line Number (2) Table 3.5-1 Line Number (4) Table 3.5-1 Line Number (5) Table 3.5-2 Line Number (59) Table 3.5-2 Line Number (60) Table 3.5-2 Line Number (61) Table 3.5-2 Line Number (63) Table 3.5-2 Line Number (64) Table 3.5-2 Line Number (65) Table 3.5-2 Line Number (66) Table 3.5-2 Line Number (69) Table 3.5-2 Line Number (70) Table 3.5-2 Line Number (71)

1. Selected instruments were conservatively included within the scope of License Renewal. Consideration was given to the consequences of an instrument housing pressure boundary failure. Where an instrument was unisolable from a pressure source and is of sufficient size that a system function would be degraded should the pressure boundary fail, that instrument is included for License Renewal review.

2.3.4.4 Turbine-Generator and Supporting Systems

System Description

The Turbine Generator and Supporting Systems function to convert the energy of the heat contained in the main steam into mechanical energy for use in turning the electric generator. These systems have no safety related functions. Turbine first-stage pressure instruments provide a signal used in Anticipated Transients Without a Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC).

The plant subsystems with boundary of the Turbine generator and Supporting systems include: the high and low pressure turbine generator and controls, the main electrical generator, the electro-hydraulic control system, the turbine lube oil system, condenser air ejector and vacuum priming, generator hydrogen cooling and generator seal oil systems.

The principal components of the Turbine Generator systems include: turbines, the main generator, pumps, tanks, heat exchanges, and the essential piping and valves. The main turbine is made up of one high-pressure and two low-pressure turbines, all mounted on a common shaft. The steam flow path is first through the high-pressure turbine, then in a parallel path to the two low-pressure units via the four moisture separator reheaters. High-pressure steam is admitted to the high-pressure turbine through two stop and four governing control valves. These valves are controlled by the electro-hydraulic control system. Turbine supervisory instrumentation is provided to monitor turbine vibration, eccentricity, and differential thermal expansion and provide alarms in the control room in the case of abnormal conditions.

The main turbine is supported by a number of auxiliary systems that improve the efficiency and safety of its operation. First and second stage air ejectors remove air and noncondensible gases from the condenser and maintain it under a vacuum, improving the efficiency of the main turbine by reducing the backpressure seen by the turbine exhaust. The gland sealing and exhaust system applies steam to a labyrinth seal around the rotor shaft to preclude air inleakage into the turbine casings and condenser and to prevent steam leakage into the turbine building. The vacuum priming system uses mechanical vacuum pumps to prevent air buildup in the condenser water boxes or tubes-a condition that would reduce condenser efficiency. The exhaust hood spray prevents overheating of the last stage low-pressure blading under low steam flow conditions. The turbine lube-oil system provides lubrication and cooling of the turbine bearings and supplies oil to the auto-stop header for turbine protection. It also provides backup oil to the seal-oil system to prevent hydrogen leakage into the turbine building. A purification system is an adjunct to the turbine lube-oil system to remove water and contaminants from the lube-oil, as well as to provide storage space for makeup oil. The generator auxiliary systems are required to ensure that the main generator will operate at its maximum rated output safely and efficiently. This is accomplished by cooling the generator rotor, stator, exciter, main output bushings, and the isophase bus ducts. Pressurized hydrogen is circulated by the internal ventilation of the generator to remove heat produced in the rotor and stator. The hydrogen then transfers this heat to hydrogen coolers which are supplied with cooling water from the condensate system. To prevent the escape of hydrogen along the generator shaft and out of the casing, a seal-oil system is utilized. The air-side seal-oil pump and the hydrogen-side seal-oil pump provide oil for sealing at pressure higher than generator hydrogen pressure. The main turbine oil system can provide a backup source of pressurized seal oil.

The Turbine Generator and Supporting Systems do not interface with any plant mechanical systems not already addressed above.

System Function Listing

In addition to the System Functions listed above, the Turbine Generator and Supporting Systems also supports additional functions (associated design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Turbine Generator and Supporting systems perform this associated design system function (augmented quality) (provide turbine trip signals which could lead to reactor trip). For the purposes of License Renewal, components within the Turbine Generator and Supporting systems that perform special capability class functions are tracked under the Criterion 3 codes (Z1 through Z5).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Turbine Generator and Supporting systems perform this associated system function.

Code Z4 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ANTICIPATED TRANSIENTS WITHOUT SCRAM (10 CFR 50.62)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
						X	

Comment: Components within the Turbine-Generator and Supporting systems perform this associated design system function. Turbine first-stage pressure instruments provide a signal used in Anticipated Transients Without a Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC).

UFSAR Reference

Additional Turbine-Generator and Supporting Systems details are provided in Section 7.2.6 and Section 10.2 of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Turbine-Generator and Supporting Systems are listed below:

33013-1232	33013-1904
33013-1233	33013-1905
33013-1235	33013-1918,1
33013-1251,1	33013-1918,2
33013-1251,2	33013-1921
33013-1252	33013-1923
33013-1874	33013-1924
33013-1894,1	33013-2249,1
33013-1894,2	33013-2249,2
33013-1895	33013-2277
33013-1896	33013-2283
33013-1901	33013-2284
33013-1903	33013-2711,1

Components Subject to an AMR

The component groups for these systems that require aging management review are indicated in Table 2.3.4-4 along with each Component Group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.3.4-4 Turbine-Generator and Supporting Systems

Component Group	Passive Function	Aging Management Reference
PIPE	PRESSURE BOUNDARY	Table 3.5-1 Line Number (2) Table 3.5-1 Line Number (5) Table 3.5-1 Line Number (6) Table 3.5-2 Line Number (35) Table 3.5-2 Line Number (39) Table 3.5-2 Line Number (41)
VALVE BODY	PRESSURE BOUNDARY	Table 3.5-1 Line Number (2) Table 3.5-1 Line Number (5) Table 3.5-1 Line Number (6) Table 3.5-2 Line Number (69) Table 3.5-2 Line Number (73) Table 3.5-2 Line Number (75)

2.4 Scoping and Screening Results: Structures

To optimize the aging management review, structures that are attached to or contained within larger structures, have been reviewed with the larger structure. Also, structural elements that have similar materials and experience similar environments have been grouped and reviewed together. The following structural groupings are addressed in this section:

- Containment Structures (Section 2.4.1)
- Essential Buildings and Yard Structures (including component supports)(Section 2.4.2)
- Non-Essential Buildings and Yard Structures (Section 2.4.3)

Note: Only the general arrangement plot drawing is included in the application. Detailed civil drawings may contain sensitive information and are therefore being made available for on site review only.

2.4.1 Containment Structures

Description

The reactor Containment Structure is a reinforced-concrete, vertical right cylinder with a flat base and a hemispherical dome. The structure houses and supports safety related equipment, provides radiation shielding, and provides a barrier against the release of radioactive nuclides. A welded steel liner is attached to the inside face of the concrete shell to ensure a high degree of leak tightness. The thickness of the liner in the cylinder and dome is 3/8-in. and in the base it is 1/4 in. The cylindrical reinforced-concrete walls are 3 ft 6 in. thick, and the concrete hemispherical dome is 2 ft 6 in. thick. The concrete base slab is 2 ft thick with an additional 2-ft-thick concrete fill over the bottom liner plate. The Containment Structure is 99 ft high to the spring line of the dome and has an inside diameter of 105 ft. The containment vessel provides a minimum free volume of approximately 972,000 ft^3. Access is provided by means of two airlocks designed with an interlocked single-door-opening feature that is leak testable at containment design pressure between doors. One airlock is removable to provide the ability to move large equipment into and out of containment.

The major components of the reactor coolant system are located within the Containment Structure. The Containment Structure provides a physical barrier to protect the equipment from natural disasters and shielding to protect personnel from radiation emitted from the reactor core while at power. Thick reinforced-concrete walls are located around the selected reactor coolant system components to serve as shielding for plant personnel. These walls also serve as a missile barrier to prevent damage to the containment wall and to components of the safety injection system should a failure occur to one of the reactor coolant system components located inside the walls. Removable shield blocks are located over the pressurizer. A movable missile barrier spans the refueling cavity over the reactor vessel head. The reactor vessel is located in the center of the Containment Structure below ground level. The space below the reactor vessel (sump A) allows for access to the bottom of the reactor vessel. Extending around the top of the reactor vessel is a stainless-steel-lined refueling cavity. Prior to refueling operations a removable seal is installed between the reactor vessel flange and the cavity. The refueling cavity is flooded with borated water to provide shielding and cooling of the irradiated fuel being removed from the reactor vessel. During post accident conditions any spilled reactor coolant, along with any water injected by the emergency core cooling systems, is ultimately collected in containment sump B for recirculation and cooling. Sump A is protected from debris fouling by a concrete curb and metal trash screen.

The Containment Structure also provides and interfaces with equipment and component supports. Major component supports include the Reactor Vessel, Steam Generators, Reactor Coolant Pumps, and the Pressurizer. The structure also supports the containment and refueling cranes. The component supports are attached either to the concrete foundation, concrete floor slabs, or shield walls through steel embedments or structural steel members encased in concrete.

The Containment Structure consists of a reinforced concrete cylinder post-tensioned in the vertical direction and reinforced circumferentially with mild steel deformed bars. The dome is hemispherical and constructed of reinforced concrete. Insulation is provided for the side walls to a point 15 ft 0 in. above the spring line so as to limit the maximum liner temperature due to the loss-of-coolant accident and thereby avoid excessive compressive stresses in the steel liner plate.

A two-foot thick reinforced concrete base slab extends radially from the reactor cavity pit to the containment cylinder wall. Except for participation in anchoring the radial tension bars at the base of the cylinder, the base slab is not an integral part of the containment shell in this design. The base slab rests directly on rock, and the loads on base slab are those from the internal structures and equipment. Near the cylinder wall the slab thickens to 6 ft. and extends beneath the wall above the concrete ring beam. The base slab and ring beam supports the dome and cylinder walls. The ring beam rests directly on rock and is the location of the end anchorage for the rock anchors. No drainage or de-watering system is provided under the Containment Structure. The base of the cylinder is supported by a neoprene pad, which provides a hinge support at the base. The vertical post-tensioning system is anchored at the base of the cylinder to rock anchors. The rock anchors are post-tensioned and grouted, which ensures that the rock acts as an integral part of the containment.

The rock anchors resist vertical axial loads in the cylinder walls and thereby avoid the transfer of vertical shear to the base slab. A sufficient physical separation is provided between wall and base slab to ensure that there is no transfer of vertical reaction to the base slab. A hinge is developed at the base of the containment cylinder by supporting the wall vertically on a series of elastomer bearing pads and anchoring the wall horizontally into the base slab with radial, high-strength steel tension bars. The tension bars resist the radial shear at the base of the containment cylinder and transfer this force as radial compression into the thickened portion of the base slab and ring beam, and thence, as a lateral load, onto the rock outboard of the ring beam and base slab.

The design for the containment provides for prestressing the concrete in the cylinder walls in the longitudinal (vertical) direction with a sufficient compressive force to ensure that upon application of the design load combinations there will be no tensile stresses in the concrete due to membrane forces. The steel tendons for prestressing consist of high tensile, bright, cold drawn and stress-relieved steel wires. The prestressed concrete is assumed to develop no tensile capacity in a direction normal to a horizontal plane.

The use of unbonded tendons gives, in addition to other advantages, accessibility for inspection or replacement. However, because the tendons are not in intimate and integral contact with surrounding concrete, the advantage of the high alkaline environment for use as a corrosion inhibitor is lost. Therefore, these tendons must be provided with a corrosion preventive medium that gives protection equivalent to concrete, but still enables withdrawal of a tendon for inspection or replacement. Consequently, one of the more important programs in connection with the tendons has been the selection of a complete corrosion protection system. The various elements involved are (1) a cathodic protection system in which all tendons are connected to the liner and then to a copper grounding system which is completed by the addition of reference cells and anodes, from which a protective potential can be generated if the need for cathodic protection is indicated by the reference cells, (2) a steel conduit surrounding each tendon providing shielding against stray electrical currents, (3) temporary shipping and erection protection of all wires in each tendon, by the application of a coating, followed by complete filling of each tendon conduit with a petroleum base wax that provides a permanent, chemically stable environment for protection from corrosion, while still giving flexibility of withdrawal for inspection. A tendon surveillance program, in accordance with Regulatory Guide 1.35, Revision 2 is required by the station Technical Specifications.

The containment leakage pressure boundary is provided by the single steel liner in the containment vessel. Each system whose piping penetrates this boundary is designed to maintain isolation of the containment from the outside environment. Provision is made to periodically monitor leakage by pressurizing the penetrations and containment. The

Containment Structure and all penetrations are designed to withstand, within design limits, the combined loadings of the design-basis accident and design seismic conditions. All piping systems, which penetrate the containment, are anchored in the penetration sleeve or the structural concrete of the Containment Structure. The penetrations for the main steam, feedwater, blowdown, and sample lines are designed so that the penetration is stronger than the piping system and the containment will not be breached due to a postulated pipe rupture. The liner thickness in the vicinity of typical penetrations is increased to a minimum of 3/4 in. All lines connected to the primary coolant system that penetrate the containment are also anchored in the secondary shield walls (i.e., walls surrounding the steam generators and reactor coolant pumps) and are each provided with at least one valve between the anchor and the reactor coolant system. For mechanical penetrations that interface with hot fluid systems, a containment penetration cooling system is used to prevent the bulk concrete temperature surrounding the penetrations from exceeding 150°F. Containment electrical penetrations are designed so the Containment Structure can, without exceeding the design leakage rate, accommodate the postulated environment resulting from a loss-of-coolant accident. The electrical penetrations have been shown to maintain structural integrity when subjected to mechanical stresses caused by large magnitude fault currents.

A fuel transfer penetration is provided for fuel movement between the refueling transfer canal in the reactor containment and the spent fuel pool. The penetration consists of a stainless steel pipe installed inside a larger pipe. The inner pipe acts as the transfer tube and connects the refueling canal with the spent fuel pool. The tube is fitted with a standard stainless steel flange in the refueling canal and a stainless steel sluice gate valve in the spent fuel pool. The outer pipe is welded to the containment liner. The fuel transfer penetration, like all other penetrations, is anchored in the containment shell. Because this anchor point moves when the containment vessel is subjected to load, expansion joints are provided where the penetration is connected to structures inside and outside of the containment vessel. Since the penetration is located on a skewed angle, not normal to the containment shell, the expansion joints are subjected to both radial and tangential (lateral) motions. The expansion bellows inside the containment vessel provide a water seal for the refueling canal and accommodate thermal growth of the penetration from the anchor, as well as the pressure and earthquake produced motion of the anchor (the containment shell). The expansion joint accommodates motion of the sleeve within the containment shell relative to the portion of the sleeve anchored in the wall of the refueling canal in the Auxiliary Building.

The Containment Structure contains racks, panels, electrical enclosures, equipment supports. Additionally the structure contains radiant heat shields and a reactor coolant pump oil collection system credited with preserving the ability to achieve safe shutdown in the event of a fire in containment. Those equipment sets receive a separate commodity group evaluation independent of the structure evaluation. Refueling equipment and the cranes located in containment also receive a separate evaluation.

System Function Listing

In addition to the System Functions described above, the Containment Structure also contains components which support additional functions (associated design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code K PROVIDE PRIMARY CONTAINMENT BOUNDARY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Containment Structure perform this primary design system function.

Code R STRUCTURALLY SUPPORT OR HOUSE SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Containment Structure perform this primary design system function.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Containment Structure perform this associated design system function (augmented quality, e.g. Reg Guide 1.97 Category 2 post accident monitoring variables). For the purposes of License Renewal, components within the Containment Structure that perform License Renewal Criterion 3 functions are tracked under the Criterion 3 codes (Z1 through Z5).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Containment Structure perform this associated design system function.

Code Z2 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ENVIRONMENTAL QUALIFICATION (10 CFR 50.49)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: Components within the Containment Structure are designated as Environmentally Qualified (electrical penetrations).

UFSAR Reference

Additional Containment Structures details are provided in Section 3.8.1, Section 3.8.2, Section 3.8.3, and Section 9.1.4.1 of the UFSAR.

License Renewal Drawings

The license renewal drawings for Containment Structures are listed below:

33013-1231
33013-1236,2
33013-1238
33013-1246,1
33013-1247
33013-1248
33013-1250,3
33013-1258
33013-1261
33013-1262,1
33013-1262,2
33013-1264
33013-1265,1
33013-1265,2
33013-1275,2
33013-1277,1
33013-1278,1
33013-1279
33013-1863
33013-1865
33013-1866
33013-1870
33013-1882
33013-1884,1
33013-1884,2
33013-1908,3
33013-1915
33013-1991
Site Plot
33013-1272,1
33013-1883
33013-1886,1
33013-1887

Components/Commodities Subject to an AMR

The component groups for the Containment Structures that require aging management review are indicated in Table 2.4.1-1 along with each component group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.4.1-1 Containment Structures

Component Group	Passive Function	Aging Management Reference
CV-BLOCK-INT This generic asset includes all masonry block walls used in the Containment Vessel protected from the weather. The containment elevator shaft is block. Mortar is included in this asset evaluation.	STRUCTURAL/ FUNCTIONAL SUPPORT NSR EQUIPMENT	Table 3.6-2 Line Number (13)

Table 2.4.1-1 Containment Structures

Component Group	Passive Function	Aging Management Reference
CV-C-BUR This generic asset includes all concrete in the Containment Vessel that is in contact with the soil and groundwater. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are evaluated in the Containment Vessel threaded fasteners and the exposed faces of plates and structural members are evaluated in Containment Vessel structural steel. This generic asset includes evaluation of sealing materials used below grade in the Containment Vessel. Water stops are embedded in selected below grade construction joints while sealants were used on exterior below grade walls.	PRESSURE BOUNDARY/LEAK BARRIER STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (7) Table 3.6-1 Line Number (8) Table 3.6-1 Line Number (9) Table 3.6-1 Line Number (10) These apply to both passive functions.

Table 2.4.1-1 Containment Structures

Component Group	Passive Function	Aging Management Reference
CV-C-EXT This generic asset includes all concrete in the Containment Vessel that is exposed to the weather. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are included in the Containment Vessel threaded fasteners and the exposed faces of plates and structural members are included in Containment Vessel structural steel. Containment tendon conduit, expansion bellows, etc. encased in concrete are included in this evaluation.	PRESSURE BOUNDARY/LEAK BARRIER RADIATION/HEAT SHIELDING STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (7) Table 3.6-1 Line Number (8) Table 3.6-1 Line Number (10) These apply to all three passive functions.

Table 2.4.1-1 Containment Structures

Component Group	Passive Function	Aging Management Reference
CV-C-INT This generic asset includes all concrete in the Containment Vessel that is protected from the weather including the biological shield walls and missile barriers. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. Grout, used under column base plates is included in this evaluation. The exposed portion of structural anchor bolts are evaluated in the Containment Vessel threaded fasteners and the exposed faces of plates and structural members are evaluated in Containment Vessel structural steel.	PIPE WHIP RESTRAINT RADIATION/HEAT SHIELDING SHELTER/PROTECT EQUIPMENT STRUCTURAL/ FUNCTIONAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (7) Table 3.6-1 Line Number (8) Table 3.6-1 Line Number (10) These apply to all five passive functions.
CV-ELAST-EXT This generic asset includes elastomer materials used in the Containment Vessel that is exposed to the weather. Included in this evaluation are the neoprene gaskets used to seal the tendon grease cans.	STRUCTURAL/ FUNCTIONAL SUPPORT NSR EQUIPMENT	Table 3.6-1 Line Number (6)

Table 2.4.1-1 Containment Structures

Component Group	Passive Function	Aging Management Reference
CV-ELAST-INT This generic asset includes elastomer materials used in the Containment Vessel that is protected from the weather. Included in this evaluation is the caulking used between thermal insulation panels and between the containment floor and the insulation.	SHELTER/PROTECT EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (6) Table 3.6-2 Line Number (5) These apply to both passive functions.
CV-EPOX-INT This generic asset includes the epoxy in the Containment Vessel that is protected from the weather. Epoxy resin is used to encapsulate the exposed tendon fill port piping.	STRUCTURAL/ FUNCTIONAL SUPPORT NSR EQUIPMENT	Table 3.6-1 Line Number (14)
CV-FAST(CS)-INT This generic asset includes the exposed portion of carbon steel threaded fasteners for the Containment Vessel that are protected from the weather. The exposed portion of high strength low alloy steel fasteners are evaluated in the component supports commodity group.	PIPE WHIP RESTRAINT SHELTER/PROTECT EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT STRUCTURAL/ FUNCTIONAL SUPPORT NSR EQUIPMENT	Table 3.6-2 Line Number (16) This applies to all four passive functions.

Table 2.4.1-1 Containment Structures

Component Group	Passive Function	Aging Management Reference
CV-INSULATION This generic asset includes the Containment Vessel thermal insulation panels.	RADIATION/HEAT SHIELDING	Table 3.6-2 Line Number (14)
Electrical Penetrations This asset represents pressure retaining boundary of the electrical penetration, including any sleeves or dissimilar metal welds.	PRESSURE BOUNDARY	Table 3.6-1 Line Number (1) Table 3.6-1 Line Number (2) Table 3.6-1 Line Number (3)
CV-SS(CS)-EXT This generic asset includes all carbon structural steel in the Containment Vessel that are exposed to the weather. The Containment tendon grease cans are carbon steel.	STRUCTURAL/ FUNCTIONAL SUPPORT NSR EQUIPMENT	Table 3.6-1 Line Number (14)

Table 2.4.1-1 Containment Structures

Component Group	Passive Function	Aging Management Reference
CV-SS(CS)-INT This generic asset includes all carbon structural steel of the Containment Vessel that is protected from the weather. Columns, posts, beams, baseplates, bracing, crane support girders, crane rails, and the exposed faces of plates and structural members are included. This does not include carbon structural steel used as component supports.	PIPE WHIP RESTRAINT SHELTER/PROTECT EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT STRUCTURAL/ FUNCTIONAL SUPPORT NSR EQUIPMENT	Table 3.6-2 Line Number (15) This applies to all four passive functions.
CV-SS(CS)-LINER This generic asset includes all carbon steel of the Containment Vessel liner that is protected from the weather. Included in this evaluation are the insulation retaining studs.	PRESSURE BOUNDARY/LEAK BARRIER	Table 3.6-1 Line Number (12)
CV-SS(CS)-ROCKANCHOR This generic asset represents the high strength carbon steel rock anchors grouted into bedrock. The bottom anchor button head is included in this evaluation. The top head is evaluated with the CV-SS(CS)-TENDONS asset.	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (14)

Table 2.4.1-1 Containment Structures

Component Group	Passive Function	Aging Management Reference
CV-SS(CS)-TENDONFILL This generic asset represents the carbon steel grease fill ports. The tendon conduits are evaluated with the CV-C-BUR asset where the steel is imbedded in concrete and the CV-SS(CS)-EXT asset where the conduit exits the top of containment.	STRUCTURAL/ FUNCTIONAL SUPPORT NSR EQUIPMENT	Table 3.6-1 Line Number (14)
CV-SS(CS)-TENDONS This generic asset represents the high strength carbon steel tensioning tendon wire cluster encapsulated in NO-OX-ID (paraffin/ mineral oil wax). The top rock anchor button head is included in this evaluation	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (14)
CV-SS(SS)-INT This generic asset includes all stainless structural steel of the Containment Vessel that is protected from the weather. Included in this evaluation is the refueling cavity and fuel transfer liners (including attachments). Stainless steel expansion bellows used with containment penetrations are addressed with the penetration.	STRUCTURAL SUPPORT SR EQUIPMENT STRUCTURAL/ FUNCTIONAL SUPPORT NSR EQUIPMENT	Table 3.6-2 Line Number (17) This applies to both passive functions.

Table 2.4.1-1 Containment Structures

Component Group	Passive Function	Aging Management Reference
Mechanical Penetrations This asset represents pressure retaining boundary of the mechanical penetration, including any penetration sleeves, bellows, and dissimilar metal welds.	PRESSURE BOUNDARY	Table 3.6-1 Line Number (1) Table 3.6-1 Line Number (2) Table 3.6-1 Line Number (3)
SPP01 This asset represent the movable hatch and mechanical wear surfaces of the personnel hatch.	PRESSURE BOUNDARY	Table 3.6-1 Line Number (5)
SPP01-GASKET This asset represents the inner and outer elastomeric seals for the hatch doors.	PRESSURE BOUNDARY	Table 3.6-1 Line Number (6)
SPP02 This asset represent the movable hatch and mechanical wear surfaces of the equipment hatch.	PRESSURE BOUNDARY	Table 3.6-1 Line Number (5)
SPP02-GASKET This asset represents the inner and outer elastomeric seals for the hatch doors as well as the containment vessel to hatch seal.	PRESSURE BOUNDARY	Table 3.6-1 Line Number (6)

Table 2.4.1-1 Containment Structures

Component Group	Passive Function	Aging Management Reference
VALVE BODY This generic asset represents the bronze manual valves attached to the tendon fill port piping.	STRUCTURAL/ FUNCTIONAL SUPPORT NSR EQUIPMENT	Table 3.6-1 Line Number (14)

2.4.2 Essential Buildings and Yard Structures

The following structures are included in this subsection:

- Auxiliary Building (Section 2.4.2.1)
- Intermediate Building (Section 2.4.2.2)
- Turbine Building (Section 2.4.2.3)
- Diesel Building (Section 2.4.2.4)
- Control Building (Section 2.4.2.5)
- All Volatile Water Treatment Building (Section 2.4.2.6)
- Screen House Building (Section 2.4.2.7)
- Standby Auxiliary Feedwater Building (Section 2.4.2.8)
- Service Building (Section 2.4.2.9)
- Cable Tunnel (Section 2.4.2.10)
- Essential Yard Structures (Section 2.4.2.11)
- Component Supports Commodity Group (Section 2.4.2.12)

The Essential Buildings and Yard Structures group is a compilation of all site civil buildings and features that perform License Renewal Intended Functions. Included within the Essential Buildings and Yard Structures evaluation boundary are civil commodity groups. Each building and commodity group has been separately scoped and screened for License Renewal with the scoping results documented below.

2.4.2.1 Auxiliary Building

Description

The Auxiliary Building is a Seismic Category I three-story rectangular structure measuring approximately 70 ft. by 214 ft. It is located south of the Containment and Intermediate Buildings and adjacent to the Service Building. The Auxiliary Building houses the major support and engineered safety features equipment required for plant operation. Portions of the structure act as fire barriers. Additionally the building contains non-safety elements whose failure can affect a safety function (portions of the structure are designed to resist, and protect equipment from, high-energy line breaks, flooding, and tornadoes).

The Auxiliary Building is part of a complex of interconnected buildings surrounding, but structurally independent of, the Containment Building. These buildings are interconnected as follows: The Seismic Category I Auxiliary Building is contiguous with the nonseismic Service Building on the west side. The Seismic Category I Intermediate Building adjoins the seismically analyzed Turbine Building to the north, and the Auxiliary Building to the south. The Turbine Building adjoins the Seismic Category I Diesel Generator Building to the north and the Seismic Category I Control Building to the south. The facade, a cosmetic rectangular structure that encloses the Containment Building, has all four sides partly or totally in common with the Auxiliary and Intermediate buildings. The Auxiliary Building adjoins the Seismic Category I Standby Auxiliary Feedwater Building on the south. In the original building analysis, each Seismic Category I structure was treated independently. During the Systematic Evaluation Program (SEP) plant evaluation it was found that the interconnected nature of the buildings was an important feature, especially in view of the lack of detailed original seismic design information. Therefore, both Seismic Category I and nonseismic category buildings were included in a complicated three-dimensional structural system reanalysis model. As part of this effort, the interconnected Turbine Building was determined to be capable of withstanding safe shutdown earthquake forces. Based on the SEP review, audits, and plant inspections, the NRC safety evaluation reports concluded that the evaluation and resolution of SEP Topics III-2, Wind and Tornado Loadings; III-4.A, Tornado Missiles; III-6, Seismic Design Considerations; and III-7.B, Load Combinations, were acceptable. The NRC also concluded that the RG&E analysis and implementation of the Structural Upgrade Program were acceptable.

Below grade the Auxiliary Building is primarily concrete, above grade the building has two roofs constructed of steel beam and bracing systems and supported by steel frame bracing systems. Insulated siding is used for most of the walls above the operating floor. The south side of the building has a combination of concrete block finished with architectural brick and siding while portions of the east and north sides contain concrete block and siding. The low roof section of the Auxiliary Buildings parapets have been provided with scuppers designed to ensure that any rainwater, resulting from a design-basis storm, would not accumulate on the roofs and cause overload. The scuppers are located so that their outflow will not damage any surrounding plant structures. The roofing and siding provide weather resistance and allow habitability control but are not designed to be wind or tornado missile resistant.

The structure has a concrete basement floor that rests on a sandstone foundation at elevation 235 ft 8 in., and two reinforced concrete floors: an intermediate floor at elevation 253 ft and an operating floor at elevation 271 ft. The refueling water storage tank extends through all three levels. The intermediate and operating level floors have a minimum thickness of 1.5 ft, and are supported by 2.5-ft thick concrete walls at the south, east, and part of the north sides of the building. There are a number of 2.5-ft to 3.5-ft thick concrete shield walls and compartments located on the floors.

The northwest corner of the building is adjacent to the circular wall of the Containment Building. The west concrete wall, which encloses the spent fuel storage pool, is 6 ft thick. The spent fuel storage pool, located in the Auxiliary Building, is a rectangular concrete structure lined with stainless steel. It contains approximately 255,000 gallons of borated water. Its bottom is at elevation 236 ft 8 in. Walls are 6-ft thick at the north and west sides and 3-ft thick at the east and south sides, which are below the ground surface and also serve as retaining walls. A fuel transfer penetration is provided for fuel movement between the refueling transfer canal in the reactor containment and the spent fuel pool. The penetration consists of a stainless steel pipe installed inside a larger pipe. The inner pipe acts as the transfer tube and connects the reactor refueling canal with the spent fuel pool. The tube is fitted with a standard stainless steel flange in the refueling canal and a stainless steel sluice gate valve in the spent fuel pool. The outer pipe is welded to the containment liner and provision is made for gas leak testing of all welds essential to the integrity of the penetration. The gasketed expansion joint accommodates motion of the sleeve within the containment shell relative to the portion of the sleeve anchored in the wall of the refueling canal in the Auxiliary Building. The expansion bellows inside the Auxiliary Building performs the same function as described for that within the containment.

The west end of the Auxiliary Building superstructure is connected with a portion of the service building and on the northwest with the Intermediate Building. The interior walls separating the buildings are block. However, the foundation of the Auxiliary Building is independent of these building foundations.

The major structures of Ginna Station have experienced no visible evidence of settlement since the construction of the station. (During the SEP and evaluation of Topic II-4.F, Settlement of Foundations and Buried Equipment, the NRC concluded that the settlement of foundations and buried equipment is not a safety concern for Ginna Station.)

In addition to structural and load bearing elements, the Auxiliary Building contains features and appurtenances credited in the licensing basis and relied upon to ensure the health and safety of the public. These features include:

a. Backdraft dampers installed in the Auxiliary Building north wall in order to eliminate the effects of differential pressures associated with the design-basis tornado.

b. Pipe whip and jet impingement protection provided for the 6-in. heating steam line riser located on the intermediate floor of the Auxiliary Building to protect safety-related electrical equipment in the vicinity of the riser.

c. Portable flood barriers for use in the event of flooding from Deer Creek. (The flood barriers consist of a panel with a pair of Pneuma-seal inflatable gaskets on the sides and across the bottom. The panels slide into frames installed around the Auxiliary Building personnel access doors and the rollup vehicle access door. Air flasks located in the Auxiliary Building adjacent to the doors which receive the barriers and are used to inflate the gaskets.) The building exterior provides flood barrier continuity between the door openings where the portable barriers are installed.

d. Curbing at the entrance to the RHR pit to prevent water from spilling from the basement floor into the pit.

e. Sealing material in between the pipe chase and RHR suction piping where it enters the RHR pit to prevent water spilling from the basement floor into the pit.

f. A fire resistant enclosure (the charging pump room) to provide separation between trains of Appendix R safe shutdown equipment.

g. Sealing material between the Intermediate Building/Auxiliary Building, and Auxiliary Building/Containment Building rattle gaps to prevent externally induced flooding of the Auxiliary Building.

h. A barrier at the cable tunnel entrance to the Auxiliary Building to prevent possible cross communication of smoke.

i. Restraining devices installed at the Intermediate Building/Auxiliary Building interface (north side, above the spent fuel pool) to ensure a block wall failure will not damage spent fuel.

j. Scuppers installed in the building parapet to ensure water can not buildup and overload roof members.

k. The building foundations and below grade walls were constructed with water stops to prevent the intrusion of ground water.

Included within the Auxiliary Building license renewal evaluation boundary are the load bearing members of the Auxiliary Building and spent fuel pool bridge cranes (NUREG-0612 cranes). The non-concrete elements of the spent fuel pool (pool and transfer canal liners) along with the new and spent fuel storage racks, spent fuel pool boraflex and borated stainless steel neutron absorbers are evaluated within the Spent Fuel Pool Cooling system.

The Auxiliary Building contains racks, panels, electrical enclosures, equipment supports and fire doors, penetration barriers and seals (in addition to those noted above). Those equipment sets receive a separate commodity group evaluation independent of the building evaluation. Building interior floor drains are evaluated within the Waste Disposal system and the non-building elements of the cranes are evaluated in the Cranes, Hoists and Lifting devices evaluation or the Fuel Handling Equipment review as appropriate.

System Function Listing

In addition to the System Functions described above, the Auxiliary Building also contains components which support additional functions (associated design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code R	Cri 1	Cri 2	Cri 3				
STRUCTURALLY SUPPORT OR HOUSE SAFETY CLASS 1, 2, 3 COMPONENTS			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Auxiliary Building perform this primary design system function.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Auxiliary Building perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Auxiliary Building that perform special capability class functions are tracked under the Criterion 2 code (Y) and Criterion 3 codes (Z1 through Z5).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Auxiliary Building perform this associated design system function.

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Auxiliary Building perform this associated design system function. Components within the Auxiliary Building resist the effects of high energy line breaks, provide internal and external flood protection, and mitigate the effects of tornados. Structural elements in the Auxiliary Building protect the Spent Fuel Pool from the effects of block wall failures.

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Auxiliary Building perform this associated design system function.

UFSAR Reference

Additional Auxiliary Building details are provided in Section 3.6.2.5.1.8, Section 3.8.4.1.9, Section 3.4.1.1.3, Section 1.2.3.3, Section 3.3.2, and Section 3.8.4.1.1 of the UFSAR.

License Renewal Drawings

The license renewal drawings for Auxiliary Building the are listed below:

Site Plot

Components/Commodities Subject to an AMR

The component groups for the Auxiliary Building that require aging management review are indicated in Table 2.4.2-1 along with each component group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.4.2-1 Auxiliary Building

Component Group	Passive Function	Aging Management Reference
AAD86	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16)
AB-ARCH-EXT AB-ARCH-EXT includes the non-load bearing building elements not relied upon in the safety analysis which provide normal habitability control and weather proofing, e.g., building siding, built up roof systems, windows, etc.	SHELTER/PROTECT EQUIPMENT	Table 3.6-2 Line Number (2)
AB-BLOCK-EXT This generic asset includes all masonry block walls of the Auxiliary Building exposed to the weather. Mortar is included in this asset evaluation.	FLOOD BARRIER SHELTER/PROTECT EQUIPMENT	Table 3.6-1 Line Number (20) This applies to both passive functions.

Table 2.4.2-1 Auxiliary Building

Component Group	Passive Function	Aging Management Reference
AB-BLOCK-INT This generic asset includes all masonry block walls of the Auxiliary Building protected from the weather. The west end of the auxiliary building superstructure is connected with a portion of the service building and on the northwest with the intermediate building. The interior walls separating the buildings are block. Mortar is included in this asset evaluation.	FIRE BARRIER	Table 3.6-1 Line Number (20)

Table 2.4.2-1 Auxiliary Building

Component Group	Passive Function	Aging Management Reference
AB-C-BUR This generic asset includes all concrete in the Auxiliary Building that is in contact with the soil and groundwater. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are evaluated in the Auxiliary Building threaded fasteners and the exposed faces of plates and structural members are evaluated in Auxiliary Building structural steel. This generic asset includes evaluation of elastomer sealing material used below grade in the Auxiliary Building. Water stops are embedded in selected below grade construction joints. Also included in this evaluation is post-construction urethane foam resin injected into seams and cracks to prevent ground water intrusion.	FLOOD BARRIER STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (17) Table 3.6-1 Line Number (21) Table 3.6-1 Line Number (22) Table 3.6-1 Line Number (23) These apply to all three passive functions.

Table 2.4.2-1 Auxiliary Building

Component Group	Passive Function	Aging Management Reference
AB-C-EXT This generic asset includes all concrete in the Auxiliary Building that is exposed to the weather. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are included in the Auxiliary Building threaded fasteners and the exposed faces of plates and structural members are included in Auxiliary Building structural steel. Also included in this evaluation is post-construction urethane foam resin injected into seams and cracks to prevent ground water intrusion.	FLOOD BARRIER STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (23) These apply to all three passive functions.

Table 2.4.2-1 Auxiliary Building

Component Group	Passive Function	Aging Management Reference
AB-C-INT This generic asset includes all concrete in the Auxiliary Building that is protected from the weather including the Spent Fuel Pool. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. Grout, used under column base plates is included in this evaluation. The exposed portion of structural anchor bolts are evaluated in the Auxiliary Building threaded fasteners and the exposed faces of plates and structural members are evaluated in Auxiliary Building structural steel.	FIRE BARRIER FLOOD BARRIER STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (23) These apply to all four passive functions.

Table 2.4.2-1 Auxiliary Building

Component Group	Passive Function	Aging Management Reference
AB-ELAST-INT This generic asset includes elastomer sealing material used in the Auxiliary Building that is protected from the weather. There is sealing material in between the pipe chase and RHR suction piping where it enters the RHR pit to prevent water spilling from the basement floor into the pit, sealing material between the Intermediate Building/Auxiliary Building, and Auxiliary Building/Containment building rattle gaps to prevent external flooding of the Auxiliary Building.	FLOOD BARRIER	Table 3.6-2 Line Number (5)
AB-FAST(CS)-EXT This generic asset includes the exposed portion of carbon steel threaded fasteners for the Auxiliary Building that are exposed to the weather.	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16)

Table 2.4.2-1 Auxiliary Building

Component Group	Passive Function	Aging Management Reference
AB-FAST(CS)-INT This generic asset includes the exposed portion of carbon steel threaded fasteners in the Auxiliary Building that are protected from the weather.	PIPE WHIP RESTRAINT SHELTER/PROTECT EQUIPMENT STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (27) These apply to all four passive functions.
AB-SS(CS)-EXT This generic asset includes all carbon structural steel in the Auxiliary Building frame that is exposed to the weather.	SHELTER/PROTECT EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) This applies to both passive functions.
AB-SS(CS)-INT This generic asset includes all carbon structural steel of the Auxiliary Building that is protected from the weather. Columns, posts, beams, baseplates, bracing, crane support girders, crane rails, and the exposed faces of plates and structural members are included. This does not include carbon structural steel used as component supports.	FLOOD BARRIER HELB SHIELDING MISSILE BARRIER SHELTER/PROTECT EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (27) These apply to all five passive functions.
FLOOD BARRIER	FLOOD BARRIER	Table 3.6-2 Line Number (1)

Table 2.4.2-1 Auxiliary Building

Component Group	Passive Function	Aging Management Reference
FLOOD BARRIER - SEAL Bladder for flood barrier.	FLOOD BARRIER	Table 3.6-2 Line Number (5)
TANK Local air flask dedicated for inflating flood barrier pneumatic seals.	FLOOD BARRIER PRESSURE BOUNDARY	Table 3.6-1 Line Number (16) This applies to both passive functions.

2.4.2.2 Intermediate Building

Description

The Intermediate Building is a Seismic Category I multi-story steel frame structure measuring approximately 136-ft. by 141-ft. The building includes a facade structure on each side. The Intermediate Building surrounds the Containment Building to the west and north and joins the Service Building, Turbine Building and Auxiliary Building. It is divided into two sections called the hot side (restricted area access) and the cold side. The Intermediate Building houses and supports safety related equipment. Portions of the structure act as fire barriers. Structural elements within the building, along with the ability to open selected door, are factors considered for heat removal during Station Blackout events. Additionally, the building contains non-safety elements whose failure can affect a safety function (portions of the structure are designed to resist high-energy line breaks and flooding).

The Intermediate Building is part of a complex of interconnected buildings surrounding, but structurally independent of, the Containment Building. These buildings are interconnected as follows: The Seismic Category I Auxiliary Building is contiguous with the nonseismic Service Building on the west side. The Seismic Category I Intermediate Building adjoins the seismically analyzed Turbine Building to the north, and the Auxiliary Building to the south. The Turbine Building adjoins the Seismic Category I Diesel Generator Building to the north and the Seismic Category I Control Building to the south. The facade, a cosmetic rectangular structure that encloses the Containment Building, has all four sides partly or totally in common with the Auxiliary and Intermediate buildings. The Auxiliary Building adjoins the Seismic Category I Standby Auxiliary Feedwater Building on the south. In the original building analysis, each Seismic Category I structure was treated independently. During the Systematic Evaluation Program (SEP) plant evaluation it was found that the interconnected nature of the buildings was an important feature, especially in view of the lack of detailed original seismic design information. Therefore, both Seismic Category I and nonseismic category buildings were included in a complicated three-dimensional structural system reanalysis model. As part of this effort, the interconnected Turbine Building was determined to be capable of withstanding safe shutdown earthquake forces. Based on the SEP review, audits, and plant inspections, the NRC safety evaluation reports concluded that the evaluation and resolution of SEP Topics III-2, Wind and Tornado Loadings; III-4.A, Tornado Missiles; III-6, Seismic Design Considerations; and III-7.B, Load Combinations, were acceptable. The NRC also concluded that the RG&E analysis and implementation of the Structural Upgrade Program were acceptable.

The Intermediate Building is located on the north and west sides of the Containment Building, and is founded on rock. The west end has a retaining wall where the floor at elevation 253 ft 6 in. is supported. The bottom of the retaining wall footing is at elevation 233 ft 6 in. Rock elevation in this area is at approximately elevation 239 ft 0 in. Foundations for interior columns are on individual column footings and embedded a minimum of 2 ft in solid rock. The building, which also encloses the cylindrical Containment Building, is north of the Auxiliary Building and is connected to the part of the Auxiliary Building that is under the high roof.

The building is a 136-ft 7-in. by 140-ft 11-in. steel frame structure with facade structures on each side. The facade structures are steel frame bracing systems covered with siding. The concrete basement floor slab (elevation 253.5 ft) is supported by a set of 2-ft 10-in. square concrete columns and a concrete retaining wall on the west side. The columns have individual concrete footings embedded in the rock foundation. The top elevations of the footings vary from 238 ft to 236.5 ft.

In the north part of the building, there are three floors at elevations 278.33 ft, 298.33 ft, and 315.33 ft, and a high roof at elevation 335.5 ft. In the south part of the building there are two floors at elevations 271 ft and 293 ft, and the low roof at elevation 318 ft. All floors are made of composite steel girders and 5-in. thick concrete slabs. Built around the circular Containment Building, the floors extend completely through the west side of the Intermediate Building, a major portion of the north side and a small portion of the south side. There are no floors on the east side. The roofs are supported by steel roof girders. The floors and roofs are also supported vertically on a set of interior steel columns which are continuous from the basement floor to the roof. Concrete block walls surround the floor space between the basement floor and the roofs. The top of the four facade structures is at elevation 387 ft. There is no roof at the top, only a horizontal truss connecting the four sides to provide out-of-plane strength. One special characteristic of the west facade is that the horizontal floor or roof girders are connected not to the bracing joints but somewhere between joints. In such a design, the columns must transform significant shears and moments when the structure is subject to lateral loads.

Cosmetic siding is used on the façade portion of the structure. The roof of the Intermediate Building has been provided with scuppers designed to ensure that any rainwater, resulting from a design-basis storm, would not accumulate on the roof and cause damage. The scuppers are located so that their outflow will not damage any surrounding plant structures. The roofing and exterior walls provide weather resistance and allow habitability control but are not designed to be tornado missile resistant.

The major structures of Ginna Station have experienced no visible evidence of settlement since the construction of the station. (During the SEP and evaluation of Topic II-4.F, Settlement of Foundations and Buried Equipment, the NRC concluded that the settlement of foundations and buried equipment is not a safety concern for Ginna Station.)

In addition to structural and load bearing elements, the Intermediate Building contains features and appurtenances credited in the licensing basis and relied upon to ensure the health and safety of the public. These features include:

a. A fire resistant enclosure at the cable tunnel entrance to the Intermediate Building to provide separation between trains of safe shutdown equipment.

b. Grating versus solid manway covers in the access holes between the cold side of the Intermediate Building and the building sub-basement to provide a dewatering path in the event of a line break.

c. Jet impingement shielding on the floor under the main steam header.

d. Missile shields to protect vital cable trays from possible Turbine Driven Auxiliary Feedwater (TDAFW) pump turbine missiles.

e. Jet impingement shields affixed to the containment wall to provide separation between vital instruments.

f. Jet impingement and missile shielding around the solenoid valves for the main steam isolation valves.

g. Sealing material between the Intermediate Building/Containment Building rattle gaps to prevent cross communication of flood volumes.

h. Restraining devices installed at the Intermediate Building/Turbine Building interface (above the main steam power operated relief valves) to ensure a block wall failure will not damage the valves.

i. Scuppers are installed in the building roof to ensure water can not buildup and overload roof members.

j. The building foundations and below grade walls were constructed with water stops to prevent the intrusion of ground water.

k. An oil containment dike around the TDAFW lube oil tank to minimize the fire risk from any spilled oil.

l. Selected structural steel building members are coated with a protective material to resist the effects of fires.

m. Specially constructed radiation shielding enclosures to minimize personnel dose during post accident sampling evolutions.

n. A Standby Auxiliary Feedwater System (SAFW) was added to further improve steam generator feedwater reliability and specifically to substitute for the preferred auxiliary feedwater in the low probability that preferred auxiliary feedwater pumps are damaged due to nearby high-energy pipe breaks within the Intermediate Building.

The Intermediate Building contains racks, panels, electrical enclosures, equipment supports and fire doors, penetration barriers and seals (in addition to those noted above). These equipment sets receive a separate commodity group evaluation independent of the building evaluation. The restricted access portion of the buildings interior floor drains are evaluated within the Waste Disposal system, the unrestricted access portion is evaluated in the Treated Water system.

System Function Listing

In addition to the System Functions described above, the Intermediate Building also contains components which support additional functions (associated design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code R	Cri 1	Cri 2	Cri 3				
STRUCTURALLY SUPPORT OR HOUSE SAFETY CLASS 1, 2, 3 COMPONENTS			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Intermediate Building perform this primary design system function.

Code S	Cri 1	Cri 2	Cri 3				
SPECIAL CAPABILITY CLASS FUNCTIONS			FP	EQ	PTS	AT	SB

Comment: Components within the Intermediate Building perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Intermediate Building that perform special capability class functions are tracked under the Criterion 2 code (Y) and Criterion 3 codes (Z1 through Z5).

Code T	Cri 1	Cri 2	Cri 3				
NON-NUCLEAR SAFETY CLASS FUNCTIONS			FP	EQ	PTS	AT	SB

Comment: Components within the Intermediate Building perform this associated design system function.

Code Y	Cri 1	Cri 2	Cri 3				
LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Intermediate Building perform this associated design system function. Components within the Intermediate Building resist the effects of high energy line breaks and provide internal and external flood protection. Structural elements in the Intermediate Building protect the Main Steam Atmospheric Relief Valves from the effects of block wall failures.

Code Z1	Cri 1	Cri 2	Cri 3				
LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Intermediate Building perform this associated design system function.

Code Z5	Cri 1	Cri 2	Cri 3				
LICENSE RENEWAL CRITERION 3 (SBO) - CONTAINS SSCS RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION’S REGULATIONS FOR STATION BLACKOUT (10 CFR 50.63)			FP	EQ	PTS	AT	SB
							X

Comment: Components within the Intermediate Building perform this associated design system function.

UFSAR Reference

Additional Intermediate Building details are provided in Section 3.6.2.4.8.1, Section 3.8.2, Section 3.6.2.1, Section 3.8.4.1.9, Section 3.8.4.1, Section 3.8.4.1.4, Section 1.2.3.4, Section 3.6.2.5.1.2, Section 8.1.4.5.2, and Section 12.3.2.2.6 of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Intermediate Building are listed below:

Site Plot

Components/Commodities Subject to an AMR

The component groups for the Intermediate Building that require aging management review are indicated in Table 2.4.2-2 along with each component group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.4.2-2 Intermediate Building

Component Group	Passive Function	Aging Management Reference
IB-ARCH-EXT IB-ARCH-EXT includes the non-load bearing building elements not relied upon in the safety analysis which provide normal habitability control and weather proofing, e.g., building siding, built up roof systems, windows, etc.	SHELTER/PROTECT EQUIPMENT	Table 3.6-2 Line Number (2)
IB-BLOCK-EXT This generic asset includes all masonry block walls of the Intermediate Building exposed to the weather. Mortar is included in this asset evaluation.	HEAT SINK SHELTER/PROTECT EQUIPMENT	Table 3.6-1 Line Number (20) This applies to both passive functions.

Table 2.4.2-2 Intermediate Building

Component Group	Passive Function	Aging Management Reference
IB-BLOCK-INT This generic asset includes all masonry block walls of the Intermediate Building that are protected from the weather. The interior walls separating the buildings rooms are block. Mortar is included in this asset evaluation.	FIRE BARRIER HEAT SINK	Table 3.6-1 Line Number (20) This applies to both passive functions.

Table 2.4.2-2 Intermediate Building

Component Group	Passive Function	Aging Management Reference
IB-C-BUR This generic asset includes all concrete in the Intermediate Building that is in contact with the soil and groundwater. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are evaluated in the Intermediate Building threaded fasteners and the exposed faces of plates and structural members are evaluated in Intermediate Building structural steel. This generic asset includes evaluation of elastomer sealing material used below grade in the Intermediate Building. Water stops are embedded in selected below grade construction joints. Also included in this evaluation is post-construction urethane foam resin injected into seams and cracks to prevent ground water intrusion.	FLOOD BARRIER STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (17) Table 3.6-1 Line Number (21) Table 3.6-1 Line Number (22) Table 3.6-1 Line Number (23) These apply to both passive functions.

Table 2.4.2-2 Intermediate Building

Component Group	Passive Function	Aging Management Reference
IB-C-EXT This generic asset includes all concrete in the Intermediate Building that is exposed to the weather. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are included in the Intermediate Building threaded fasteners and the exposed faces of plates and structural members are included in Intermediate Building structural steel.	HEAT SINK STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (23) These apply to both passive functions.

Table 2.4.2-2 Intermediate Building

Component Group	Passive Function	Aging Management Reference
IB-C-INT This generic asset includes all concrete in the Intermediate Building that is protected from the weather. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. Grout, used under column base plates is included in this evaluation. The exposed portion of structural anchor bolts are evaluated in the Intermediate Building threaded fasteners and the exposed faces of plates and structural members are evaluated in Intermediate Building structural steel.	FIRE BARRIER FLOOD BARRIER HEAT SINK MISSILE BARRIER STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (23) These apply to all five passive functions.
IB-ELAST-INT This generic asset includes elastomer sealing material used in the Intermediate Building that is protected from the weather. Elastomers are used as a sealing material in the seismic gap between Containment and Intermediate Building.	FLOOD BARRIER	Table 3.6-2 Line Number (5)

Table 2.4.2-2 Intermediate Building

Component Group	Passive Function	Aging Management Reference
IB-FAST(CS)-EXT This generic asset includes the exposed portion of carbon steel threaded fasteners for the Intermediate Building that are exposed to the weather.	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16)
IB-FAST(CS)-INT This generic asset includes the exposed portion of carbon steel threaded fasteners in the Intermediate Building that are protected from the weather.	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (27)
IB-LEAD-INT This generic asset represents the shielded enclosure constructed over the primary sample Containment isolation valves in the Intermediate Building hot side. This generic asset includes leaded glass and lead bricks. The lead bricks are sandwiched between galvanized steel sheets which are evaluated under the IB-SS(CS)-INT asset.	STRUCTURAL SUPPORT NSR EQUIPMENT	Table 3.6-2 Line Number (8)

Table 2.4.2-2 Intermediate Building

Component Group	Passive Function	Aging Management Reference
IB-SS(CS)-EXT This generic asset includes all carbon structural steel in the Intermediate Building frame that is exposed to the weather.	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16)
IB-SS(CS)-INT This generic asset includes all carbon structural steel of the Intermediate Building that is protected from the weather. Columns, posts, beams, baseplates, bracing, and the exposed faces of plates and structural members are included. Structural steel is used as missile barriers, jet impingement shields and an oil containment dike around the Turbine Driven Auxiliary Feed Pump oil system. This does not include carbon structural steel used as component supports.	HEAT SINK HELB SHIELDING MISSILE BARRIER SHELTER/PROTECT EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (27) These apply to all five passive functions.

2.4.2.3 Turbine Building

Description

Though the Turbine Building was not originally designed to be Seismic Category I, it is part of complex of interconnected buildings and is connected to Seismic Category I structures. As such, the Turbine Building is considered a structure that supports nuclear safety related equipment. Subsequently, the Turbine Building has been modified and re-evaluation found the building capable of withstanding safe shutdown earthquake forces. Portions of structure also act as fire barriers and oil confinement systems. Additionally, the building contains non-safety elements whose failure can affect a safety function (portions of the structure are designed to resist, and protect equipment from, high-energy line breaks, flooding, and tornadoes).

The Turbine Building is part of a complex of interconnected buildings surrounding, but structurally independent of, the Containment Building. These buildings are interconnected as follows: The Seismic Category I Auxiliary Building is contiguous with the nonseismic Service Building on the west side. The Seismic Category I Intermediate Building adjoins the seismically analyzed Turbine Building to the north, and the Auxiliary Building to the south. The Turbine Building adjoins the Seismic Category I Diesel Generator Building to the north and the Seismic Category I Control Building to the south. The facade, a cosmetic rectangular structure that encloses the Containment Building, has all four sides partly or totally in common with the Auxiliary and Intermediate buildings. The Auxiliary Building adjoins the Seismic Category I Standby Auxiliary Feedwater Building on the south. In the original building analysis, each Seismic Category I structure was treated independently. During the Systematic Evaluation Program (SEP) plant evaluation it was found that the interconnected nature of the buildings was an important feature, especially in view of the lack of detailed original seismic design information. Therefore, both Seismic Category I and nonseismic category buildings were included in a complicated three-dimensional structural system reanalysis model. As part of this effort, the interconnected Turbine Building was determined to be capable of withstanding safe shutdown earthquake forces. Based on the SEP review, audits, and plant inspections, the NRC safety evaluation reports concluded that the evaluation and resolution of SEP Topics III-2, Wind and Tornado Loadings; III-4.A, Tornado Missiles; III-6, Seismic Design Considerations; and III-7.B, Load Combinations, were acceptable. The NRC also concluded that the RG&E analysis and implementation of the Structural Upgrade Program were acceptable.

The Turbine Building is a 257.5-ft by 124.5-ft rectangular building on the north side of the building complex. The Turbine Building foundation is a concrete mat supported by compacted fill material. In addition to the concrete basement, it has two concrete floors. The building roof includes a roof truss structure composed of top and bottom chords connected by vertical bracing. The roof and floors are supported by steel framing and bracing systems on all four sides of the building. The floors are also supported by additional interior framing at various locations under the floors. Part of the south wall frame also serves as the north wall of the Intermediate Building. The north facade structure is actually on the top of the south frame of the Turbine Building. The west frame is the continuation of the west facade structure of the Intermediate Building. This west frame is also part of the Service Building. Except between buildings, the walls of the Turbine Building have insulated siding. The Turbine Building parapets have been provided with scuppers designed to ensure that any rainwater, resulting from a design-basis storm, would not accumulate on the roof and cause damage. The scuppers are located so that their outflow will not damage any surrounding plant structures. The roofing and siding provide weather resistance and allow habitability control but are not designed to be tornado missile resistant.

Inside the building and parallel to the south and north frames, there is an interior frame system supporting the crane from the basement elevation. The crane frame is designed like the exterior frame system with vertical columns, horizontal beams, and cross bracing bolted to columns. Each interior column is welded to the corresponding exterior column at the joints and mid-points of columns by a series of girder connections. The south frame of the Turbine Building is designed like the west facade structure of the Intermediate Building; that is, horizontal floor girders are connected to columns somewhere between joints. For the purposes of License Renewal review, the walls separating the Turbine Building from the Intermediate Building, the Diesel Generator Building, the Service Building, the Control Building and the All-Volatile-Treatment (AVT) building will be evaluated as part of those structures as appropriate. Additionally, the main feedwater pumps are surrounded by a block wall enclosure. (This enclosure is not credited for fire protection or high energy line break mitigation.)

The major structures of Ginna Station have experienced no visible evidence of settlement since the construction of the station. (During the SEP and evaluation of Topic II-4.F, Settlement of Foundations and Buried Equipment, the NRC concluded that the settlement of foundations and buried equipment is not a safety concern for Ginna Station.)

In addition to structural and load bearing elements, the Turbine Building contains features and appurtenances credited in the licensing basis and relied upon to ensure the health and safety of the public. These features include:

a. The Turbine Building includes a barrier installed around the turbine lube-oil reservoir area to contain possible oil spillage.

b. Selected structural steel building members are coated with a protective material to resist the effects of fires.

c. Some building structural members also interface with the pressure-shielding steel diaphragm walls that were installed at the Control Building-Turbine Building wall and at the diesel building-Turbine Building wall to ensure continued operability of safety-related equipment following a postulated high-energy pipe break in the Turbine Building.

d. The turbine seal oil unit is enclosed in a fire resistant shelter.

e. The building foundations and below grade walls were constructed with water stops to prevent the intrusion of ground water.

The Turbine Building contains racks, panels, electrical enclosures, equipment supports and fire doors, penetration barriers, seals and coatings. These equipment sets receive a separate commodity group evaluation independent of the building evaluation.

System Function Listing

In addition to the System Functions described above, the Turbine Building also contains components which support additional functions (associated design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code R	Cri 1	Cri 2	Cri 3				
STRUCTURALLY SUPPORT OR HOUSE SAFETY CLASS 1, 2, 3 COMPONENTS			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Turbine Building perform this primary design system function. Even though the Turbine Building was not originally designed to be Seismic Category I, it is part of complex of interconnected buildings surrounding the containment building and is connected to Seismic Category I structures. As such, the building is considered a structure that supports nuclear safety related equipment. Because of its importance, as part of the Structural Upgrade Program, the building was modified to add additional bracing.

Code S	Cri 1	Cri 2	Cri 3				
SPECIAL CAPABILITY CLASS FUNCTIONS			FP	EQ	PTS	AT	SB

Comment: Components within the Turbine Building perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Turbine Building that perform special capability class functions are tracked under the Criterion 2 code (Y) and Criterion 3 codes (Z1 through Z5).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Turbine Building perform this associated design system function.

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Turbine Building perform this associated design system function. Components within the Turbine Building resist the effects of high energy line breaks.

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Turbine Building perform this associated design system function.

UFSAR Reference

Additional Turbine Building details are provided in Section 1.2.3.5, Section 3.8.4.1.7, Section 3.7.1.5, Section 3.8.4.1, Section 3.8.4.4.2.3, and Section 3.8.4.1.9 of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Turbine Building are listed below:

Site Plot

Components/Commodities Subject to an AMR

The component groups for the Turbine Building that require aging management review are indicated in Table 2.4.2-3 along with each component group’s passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.4.2-3 Turbine Building

Component Group	Passive Function	Aging Management Reference
TB-ARCH-EXT TB-ARCH-EXT includes the non-load bearing building elements not relied upon in the safety analysis which provide normal habitability control and weather proofing, e.g., building siding, built up roof systems, windows, etc.	SHELTER/PROTECT EQUIPMENT	Table 3.6-2 Line Number (2)

Table 2.4.2-3 Turbine Building

Component Group	Passive Function	Aging Management Reference
TB-C-BUR This generic asset includes all concrete in the Turbine Building that is in contact with the soil and groundwater. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are evaluated in the Turbine Building threaded fasteners and the exposed faces of plates and structural members are evaluated in Turbine Building structural steel. This generic asset includes evaluation of elastomer sealing material used below grade in the Turbine Building. Water stops are embedded in selected below grade construction joints. Also included in this evaluation is post-construction urethane foam resin injected into seams and cracks to prevent ground water intrusion.	STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (17) Table 3.6-1 Line Number (21) Table 3.6-1 Line Number (22) Table 3.6-1 Line Number (23) These apply to both passive functions.

Table 2.4.2-3 Turbine Building

Component Group	Passive Function	Aging Management Reference
TB-C-EXT This generic asset includes concrete exposed to the weather that acts as part of the building siding system. TB-C-EXT includes the non-load bearing building elements not relied upon in the safety analysis which provide normal habitability control and weather proofing, e.g., building siding, built up roof systems, windows, etc.	SHELTER/PROTECT EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (23)

Table 2.4.2-3 Turbine Building

Component Group	Passive Function	Aging Management Reference
TB-C-INT This generic asset includes all concrete in the Turbine Building that is protected from the weather. This includes oil confinement curbing around the seal oil unit. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. Grout, used under column base plates is included in this evaluation. The exposed portion of structural anchor bolts are evaluated in the Turbine Building threaded fasteners and the exposed faces of plates and structural members are evaluated in Turbine Building structural steel.	STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) This applies to both passive functions.
TB-FAST(CS)-EXT This generic asset includes the exposed portion of carbon steel threaded fasteners for the Turbine Building that are exposed to the weather.	SHELTER/PROTECT EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) This applies to both passive functions.

Table 2.4.2-3 Turbine Building

Component Group	Passive Function	Aging Management Reference
TB-FAST(CS)-INT This generic asset includes the exposed portion of carbon steel threaded fasteners in the Turbine Building that are protected from the weather.	STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) This applies to both passive functions.
TB-SS(CS)-EXT This generic asset includes all carbon structural steel in the Turbine Building frame that is exposed to weather.	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16)
TB-SS(CS)-INT This generic asset includes all carbon structural steel of the Turbine Building that is protected from the weather. Columns, posts, beams, baseplates, bracing, crane support girders, crane rails, and the exposed faces of plates and structural members are included. This does not include carbon structural steel used as component supports.	STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) This applies to both passive functions.

2.4.2.4 Diesel Building

Description

The Diesel Building is a Seismic Category I structure that houses and supports nuclear safety related equipment. The Diesel Building also protects connections for an alternate diesel cooling water source should the service water system be disabled. The structure also acts as a fire barrier and contains non-safety elements whose failure can affect a safety function (portions of the structure are designed to resist, and protect equipment from, high-energy line breaks, flooding, and tornadoes).

The Diesel Building is part of a complex of interconnected buildings surrounding, but structurally independent of, the Containment Building. These buildings are interconnected as follows: The Seismic Category I Auxiliary Building is contiguous with the nonseismic Service Building on the west side. The Seismic Category I Intermediate Building adjoins the seismically analyzed Turbine Building to the north, and the Auxiliary Building to the south. The Turbine Building adjoins the Seismic Category I Diesel Generator Building to the north and the Seismic Category I Control Building to the south. The facade, a cosmetic rectangular structure that encloses the Containment Building, has all four sides partly or totally in common with the Auxiliary and Intermediate buildings. The Auxiliary Building adjoins the Seismic Category I Standby Auxiliary Feedwater Building on the south. In the original building analysis, each Seismic Category I structure was treated independently. During the Systematic Evaluation Program (SEP) plant evaluation it was found that the interconnected nature of the buildings was an important feature, especially in view of the lack of detailed original seismic design information. Therefore, both Seismic Category I and nonseismic category buildings were included in a complicated three-dimensional structural system reanalysis model. As part of this effort, the interconnected Turbine Building was determined to be capable of withstanding safe shutdown earthquake forces. Based on the SEP review, audits, and plant inspections, the NRC safety evaluation reports concluded that the evaluation and resolution of SEP Topics III-2, Wind and Tornado Loadings; III-4.A, Tornado Missiles; III-6, Seismic Design Considerations; and III-7.B, Load Combinations, were acceptable. The NRC also concluded that the RG&E analysis and implementation of the Structural Upgrade Program were acceptable.

The Diesel Building adjoins the Turbine Building on the east end of the north wall, opposite the Control Building. The Diesel Building is a one-story reinforced-concrete structure divided into two rooms, each with a cable vaults underneath the floor. The south wall, which is common with the turbine building, is reinforced to be a pressurization wall to protect the areas adjacent to the turbine building from the effects of high-energy line breaks. The foundations of the diesel generator buildings were excavated to the surface of bedrock. Lean concrete or compacted backfill was placed on the rock surface to a depth whereby the elevation of the top of the fill material was coincident with the elevation of the bottom of the concrete foundation.

The Diesel Building was modified as part of the Structural Upgrade Program to withstand tornado winds and missiles, external flooding, seismic loads, and extreme snow loads. A new reinforced-concrete north wall was constructed 4 ft north of the existing north wall. Reinforced-concrete wing walls were constructed that extended the east and west walls to meet the new north wall, enclosing the space between the existing and new north wall. The new wall includes missile-resistant watertight equipment and personnel doors. A new reinforced-concrete slab roof with a reinforced-concrete parapet was constructed covering the entire Diesel Building. The existing north wall and portions of the existing roof were left in place. The building as modified was designed to remain undamaged during and after an operating basis earthquake and remain functional during and after a safe shutdown earthquake.

The major structures of Ginna Station have experienced no visible evidence of settlement since the construction of the station. (During the SEP and evaluation of Topic II-4.F, Settlement of Foundations and Buried Equipment, the NRC concluded that the settlement of foundations and buried equipment is not a safety concern for Ginna Station.)

In addition to structural and load bearing elements, the Diesel Building contains features and appurtenances credited in the licensing basis and relied upon to ensure the health and safety of the public. These features include:

a. The B diesel room vault contains a fire resistant enclosure to provide electrical train separation.

b. The building foundations and below grade walls were constructed with water stops to prevent the intrusion of ground water.

c. The common wall between the Diesel Building and Turbine Building is reinforced with heavy sheet piling and stiffeners to form a pressurization wall to resist the effects of a high energy line break in the Turbine Building.

d. The Diesel Building exterior has been modified during the Structural Upgrade Program to withstand the effects of tornado wind, tornado differential pressure, tornado missiles, and flooding of Deer Creek.

e. Scuppers are installed in the building roof to ensure water can not buildup and overload roof members.

The Diesel Building contains sump pumps to remove any ground water that may leak into the vaults. This equipment is not considered part of the structure and is evaluated within the Treated Water System. The building also contains racks, panels, electrical enclosures, equipment supports and fire doors, penetration barriers and seals. These equipment sets receive a separate commodity group evaluation independent of the building evaluation.

System Function Listing

In addition to the System Functions described above, the Diesel Building also contains components which support additional functions (associated design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code R STRUCTURALLY SUPPORT OR HOUSE SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Diesel Generator Building perform this primary design system function.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Diesel Generator Building perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Diesel Generator Building that perform special capability class functions are tracked under the Criterion 3 codes (Z1 through Z5).

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Diesel Generator Building perform this associated design system function. The Diesel Generator Building contains structural elements designed to resist the effects of high energy line breaks, tornadoes, and flooding events.

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Diesel Generator Building perform this associated design system function.

UFSAR Reference

Additional Diesel Building details are provided in Section 3.8.5, Section 3.8.4.4, Section 3.8.4.1.3, and Section 3.8.4.1.9 of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Diesel Building are listed below:

Site Plot

Components/Commodities Subject to an AMR

The component groups for the Diesel Building that require aging management review are indicated in Table 2.4.2-4 along with each component group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.4.2-4 Diesel Building

Component Group	Passive Function	Aging Management Reference
DB-ARCH-EXT DB-ARCH-EXT includes the non-load bearing building elements not relied upon in the safety analysis which provide normal habitability control and weather proofing, e.g., building siding, built up roof systems, windows, etc.	SHELTER/PROTECT EQUIPMENT	Table 3.6-2 Line Number (2)

Table 2.4.2-4 Diesel Building

Component Group	Passive Function	Aging Management Reference
DB-C-BUR This generic asset includes all concrete in the Diesel Building that is in contact with the soil and groundwater. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are evaluated in the Diesel Building threaded fasteners and the exposed faces of plates and structural members are evaluated in Diesel Building structural steel. This generic asset includes evaluation of elastomer sealing material used below grade in the Diesel Generator Building. Water stops are embedded in selected below grade construction joints. Also included in this evaluation is post-construction urethane foam resin injected into seams and cracks to prevent ground water intrusion.	SHELTER/PROTECT EQUIPMENT SHELTER/PROTECT EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (17) Table 3.6-1 Line Number (21) Table 3.6-1 Line Number (22) Table 3.6-1 Line Number (23) These apply to all three passive functions.

Table 2.4.2-4 Diesel Building

Component Group	Passive Function	Aging Management Reference
DB-C-EXT This generic asset includes all concrete in the Diesel Generator Building that is exposed to the weather. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are included in the Diesel Generator Building threaded fasteners and the exposed faces of plates and structural members are included in Diesel Generator Building structural steel.	FLOOD BARRIER MISSILE BARRIER SHELTER/PROTECT EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (23) These apply to all four passive functions.

Table 2.4.2-4 Diesel Building

Component Group	Passive Function	Aging Management Reference
DB-C-INT This generic asset includes all concrete in the Diesel Generator Building that is protected from the weather. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. Grout, used under column base plates is included in this evaluation. The exposed portion of structural anchor bolts are evaluated in the Diesel Generator Building threaded fasteners and the exposed faces of plates and structural members are evaluated in Diesel Generator Building structural steel.	FIRE BARRIER FLOOD BARRIER STRUCTURAL SUPPORT SR EQUIPMENT SHELTER/PROTECT EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (23) These apply to all four passive functions.
DB-ELAST-INT This generic asset includes all elastomer sealing material in the Diesel Generator Building that is protected from the weather. Elastomers are used as door seals.	FLOOD BARRIER	Table 3.6-2 Line Number (5)

Table 2.4.2-4 Diesel Building

Component Group	Passive Function	Aging Management Reference
DB-FAST(CS)-EXT This generic asset includes the exposed portion of carbon steel threaded fasteners for the Diesel Generator Building that are exposed to the weather.	STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) This applies to both passive functions.
DB-FAST(CS)-INT This generic asset includes the exposed portion of carbon steel threaded fasteners in the Diesel Generator Building that are protected from the weather.	STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) This applies to both passive functions.
DB-FAST(HSLAS)-INT This generic asset includes the exposed portion of high strength carbon steel threaded fasteners in the Diesel Generator Building that is protected from the weather.	FLOOD BARRIER HELB SHIELDING MISSILE BARRIER	Table 3.6-2 Line Number (12) This applies to all three passive functions.
DB-SS(CS)-EXT This generic asset includes all structural carbon steel for the Diesel Generator Building (e.g., missile barriers) that is exposed to the weather.	MISSILE BARRIER STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) This applies to both passive functions.

Table 2.4.2-4 Diesel Building

Component Group	Passive Function	Aging Management Reference
DB-SS(CS)-INT This generic asset includes all structural carbon steel for the Diesel Generator Building (e.g., plates, beams, columns, grating, high energy line break pressurization wall, etc.) that is protected from the weather.	FIRE BARRIER FLOOD BARRIER MISSILE BARRIER STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) This applies to all five passive functions.
EXT-DOOR Carbon steel exterior doors are designed to resist tornados and floods.	FLOOD BARRIER MISSILE BARRIER SHELTER/PROTECT EQUIPMENT	Table 3.6-1 Line Number (16) This applies to all three passive functions.
INT-DOOR Carbon steel interior doors are designed to resist high energy line breaks and floods.	FLOOD BARRIER HELB SHIELDING	Table 3.6-1 Line Number (16) This applies to both passive functions.

2.4.2.5 Control Building

Description

The Control Building is a Seismic Category I three-story structure measuring approximately 41 ft. by 54 ft. It is located north of the Containment Building and adjacent to the Turbine Building. The Control Building houses and supports the safety related control room, vital battery rooms, the relay room, and the mechanical equipment room. These rooms provide power and controls for the engineered safety features equipment and most other equipment required for plant operation. The control room portion of the Control Building functions in concert with the Control Room Emergency Air Treatment ventilation equipment to maintain a habitable environment for plant operators during design basis events. Portions of the structure act as fire barriers. Some structural elements of the building are credited with providing a heat sink to ensure vital equipment can function for the required coping duration of a Station Blackout. Additionally, the building contains non-safety elements whose failure can affect a safety function (portions of the structure are designed to resist, and protect equipment from, high-energy line breaks, flooding, and tornadoes).

The Control Building is part of a complex of interconnected buildings surrounding, but structurally independent of, the Containment Building. These buildings are interconnected as follows: The Seismic Category I Auxiliary Building is contiguous with the nonseismic Service Building on the west side. The Seismic Category I Intermediate Building adjoins the seismically analyzed Turbine Building to the north, and the Auxiliary Building to the south. The Turbine Building adjoins the Seismic Category I Diesel Generator Building to the north and the Seismic Category I Control Building to the south. The facade, a cosmetic rectangular structure that encloses the Containment Building, has all four sides partly or totally in common with the Auxiliary and Intermediate buildings. The Auxiliary Building adjoins the Seismic Category I Standby Auxiliary Feedwater Building on the south. In the original building analysis, each Seismic Category I structure was treated independently. During the Systematic Evaluation Program (SEP) plant evaluation it was found that the interconnected nature of the buildings was an important feature, especially in view of the lack of detailed original seismic design information. Therefore, both Seismic Category I and nonseismic category buildings were included in a complicated three-dimensional structural system reanalysis model. As part of this effort, the interconnected Turbine Building was determined to be capable of withstanding safe shutdown earthquake forces. Based on the SEP review, audits, and plant inspections, the NRC safety evaluation reports concluded that the evaluation and resolution of SEP Topics III-2, Wind and Tornado Loadings; III-4.A, Tornado Missiles; III-6, Seismic Design Considerations; and III-7.B, Load Combinations, were acceptable. The NRC also concluded that the RG&E analysis and implementation of the Structural Upgrade Program were acceptable.

The Control Building is located adjacent to the south side of the Turbine Building. The foundation of the Control Building is supported on lean concrete or compacted backfill. The foundation of the Control Building was excavated to the surface of the bedrock. The fill material was placed on the rock surface to a depth coincident with the control building foundation. The bottom elevation of the deepest portion of the foundation mat is at elevation 245 ft 4 in., with a structural slab supported at elevation 250 ft 6 in. with a thickened slab for column footings. The south and west sides have reinforced-concrete walls, and the roof is also reinforced concrete. The roof of the Control Building has a parapet provided with a scupper designed to ensure that any rainwater, resulting from a design-basis storm, would not accumulate on the roof and cause damage. The scupper is located so that its outflow will not damage any surrounding plant structures or equipment. The building siding is entirely cosmetic.

The control room floor and the relay room floor are 6-in. thick reinforced-concrete slabs supported by steel girders that are tied to Turbine Building floors at the respective elevations. The relay room east interior wall is primarily insulated siding and some concrete block. The east relay room exterior wall was installed during the Structural Upgrade Program and is designed to withstand the effects of tornado wind, tornado differential pressure, tornado missiles, and flooding of Deer Creek. The modification consisted of installing a reinforced-concrete Seismic Category I structure adjoining the east wall of the relay room. The entire north wall of the Control Building is protected from the effects of a high energy line break in the Turbine Building by a steel barrier.

In the basement of the structure are the battery rooms and the Control Building mechanical equipment room. Analysis concluded that a service water system or fire main system postulated failure in the mechanical equipment room was considered capable of flooding both battery rooms. To preclude that event the original door between the air handling room and the B battery room has been replaced by a wall and a water relief valve has been installed between the mechanical equipment room and the turbine building. The relief valve will ensure that the room can de-water sufficiently to prevent wall collapse.

The control room portion of the structure acts in conjunction with the control room ventilation system to accomplish Control Room Emergency Air Treatment (CREATS). The control room’s role is to provide a non-leak tight pressure boundary envelope to support emergency air treatment. The CREATS boundary encompasses the entire room interior boundary, including the room access doors. The false ceiling is not included in the CREATS boundary but the structural ceiling and the room's interface with the ventilation ductwork is. The false ceiling panels are non-safety equipment whose failure can effect a safety function and are seismically restrained. Additionally, the panels can be removed during station blackout events in order to allow optimum heat transfer to the structural members.

The major structures of Ginna Station have experienced no visible evidence of settlement since the construction of the station. (During the SEP and evaluation of Topic II-4.F, Settlement of Foundations and Buried Equipment, the NRC concluded that the settlement of foundations and buried equipment is not a safety concern for Ginna Station.)

In addition to structural and load bearing elements, the Control Building contains features and appurtenances credited in the licensing basis and relied upon to ensure the health and safety of the public. These features include:

a. The common wall between the Control Building and Turbine Building is reinforced with heavy sheet piling and stiffeners to form a pressurization wall to resist the effects of a high energy line break in the Turbine Building.

b. The north and east wall of the control room has 1/4-in. armor plate to resist the effects of tornado missiles and malicious acts.

c. The east wall of the relay room was been modified during the Structural Upgrade Program to withstand the effects of tornado wind, tornado differential pressure, tornado missiles, and flooding of Deer Creek.

d. Scuppers installed in the building parapet to ensure water can not buildup and overload roof members.

e. Selected below grade construction joints, seams and cracks are sealed to prevent ground water intrusion.

f. Battery room and mechanical equipment room doors at the Turbine Building entrances are elevated to preclude water intrusion into the rooms from floods.

g. A barrier exists at the cable tunnel entrance to the Control Building to provide fire area separation.

h. Selected structural steel building members are coated with a protective material to resist the effects of fires.

i. Design configuration control is strictly maintained to ensure the building structural features credited in heat sink calculations for Station Blackout are maintained.

j. The south and west above grade concrete walls provide radiation shielding for plant operators.

k. Block walls were evaluated and upgraded as necessary to ensure their continued functioning during a design basis earthquake.

l. The building foundations and below grade walls were constructed with water stops to prevent the intrusion of ground water.

The Control Building contains racks, panels, electrical enclosures, equipment supports and fire doors, penetration barriers and seals (in addition to those noted above). These equipment sets receive a separate commodity group evaluation independent of the building evaluation. Building interior floor drains are evaluated within the Treated Water system.

System Function Listing

In addition to the System Functions described above, the Control Building also contains components which support additional functions (associated design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code R STRUCTURALLY SUPPORT OR HOUSE SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Control Building perform this primary design system function.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Control Building perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Control Building that perform special capability class functions are tracked under the Criterion 3 codes (Z1 through Z5).

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Control Building perform this associated design system function. The Control Building contains structural elements designed to resist the effects of high energy line breaks, tornadoes, and flooding events.

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Control Building perform this associated design system function.

Code Z5 LICENSE RENEWAL CRITERION 3 (SBO) - CONTAINS SSCS RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR STATION BLACKOUT (10 CFR 50.63)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: Components within the Control Building perform this associated design system function.

UFSAR Reference

Additional Control Building details are provided in Section 3.8.4.1.2, Section 3.6.2.5.1.5, Section 3.8.2, Section 1.2.3.6, Section 3.8.4.1, Section 3.3.5.4.4, Section 3.7.1.5, and Section 3.8.5 of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Control Building are listed below:

Site Plot

Components/Commodities Subject to an AMR

The component groups for the that require aging management review are indicated in Table 2.4.2-5 along with each component group’s passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.4.2-5 Control Building

Component Group	Passive Function	Aging Management Reference
CB-ARCH-EXT CB-ARCH-EXT includes the non-load bearing building elements not relied upon in the safety analysis which provide normal habitability control and weather proofing, e.g., building siding, built up roof systems, windows, etc.	SHELTER/PROTECT EQUIPMENT	Table 3.6-2 Line Number (2)
CB-BLOCK-INT This generic asset includes all masonry block walls of the Control Building that are protected from the weather. The interior walls separating the buildings rooms are block. Mortar is included in this asset evaluation.	FIRE BARRIER FLOOD BARRIER	Table 3.6-1 Line Number (20) This applies to both passive functions.

Table 2.4.2-5 Control Building

Component Group	Passive Function	Aging Management Reference
CB-C-BUR This generic asset includes all concrete in the Control Building that is in contact with the soil and groundwater. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are evaluated in the Control Building threaded fasteners and the exposed faces of plates and structural members are evaluated in Control Building structural steel. This generic asset includes evaluation of elastomer sealing material used below grade in the Control Building. Water stops are embedded in selected below grade construction joints. Also included in this evaluation is post-construction urethane foam resin injected into seams and cracks to prevent ground water intrusion.	SHELTER/PROTECT EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (17) Table 3.6-1 Line Number (21) Table 3.6-1 Line Number (22) Table 3.6-1 Line Number (23) These apply to both passive functions.

Table 2.4.2-5 Control Building

Component Group	Passive Function	Aging Management Reference
CB-C-EXT This generic asset includes all concrete in the Control Building that is exposed to the weather. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are included in the Control Building threaded fasteners and the exposed faces of plates and structural members are included in Control Building structural steel.	FLOOD BARRIER HEAT SINK MISSILE BARRIER RADIATION/HEAT SHIELDING SHELTER/PROTECT EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (23) These apply to all six passive functions.

Table 2.4.2-5 Control Building

Component Group	Passive Function	Aging Management Reference
CB-C-INT This generic asset includes all concrete in the Control Building that is protected from the weather. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. Grout, used under column base plates is included in this evaluation. The exposed portion of structural anchor bolts are evaluated in the Control Building threaded fasteners and the exposed faces of plates and structural members are evaluated in Control Building structural steel.	HEAT SINK RADIATION/HEAT SHIELDING SHELTER/PROTECT EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (23) These apply to all four passive functions.
CB-ELAST-INT This generic asset includes elastomer sealing material used in the Control Building that is protected from the weather. Elastomers are used as door seals. Elastomer is also used for the dewatering valve closure seal.	FLOOD BARRIER HELB SHIELDING	Table 3.6-2 Line Number (5) This applies to both passive functions.

Table 2.4.2-5 Control Building

Component Group	Passive Function	Aging Management Reference
CB-FAST(CS)-EXT This generic asset includes the exposed portion of carbon steel threaded fasteners for the Control Building that are exposed to the weather.	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16)
CB-FAST(CS)-INT This generic asset includes the exposed portion of carbon steel threaded fasteners in the Control Building that is protected from the weather.	HELB SHIELDING MISSILE BARRIER STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) This applies to all three passive functions.
CB-FAST(HSLAS)-INT This generic asset includes the exposed portion of high strength carbon steel threaded fasteners in the Control Building that is protected from the weather.	HELB SHIELDING	Table 3.6-2 Line Number (12)
CB-SS(CS)-EXT This generic asset includes all carbon structural steel in the Control Building frame that is exposed to the weather.	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16)

Table 2.4.2-5 Control Building

Component Group	Passive Function	Aging Management Reference
CB-SS(CS)-INT This generic asset includes all carbon structural steel of the Control Building that is protected from the weather. Columns, posts, beams, baseplates, bracing, and the exposed faces of plates and structural members are included. This does not include carbon structural steel used as component supports.	FIRE BARRIER SHELTER/PROTECT EQUIPMENT HEAT SINK HELB SHIELDING MISSILE BARRIER SHELTER/PROTECT EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) This applies to all seven passive functions.
S51F This carbon steel interior door is designed to resist high energy line breaks. The door, along with the water curtain, provides a fire barrier.	FIRE BARRIER HELB SHIELDING MISSILE BARRIER	Table 3.6-1 Line Number (16) This applies to all three passive functions.
EXT-DOOR Carbon steel interior doors are designed to resist high energy line breaks and floods.	FLOOD BARRIER HELB SHIELDING MISSILE BARRIER	Table 3.6-1 Line Number (16) This applies to all three passive functions.
INT-DOOR Carbon steel interior doors are designed to resist high energy line breaks.	HELB SHIELDING MISSILE BARRIER	Table 3.6-1 Line Number (16) This applies to both passive functions.

Table 2.4.2-5 Control Building

Component Group	Passive Function	Aging Management Reference
VALVE BODY This valve dewaters the Control Building in the event of a Service Water or Fire Water line break in the mechanical equipment room.	FLOOD BARRIER HELB SHIELDING	Table 3.6-2 Line Number (3) This applies to both passive functions

2.4.2.6 All Volatile Water Treatment Building

Description

The All-Volatile-Treatment (AVT) Building is a nonseismic structure that houses non-nuclear safety equipment considered important to safety. Specifically, the AVT Building houses and supports the technical support center diesel generator and battery which may be used to mitigate the effects of fires and station blackouts. Additionally, building walls act as fire barriers. Accordingly, the AVT Building is considered a non-safety structure whose failure could affect a safety function.

The AVT Building houses demineralizers and other equipment necessary for the condensate polishing system to allow all-volatile-treatment of secondary water. The technical support center is located on the second floor of the all-volatile-treatment building and houses the computers and equipment, including emergency power supplies (diesel generator and batteries), necessary to provide the staff technical support during an emergency event. The AVT Building is founded on a concrete mat. The building abuts the Turbine Building at the east end of the Turbine Building. A small section of masonry block separates the Turbine Building from the AVT Building. Some exterior portions of the east and north sides of the structure are masonry block. The load bearing portions of the building include steel support framing and reinforced concrete. Select portions of the bottom floor of the building have 2-ft-thick concrete walls and ceiling to minimize operator exposure in case of radioactivity buildup in the resin beds should a steam-generator tube leak occurs. The building concrete roof is supported by steel decking and trusses. The AVT building is not designed to be high wind or tornado missile resistant. The technical support center is above the maximum external flood water level.

The AVT Building contains racks, panels, electrical enclosures, equipment supports and fire doors, penetration barriers and seals. Those equipment sets receive a separate commodity group evaluation independent of the building evaluation. Building interior floor drains are evaluated within the Treated Water system.

System Function Listing

In addition to the System Functions described above, the All-Volatile-Treatment Building also contains components which support additional functions (associated design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the AVT/TSC Building perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the AVT/TSC Building that perform special capability class functions are tracked under the Criterion 2 code (Y) and Criterion 3 codes (Z1 through Z5).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the AVT/TSC Building perform this associated design system function.

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the AVT/TSC Building perform this associated design system function. The AVT/TSC Building shelters and support the TSC diesel generator and battery charger both of which are used for post fire safe shutdown activities.

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the AVT/TSC Building perform this associated design system function.

Code Z5 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS STATION BLACKOUT (10 CFR 50.63)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: Components within the AVT/TSC Building perform this associated design system function.

UFSAR Reference

Additional All Volatile Water Treatment Building details are provided in Section 10.7.7.4.1, Section 13.3, Section 1.2.3.7, and Section 3.8.5 of the UFSAR.

License Renewal Drawings

The license renewal drawings for the All Volatile Water Treatment Building are listed below:

Site Plot

Components/Commodities Subject to an AMR

The component groups for the All Volatile Water Treatment Building that require aging management review are indicated in Table 2.4.2-6 along with each component group’s passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.4.2-6 All Volatile Water Treatment Building

Component Group	Passive Function	Aging Management Reference
AVT-ARCH-EXT AVT-ARCH-EXT includes the non-load bearing building elements not relied upon in the safety analysis which provide normal habitability control and weather proofing, e.g., building siding, built up roof systems, windows, etc.	SHELTER/PROTECT EQUIPMENT	Table 3.6-2 Line Number (2)
AVT-BLOCK-EXT This generic asset includes masonry block exposed to the weather that acts as part of the building siding system. Mortar is included in this evaluation. AVT-BLOCK-EXT includes the non-load bearing building elements not relied upon in the safety analysis which provide normal habitability control and weather proofing, e.g., building siding, built up roof systems, windows, etc.	SHELTER/PROTECT EQUIPMENT	Table 3.6-1 Line Number (20)

Table 2.4.2-6 All Volatile Water Treatment Building

Component Group	Passive Function	Aging Management Reference
AVT-BLOCK-INT This generic asset includes all masonry block walls of the AVT/TSC Building protected from the weather. The west side of the AVT/TSC Building abuts the Turbine Building. Some interior walls separating the buildings are block. Mortar is included in this evaluation.	FIRE BARRIER	Table 3.6-1 Line Number (20)

Table 2.4.2-6 All Volatile Water Treatment Building

Component Group	Passive Function	Aging Management Reference
AVT-C-BUR This generic asset includes all concrete in the AVT/TSC Building that is in contact with the soil and groundwater. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are evaluated in the AVT/TSC Building threaded fasteners and the exposed faces of plates and structural members are evaluated in AVT/TSC Building structural steel. This generic asset includes evaluation of elastomer sealing material used below grade in the AVT/TSC Building. Water stops are embedded in selected below grade construction joints. Also included in this evaluation is post-construction urethane foam resin injected into seams and cracks to prevent ground water intrusion.	STRUCTURAL SUPPORT NSR EQUIPMENT	Table 3.6-1 Line Number (17) Table 3.6-1 Line Number (21) Table 3.6-1 Line Number (22) Table 3.6-1 Line Number (23)

Table 2.4.2-6 All Volatile Water Treatment Building

Component Group	Passive Function	Aging Management Reference
AVT-C-EXT This generic asset includes all concrete in the AVT/TSC Building that is exposed to the weather. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are included in the AVT/TSC Building threaded fasteners and the exposed faces of plates and structural members are included in AVT/TSC Building structural steel.	SHELTER/PROTECT EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (23)

Table 2.4.2-6 All Volatile Water Treatment Building

Component Group	Passive Function	Aging Management Reference
AVT-C-INT This generic asset includes all concrete in the AVT/TSC Building that is protected from the weather. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. Grout, used under column base plates is included in this evaluation. The exposed portion of structural anchor bolts are evaluated in the AVT/TSC Building threaded fasteners and the exposed faces of plates and structural members are evaluated in AVT/TSC Building structural steel. This asset includes the fuel oil confinement curb for the TSC Diesel.	STRUCTURAL SUPPORT NSR EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (23)
AVT-FAST(CS)-INT This generic asset includes the exposed portion of carbon steel threaded fasteners in the AVT/TSC Building that are protected from the weather.	STRUCTURAL SUPPORT NSR EQUIPMENT	Table 3.6-1 Line Number (16)

Table 2.4.2-6 All Volatile Water Treatment Building

Component Group	Passive Function	Aging Management Reference
AVT-SS(CS)-INT This generic asset includes all structural carbon steel of the AVT/TSC Building that is protected from the weather. Columns, posts, beams, baseplates, bracing, crane support girders, crane rails, and the exposed faces of plates and structural members are included. This does not include carbon structural steel used as component supports.	STRUCTURAL SUPPORT NSR EQUIPMENT	Table 3.6-1 Line Number (16)

2.4.2.7 Screen House Building

Description

The Screen House Building is partially a Seismic Category 1 structure. It is located north of the Turbine Building and is not immediately adjacent to any other major structure. The Screen House structurally supports and houses safety related equipment, equipment used to mitigate fires and components used for safe shutdown following fires and station blackout events. Included within the Screen House evaluation boundary is the Circulating Water system discharge canal which functions to ensure the availability of essential service water from the ultimate heat sink. Additionally, the Screen House also contains non-safety equipment whose failure could prevent the satisfactory accomplishment of a safety function (internal flood protection).

The Screen House is located 115 ft north of the Turbine Building and 80 ft south of the lake shore. The structural configuration of the Screen House is integral to the functioning of the Service Water System, the Circulating Water System and the Fire Protection System. The below grade and submerged portions of the structure make available the flow path from the ultimate heat sink to the referenced system pump suctions. Should the Circulating Water System intake tunnels be lost, the structure supports provisions for an alternate lake suction path by providing a flow path from the discharge canal to the Service Water and Fire System water pump bay. The components supporting this feature are evaluated within the Service Water system.

The Screen House building is comprised of two structural steel superstructures, one on the service water system side and one on the circulating water system side. The superstructures share a reinforced concrete substructure. The service water portion of the building (both below and above grade) is a Seismic Category I structure. The service water portion houses four safety related service water pumps and safety related electric switchgear. The circulating water side houses the traveling water screens and circulating water pumps. The entire screen house-service water building is founded in or on bedrock with the exception of the basement of the service water portion, which is founded approximately 4 ft above bedrock. The service water portion of the Screen House consists of four rigid frame bents in the east-west direction with bracing for wind and seismic loads in the north-south direction. The roof system is designed as a rigid bent to transmit horizontal seismic loads to the frame columns and through the bracing to the foundation. Insulated siding is used for most of the walls above the operating floor. The exterior walls contain windows, doors and louvered ventilation openings. The roof has been provided with scuppers designed to ensure that any rainwater, resulting from a design-basis storm, would not accumulate on the roofs and cause damage. The roofing and siding provide weather resistance and allow habitability control but are not designed to be tornado missile resistant.

The Screen House Building is not designed to resist, or protect housed components, against all possible external flooding, high wind, fire, or high or moderate energy line break events. Complete protection against these low probability events is not needed because alternative shutdown means are available, which do not rely upon service water from the Screen House. In the safety evaluation report for SEP Topic III-5.B, Pipe Breaks Outside Containment, the NRC concluded that any further modification of the Screen House to provide additional protection from pipe break effects for service water system components or for buses 17 and 18 is not required. The mitigative strategy developed and approved for pipe breaks was subsequently applied with respect to external flooding and tornado events. After completion of the Structural Upgrade Program the NRC concluded that the station could achieve safe shutdown given the effects of loss of the Screen House from external events.

The discharge canal, included in the Screen House evaluation boundary, is a reinforced concrete structure that directs Circulating Water and Service Water effluent back to the lake and the end of the open loop cooling cycle. As noted above the Screen House and discharge canal have features which provide water intake from the discharge canal to the Service Water system should the Circulating Water intake tunnel become unavailable. Over topping of the discharge canal from storm effects in the lake is prevented by a revetment. The revetment is evaluated separately within the Essential Yard Structures group.

In addition to structural and load bearing elements, the Screen House contains features and appurtenances credited in the licensing basis and relied upon to ensure the health and safety of the public. These features include:

a. Protection of safety-related equipment from flooding due to a break or leakage in the circulating water system. The first protective feature consists of tripping the circulating water pumps when a leak is detected. The tripping of the circulating water pumps is accomplished by redundant two-out-of-three logic receiving level information from the circulating water pump pit in the screen house and from the condenser pit in the Turbine Building. Electrical components that perform this function are evaluated within the Reactor Protection System. The second part of the flood mitigative system is a permanently installed, non-movable Seismic Category I dike in the Screen House, which have been built to contain the water that may escape from the circulating water system. The dike is 30 in. in height and is situated to prevent water from reaching safety-related equipment.

b. A curb has been installed around the diesel fire pump and the diesel oil storage tank to control any diesel oil leaks. The curbed area is equipped with a floor drain which drains to a holding tank buried outside the Screen House.

c. The building foundations and below grade walls were constructed with water stops to prevent the intrusion of ground water.

d. Cable entrances are sealed to prevent the intrusion of ground water.

The Screen House contains racks, panels, electrical enclosures, equipment supports and fire penetration barriers and seals. Those equipment sets receive a separate commodity group evaluation independent of the building evaluation. Building interior floor drains are evaluated within the Circulating Water System.

System Function Listing

In addition to the System Functions described above, the Screen House also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code R	Cri 1	Cri 2	Cri 3				
STRUCTURALLY SUPPORT OR HOUSE SAFETY CLASS 1, 2, 3 COMPONENTS			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Screen House Building perform this primary design system function.

Code S	Cri 1	Cri 2	Cri 3				
SPECIAL CAPABILITY CLASS FUNCTIONS			FP	EQ	PTS	AT	SB

Comment: Components within the Screen House Building perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Screen House Building that perform special capability class functions are tracked under the Criterion 2 code (Y) and Criterion 3 codes (Z1 through Z5).

Code T	Cri 1	Cri 2	Cri 3				
NON-NUCLEAR SAFETY CLASS FUNCTIONS			FP	EQ	PTS	AT	SB

Comment: Components within the Screen House Building perform this associated design system function.

Code Y	Cri 1	Cri 2	Cri 3				
LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Screen House Building perform this associated design system function. Components within the Screen House Building resist the effects of flooding.

Code Z1	Cri 1	Cri 2	Cri 3				
LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Screen House Building perform this associated design system function.

Code Z5 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS STATION BLACKOUT (10 CFR 50.63)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: Components within the Screen House Building perform this associated design system function.

UFSAR Reference

Additional Screen House Building details are provided in Section 3.6.2.5.1.1 and Section 2.2.2.3 of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Screen House Building are listed below:

Site Plot

Components/Commodities Subject to an AMR

The component groups for the Screen House Building that require aging management review are indicated in Table 2.4.2-7 along with each component group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.4.2-7 Screen House Building

Component Group	Passive Function	Aging Management Reference
SH-ARCH-EXT SH-ARCH-EXT includes the non-load bearing building elements not relied upon in the safety analysis which provide normal habitability control and weather proofing, e.g., building siding, built up roof systems, windows, etc.	SHELTER/PROTECT EQUIPMENT	Table 3.6-2 Line Number (2)

Table 2.4.2-7 Screen House Building

Component Group	Passive Function	Aging Management Reference
SH-C-BUR This generic asset includes all concrete in the Screenhouse that is in contact with the soil and groundwater. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are evaluated in the Screenhouse threaded fasteners and the exposed faces of plates and structural members are evaluated in Screenhouse structural steel. This generic asset includes evaluation of elastomer sealing material used below grade in the Screenhouse. Water stops are embedded in selected below grade construction joints. Also included in this evaluation is post-construction urethane foam resin injected into seams and cracks to prevent ground water intrusion.	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (17) Table 3.6-1 Line Number (21) Table 3.6-1 Line Number (22) Table 3.6-1 Line Number (23) Table 3.6-2 Line Number (7)

Table 2.4.2-7 Screen House Building

Component Group	Passive Function	Aging Management Reference
SH-C-EXT This generic asset includes all of the concrete in the Screenhouse that is exposed to the weather. Included in this evaluation is the concrete used in the discharge canal. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are included in the Screenhouse threaded fasteners and the exposed faces of plates and structural members are included in Screenhouse structural steel.	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (23) Table 3.6-2 Line Number (7)

Table 2.4.2-7 Screen House Building

Component Group	Passive Function	Aging Management Reference
SH-C-INT This generic asset includes all concrete in the Screenhouse that is protected from the weather. Interior concrete provides flood protection curbing for the sub-basement and fire protection curbing to contain diesel fuel oil spills. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. Grout, used under column base plates is included in this evaluation. The exposed portion of structural anchor bolts are evaluated in the Screenhouse threaded fasteners and the exposed faces of plates and structural members are evaluated in Screenhouse structural steel.	COOLING WATER SOURCE FIRE BARRIER STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (23) These apply to all four passive functions.

Table 2.4.2-7 Screen House Building

Component Group	Passive Function	Aging Management Reference
SH-C-RW This generic asset includes all of the concrete in the Screenhouse that is submerged. Included in this evaluation is the concrete used in the discharge canal. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are included in the Screenhouse threaded fasteners and the exposed faces of plates and structural members are included in Screenhouse structural steel.	COOLING WATER SOURCE STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-2 Line Number (7) This applies to all three passive functions.
SH-ELAST-INT This generic asset includes all elastomer sealing material in the Screenhouse that is protected from the weather. Elastomers are used as gasketing material for flood barriers.	FLOOD BARRIER	Table 3.6-2 Line Number (5)

Table 2.4.2-7 Screen House Building

Component Group	Passive Function	Aging Management Reference
SH-FAST(CS)-INT This generic asset includes the exposed portion of carbon steel threaded fasteners in the Screenhouse that are protected from the weather.	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16)
SH-SS(CS)-INT This generic asset includes all structural carbon steel for the Screenhouse (e.g., plates, beams, columns, grating, etc., and the barrier between the CW and SW bays to protect the SW pumps from a break in the C/W system and to prevent flooding of other vital equipment) that is protected from the weather.	FLOOD BARRIER SHELTER/PROTECT EQUIPMENT STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) This applies to all four passive functions.

2.4.2.8 Standby Auxiliary Feedwater Building

Description

The Standby Auxiliary Feedwater (SAFW) Building is a Seismic Category I high wind and tornado missile resistant structure located south of the Auxiliary Building. The SAFW Building houses and supports a safety related feedwater system that is completely diverse from the preferred auxiliary feedwater system (located in the Intermediate Building). The SAFW Building also protects connections for an alternate pump suction source should the service water system be disabled.

The SAFW Building is a concrete structure utilizing reinforced concrete for the walls, roof, and base mat. The building is supported by 12 caissons that are socketed into competent rock. The exterior of the building is sheathed with a combination of architectural brick and siding. The SAFW building does not need to be external flood resistant because the safety related equipment in the room is mounted above the maximum flood level. The seismic portion of the building is connected to an entry way enclosure that allows for building environmental control. The major structures of Ginna Station have experienced no visible evidence of settlement since the construction of the station. (During the SEP and evaluation of Topic II-4.F, Settlement of Foundations and Buried Equipment, the NRC concluded that the settlement of foundations and buried equipment is not a safety concern for Ginna Station.)

In addition to structural and load bearing elements, the SAFW Building contains features and appurtenances credited in the licensing basis and relied upon to ensure the health and safety of the public. These features include:

a. Safety related equipment is mounted above the maximum external flood level.

b. An internal missile barrier separates the A and B SAFW pumps.

c. The structure provides a high wind and tornado missile protected feedwater source.

d. The structure, in conjunction with the Standby Auxiliary Feedwater System, was added to further improve steam generator feedwater reliability and specifically to substitute for the preferred auxiliary feedwater in the low probability that preferred auxiliary feedwater pumps are damaged due to nearby high-energy pipe breaks within the Intermediate Building.

The SAFW Building contains racks, panels, electrical enclosures, equipment supports and fire, penetration barriers and seals (in addition to those noted above). These equipment sets receive a separate commodity group evaluation independent of the building evaluation.

System Function Listing

In addition to the System Functions described above, the SAFW Building also contains components which support additional functions (associated design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code R STRUCTURALLY SUPPORT OR HOUSE SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Standby Auxiliary Feedwater Building perform this primary design system function.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Standby Auxiliary Feedwater Building perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Standby Auxiliary Feedwater Building that perform special capability class functions are tracked under the Criterion 2 code (Y) and Criterion 3 codes (Z1 through Z5).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Standby Auxiliary Feedwater Building perform this associated design system function.

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Standby Auxiliary Feedwater Building perform this associated design system function. The Standby Auxiliary Feedwater Building entrance way provides an enclosure necessary for habitability control.

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Standby Auxiliary Feedwater Building perform this associated design system function.

UFSAR Reference

Additional Standby Auxiliary Feedwater Building details are provided in Section 3.11.3.8, Section 3.8.4.1.5, Section 3.8.5, Section 1.2.3.8, and Section 3.8.4.1 of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Standby Auxiliary Feedwater Building are listed below:

Site Plot

Components/Commodities Subject to an AMR

The component groups for the Standby Auxiliary Feedwater Building that require aging management review are indicated in Table 2.4.2.8 along with each component group’s passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.4.2-8 Standby Auxiliary Feedwater Building

Component Group	Passive Function	Aging Management Reference
AF-ARCH-EXT AF-ARCH-EXT includes the non-load bearing building elements not relied upon in the safety analysis which provide normal habitability control and weather proofing, e.g., building siding, built up roof systems, windows, etc.	SHELTER/PROTECT EQUIPMENT	Table 3.6-2 Line Number (2)

Table 2.4.2-8 Standby Auxiliary Feedwater Building

Component Group	Passive Function	Aging Management Reference
AF-C-BUR This generic asset includes all concrete in the Standby Aux Feedwater Building that is in contact with the soil and groundwater. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are evaluated in the Standby Aux Feedwater Building threaded fasteners and the exposed faces of plates and structural members are evaluated in Standby Aux Feedwater Building structural steel. This generic asset includes evaluation of elastomer sealing material used below grade in the Auxiliary Building. Water stops are embedded in selected below grade construction joints. Also included in this evaluation is post-construction urethane foam resin injected into seams and cracks to prevent ground water intrusion.	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (17) Table 3.6-1 Line Number (21) Table 3.6-1 Line Number (22) Table 3.6-1 Line Number (23)

Table 2.4.2-8 Standby Auxiliary Feedwater Building

Component Group	Passive Function	Aging Management Reference
AF-C-EXT This generic asset includes all concrete in the Standby Aux Feedwater Building that is exposed to the weather. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are included in the Standby Aux Feedwater Building threaded fasteners and the exposed faces of plates and structural members are included in Standby Aux Feedwater Building structural steel.	SHELTER/PROTECT EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (23)

Table 2.4.2-8 Standby Auxiliary Feedwater Building

Component Group	Passive Function	Aging Management Reference
AF-C-INT This generic asset includes all concrete in the Standby Aux Feedwater Building that is protected from the weather. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. Grout, used under column base plates is included in this evaluation. The exposed portion of structural anchor bolts are evaluated in the Standby Aux Feedwater Building threaded fasteners and the exposed faces of plates and structural members are evaluated in Standby Aux Feedwater Building structural steel.	FIRE BARRIER MISSILE BARRIER SHELTER/PROTECT EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (23) These apply to all four passive functions.
AF-FAST(CS)-INT This generic asset includes the exposed portion of carbon steel threaded fasteners in the Standby Aux Feedwater Building that are protected from the weather.	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16)

Table 2.4.2-8 Standby Auxiliary Feedwater Building

Component Group	Passive Function	Aging Management Reference
AF-SS(CS)-INT This generic asset includes all structural carbon steel of the Standby Aux Feedwater Building that is protected from the weather. Columns, posts, beams, baseplates, bracing, and the exposed faces of plates and structural members are included. This does not include carbon structural steel used as component supports.	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16)

2.4.2.9 Service Building

Description

The Service Building is a nonseismic structure that houses non-nuclear safety equipment considered important to safety. Specifically, the Service Building houses and supports the condensate storage tanks which may be used to mitigate the effects of fires and station blackouts and is the preferred suction source for the auxiliary feedwater system. Additionally, building walls act as fire barriers. Accordingly, the Service Building is considered a non-safety structure whose failure could affect a safety function.

The Service Building is part of a complex of interconnected buildings surrounding, but structurally independent of, the Containment Building. These buildings are interconnected as follows: The Seismic Category I Auxiliary Building is contiguous with the nonseismic Service Building on the west side. The Seismic Category I Intermediate Building adjoins the seismically analyzed Turbine Building to the north, and the Auxiliary Building to the south. The Turbine Building adjoins the Seismic Category I Diesel Generator Building to the north and the Seismic Category I Control Building to the south. The facade, a cosmetic rectangular structure that encloses the Containment Building, has all four sides partly or totally in common with the Auxiliary and Intermediate Buildings. The Auxiliary Building adjoins the Seismic Category I Standby Auxiliary Feedwater Building on the south. In the original building analysis, each Seismic Category I structure was treated independently. During the Systematic Evaluation Program (SEP) plant evaluation it was found that the interconnected nature of the buildings was an important feature, especially in view of the lack of detailed original seismic design information. Therefore, both Seismic Category I and nonseismic category buildings were included in a complicated three-dimensional structural system reanalysis model. As part of this effort, the interconnected Turbine Building was determined to be capable of withstanding safe shutdown earthquake forces. Based on the SEP review, audits, and plant inspections, the NRC safety evaluation reports concluded that the evaluation and resolution of SEP Topics III-2, Wind and Tornado Loadings; III-4.A, Tornado Missiles; III-6, Seismic Design Considerations; and III-7.B, Load Combinations, were acceptable. The NRC also concluded that the RG&E analysis and implementation of the Structural Upgrade Program were acceptable.

The Service Building is a nonseismic structure. It extends from the south end of the Auxiliary Building, through the Intermediate Building, and ends at the north end of the Turbine Building. The building is a two-story steel structure with spread footings, steel columns, and concrete-steel framing floors and roof. The basement is at elevation 253.66 ft, the floor is at elevation 271 ft, and the roof is at elevation 287.33 ft. The walls between the Service Building and the other buildings as well as the partitions in the building are made of concrete blocks.

The building has a combination of architectural brick siding and glass windows. The roofing, siding and windows provide weather resistance and allow habitability control but are not designed to be wind or tornado missile resistant. During high wind or tornado events the siding on the superstructure above elevation 271 ft would blow outward, thus relieving the pressure and wind loads. The components that might be affected by a tornado are the two condensate storage tanks. There is reasonable assurance that the feedwater supply will be maintained because of the available redundancy and the fact that two-thirds of the tank volume is below grade. If the tanks are damaged alternate protected feedwater sources are available.

The Service Building contains racks, panels, electrical enclosures, equipment supports and fire doors, penetration barriers and seals. Those equipment sets receive a separate commodity group evaluation independent of the building evaluation. Building interior floor drains are evaluated within the Treated Water system.

System Function Listing

In addition to the System Functions described above, the Service Building also contains components which support additional functions (associated design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Service Building perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Service Building that perform special capability class functions are tracked under the Criterion 2 code (Y) and Criterion 3 codes (Z1 through Z5).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Service Building perform this associated design system function.

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Service Building perform this associated design system function. The Service Building provides shelter and support for the condensate storage tanks which are the preferred suction source for Auxiliary Feedwater.

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Service Building perform this associated design system function.

Code Z5 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS STATION BLACKOUT (10 CFR 50.63)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: Components within the Service Building perform this associated design system function.

UFSAR Reference

Additional Service Building details are provided in Section 3.8.4.1, Section 1.2.3.10, Section 3.8.4.1.8, and Appendix 3A.3.8 of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Service Building are listed below:

Site Plot

Components/Commodities Subject to an AMR

The component groups for the Service Building that require aging management review are indicated in Table 2.4.2.9 along with each component group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.4.2-9 Service Building

Component Group	Passive Function	Aging Management Reference
SB-ARCH-EXT SB-ARCH-EXT includes the non-load bearing building elements not relied upon in the safety analysis which provide normal habitability control and weather proofing, e.g., building siding, built up roof systems, windows, etc.	SHELTER/PROTECT EQUIPMENT	Table 3.6-2 Line Number (2)

Table 2.4.2-9 Service Building

Component Group	Passive Function	Aging Management Reference
SB-BLOCK-INT This generic asset includes all masonry block walls of the Service Building protected from the weather. The east side of the Service building abuts the Intermediate Building and Turbine Building. The interior walls separating the buildings are block. Mortar is included in this asset evaluation.	FIRE BARRIER	Table 3.6-1 Line Number (20)

Table 2.4.2-9 Service Building

Component Group	Passive Function	Aging Management Reference
SB-C-BUR This generic asset includes all concrete in the Service Building that is in contact with the soil and groundwater. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are evaluated in the Service Building threaded fasteners and the exposed faces of plates and structural members are evaluated in Service Building structural steel. This generic asset includes evaluation of elastomer sealing material used below grade in the Service Building. Water stops are embedded in selected below grade construction joints. Also included in this evaluation is post-construction urethane foam resin injected into seams and cracks to prevent ground water intrusion.	STRUCTURAL SUPPORT NSR EQUIPMENT	Table 3.6-1 Line Number (17) Table 3.6-1 Line Number (21) Table 3.6-1 Line Number (22) Table 3.6-1 Line Number (23)

Table 2.4.2-9 Service Building

Component Group	Passive Function	Aging Management Reference
SB-C-EXT This generic asset includes all concrete in the Service Building that is exposed to the weather. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are included in the Service Building threaded fasteners and the exposed faces of plates and structural members are included in Service Building structural steel.	STRUCTURAL SUPPORT NSR EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (23)

Table 2.4.2-9 Service Building

Component Group	Passive Function	Aging Management Reference
SB-C-INT This generic asset includes all concrete in the Service Building that is protected from the weather. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. Grout, used under column base plates is included in this evaluation. The exposed portion of structural anchor bolts are evaluated in the Service Building threaded fasteners and the exposed faces of plates and structural members are evaluated in Service Building structural steel.	STRUCTURAL SUPPORT NSR EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (23)
SB-FAST(CS)-INT This generic asset includes the exposed portion of carbon steel threaded fasteners in the Service Building that are protected from the weather.	STRUCTURAL SUPPORT NSR EQUIPMENT	Table 3.6-1 Line Number (16)

Table 2.4.2-9 Service Building

Component Group	Passive Function	Aging Management Reference
SB-SS(CS)-INT This generic asset includes all structural carbon steel of the Service Building that is protected from the weather. Columns, posts, beams, baseplates, bracing, crane support girders, crane rails, and the exposed faces of plates and structural members are included. This does not include carbon structural steel used as component supports	STRUCTURAL SUPPORT NSR EQUIPMENT	Table 3.6-1 Line Number (16)

2.4.2.10 **Cable Tunnel**

Description

The Cable Tunnel is a safety related structure which houses and supports the electrical control circuits for most safety related equipment. The Cable Tunnel includes a cofferdam, placed to protect the structure from the effects of external flooding. Accordingly, the Cable Tunnel contains non-nuclear safety equipment whose failure could affect a safety function.

The Cable Tunnel is a below grade reinforced concrete structure supported by steel piles that has openings in the Control Building, the Intermediate Building and the Auxiliary Building. The tunnel allows cables to be routed between these structures. The roof of the portion of the tunnel that extends to the Control Building is level with the yard grade. This section houses an escape hatch. The tunnel is protected from the effects of external flooding by a cofferdam surrounding the escape hatch. The Cable Tunnel is resistant to the effects of high winds and internally or externally generated missiles due to its underground configuration, the orientation of its openings, and the shielding provided by adjacent structures and components.

The Cable Tunnel contains equipment supports and is also associated with fire penetration barriers and seals where it interfaces with the other structures. These equipment sets receive a separate commodity group evaluation independent of the structure evaluation. The structure interior floor drains are evaluated within the Treated Water system.

System Function Listing

In addition to the System Functions described above, the Cable Tunnel also contains components which support additional functions (associated design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code R STRUCTURALLY SUPPORT OR HOUSE SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Cable Tunnel perform this primary design system function.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Cable Tunnel perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Cable Tunnel that perform special capability class functions are tracked under the Criterion 3 codes (Z1 through Z5).

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Cable Tunnel perform this associated design system function. The Cable Tunnel escape hatch is equipped with a cofferdam that acts as an external flood barrier.

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Cable Tunnel perform this associated design system function.

UFSAR Reference

Additional Cable Tunnel details are provided in Section 3.8.2.1.2.5, Section 3.8.2.1.1.1, Section 3.11.3.4, Section 3.3.3.3.10, and Section 3.6.2.1.1 of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Cable Tunnel are listed below:

Site Plot

Components/Commodities Subject to an AMR

The component groups for the Cable Tunnel that require aging management review are indicated in Table 2.4.2.10 along with each component group’s passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.4.2-10 Cable Tunnel

Component Group	Passive Function	Aging Management Reference
PIPE Cable tunnel escape hatch gutter drain.	PRESSURE BOUNDARY	Table 3.6-2 Line Number (10)

Table 2.4.2-10 Cable Tunnel

Component Group	Passive Function	Aging Management Reference
TUNNEL-C-BUR This generic asset includes all concrete in the Cable Tunnel that is in contact with the soil and groundwater. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are evaluated in the Cable Tunnel threaded fasteners. This generic asset includes evaluation of elastomer sealing material used below grade in the Cable Tunnel. Water stops are embedded in selected below grade construction joints. Also included in this evaluation is post-construction urethane foam resin injected into seams and cracks to prevent ground water intrusion.	SHELTER/PROTECT EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (17) Table 3.6-1 Line Number (21) Table 3.6-1 Line Number (22) Table 3.6-1 Line Number (23) These apply to both passive functions.

Table 2.4.2-10 Cable Tunnel

Component Group	Passive Function	Aging Management Reference
TUNNEL-C-EXT This generic asset includes all concrete in the Cable Tunnel that is exposed to the weather. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included. The exposed portion of structural anchor bolts are included in the Cable Tunnel threaded fasteners.	FLOOD BARRIER SHELTER/PROTECT EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (23) These apply to all three passive functions.
TUNNEL-C-INT This generic asset includes all concrete in the Cable Tunnel that is protected from the weather. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included.	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (23)
TUNNEL-ELAST-EXT This generic asset includes all elastomer sealing material in the Cable Tunnel that is exposed to the weather. Elastomers are used between the escape hatch cofferdam and the exterior concrete.	FLOOD BARRIER	Table 3.6-2 Line Number (6)

Table 2.4.2-10 Cable Tunnel

Component Group	Passive Function	Aging Management Reference
TUNNEL-FAST(CS)-EXT This generic asset includes the exposed portion of carbon steel threaded fasteners for the Cable Tunnel that are exposed to the weather.	FLOOD BARRIER	Table 3.6-1 Line Number (16)
TUNNEL-SS(CS)-EXT This generic asset includes all structural carbon steel for the Cable Tunnel (e.g., cofferdam) that is exposed to the weather.	FLOOD BARRIER	Table 3.6-1 Line Number (16)
TUNNEL-SS(CS)-PILES-BURIED This generic asset represents the carbon steel piles that comprise portions of the cable tunnel foundation.	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-2 Line Number (9)
VALVE BODY Cable tunnel escape hatch gutter drain.	PRESSURE BOUNDARY	Table 3.6-2 Line Number (10)

2.4.2.11 Essential Yard Structures

Description

Within the site boundary there are structures and subterranean yard components that are necessary to support safety related functions. These civil features emanate from design considerations that are typically independent of the function of the plant system associated with the feature. Accordingly, components evaluated within the Essential Yard Structures group house and support or provide shelter and protection for safety related or essential equipment.

The Essential Yard Structures group is a listing of the major civil components found onsite but not be included within any other License Renewal review boundary. (Note: Above and below grade tanks are evaluated within the mechanical system they serve while the discharge canal is evaluated with the Screen House.)

Structures and Components evaluated in this grouping include:

a. Service Water Alternative Discharge Structure

b. Vital AC and DC Duct Banks, including their manholes and covers

c. Revetment Armor Stone

d. Transformer support pads

The primary service water discharge line discharges to the discharge canal and then to Lake Ontario. The redundant service water discharge line discharges to a Seismic Category I discharge structure, then to Deer Creek and to Lake Ontario. The redundant service water discharge line is normally in standby; however, it is occasionally placed in service for such activities as surveillance testing or maintenance work.

Direct current control power from the station batteries is run in underground duct, separated, and apart from the cable tunnel, in order to maintain the necessary control in the event of an emergency. The electrical connections from the diesels to buses 17 and 18 are routed inside a separate underground duct bank from the diesel-generator building to the screen house.

The breakwater that protects the plant from lake flooding is a stone revetment constructed in two reaches. The stone revetment was initially constructed with two layers of 5-ton minimum armor stones laid upon a 1.0 vertical to a 1.5 horizontal sideslope to a minimum elevation of 257.0 ft msl. Because of the high lake levels that were predicted for Lake Ontario during the early 1970s, the crest elevation of the revetment was raised to a minimum of 261.0 ft msl by placement of cap stone along the top of the revetment.

System Function Listing

In addition to the System Functions described above, the Essential Yard Structures group also contains components which support additional functions (associated design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the, or specific components contained in the group, is provided within the summary below.

Code R STRUCTURALLY SUPPORT OR HOUSE SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Essential Yard Structures group perform this primary design system function.

UFSAR Reference

Additional Essential Yard Structures details are provided in Appendix 3A.4.5.2, Section 9.2.1.2.2, Section 3.4.1.1.2, and Appendix 3A.4.1.3 of the UFSAR.

License Renewal Drawings

The license renewal drawings for the Essential Yard Structures are listed below:

Site Plot

Components/Commodities Subject to an AMR

The component groups for the Essential Yard Structures that require aging management review are indicated in Table 2.4.2.11 along with each component group’s passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.4.2-11 Essential Yard Structures

Component Group	Passive Function	Aging Management Reference
YARD-CASTIRON-EXT This generic asset includes all cast iron for the Essential Yard Structures (e.g., manhole covers) that are exposed to the weather.	SHELTER/PROTECT EQUIPMENT	Table 3.6-2 Line Number (4)
YARD-C-BUR This generic asset includes all concrete in Essential Yard Structures that is in contact with the soil and groundwater. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included.	COOLING WATER SOURCE SHELTER/PROTECT EQUIPMENT	Table 3.6-1 Line Number (17) Table 3.6-1 Line Number (21) Table 3.6-1 Line Number (22) Table 3.6-1 Line Number (23) These apply to both passive functions.
YARD-C-EXT This generic asset includes all of the concrete in the Essential Yard Structures that is exposed to the weather. Included in this evaluation is the concrete used in the Service Water alternate discharge structure and the exposed portions of duct bank manholes. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included.	SHELTER/PROTECT EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (23)

Table 2.4.2-11 Essential Yard Structures

Component Group	Passive Function	Aging Management Reference
YARD-C-INT This generic asset includes all of the concrete in the Essential Yard Structures that is protected from the weather. Included in this evaluation is the concrete used in the duct bank manholes. Embedded steel, reinforcement, and the embedded portion of anchor bolts are included.	SHELTER/PROTECT EQUIPMENT	Table 3.6-1 Line Number (16) Table 3.6-1 Line Number (23)
YARD-STONE-EXT This generic asset includes all armor stone used in the revetment.	SHELTER/PROTECT EQUIPMENT	Table 3.6-2 Line Number (7)

2.4.2.12 **Component Supports Commodity Group**

Description

System components physically interface with civil structures. The interface takes place in the form of Component Supports that position and bear the weight of the component assemblies and provide the proper amount of resistance to motion during normal operating conditions, accidents, transients and off normal events. Component Supports are located throughout the plant. Included in the scope of the Component Support Commodity group are supports for safety related components and non-safety related components whose failure could affect a safety function (typically referred to as seismic II/I). The Component Support Commodity Group does not include evaluation of snubbers (considered active per NEI 95-10) or the reactor vessel, reactor coolant pumps, steam generators, pressurizer and other reactor coolant system supports, all of which receive a separate evaluation.

The Component Support Commodity Group includes those structural elements that are connected to civil structures and which extend to a system or system components for the purpose of providing support or restraint. Inclusive in this boundary definition is any vibration dampeners or other passive connective appurtenances intrinsic to the functioning of the support. The commodity group also includes spray or drip shields attached to equipment and electrical system rack, panels and enclosures.

For mechanical systems the evaluation boundary includes the connections to or around piping systems, bracing and framing for tanks, pumps and skids, etc. Component Supports provide the connection between a system’s equipment or component and a plant structural member (e.g., wall, floor, ceiling, column, beams, etc.). They provide support for distributed loads (e.g., piping, tubing, HVAC ducting,) and localized loads (e.g., individual equipment). Pipe restraints, consisting of failure restraints and seismic restraints, limit pipe movement during postulated events.

For electrical systems the evaluation boundary includes the connections to raceways, cable trays and conduits. The evaluation boundary also includes the raceways, cable trays, and conduits as well as racks, panels or enclosures which house or support system components within the scope of license renewal. Raceways and cable trays identify a general component type that is designed specifically for holding electrical wires and cables. Like mechanical system supports, electrical supports provide support for distributed loads (e.g., cable trays, raceways, conduit) and localized loads (e.g., individual equipment, cabinets, junction boxes, etc.).

Only seismically analyzed supports for system piping greater than or equal to four inches is uniquely field labeled and tracked in the plant database. These supports are not all inclusive of the supports that are in-scope to License Renewal. Because of the difficulty in uniquely distinguishing supports, all supports for safety related equipment and all supports for any equipment contained within a safety related structure, regardless of the equipment’s seismic classification, shall be considered in-scope to License Renewal unless a support is specifically excepted and that exception documented. Additionally, other structures also house and support equipment that is in-scope to the rule. Component supports for those equipment sets are also in-scope to License Renewal.

System Function Listing

In addition to the System Functions listed above, the Component Support Commodity Group also supports additional functions (associated design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code R STRUCTURALLY SUPPORT OR HOUSE SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Component Supports commodity group perform this primary design system function. Component Supports commodity group provides structural support, including the fasteners and anchorages, for safety related system piping, ventilation ducting, raceways, and equipment. This commodity group also contains electrical enclosures and raceways that can house safety related system electrical components.

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Component Supports commodity group perform this associated design system function. Component Supports commodity group provides structural support to non-safety related system piping, ventilation ducting, raceways, and equipment, whose failure would not prevent satisfactory accomplishment of safety related functions.

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSCS WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Component Supports commodity group perform this associated design system function. Component Supports commodity group provides structural support, including the fasteners and anchorages, for non-safety related system piping, ventilation ducting, raceways, and equipment, whose failure could prevent satisfactory accomplishment of safety related functions.

UFSAR Reference

Additional Component Supports Commodity Group details are provided in Section 3.9.2, Section 3.9.3, and Section 3.10 of the UFSAR.

Components/Commodities Subject to an AMR

The component groups for the Component Supports Commodity Group that require aging management review are indicated in Table 2.4.2.12 along with each component group's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.4.2-12 Component Supports Commodity Group

Component Group	Passive Function	Aging Management Reference
CSUPP-AL-INT This generic asset includes aluminum alloy electrical conduit and conduit supports that are not exposed to the weather.	STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-2 Line Number (1) This applies to both passive functions.
CSUPP-ASME(CS)-EXT This generic asset includes all structural carbon steel used in NSSS pipe and component supports that is outdoors (i.e., exposed to the weather).	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (25) Table 3.6-1 Line Number (28)
CSUPP-ASME(CS)-INT This generic asset includes all structural carbon steel used in NSSS pipe and component supports that is indoor (i.e., protected from the weather).	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (25) Table 3.6-1 Line Number (27) Table 3.6-1 Line Number (28)

Table 2.4.2-12 Component Supports Commodity Group

Component Group	Passive Function	Aging Management Reference
CSUPP-ELAST-INT This generic asset includes all elastomer (e.g., vibration isolator equipment mounts) that is indoor (i.e., protected from the weather). Also included in this evaluation are cabinet door seals, gaskets, and other seals. Fire barrier sealing material is evaluated as a separate commodity group.	STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (25) Table 3.6-2 Line Number (5) These apply to both passive functions.
CSUPP-EXP(CS)-EXT This generic asset includes all carbon steel expansion/grouted anchors that are exposed to the weather.	STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (25) This applies to both passive functions.
CSUPP-EXP(CS)-INT This generic asset includes all carbon steel expansion/grouted anchors that are not exposed to the weather.	STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (25) Table 3.6-1 Line Number (27) These apply to both passive functions.
CSUPP-EXP(SS)-RW This generic asset includes all stainless steel expansion/grouted anchors that are submerged in raw water.	STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (25) This applies to both passive functions.

Table 2.4.2-12 Component Supports Commodity Group

Component Group	Passive Function	Aging Management Reference
CSUPP-FAST(CS)-EXT This generic asset includes all carbon steel structural fasteners (e.g., bolts, studs, nuts) that are exposed to the weather.	STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (25) This applies to both passive functions.
CSUPP-FAST(CS)-INT This generic asset includes all carbon steel structural fasteners (e.g., bolts, studs, nuts) that are indoor (i.e., protected from the weather). Indoor air is considered to be non-air conditioned (bounding condition), even though some fasteners are within boundaries for air conditioned areas (e.g., Control Building).	STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (25) Table 3.6-1 Line Number (27) These apply to both passive functions.
CSUPP-FAST(HSLAS)-INT This generic asset includes all high strength carbon steel structural fasteners (e.g., bolts, studs, nuts whose yield strength is greater than 150 ksi) that are indoor (i.e., protected from the weather) (e.g., used for selected electrical enclosures and a limited number of structural steel component supports).	STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (25) Table 3.6-1 Line Number (27) Table 3.6-1 Line Number (29) These apply to both passive functions.

Table 2.4.2-12 Component Supports Commodity Group

Component Group	Passive Function	Aging Management Reference
CSUPP-FAST(SS)-RW This generic asset includes all stainless steel fasteners (e.g., bolts, studs, nuts, etc.) that are submerged in raw water.	STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (25) This applies to both passive functions.
CSUPP-G-INT This generic asset includes all grout used in expansion/grouted anchors within the plant (expansion/grouted anchors include Hilti bolts, but do not include Drillco Maxi-Bolts). Also included in this evaluation is grout used as a component support under equipment bases. Grout used as a fire barrier is evaluated as a separate commodity group.	STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (25) This applies to both passive functions.
CSUPP-SS(CS)-EXT This generic asset includes all structural carbon steel (e.g., plates, channels, support members) that is exposed to the weather.	STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (25) This applies to both passive functions.

Table 2.4.2-12 Component Supports Commodity Group

Component Group	Passive Function	Aging Management Reference
CSUPP-SS(CS)-INT This generic asset includes all structural carbon steel (e.g., plates, beams, support members) that is indoor (i.e., protected from the weather). Indoor air is considered to be non-air conditioned (bounding condition), even though some steel surfaces are within boundaries for air conditioned areas (e.g., Control Building). Also included in this evaluation are electrical enclosure drip guards and spray shields	STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (25) . Table 3.6-1 Line Number (27) These apply to both passive functions.
CSUPP-SS(SS)-RW This generic asset includes all structural stainless steel (e.g., plates, beams, support members) that is submerged in raw water.	STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (25) This applies to both passive functions.
CSUPP-SURFACE-ELAST-EXT This generic asset includes all non-metallic materials used in NSSS pipe and component supports that is outdoors (i.e., exposed to the weather).	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (28)

Table 2.4.2-12 Component Supports Commodity Group

Component Group	Passive Function	Aging Management Reference
CSUPP-SURFACE-ELAST-INT This generic asset includes all non-metallic materials used in NSSS pipe and component supports that is indoors (i.e.,protected from the weather).	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (28)
CSUPP-SURFACE-METAL-EXT This generic asset includes all metallic surfaces used in NSSS pipe and component supports that is outdoors (i.e., exposed to the weather).	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (28)
CSUPP-SURFACE-METAL-INT This generic asset includes all metallic surfaces used in NSSS pipe and component supports that is indoors (i.e., protected from the weather).	STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-1 Line Number (27) Table 3.6-1 Line Number (28)
CSUPP-WOOD-INT This generic asset includes wood used in electrical cable support spacers and the Intermediate Building supply fan support frame.	STRUCTURAL SUPPORT NSR EQUIPMENT STRUCTURAL SUPPORT SR EQUIPMENT	Table 3.6-2 Line Number (11) This applies to both passive functions.

2.4.3 Non-Essential Buildings and Yard Structures - Not within Scope of License Renewal

Description

Within the site boundary there are numerous buildings, structures and yard components that are necessary to support the commercial operation of the facility. These civil features differ than those evaluated in the other building and yard component groups in that they do not perform License Renewal intended functions nor do they have any failure modes or effects that prevent an intended function from being performed. Accordingly, components within the Non-Essential Buildings, Structures and Yard Components group do not perform any License Renewal intended functions.

The Non-Essential Buildings, Structures and Yard Components group is a listing of the major civil components found onsite but not included within any other License Renewal review boundary. (Note: Above and below grade tanks are evaluated within the mechanical system they serve. Some external tanks have dikes and spill abatement features included in this list.) A review of the Update Final Safety Analysis Report (UFSAR), other current licensing basis documents, as well as field verifications were performed to ensure that these civil features do not meet the criteria to be considered in-scope to License Renewal.

Structures and Components evaluated in this grouping include:

a) Nuclear Engineering Services Building located on the northwest side of site boundary and used for office space (also contains the Emergency Plan Engineering Support Facility).

b) Miscellaneous Storage Building located east of Engineering Building and use for equipment and spare parts storage.

c) Office Trailers at various locations around the site used for office space and storage.

d) Steam Generator Facilities Building located northeast of plant used as office space (the building was previously used for Steam Generator repair training).

e) Radwaste Storage Building located northeast of plant used for contaminated waste storage.

f) Sodium Hypochlorite Tank located north of plant and east of Screenhouse used for chemical storage for secondary water treatment. Tank has a spill containment dike.

g) Ammonia Storage Tank located north of plant and south of Screenhouse used for chemical storage for secondary water treatment. Tank has a spill containment dike.

h) Roadways, Paths, and Sidewalks located variously around the site used for personnel and equipment access/egress.

i) Contaminated Storage Building located south of Auxiliary Building used for personnel and equipment access/egress.

j) Guard House located on the south side of site boundary, centerline, used for personnel access/egress point.

k) Warehouse / Construction Office (Butler Building) located west of plant used for office space, the wellness center and the machine shop.

l) Miscellaneous Storage Building located south of Engineering Building used for equipment and spare parts storage.

m) Off Load Portal located west of Guardhouse on the south site boundary used as a shipping transfer point from offsite warehouse into the secure area.

n) Hydrogen Building located south of Auxiliary Building contains Hydrogen and Nitrogen bottled gas for the Volume Control Tank.

o) High Mast Lighting located in variously throughout the site used for security Lighting.

p) Security Fences and Structures at various locations around the site parameter used for site access control.

q) Storm Drainage Structures various located throughout the site used for ground water runoff control.

r) Lube Oil Storage Building located north of Turbine Building used for Oil storage.

s) Hydrogen Bottle House located north of Turbine Building contains hydrogen and carbon dioxide bottled gas used for the main electrical generator.

t) High Integrity Container (HIC) storage facility located west of the Radwaste Storage Building used for shielding for containerized spent resin prior to shipment.

u) Old Steam Generator Storage facility located northwest of the plant outside the security fence. This facility houses the old steam generators and is designed for long-term storage.

System Function Listing

A comprehensive listing of all functions associated with the Non-Essential Buildings, Structures and Yard Components, or specific components contained in the system, is provided within the summary below.

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Non-Essential Buildings and Yard Structures perform this associated design system function.

UFSAR Reference

Additional Non-Essential Buildings, Structures and Yard Components details are provided in Section 1.2.3 of the UFSAR.

2.5 Screening Results: Electrical and Instrumentation and Controls Systems

As stated in Section 2.1.7.4, the electrical and I&C components have been screened and evaluated on a plant-wide basis rather than on a system basis. The generic list of long-lived, passive components discussed in Section 2.1.7.4. The revised list is provided below:

- Medium Voltage Insulated Cables and Connections
- Low Voltage Insulated Cables and Connections
- Electrical Portions of Electrical and I&C Penetration Assemblies
- Electrical Phase Bus
- Switchyard Bus
- Transmission Conductors
- Uninsulated Ground Conductors
- High Voltage Insulators

Each component commodity group is described below. Following the component commodity group descriptions is electrical and I&C system descriptions. There is no technical or regulatory requirement to perform plant system level scoping on the electrical and I&C systems. Although no technical benefit is derived from performing plant system level electrical scoping, system information is included in the License Renewal Application for consistency and flow of the application and is intended to aid a reviewer in understanding electrical system functions, boundaries, interfaces and nomenclature. Reviewers are cautioned to understand that the system level descriptions presented in this section were not required, needed, or used in the performance of the electrical and I&C components aging management reviews. The objective of the information contained in the system component group tables is to provide a cross-reference between plant electrical and I&C systems and the commodity groups that receive aging management. Each system description contains a table listing the associated component commodity group.

2.5.1 Commodity Group Descriptions

Medium Voltage Insulated Cables and Connections

An insulated cable is an assembly of a single electrical conductor (wire) with an insulation covering or a combination of conductors insulated from one another. Medium voltage cables operate between 1000V and 15000V and are normally shielded. Power cables at Ginna Station operate at a nominal voltage of 4160V. Connections (or terminations) are used to connect the cable conductors to other cables or electrical devices. Connections include, but are not limited to; mechanical connections, splices, terminal blocks and fuse holders not included within other assemblies. The License Renewal intended function of these components is to provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals. The aging management review of medium voltage cables and connections is performed in application Section 3.7.

Low Voltage Insulated Cables and Connections

An insulated cable is an assembly of a single electrical conductor (wire) with an insulation covering or a combination of conductors insulated from one another. Low voltage cables operate at voltage levels below 600 VAC. This includes power, instrumentation, control, and communications cables. Connections (or terminations) are used to connect the cable conductors to other cables or electrical devices. Connections include, but are not limited to; mechanical connections, splices, terminal blocks and fuse holders not included within other assemblies. The License Renewal intended function of these components to provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals. The aging management review of low voltage cables and connections is performed in application Section 3.7.

Electrical Penetration Assemblies

Electrical penetration assemblies consist of one or more electrical conductors and a pressure boundary between the inboard and outboard sides of the penetration capable of maintaining electrical continuity through the boundary. These penetrations are used to transmit electrical power and signals, through the containment wall. The License Renewal intended function of these components to provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals. The review of the Electrical Penetration Assemblies does not include the pressure boundary function, which is discussed in the containment structural review. All primary containment electrical penetration assemblies at Ginna Station are included in the scope of the environmental qualification program (10 CFR 50.49). The Time-Limited Aging Analysis of Electrical Penetration Assemblies is performed in application Section 4.4.3.

Electrical Phase Bus

Phase bus consists of rigid electrical conductors that are enclosed within their own enclosure or vault, and are not part of an active component such as switchgear, load center or motor control center. Phase bus is discussed as three distinct types: isolated-phase bus, non-segregated phase bus, and segregated phase bus. Only non-segregated phase bus is within the scope of License Renewal at Ginna Station. This includes the 480V diesel generator bus and the portions of the 4.16 KV bus that provide a normal source of power for the 480V Class 1E power system. The License Renewal intended function of these components to provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals. The aging management review of the electrical phase bus is performed in application Section 3.7.

Switchyard Bus

Switchyard bus is uninsulated, unenclosed, rigid electrical conductor used in switchyards and switching stations to connect two or more elements of an electrical power circuit such as disconnect switches and transmission conductors. The switchyard bus at Ginna station is used to distribute 34.5KV power from the offsite power circuits (751 and 767) to oil circuit breakers and then to the station auxiliary transformers. The License Renewal intended function of these components is to provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals. The aging management review of the switchyard bus is performed in application Section 3.7.

Transmission Conductors

Transmission conductors are uninsulated, stranded electrical cables used in switchyards, switching stations and transmission lines to connect two or more elements of an electrical power circuit such as disconnect switches, power circuit breakers, transformers, and switchyard bus. At Ginna station, transmission conductors are used to supply the onsite 34.5 KV switchyard from multiple offsite sub-stations. These components are not included in the license renewal boundary because they are primarily located off-site. Therefore an aging management review of transmission conductors is not required.

Uninsulated Ground Conductors

Uninsulated ground conductors are electrical conductors (e.g., copper cable, copper bar, steel bar) that are uninsulated (bare) and are used to make ground connections for electrical equipment. Uninsulated ground conductors are connected to electrical equipment housings and electrical enclosures as well as metal structural features such as the cable tray system and building structural steel. Uninsulated ground conductors are always isolated or insulated from the electrical operating circuits. Uninsulated ground conductors enhance the capability of the electrical system to withstand electrical system faults disturbances (e.g., electrical faults, lightening surges) for equipment and personnel protection. There are no License Renewal intended functions for uninsulated ground conductors used at Ginna Station.

High Voltage Insulators

An insulator is a material in a form designed to (a) support a conductor physically and (b) separate the conductor electrically from another conductor or object. The insulators evaluated for License Renewal are those used to support and insulate “high voltage” electrical components in switchyards, switching stations and transmissions such as transmission conductors and switchyard bus. The License Renewal intended function of these components is to insulate and support electrical conductors. The aging management review of high voltage insulators is performed in application Section 3.7

2.5.2 120 VAC Vital Instrument Buses - Within Scope of License Renewal

System Description

The 120 VAC Vital Instrument Bus System is a source of uninterruptible power to safety related components. In addition to providing power to reactor protection and engineered safety features actuation equipment, the system provides power to components used for safe shutdown following fire and Station Blackout Events. The vital instrument buses also provide power to instrumentation used to detect and mitigate Anticipated Transients without Scram.

The principal components of the 120 VAC Vital Instrument Bus system are four instrument buses, three inverters, two static switches/ regulating transformers, three constant voltage transformers, eight distribution panels and the corresponding essential breakers, fuses, metering, relays and wiring. The A and C buses normally receive power from inverters powered from the vital batteries, while the B and D buses receive power from constant voltage transformers connected to separate 480 VAC buses. Thus Instrument Buses A and C have two power sources, with automatic transfer from the primary to backup supply. Instrument Buses A, B, and C provide power to vital plant instrumentation. All three buses are backed up by safety related emergency supplies. Mechanical interlocks prevent the inadvertent paralleling of the instrument buses. In addition to the four instrument buses, inverter MQ-483 supplies power to one channel each of containment pressure and steam generator pressure (PT-950 and PT-479).

Included within the system review boundary is selected 120 VAC non-vital power distributed from the technical support center uninterruptible power supply. This power is used for the PPCS/SAS computer systems which do not perform any control functions. This distribution system is electrically diverse from the 120 VAC Vital Instrument Buses but was included for review to ensure devices associated with the plant computer system are accounted for.

The following electrical systems interface with the 120 VAC Vital Instrument Buses:

Reactor Protection System	Engineered Safety Feature Actuation System
125 VDC System	4160 VAC Power
480 VAC Power	

System Function Listing

In addition to the System Functions described above, the 120 VAC Vital Instrument Buses also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code Q PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the 120 VAC Vital Instrument Buses system perform this primary system function.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the 120 VAC Vital Instrument Buses system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the 120 VAC Vital Instrument Buses system that perform special capability class functions are tracked under the Criterion 3 codes (Z1 through Z5). Included within the system boundary is non-vital 120 VAC power (TSC) providing power to RG 1.97 Category 2 and 3 Post Accident Monitoring variables and displays (e.g., SAS/PPCS).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the 120 VAC Vital Instrument Buses system perform this associated system function.

Code X SFR FUNCTION NOT APPLICABLE AT COMPONENT LEVEL	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the 120 VAC Vital Instrument Buses system perform specific safety related functions different from and in addition to the system level functions (e.g. electrical isolation devices protecting safety related sources from non-safety loads).

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the 120 VAC Vital Instrument Buses system perform this associated system function.

Code Z4 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ANTICIPATED TRANSIENTS WITHOUT SCRAM (10 CFR 50.62)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
						X	

Comment: Components within the 120 VAC Vital Instrument Buses system perform this associated system function.

Code Z5 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR STATION BLACKOUT (10 CFR 50.63)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: Components within the 120 VAC Vital Instrument Buses system perform this associated system function.

UFSAR Reference

Additional 120 VAC Vital Instrument Buses system details are provided in Section 8.3.1.1.5 and Section 8.3.1.2.5 of the UFSAR.

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.5.2-1 along with each components passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.5.2-1 120 VAC Vital Instrument Buses

Component/Commodity Group	Passive Function	Aging Management Reference
Low Voltage Insulated Cables and Connections	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals	Table 3.7-1 Line Number (2) Table 3.7-1 Line Number (3) Table 3.7-1 Line Number (5)

2.5.3 125 VDC Power - Within Scope of License Renewal

System Description

The 125 VDC Power system provides direct current (dc) electrical power to safety related components. Equipment contained within the system boundary is used for safe shutdown following fire and Station Blackout events. Additionally, some components within the system boundary are Environmentally Qualified. The system also supplies non-safety power to Anticipated Transient-Without- Scram (ATWS) mitigation system actuation circuitry.

The principal components of the 125 VDC Electrical System are four station battery chargers, one technical support center battery charger, two station batteries, one technical support center battery, fuse cabinets, distribution panels, switches, fuses, relays and the essential wire and cabling. The 125-V dc system is divided into two buses with one battery and two battery chargers (supplied from the 480-V system) serving each. The battery chargers supply the normal dc loads as well as maintaining proper charges on the batteries. Two 60-cell batteries are provided for power supply for control, emergency lighting, and the inverters for critical 60-cycle instrument power. Control power for 4160-V and 480-V switchgear sections and for each diesel-generator can be supplied from either battery.

Two vital batteries provide separate sources of dc power. The train A engineered safety features equipment is supplied from battery A while train B engineered safety features equipment is supplied from battery B. In addition, the 480-V engineered safety features switchgear and diesel-generator control panels are supplied from either battery by means of an automatic transfer circuit in the switchgear and control panels. The normal supply from train A (switchgear buses 14 and 18 and diesel generator A) is from dc distribution panels A in the auxiliary building, diesel generator building, and screen house. These panels also provide the emergency dc supply for train B (switchgear buses 16 and 17 and diesel generator B). Similarly, dc distribution panels 1B in the auxiliary building, diesel generator building and screen house provide the normal supply for switchgear buses 16 and 17 and diesel generator B and the emergency supply for switchgear buses 14 and 18 and diesel generator A. In the event of loss of the normal Class 1E battery supply, throwover contactors automatically transfer the load to the emergency supply (other battery). The DC system also has provisions to ensure that equipment required for post fire alternative safe shutdown are supplied appropriately and that the power supply can be maintained for the required time period.

A third 60-cell non-vital battery is installed in the technical support center. This battery and charger supply power to the uninterruptible power supply to the plant process computer. The uninterruptible power supply is designed to provide continuous power for up to 3 hours during loss of its normal power supply and failure of the technical support center diesel generator to start. It also supplies dc power to the anticipated transient without scram (ATWS) mitigation system actuation circuitry (AMSAC) inverter. The technical support center battery is capable of supplying both safeguards dc trains in the event of an emergency. The system is designed with an intertie between each of the two main distribution panels and the technical support center panel so that either Class 1E battery and its chargers can be removed from service. This intertie is utilized only during maintenance, testing, or abnormal plant conditions. The intertie is also configured so both Class 1E-battery systems can be paralleled simultaneously through the technical support center battery. Procedures permit this condition only during specific 10 CFR 50 Appendix R conditions in which some process instrumentation from both trains is required for long-term cooldown. Paralleling both safety-related dc trains is restricted by two separate key locks on the throwover switches and separate locked disconnect switches in each battery room.

The following Electrical Systems interface with the 125 VDC Power System:

Reactor Protection	480 VAC Power
4160 VAC Power	120 VAC Vital Instrument Buses
Plant Communications	Misc. AC Power and Lighting
Plant Computer	Engineered Safety Features Actuation
Emergency Power	Control Rod Drive and Nuclear Process Instrumentation

System Function Listing

In addition to the System Functions described above, the 125 VDC Power System also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code Q PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the 125 VDC Power system perform this primary design system function.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the 125 VDC Power system perform this associated design system function (augmented quality, e.g. RG 1.97 Category 2 and 3 Post Accident Monitoring variables). For the purposes of License Renewal, components within the 125 VDC Power system that perform License Renewal Criterion 3 functions are tracked under the Criterion 3 codes (Z1 through Z5).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the 125 VDC Power system perform this associated design system function.

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the 125 VDC Power system perform this associated design system function.

Code Z4 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ANTICIPATED TRANSIENTS WITHOUT SCRAM (10 CFR 50.62)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
						X	

Comment: Components within the 125 VDC Power system perform this associated design system function.

Code Z5 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR STATION BLACKOUT (10 CFR 50.63)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: Components within the 125 VDC Power system perform this associated design system function.

UFSAR Reference

Additional 125 VDC Power System details are provided in Section 8.3.2 and Section 7.2.6 of the UFSAR.

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.5.3-1 along with each components passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.5.3-1 125 VDC Power

Component/Commodity Group	Passive Function	Aging Management Reference
Low Voltage Insulated Cables and Connections	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals	Table 3.7-1 Line Number (2) Table 3.7-1 Line Number (5)

2.5.4 **4160 VAC Power- Within Scope of License Renewal**

System Description

The 4160 VAC power system consists of four buses that are classified as non-Class 1E. Discrete relaying is used to provide overcurrent, undervoltage, and underfrequency protection as required. The two buses connected to the No. 11 transformer supply normal plant auxiliary loads (non-Class 1E) and are designated 11A and 11B. The two buses connected to the offsite system through transformers 12A and 12B supply the startup power and also feed the Class 1E loads on the 480-V safeguards system through four station service transformers. Ginna Stations licensing basis does not rely on the recovery of offsite power to mitigate Station Blackout (SBO) events. License Renewal regulatory guidance also mandates the inclusion of power system SSCs used for SBO recovery beyond those identified in the regulatory commitments made to satisfy 10 CFR 50.63 criteria. In accordance with the current License Renewal regulatory interpretation, 4160 VAC Power systems and structures that provide a function for SBO coping and systems or structures that provide a function for recovery from an SBO condition are being evaluated within the system boundary.

The principal components of the 4160 VAC System are four 4160 Volt supply buses, six station service transformers, breakers, relays and the essential cables and wiring. The buses in this system are 11A, 11B, 12A, 12B. The six station service transformers are numbers 13,14,15,16,17 and 18. Buses 11A and 11B are connected to the generator leads via bus main breakers and the unit auxiliary transformer. Buses 11A and 12A or buses 11B and 12B can be tied together via bus tie breakers. When off-line, a tie breaker is also supplied between buses 11A and 11B, which may be closed under administrative control so as to perform certain maintenance activities. All 4160-V auxiliaries except condensate booster pump 1A are split between buses 11A and 11B. In addition, buses 11A and 11B each serve one 4160/480-V station service transformer. Buses 12A and 12B each serve two 4160/480-V station service transformers. Bus 12A also feeds condensate booster pump 1A. Buses 11A and 11B are provided with solid-state underfrequency relays and undervoltage relays to provide protection against a loss-of-flow transient. The relays are set to give a reactor trip before decreasing bus frequency can degrade primary system flow below the level assumed in the steady-state or transient analyses.

The following electrical systems interface with the 4160 VAC Power System:

Reactor Protection	480 VAC Power
125 VDC Power	Offsite Power

System Function Listing

In addition to the System Functions described above, the 4160 VAC Power system also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the 4160 VAC Power system perform this associated design system function (augmented quality, e.g. components within the system provide non-nuclear safety source trip signals to the Reactor Protection system). For the purposes of License Renewal, components within the 4160 VAC Power system that perform License Renewal Criterion 3 functions are tracked under the Criterion 3 codes (Z1 through Z5).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the 4160 VAC Power system perform this associated design system function.

Code Z5 LICENSE RENEWAL CRITERION 3 - SSCS RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION’S REGULATIONS FOR STATION BLACKOUT (10 CFR 50.63)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: Ginna Stations licensing basis does not rely on the recovery of offsite power to mitigate Station Blackout (SBO) events. That notwithstanding, recent regulatory guidance suggests the scope of SBO mitigation equipment be expanded. Accordingly, components within the 4160 VAC power system have been evaluated with regard to SBO.

UFSAR Reference

Additional 4160 VAC Power System details are provided in Section 8.3 of the UFSAR.

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.5.4-1 along with each components passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.5.4-1 4160 VAC Power

Component/Commodity Group	Passive Function	Aging Management Reference
Medium Voltage Insulated Cables and Connections	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	Table 3.7-1 Line Number (2) Table 3.7-1 Line Number (3) Table 3.7-1 Line Number (5)
Low Voltage Insulated Cables and Connections	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	Table 3.7-1 Line Number (2) Table 3.7-1 Line Number (3) Table 3.7-1 Line Number (5)
Phase Bus	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	Table 3.7-2 Line Number (1)

2.5.5 **480 VAC Power - Within Scope of License Renewal**

Description

The 480-VAC System provides electrical power to safety related and non-safety related loads during both normal and emergency modes of plant operation. The 480-VAC System is also credited for use in safe shutdown following fires and contains components that are part of the Environmental Qualification Program. Restoration of the 480-VAC system is the means by which the plant recovers from Station Black Out events. The 480 VAC Power system also contains non safety related relays, fuses or supplementary breakers used to provide backup protection schemes such that non safety circuits will not have failures that could prevent satisfactory accomplishment of a safety related function (containment penetration integrity).

The principal components of the 480-VAC system include buses, breakers, starter/controllers, relays, transformers, switches, and the essential cables, wires, and connectors necessary for their functioning.

The 480-VAC system is divided into separate electrical distribution buses. Each bus is normally supplied by a separate 4160/480-V station service transformer. Two Class 1E independent trains provide the necessary redundancy for safety related electrical loads on the 480-VAC safeguards system. Each safeguard 480-VAC train consists of two buses. The plant equipment is arranged electrically so that multiple items receive their power from the two different sources. The electrical system equipment is arranged so that no single contingency can inactivate enough engineered safety features equipment to jeopardize the plant safety.

Each safeguards bus has undervoltage detection. Activation of the undervoltage detection system will result in bus load shedding and an associated onsite Emergency Power generator start and bus connection. In the event of a loss of offsite power, or abnormal offsite power, the emergency generators are started concurrent with load shedding. When the generators come up to speed and close onto the buses, the undervoltage relays reset, thus allowing the operator to manually load any of the motors that are required. Some loads may also be automatically loaded onto the bus. The automatic load sequencer is not activated unless a safety injection signal is present.

Additional non-class 1E 480-VAC buses provide power to non-safety related electrical loads. Tie breakers are provided between 480-VAC buses to provide the cross-connect capability necessary for maintenance purposes.

The following electrical systems interface with the 480 VAC System:.

120 VAC Vital Instrument Buses	Misc. AC Power and Lighting
125 VDC Power	4160 VAC Power
Offsite Power	

System Function Listing

In addition to the System Functions described above, the 480-VAC System also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code B	Cri 1	Cri 2	Cri 3				
INTRODUCE EMERGENCY NEGATIVE REACTIVITY TO MAKE THE REACTOR SUBCRITICAL			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the 480 VAC Power system perform this associated design system function.

Code D	Cri 1	Cri 2	Cri 3				
SENSE OR PROVIDE PROCESS CONDITIONS AND GENERATE SIGNALS FOR REACTOR TRIP AND ENGINEERED SAFETY FEATURES ACTUATION			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the 480 VAC Power system perform this associated design system function (MCC load shed).

Code F	Cri 1	Cri 2	Cri 3				
REMOVE RESIDUAL HEAT FROM THE RCS			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the 480 VAC Power system perform this associated design system function.

Code G	Cri 1	Cri 2	Cri 3				
PROVIDE EMERGENCY CORE COOLANT WHERE THE ECCS PROVIDES COOLANT DIRECTLY TO THE CORE			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the 480 VAC Power system perform this associated design system function.

Code H	Cri 1	Cri 2	Cri 3				
PROVIDE EMERGENCY HEAT REMOVAL FROM THE REACTOR COOLANT SYSTEM USING SECONDARY HEAT REMOVAL CAPABILITY			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the 480 VAC Power system perform this associated design system function.

Code J	Cri 1	Cri 2	Cri 3				
PROVIDE HEAT REMOVAL FROM SAFETY RELATED HEAT EXCHANGERS			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the 480 VAC Power system perform this associated design system function.

Code L	Cri 1	Cri 2	Cri 3				
PROVIDE EMERGENCY HEAT REMOVAL FROM PRIMARY CONTAINMENT AND PROVIDE CONTAINMENT PRESSURE CONTROL			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the 480 VAC Power system perform this associated design system function.

Code M	Cri 1	Cri 2	Cri 3				
PROVIDE EMERGENCY REMOVAL OF RADIOACTIVE MATERIAL FROM THE PRIMARY CONTAINMENT ATMOSPHERE			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the 480 VAC Power system perform this associated design system function.

Code O	Cri 1	Cri 2	Cri 3				
MAINTAIN EMERGENCY TEMPERATURES WITHIN AREAS CONTAINING SAFETY CLASS 1, 2, 3 COMPONENTS			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the 480 VAC Power system perform this associated design system function.

Code P	Cri 1	Cri 2	Cri 3				
ENSURE ADEQUATE COOLING IN THE SPENT FUEL POOL			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the 480 VAC Power system perform this associated design system function.

Code Q	Cri 1	Cri 2	Cri 3				
PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the 480 VAC Power system perform this primary design system function.

Code S	Cri 1	Cri 2	Cri 3				
SPECIAL CAPABILITY CLASS FUNCTIONS			FP	EQ	PTS	AT	SB

Comment: Components within the 480 VAC Power system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the 480 VAC Power system that perform License Renewal Criterion 3 functions are tracked under the Criterion 3 codes (Z1 through Z5).

Code T	Cri 1	Cri 2	Cri 3				
NON-NUCLEAR SAFETY CLASS FUNCTIONS			FP	EQ	PTS	AT	SB

Comment: Components within the 480 VAC Power system perform this associated design system function.

Code X	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
SFR FUNCTION NOT APPLICABLE AT COMPONENT LEVEL	X						

Comment: Components within the 480 VAC Power system perform specific safety related functions different from and in addition to the system level functions (e.g. electrical isolation devices protecting safety related sources from non-safety loads).

Code Y	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION		X					

Comment: Components within the 480 VAC Power system perform this associated design system function. (In addition to the circuit protection afforded by safety related isolation devices, circuits serving some non safety related loads have been modified to add relays, fuses or supplementary breakers to provide backup protection schemes. This was done to ensure containment penetrations are not affected by possible fault currents should the non safety load fail.)

Code Z1	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)			X				

Comment: Components within the 480 VAC Power system perform this associated design system function.

Code Z2	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ENVIRONMENTAL QUALIFICATION (10 CFR 50.49)				X			

Comment: Components within the 480 VAC Power system are designated as Environmentally Qualified.

UFSAR Reference

Additional 480 VAC Power system details are provided in Section 8.3.1.1.4 of the UFSAR.

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.5.5-1 along with each components passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.5.5-1 480 VAC Power

Component/Commodity Group	Passive Function	Aging Management Reference
Low Voltage Insulated Cables and Connections	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	Table 3.7-1 Line Number (2) Table 3.7-1 Line Number (3) Table 3.7-1 Line Number (5)
Phase Bus	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	Table 3.7-2 Line Number (1)

2.5.6 Control Rod Drive and Nuclear Process Instrumentation - Within Scope of License Renewal

Description

The Control Rod Drive and Nuclear Process Indication System evaluation boundary includes Rod Control, Rod Position indication, in-core and ex-core Nuclear Instrumentation, and Core Exit Thermocouples. The ex-core Nuclear Instrumentation portion of the system generates signals used for reactor trip and indication used for safe shutdown following fire events and Station Blackout events. Core Exit Thermocouples are used for Station Blackout Events and are Environmentally Qualified. Note: This system has an interface with the Reactor Coolant Pressure boundary. The mechanical components that provide that interface are not within the boundary of the Control Rod Drive and Nuclear Process Indication System. For information relating to the Reactor Coolant Pressure Boundary see the Class 1 Reactor Coolant System and Non-Class 1 RCS Components system evaluations.

The principal components of the Control Rod Drive and Nuclear Process Indication System include rod drive motor-generators, rod control power supplies, magnetic latch and step assemblies, breakers, thermocouples, relays, power supplies, isolators, indicators, neutron detectors, display modules, switches, and the essential wiring.

The control rod drive mechanisms are used for withdrawal and insertion of the control rods into the reactor core and to provide sufficient holding power for stationary support. Fast total insertion (reactor trip) is obtained by simply removing the electrical power allowing the rods to fall by gravity. The complete drive mechanism, consists of the internal (latch) assembly, the pressure vessel, the operating coil stack, the drive shaft assembly, and the position indicator coil stack. Each assembly is an independent unit which can be dismantled or assembled separately. Reactor coolant fills the pressure containing parts of the drive mechanism. All working components and the shaft are immersed in the main coolant. Three magnetic coils, which form a removable electrical unit and surround the rod drive pressure housing, induce magnetic flux through the housing wall to operate the working components. They move two sets of latches which lift or lower the grooved drive shaft. The three operating coils are sequenced by solid-state switches for the control rod drive assemblies. The position indicator coil stack slides over the rod travel housing section of the pressure vessel. It detects drive rod position by means of discrete cylindrically wound coils that are spaced at 7.5 in. (12 step) intervals along the rod travel (144 in.). The rod control system is a solid-state electronic control system that moves and holds the control rods according to system input orders. The rod drive mechanism is an electromagnetic stepping type mechanism with three actuating coils for holding and movement. To hold a control rod, the system keeps a gripper coil energized. To move a rod, the system sequentially energizes and deenergizes the three coils causing the rod to move in discrete steps. In automatic control the rod control system maintains a programmed reactor coolant average temperature with adjustments of control rod position for equilibrium plant conditions. The reactor control system is capable of restoring programmed average temperature following a scheduled or transient change in load. The coolant average temperature increases linearly from zero power to the full power conditions. In manual control the operator maintains control of the reactor through bypassing the reactor control unit. By using the bank selector and the IN-HOLD-OUT switches the operator can move the rods either by individual banks or in manual with bank overlap.

Two separate systems are provided to sense and display control rod position. The microprocessor rod position indication (MRPI) system consists of a digital detector assembly for each rod, a data cabinet located inside containment, and display racks located in the relay room. Rod position data is displayed on a color cathode ray tube

(CRT) in the control room and also transmitted to the plant process computer system. The data cabinet inside containment contains two multiplexers, which take rod position information from each of the rods and transmit it to the processors, which are in the display racks located in the relay room. One processor supplies information to the CRT located on the control board, the other processor supplies information to the plant process computer system. Both processors are required to produce a block rod withdrawal signal. The plant process computer system backup can be used if the CRT in the MRPI system becomes inoperable. The digital system counts pulses generated in the rod drive control system. One counter is associated with each group of rods within a bank, making a total of 10 for the four control banks and one shutdown bank. Readout of the digital system is in the form of electromechanical add-subtract counters reading the number of steps of rod withdrawal with one display for each. These readouts are mounted on the control panel.

Thirty-nine chromel-alumel thermocouples are threaded into guide tubes that penetrate the reactor vessel head through seal assemblies (36 terminate at the exit flow end of the fuel assemblies and three are located in the upper head). The thermocouples are enclosed in stainless steel sheaths within the above tubes to allow replacement if necessary. Thermocouple readings are indicated in the control room on scanning digital display units.

The movable detector flux monitoring system has miniature neutron flux detectors are remotely positioned in the core and provide remote readout for flux mapping. Retractable thimbles are provided into which the miniature detectors are driven. Four movable detectors are provided, with separate drives and readouts. This allows four locations to be monitored simultaneously. The control room contains the necessary equipment for control, position indication, and flux recording. Panels are provided to indicate the core position of the detectors and for recording the flux level versus the detector position. The detectors are driven or inserted to the top of the core and stopped automatically.

The nuclear instrumentation system is provided to monitor the reactor power from source range through the intermediate range and power range up to 120% of full power. The system provides indication, control, and alarm signals for reactor operation and protection. The power range nuclear instrumentation system has four channels, each consisting of two long ion chambers (top and bottom detectors). The intermediate range is composed of two independent channels. The lowest level of intermediate range indication corresponds to ~103 cps on the source range and the highest level corresponds to full power operation The intermediate range neutron detectors are also compensated ionization chambers. The source range is composed of two independent channels, N-31 and N-32. The neutron detectors are proportional counters that are filled with boron trifluoride (BF3) gas. Neutron flux, as measured in the primary shield area, produces current pulses in the detectors. These preamplified pulses are applied to transistor amplifiers.

The following electrical distribution systems interface with the Control Rod Drive and Nuclear Process Indication System:

120 VAC Vital Instrument Buses	Plant Process Computers
125 VDC Power	480 VAC Power
Reactor Protection	

System Function Listing

In addition to the System Functions described above, the Control Rod Drive and Nuclear Process Indication System also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code D	Cri 1	Cri 2	Cri 3				
SENSE OR PROVIDE PROCESS CONDITIONS AND GENERATE SIGNALS FOR REACTOR TRIP AND ENGINEERED SAFETY FEATURES ACTUATION			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Control Rod Drive and Nuclear Process Instruments system perform this primary design system function (Nuclear Instrumentation).

Code Q	Cri 1	Cri 2	Cri 3				
PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Control Rod Drive and Nuclear Process Instruments system perform this associated design system function

Code S	Cri 1	Cri 2	Cri 3				
SPECIAL CAPABILITY CLASS FUNCTIONS			FP	EQ	PTS	AT	SB

Comment: Components within the Control Rod Drive and Nuclear Process Instruments system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Control Rod Drive and Nuclear Process Instruments system that perform License Renewal Criterion 3 functions are tracked under the Criterion 3 codes (Z1 through Z5) (e.g., RG-1.97 Category 2 and 3 Post Accident Monitoring, Rod Position Indication)

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Control Rod Drive and Nuclear Process Instruments system perform this associated design system function.

Code X SFR FUNCTION NOT APPLICABLE AT COMPONENT LEVEL	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Control Rod Drive and Nuclear Process Instruments system perform specific safety related functions different from and in addition to the system level functions (RG-1.97 Category 1 Post Accident Monitoring).

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Control Rod Drive and Nuclear Process Instruments system perform this associated design system function.

Code Z2 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ENVIRONMENTAL QUALIFICATION (10 CFR 50.49)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: Components within the Control Rod Drive and Nuclear Process Instruments system are Environmentally Qualified (Core Exit Thermocouple cables and connectors).

Code Z5 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS STATION BLACKOUT (10 CFR 50.63)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: Components within the Control Rod Drive and Nuclear Process Instruments system perform this associated design system function (Core Exit Thermocouple Indication).

UFSAR Reference

Additional Control Rod Drive and Nuclear Process Instrumentation System details are provided in Section 3.9.4, Section 7.7.1.2, Section 4.2.1.3.4.4, Section 7.7.2, Section 7.7.3, and Section 7.7.4 of the UFSAR.

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.5.6-1 along with each components passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

.

Table 2.5.6-1 Control Rod Drive and Nuclear Process Instrumentation

Component/Commodity Group	Passive Function	Aging Management Reference
Medium Voltage Insulated Cables and Connections	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	Table 3.7-1 Line Number (2) Table 3.7-1 Line Number (3) Table 3.7-1 Line Number (5)
Low Voltage Insulated Cables and Connections	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	Table 3.7-1 Line Number (2) Table 3.7-1 Line Number (3) Table 3.7-1 Line Number (5)

2.5.7 Misc. AC Power and Lighting - Within Scope of License Renewal

Description

The Miscellaneous AC Power and Lighting System includes the source of power to a majority of the low voltage non-safety related power receptacles, transformers, circuit breakers, lighting panels, and various pieces of office and shop equipment within the plant. The alternating current (AC) equipment grouping receives power from various AC power sources via isolation transformers that serve to convert the voltage level and protect the source. (Those isolation transformers, as well as their source, are not included in the Miscellaneous AC Power and Lighting System evaluation boundary.) Components evaluated within the system provide power to non-safety sump pumps used to mitigate internal flooding events and whose failure could therefore prevent the satisfactory accomplishment of a safety related function. The system evaluation boundary also includes the direct current (DC) permanently mounted emergency lights used to ensure safe shutdown can be achieved during fire events.

The principal components of the Miscellaneous AC Power and Lighting System are the AC power Distribution Panels, Power Receptacles, Transformers, Circuit Breakers, and Lighting Panels, and battery operated DC lights. Fixed emergency lighting units are provided in safety-related areas and other areas which contain fire hazards to facilitate emergency operations, manual fire fighting, and access to and egress from each designated fire area.

The following electrical systems interface with the Misc. AC Power and Lighting System:

125 VDC Power	480 VAC Power
Offsite Power	

System Function Listing

In addition to the System Functions described above, the Miscellaneous AC Power and Lighting System also contains components which support additional components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code Q PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Misc AC Power and Lighting system perform this associated system function. Control Room direct current lighting is powered from the vital batteries.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Misc. AC Power and Lighting system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Misc. AC Power and Lighting system that perform special capability class functions are tracked under the Criterion 3 codes (Z1 through Z5). Included within the system boundary is non-vital 120 VAC power (TSC) providing power to RG 1.97 Category 2 and 3 Post Accident Monitoring variables and displays (e.g., SAS/PPCS).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Misc. AC Power and Lighting system perform this associated design system function.

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Misc. AC Power and Lighting system perform this associated design system function. Misc. AC Power and Lighting provides power to dewatering sump pumps.

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Misc. AC Power and Lighting system perform this associated design system function.

UFSAR Reference

Additional Misc. AC Power and Lighting system details are provided in Section 9.5.3 and Section 8.1 of the UFSAR.

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.5.7-1 along with each components passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.5.7-1 Misc. AC Power and Lighting

Component/Commodity Group	Passive Function	Aging Management Reference
Low Voltage Insulated Cables and Connections	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	Table 3.7-1 Line Number (2) Table 3.7-1 Line Number (3) Table 3.7-1 Line Number (5)

2.5.8 Offsite Power - Within Scope of License Renewal

Description

Offsite power is supplied by two separate sources that are classified as non-Class 1E. One source comes directly from the RG&E 34.5-kV system through station auxiliary (startup) transformer 12A and the second from the 115-kV system through a 115-kV to 34.5-kV step-down transformer and station auxiliary (startup) transformer 12B. (The station auxiliary transformers (12A and 12B) are the normal offsite power sources to the safeguards buses.) In the event of a failure of both station auxiliary transformers, the unit auxiliary transformer (11) can be used as a backup supply. Ginna Stations licensing basis does not rely on the recovery of offsite power to mitigate Station Blackout (SBO) events. License Renewal regulatory guidance also mandates the inclusion of the plant system portion of the offsite power SSCs used for SBO recovery beyond those identified in the regulatory commitments made to satisfy 10 CFR 50.63 criteria. In accordance with the current License Renewal regulatory interpretation, Offsite Power systems and structures that provide a function for SBO coping and systems or structures that provide a function for recovery from an SBO condition are being evaluated within the system boundary.

The principal components of the Offsite Power System include station auxiliary transformers (12A, 12B, 11, and the GSU), and circuit breakers, relays and the essential cable and wiring. The components that are part of the Switchyard 115 KV (Station 13A) are not included in this system because they are not within the R.E. Ginna station site. The external transmission system provides two basic and interrelated functions for the station. It supplies auxiliary power for startup and normal shutdown and Class 1E auxiliary loads during MODES 1 and 2 via the station auxiliary (startup) transformers, and it delivers the output of the station to the grid.

During normal startup and operation, the station auxiliary (startup) transformers are supplied from two separate offsite feeders. Each of these feeders is capable of supplying the entire auxiliary power load. During normal shutdown, auxiliary loads are transferred to the station auxiliary (startup) transformers prior to securing the main generator. Two independent offsite power sources are available to supply the engineered safety features equipment. These offsite sources each feed an independent auxiliary (startup) transformer. Offsite circuit 751 feeds transformer 12A. Offsite circuit 767 feeds transformer 12B. Each transformer is capable of supplying all plant engineered safety features equipment. Breakers permit the station auxiliary transformers to be lined up so that transformer 12A supplies one engineered safeguards bus and transformer 12B supplies the other (50/50 mode), transformer 12A supplies both safeguards buses (0/100 mode), or transformer 12B supplies both safeguards buses (100/0 mode). The 50/50 mode is the normal configuration. The Offsite Power System, in conjunction with onsite power sources, is provided to permit functioning of structures, systems, and components important to safety. Each system provides sufficient capacity and capability to ensure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences, and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The following electrical systems interface with the Offsite Power System:

4160 VAC Power

125 VDC Power

System Function Listing

In addition to the System Functions described above, the Offsite Power System also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Offsite Power system perform this associated design system function (augmented quality, e.g. components within the system are required to be operable as per Technical Specifications). For the purposes of License Renewal, components within the Offsite Power system that perform License Renewal Criterion 3 functions are tracked under the Criterion 3 codes (Z1 through Z5).

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Offsite Power system perform this associated design system function.

Code Z5 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR STATION BLACKOUT (10 CFR 50.63)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: Ginna Stations licensing basis does not rely on the recovery of offsite power to mitigate Station Blackout (SBO) events. That notwithstanding, recent regulatory guidance suggests the scope of SBO mitigation equipment be expanded. Accordingly, components within the Offsite power system have been evaluated with regard to SBO.

UFSAR Reference

Additional Offsite Power System details are provided in Section 8.2 of the UFSAR.

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.5.8-1 along with each components passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.5.8-1 Offsite Power

Component/Commodity Group	Passive Function	Aging Management Reference
Medium Voltage Insulated Cables and Connections	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	Table 3.7-1 Line Number (2) Table 3.7-1 Line Number (3) Table 3.7-1 Line Number (5)
Low Voltage Insulated Cables and Connections	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	Table 3.7-1 Line Number (2) Table 3.7-1 Line Number (3) Table 3.7-1 Line Number (5)

Table 2.5.8-1 Offsite Power

Component/Commodity Group	Passive Function	Aging Management Reference
Switchyard Bus	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	Table 3.7-2 Line Number (2)
High Voltage Insulators	To insulate and support an electrical conductor.	Table 3.7-2 Line Number (3)

2.5.9 Reactor Protection - Within Scope of License Renewal

Description

The Reactor Protection System senses or provides the process conditions and generates signals for Reactor Trip and Engineered Safety features actuation. The Reactor Protection system includes non-safety component whose failure could prevent the accomplishment of a safety function (the circulating water flood detection and circulating water pump trip). Components evaluated within the boundary of the Reactor Protection system are used for safe shutdown following fires and Station Blackout Events. The Reactor Protection System also contains components used in ATWS mitigation and components that are Environmentally Qualified.

The principal components of the Reactor Protection System include: transmitters, detectors, relays, power supplies, bistables, switches, converters, indicators and their essential wiring and connectors. The Reactor Protection System senses accidents or impending accident conditions and responds by supplying a reactor trip signal, initiates a rod stop, and initiates a load runback depending upon the severity of the condition. The Reactor Protection System provides the introduction of negative reactivity by opening reactor trip breakers, causing gravity insertion of control rods into the reactor core. There are two complete and independent sets of logic circuits to the reactor protection system cabinets. Each set constitutes a logic train. When the setpoint values are sensed, a trip signal is sent to the protection cabinets. If a reactor trip is required, the protection cabinets will send a signal to the reactor trip breakers. Tripping of these breakers will remove power from the control rod drive mechanisms allowing the rods to drop into the reactor core. Additionally the protection cabinets will actuate any required safeguards devices and also provide appropriate permissive signals to the logic trains to allow automatic or manually initiated interlocks and blocks. In addition, the Reactor Protection System initiates a rod stop signal, which prevents control rod withdrawal, and also initiates a turbine runback, which limits the maximum reactor overpower.

The Reactor Protection System also receives input from non-nuclear safety signals that provide anticipatory reactor trips (e.g. Turbine trip causing a reactor trip is provided to anticipate probable plant transients and to avoid the resulting thermal transients. If the reactor were not tripped by the turbine trip, the overtemperature delta T or high pressure trip would prevent reactor safety limits from being exceeded. By utilizing this trip, undesirable excursions are prevented rather than terminated). Additionally, 10 CFR 50.62 requires that all PWRs provide a means that is diverse and independent from the existing Reactor Trip System (RTS) for tripping the main steam turbine and initiating auxiliary feedwater flow following an anticipated transient without scram (ATWS) event. Anticipated transients include loss of normal feedwater flow, loss of electrical load that results in closure of the turbine stop valves, and loss of offsite power. Rochester Gas & Electric has installed a system providing ATWS mitigation system actuation circuitry (AMSAC) at Ginna Station that satisfies the requirements. The AMSAC, whose functionality is based on low feedwater flow logic, is evaluated within the Reactor Protection System boundary. AMSAC is a non-Class 1E system designed to trip the turbine and start the motor-driven (MDAFW) and turbine-driven (TDAFW) auxiliary feedwater pumps if main feedwater flow is lost with reactor power above 40%.

The following electrical systems interface with the Reactor Protection System:

480 VAC Power	Control Rod Drive and Nuclear Process Inst.
120 VDC Power	Plant Process Computer System
125 VAC Power	Engineered Safety Feature Actuation
Nuclear Process Instrumentation	

System Function Listing

In addition to the System Functions described above, the Reactor Protection System also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code D	Cri 1	Cri 2	Cri 3				
SENSE OR PROVIDE PROCESS CONDITIONS AND GENERATE SIGNALS FOR REACTOR TRIP AND ENGINEERED SAFETY FEATURES ACTUATION			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Reactor Protection system perform this primary design system function.

Code E	Cri 1	Cri 2	Cri 3				
PROVIDE REACTOR COOLANT PRESSURE BOUNDARY			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Reactor Protection system perform this associated design system function. Reactor Protection system transmitters interface with the Reactor Coolant system. Associated valves, piping, etc., are evaluated within the Reactor Coolant system.

Code H	Cri 1	Cri 2	Cri 3				
PROVIDE EMERGENCY HEAT REMOVAL FROM THE REACTOR COOLANT SYSTEM USING SECONDARY HEAT REMOVAL CAPABILITY			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Reactor Protection system perform this associated design system function (steam generator level inputs into RPS).

Code K	Cri 1	Cri 2	Cri 3				
PROVIDE PRIMARY CONTAINMENT BOUNDARY			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Reactor Protection system perform this associated design system function.

Code Q	Cri 1	Cri 2	Cri 3				
PROVIDE ELECTRICAL POWER TO SAFETY CLASS 1, 2, 3 COMPONENTS			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Reactor Protection system perform this associated design system function.

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Reactor Protection system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Reactor Protection system that perform License Renewal Criterion 3 functions are tracked under the Criterion 3 codes (Z1 through Z5) (e.g., RG-1.97 Post Accident Monitoring category 2 or 3 variable, non safety trip signals to Reactor Protection, feedwater control signals, etc.)

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Reactor Protection system perform this associated design system function.

Code X SFR FUNCTION NOT APPLICABLE AT COMPONENT LEVEL	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Reactor Protection system perform specific safety related functions different from and in addition to the system level functions (e.g. RG 1.97 category 1 Post Accident Monitoring variables).

Code Y LICENSE RENEWAL CRITERION 2 - NON SAFETY RELATED SSC'S WHOSE FAILURE COULD PREVENT SATISFACTORY ACCOMPLISHMENT OF A SAFETY RELATED FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Components within the Reactor Protection system associated with the Circulating Water system perform this associated design system function. These components provide internal flooding protection measures by tripping the Circulating Water Pumps on high level in the screenhouse basement and turbine building condenser pit.

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Reactor Protection system perform this associated design system function.

Code Z2 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ENVIRONMENTAL QUALIFICATION (10 CFR 50.49)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: Components within the Reactor Protection system are designated as Environmentally Qualified.

Code Z4 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ANTICIPATED TRANSIENTS WITHOUT SCRAM (10 CFR 50.62)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
						X	

Comment: Components within the Reactor Protection system perform this associated design system function. The ATWS Mitigating System Actuation Circuitry shall be capable of providing a diverse signal to trip the main turbine and start MDAFWP and TDAFWP when a loss of MFW is detected.

Code Z5 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR STATION BLACKOUT (10 CFR 50.63)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: Components within the Reactor Protection system perform this associated design system function.

UFSAR Reference

Additional Reactor Protection system details are provided in Section 7.2 of the UFSAR.

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.5.9-1 along with each components passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.5.9-1 Reactor Protection

Component/Commodity Group	Passive Function	Aging Management Reference
Low Voltage Insulated Cables and Connections	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	Table 3.7-1 Line Number (2) Table 3.7-1 Line Number (3) Table 3.7-1 Line Number (5)

2.5.10 Engineered Safety Features Actuation System - Within Scope of License Renewal

Description

The Engineered Safety Features Actuation System (ESFAS) senses process conditions and generates actuation signals for reactor trip and engineered safety features. The following actions are initiated by ESFAS: safety injection, containment isolation, steam line isolation, containment spray and feedwater isolation, emergency diesel generator start, and preferred auxiliary feedwater pump start. Instrumentation included within the ESFAS system boundary is also used for safe shutdown following some fire events and Station Blackout events.

The principal components of the ESAFS system include: transmitters, detectors, relays, power supplies, bistables, switches, converters, indicators and their essential wiring and connectors. The ESFAS detects plant conditions that require automatic Engineered Safety Features (ESF) equipment operation, and actuates the appropriate ESF equipment when preset limits are reached. ESFAS subsystems monitor plant parameters indicative of different accidents. When the minimum number of channels of a monitored variable reaches a preset limit, trip bistables satisfy coincidence logic for an individual subsystem and the subsystem is automatically initiated. On the channel level, the four ESFAS channels share protection racks with the four Reactor Protection System channels, because some of the same plant variables used to initiate reactor trip also actuate ESFAS subsystems. Not all four channels are used for each ESFAS variable, because most ESFAS subsystem coincidence logic relies on less than four channels to actuate. Each channel is energized from a separate AC power feed. On the train level, the racks for the two ESFAS logic trains are independent and separate from the racks for the two Reactor Protection System logic trains. Each train is energized from a separate DC power feed.

The following electrical systems interface with the ESFAS System:

120 VAC Vital Instrument Buses	125 VDC Power
Reactor Protection	Plant Process Computers

System Function Listing

In addition to the System Functions described above, the Engineered Safety Features Actuation System also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below.

Code D SENSE OR PROVIDE PROCESS CONDITIONS AND GENERATE SIGNALS FOR REACTOR TRIP AND ENGINEERED SAFETY FEATURES ACTUATION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Engineered Safety Features Actuation system perform this primary design system function.

Code H PROVIDE EMERGENCY HEAT REMOVAL FROM THE REACTOR COOLANT SYSTEM USING SECONDARY HEAT REMOVAL CAPABILITY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Engineered Safety Features Actuation system perform this associated design system function.

Code K PROVIDE PRIMARY CONTAINMENT BOUNDARY	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Engineered Safety Features Actuation system perform this associated design system function (transmitters in this system provide signals that initiate containment isolation).

Code S SPECIAL CAPABILITY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Engineered Safety Features Actuation system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Engineered Safety Features Actuation system that perform License Renewal Criterion 3 functions are tracked under the Criterion 3 codes (Z1 through Z5). Transmitters in the Engineered Safety Features Actuation system also provide feedwater control signals.

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Engineered Safety Features Actuation system perform this associated design system function.

Code X SFR FUNCTION NOT APPLICABLE AT COMPONENT LEVEL	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Engineered Safety Features Actuation system perform specific safety related functions different from and in addition to the system level functions (RG 1.97 Type A post-accident monitoring variables).

Code Z1 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Engineered Safety Features Actuation system perform this associated design system function.

Code Z2 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR ENVIRONMENTAL QUALIFICATION (10 CFR 50.49)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: Components within the Engineered Safety Features Actuation system are designated as Environmentally Qualified.

Code Z5 LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS STATION BLACKOUT (10 CFR 50.63)	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: Components within the Engineered Safety Features Actuation system perform this associated design system function.

UFSAR Reference

Additional ESFAS system details are provided in Section 7.3 and Section 8.3.1.2.4.4 of the UFSAR.

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.5.10-1 along with each components passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.5.10-1 Engineered Safety Features Actuation System

Component/Commodity Group	Passive Function	Aging Management Reference
Low Voltage Insulated Cables and Connections	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	Table 3.7-1 Line Number (2) Table 3.7-1 Line Number (3) Table 3.7-1 Line Number (5)

2.5.11 Plant Communications - Within Scope of License Renewal

Description

The Plant Communications System is comprised of a broad range of communications equipment. Included within the evaluation boundary is the radio and repeater system used for fire fighting and safe shutdown activities following some fire events.

The principal components of the Plant Communications System include various different 8” speaker assemblies, speaker amplifier assemblies, GAI-tronics Model 350 Tone Generator or equivalent, a Motorola radio equipment to include handsets, base station, and repeater, and a commercial telephone system. In addition there are Grey Page telephones situated at every level throughout the plant. This equipment works together to ensure good communications throughout the plant during an emergency.

The Plant Communications System consists of several different communication systems. The primary system is the combination paging and party system; in addition, there is a sound powered phone system and a radio paging system. The sound powered system is hard wired with separate wires from the combination paging and party system. The radio paging system provides communication with areas inside the containment with the help of a radio antenna mounted in the containment. Additionally, a repeater located in the yard area allows for greater flexibility with radio communications. There is adequate redundancy with these three systems to ensure good communications throughout the plant during an emergency. There are also emergency telephones within the Technical Support Center (TSC) that provide a direct line to the NRC, New York State, and Monroe and Wayne Counties. The TSC is also equipped for direct communications with the Control Room, Survey Center, Operational Support Center, and the Emergency Operations Facility. Communications between the control room, technical support center, emergency survey center, and other operations centers can be established using either telephone, two-way intercom, radio, or the plant public address system. The telephone system at Ginna affords a great deal of flexibility and capacity. Calls can be received or made to either the Rochester telephone system or the Ontario telephone system. The telephone system has its own power supply located onsite which could maintain house phones independent of offsite lines. Additionally, the plant commercial telephone system can access the plant sound powered phone system.

The following are electrical systems that interface with the Plant Communications System:

Misc. AC Power and Lighting	125 VDC Power

System Function Listing

In addition to the System Functions described above, the Plant Communications System also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the system, is provided in the summary below

Code S	Cri 1	Cri 2	Cri 3				
SPECIAL CAPABILITY CLASS FUNCTION			FP	EQ	PTS	AT	SB

Comment: Components within the Plant Communications system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Plant Communications system that perform special capability class functions are tracked under the Criterion 3 codes (Z1 through Z5).

Code T	Cri 1	Cri 2	Cri 3				
NON-NUCLEAR SAFETY CLASS FUNCTIONS			FP	EQ	PTS	AT	SB

Comment: Components within the Plant Communications system perform this associated design system function.

Code Z1	Cri 1	Cri 2	Cri 3				
LICENSE RENEWAL CRITERION 3 - SSC'S RELIED UPON IN SAFETY ANALYSES OR PLANT EVALUATIONS TO PERFORM A FUNCTION THAT DEMONSTRATES COMPLIANCE WITH THE COMMISSION'S REGULATIONS FOR FIRE PROTECTION (10 CFR 50.48)			FP	EQ	PTS	AT	SB
			X				

Comment: Components within the Plant Communications system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Plant Communications system that perform special capability class functions are tracked under the Criterion 3 codes (Z1 through Z5).

UFSAR Reference

Additional information on the plant communications system is provided in Section 9.5.1.2.4.7 and Section 9.5.2 of the UFSAR.

Components Subject to an AMR

The component groups for this system that require aging management review are indicated in Table 2.5.11-1 along with each components passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.5.11-1 Plant Communications

Component/Commodity Group	Passive Function	Aging Management Reference
Low Voltage Insulated Cables and Connections	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	Table 3.7-1 Line Number (2) Table 3.7-1 Line Number (3) Table 3.7-1 Line Number (5)

2.5.12 Plant Process Computer - Not Within Scope of License Renewal

Description

The Plant Process Computer System (PPCS) and Safety Assessment System (SAS) are integrated data acquisition and display systems, which include common multiplexer cabinets for the termination of field inputs. The Plant Process Computer System (PPCS) provides information to the plant operator to effectively assist in the operation of the nuclear steam supply system and to informing the operator of specific abnormal conditions by comparison with preset or calculated limits. Basic to the design of this computer system is the requirement that the conventional plant instrumentation systems and control room instrumentation and control functions permit operation of the plant with the computer out of service. The computer system reduces the burden to the plant operator in maintaining surveillance over the nuclear steam supply system to ensure that operating conditions are maintained within normal bounds. The PPCS does not control any part of the plant, it is used as an aid in alerting operators to deviations in plant operating parameters. The Safety Assessment System (SAS) is designed to provide an integrated display of critical plant safety parameters and perform reference diagnostics during emergencies. The Plant Process Computer System contains no safety-related or safety significant components. The equipment contained within the Plant Process Computers system boundary does not perform any license renewal intended function.

The principal components of the Plant Process Computer System and SAS are two redundant processing units, multiplexers, display and output devices and the essential wiring. The Plant process computer system is a dedicated real-time system consisting of a central processing unit with parametric inputs from the reactor coolant system, the secondary system, the effluent monitoring system, and auxiliary service systems throughout the plant. These inputs are stored as discrete, addressable data points that are used to perform specific computations (e.g., compute subcooling margin), generate alarms, indicate digital and analog information, and to provide pre-trip and post-trip data.

System Function Listing

In addition to the System Functions described above, the Plant Process Computers System also contains components which support additional functions (associated system design functions) that may or may not be License Renewal Intended Functions. A comprehensive listing of all functions associated with the system, or specific components contained in the in the system, is provided in the summary below.

Code S SPECIAL CAPABILITY CLASS FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Plant Process Computers system perform this associated design system function (augmented quality). For the purposes of License Renewal, components within the Plant Process Computers system that perform special capability class functions are tracked under the Criterion 3 codes (Z1 through Z5). The PPCS portion of the system displays and records Reg Guide 1.97 post

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Plant Process Computers system perform this associated design system function

UFSAR Reference

Additional information on the plant process computer system is provided in Section 7.7.6 of the UFSAR.

Components Subject to an AMR

System level scoping shows that there are no components subject to AMR in the Plant Process Computers system. However, the basic philosophy used in the electrical components IPA process does not account for system level scoping (all electrical components are included in the IPA review unless they are specifically scoped out or screened out in the electrical IPA process). Therefore the component groups associated with Plant Process Computers are conservatively included in the population of components subject to AMR.

The component groups for this system that require aging management review are indicated in Table 2.5.12-1 along with each components passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.5.12-1 Plant Process Computers

Component/Commodity Group	Passive Function	Aging Management Reference
Low Voltage Insulated Cables and Connections	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	Table 3.7-1 Line Number (2) Table 3.7-1 Line Number (3) Table 3.7-1 Line Number (5)

2.5.13 Plant Security - Not Within Scope of License Renewal

Description

The plans for physical protection of Ginna Station are described in the NRC-approved plans which are withheld from public disclosure pursuant to 10 CFR 2.790(d) and 10 CFR 73.21 The plans conform to the requirements of appropriate federal regulations governing security activities of nuclear power reactors. Some plant security physical barrier features interfaces with station building and structures. Those features are evaluated within the system boundary of the appropriate plant structure (e.g. security doors that act as external missile shield, etc.). Thus the remaining equipment contained within the Plant Security System boundary does not perform any license renewal intended function.

The principal components of the Plant Security system include intrusion detection systems (visual and electronic perimeter monitoring, access control and door position), the vehicle barrier system, and perimeter fencing. Additional protection features include defensive positions and strategically placed armor and bullet resistant ballistic steel. The security system includes a dedicated diesel generator that provides backup power to security computers, detection systems and the high mast lighting system. Security computers also have an additional diverse backup power supply. Plant Security features have been evaluated to ensure that they do not impede the safe operation of the facility. The equipment identified in the system scoping and screening report does not represent any safeguards information and functional descriptions and references are withheld. The list is simply the collection of identification numbers contained in the plant database.

System Function Listing

A comprehensive listing of all functions associated with the Plant Security system, or specific components contained in the system, is provided in the summary below.

Code T	Cri 1	Cri 2	Cri 3				
NON-NUCLEAR SAFETY CLASS FUNCTIONS			FP	EQ	PTS	AT	SB

Comment: Components within the Plant Security system perform this associated design system function

UFSAR Reference

Additional functional information or descriptions are available in Section 1.8.1.17 and Section 13.6 of the UFSAR.

Components Subject to an AMR

System level scoping shows that there are no components subject to AMR in the Plant Security system. However, the basic philosophy used in the electrical components IPA process does not account for system level scoping (all electrical components are included in the IPA review unless they are specifically scoped out or screened out in the electrical IPA process). Therefore the component groups associated with Plant Security are conservatively included in the population of components subject to AMR.

The component groups for this system that require aging management review are indicated in Table 2.5.13-1 along with each components passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.5.13-1 Plant Security

Component/Commodity Group	Passive Function	Aging Management Reference
Low Voltage Insulated Cables and Connections	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	Table 3.7-1 Line Number (2) Table 3.7-1 Line Number (3) Table 3.7-1 Line Number (5)

2.5.14 Seismic and Meteorological Instrumentation System- Not Within Scope of License Renewal

Description

The Seismic and Meteorological Instrumentation System provides operators with current weather conditions, including wind speeds, wind direction and atmospheric stability, which are Regulatory Guide 1.97, category 3 variables. Variable readout of this information is displayed in the control room on the plant process computer. A strong motion accelerograph, used to measure seismic activity, is installed in the subbasement of the intermediate building. Components within the Seismic and Meteorological Instrumentation System do not perform any License Renewal intended functions.

The principal components of the Seismic and Meteorological Instrumentation System are the Primary Tower, the Backup Tower, and the Seismic Monitoring Station. The primary tower, a 250 foot tower located approximately 850 feet Northwest of the of the containment building, measures wind speed, wind direction, and temperatures. The primary tower has a gasoline powered backup AC generator. The backup tower, located at substation 13A, approximately 0.5 miles south of the Ginna site, measures wind speed and direction. Precipitate is measured by means of a rainfall bucket, on a separate pad near the primary tower. The seismic monitoring station, accelerograph, is located in the intermediate building subbasement, and is used to take shock recordings. This location was chosen rather than the basement of the containment since it more easily facilitates periodic surveillance of the instrument (this would be difficult should the instrument be located in the basement of the containment) and the retrieval of the shock record can more readily be made. Instrumentation included within the Seismic and Meteorological instrumentation boundary include wind speed, wind direction, and atmospheric stability equipment.

The following electrical systems interface with the Seismic and Meteorological Instrumentation System:

Plant Process Computer System	Misc. AC Power and Lighting

System Function Listing

A comprehensive listing of all functions associated with the Seismic and Meteorological Instrumentation System, or specific components contained in the system, is provided within the summary below.

Code S SPECIAL CAPABILITY CLASS FUNCTION	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Seismic and Meteorological Instrumentation system perform this associated design system function (augmented quality). The Seismic and Meteorological Instrumentation system provides RG 1.97 Category 3 Post Accident monitoring variables.

Code T NON-NUCLEAR SAFETY CLASS FUNCTIONS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components within the Seismic and Meteorological Instrumentation system perform this associated design system function.

UFSAR Reference

Additional information on the Seismic and Meteorological Instrumentation System is provided in Section 2.3.3 and Section 3.7.4 of the UFSAR.

Components Subject to an AMR

System level scoping shows that there are no components subject to AMR in the Seismic and Meteorological Instrumentation system. However, the basic philosophy used in the electrical components IPA process does not account for system level scoping (all electrical components are included in the IPA review unless they are specifically scoped out or screened out in the electrical IPA process). Therefore the component groups associated with Seismic and Meteorological Instrumentation are conservatively included in the population of components subject to AMR.

The component groups for this system that require aging management review are indicated in Table 2.5.14-1 along with each components passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3.

Table 2.5.14-1 Seismic and Meteorological

Component/Commodity Group	Passive Function	Aging Management Reference
Low Voltage Insulated Cables and Connections	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	Table 3.7-1 Line Number (2) Table 3.7-1 Line Number (3) Table 3.7-1 Line Number (5)

3.0 AGING MANAGEMENT REVIEW RESULTS

For those structures and components that are subject to aging management review, paragraph 54.21(a)(3) of the license renewal rule requires demonstration that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.

This chapter describes the results of the aging management reviews for structures and components that were identified in Chapter 2, Structures and Components Subject to Aging Management Review.

3.1 Review Methodology

The overall process by which aging effects requiring management were identified and evaluated is summarized in this section.

3.1.1 Determination of Materials of Construction

Material(s) of construction were identified for all systems, structures and components subject to aging management review. Sources of information used to identify materials of construction included original Westinghouse and Gilbert Associates, Inc. (GAI) equipment and material specifications, vendor technical manuals and drawings, fabrication drawings, piping and instrument drawings, and piping line specifications. Field walkdowns were also used to identify/verify materials of construction for some components.

3.1.2 Determination of Operating Environments

Internal operating environments were defined by fluid medium and chemistry (i.e., treated water, raw water, lubricating oil and fuel oil, air/gas, etc.), service temperature, and condition of fluid flow. External environments were defined by plant location, including temperature and humidity (i.e., indoor with no air-conditioning, outdoor with exposure to weather), exposure to soil/soil water (i.e., buried), embedment in concrete, and exposure to borated water leaks.

Table 3.1-1 and Table 3.1-2 contain descriptions of the internal and external service environments at Ginna Station which are used in subsequent sections of this chapter. Within this Application, some of the internal environments have been subdivided into subgroups based on the fluid chemistry or flow rate. The subgroups are identified in the Description column in Table 3.1-1.

3.1.3 Component Grouping by Material/Environment Combination

The aging mechanisms and effects that apply to a structure or component are determined by the material(s) of construction and operating environment (including temperature and stress) to which the material is exposed. Structures or components constructed of the same material and exposed to the same environment would therefore be susceptible to the same aging mechanisms and effects. As a result, structures and components were grouped together according to material/environment combinations. This facilitated the aging management review process, in that a single aging management review could be performed for an entire group of structures or components.

3.1.4 Aging Effects Analysis - Non-Class 1 Mechanical Systems and Components

Aging effects requiring management for Non-Class 1 systems and components were determined using the evaluation processes described in standard industry guidance for aging evaluation of mechanical systems and components. Systems and components were evaluated by applying a set of material/environment-based rules derived from known age-related degradation mechanisms documented in the technical literature and published industry operating experience. A plant-specific review of this guidance document was conducted to demonstrate applicability of this document at Ginna Station and to provide corrections and/or enhancements to criteria for evaluating aging of specific materials in certain environments (see Section 3.1.9).

3.1.4.1 Treated Water Systems

In accordance with NUREG-1801 for treated water systems, aging mechanisms and effects were identified and evaluated without crediting the mitigative effects of water chemistry controls.

3.1.4.2 Protective Coatings

Coatings are used at Ginna Station to protect the surfaces of steel components in mechanical systems and structures. Although the benefits derived from protective coatings are recognized, coatings, in and of themselves, do not perform License Renewal intended functions. Therefore, protective coatings are not credited with managing the effects of aging. However, the condition of steel surfaces protected by coatings is evaluated during inspections directed by aging management programs at Ginna Station. When evidence of superficial surface corrosion caused by coating degradation is found, the coating is evaluated and repaired in accordance with plant procedures. That notwithstanding, protective coatings applied to surfaces in containment are managed within the current licensing basis. This is further discussed in Appendix B2.1.24.

3.1.5 Aging Effect Analysis - Essential Structures

Aging effects requiring management for Essential Structures (including Yard Structures) were determined using the evaluation processes described in standard industry guidance for aging evaluation of structures and structural components. Aging mechanisms and effects identified in the EPRI document for structural materials were derived from a number of sources, including collective nuclear plant operating experience and relevant operating experience from other industries. A plant-specific review of this guidance document was conducted to evaluate applicability of various aging mechanisms at Ginna Station (see Section 3.1.9).

3.1.6 Aging Effects Analysis - Class 1 Systems, Structures and Components

Aging effects requiring management for Class 1 mechanical systems, components and the Containment Structure were determined using the information and guidance presented in Westinghouse Generic Topical Reports (GTRs). The following Class 1 components were evaluated using the GTRs:

- Containment Structure
- Reactor Pressure Vessel
- Reactor Vessel Internals
- Reactor Coolant System Piping
- Reactor Coolant System Supports
- Steam Generators
- Pressurizer

The GTRs have undergone extensive peer review and, in some cases, NRC review. In addition, they contain thorough reviews of equipment maintenance histories as well as discussions and assessments of industry/regulatory issues. For those GTRs with U. S. NRC Final Safety Evaluation Reports (FSERs), detailed responses to all Applicant Action Items were prepared.

3.1.7 Industry and Plant-Specific Operating Experience Review

A thorough review of appropriate industry and plant-specific operating experience was conducted to confirm that applicable aging effects had been identified. Industry operating experience sources included NRC Generic Publications, INPO Significant Operating Event Reports (SOER), EPRI Technical Reports, Westinghouse Generic Technical Reports (GTRs), and NUREG-1801 (Generic Aging Lessons Learned (GALL) report). Plant-specific operating experience sources included Reports to AEC/NRC, Abnormal Occurrence and Licensee Event Reports (LERs), Non-Conformance Reports (NCRs), Corrective Action

Reports (CARs), Refueling, Inspection and Overhaul Reports (RIOs), Inservice Inspection (ISI) Reports, Identified Deficiency Reports (IDRs), and ACTION Reports (ARs) from 1969 to the present. This review was conducted not only to confirm that the aging effects determined by material/environment-based rules were appropriate, but also to assure that any additional plant-specific aging mechanisms and related effects were identified for management.

3.1.8 Assignment of Aging Management Programs

Appropriate aging management program(s) credited for managing each aging effect were assigned to each structure or component evaluated in the aging management review process. Aging management programs are described in Appendix B of this Application.

3.1.9 Standard Industry Guidance Document Review (Mechanical Systems and Components)

A technical review of standard industry guidance for aging evaluation of mechanical systems and components was conducted to demonstrate that the materials and internal/external operating environments evaluated in the document were applicable and bounding for Ginna Station. In addition, an evaluation of the aging mechanisms identified for specific materials in certain environments was performed. Positions were developed for mechanisms such as stress corrosion cracking (SCC) and intergranular attack/stress corrosion cracking (IGA/IGSCC) of austenitic stainless steels in treated and raw water environments. A position was also established for SCC of bolting materials.

3.1.9.1 Position on SCC of Austenitic Stainless Steel

The threshold temperature for the onset of stress corrosion cracking of austenitic stainless steels in the presence of halides (>150 ppb) and sulfates (>100 ppb) is generally agreed to be approximately 140°F (Reference 1 and Reference 2). The validity of this threshold temperature is also supported by industry operating experience. This threshold temperature has been applied to austenitic stainless steels in all environments evaluated in this LRA. However, it should be noted that the Water Chemistry Control Program (supplemented by one-time inspections in stagnant or low-flow areas) is the aging management program credited for managing cracking due to SCC in treated water systems. This aging management approach is consistent with NUREG-1801.

3.1.9.2 Position on IGA/IGSCC of Austenitic Stainless Steels

Cracking of austenitic stainless steels due to IGA/IGSCC requires a threshold level of grain boundary sensitization and a threshold temperature of approximately 140°F (Reference 1 and Reference 2). IGA/IGSCC is not a credible aging mechanism for welded austenitic stainless steel piping and components at Ginna Station due to controls imposed on heat input and interpass temperature during fabrication which limited grain boundary sensitization in heat affected zones of welded joints. Susceptibility of austenitic stainless steels to IGA/IGSCC may be increased only after prolonged exposure to elevated temperatures above 482°F.

3.1.9.3 Position on SCC of SA 193 Grade B7 Bolting Materials

Although there have been a few reported cases of cracking of bolting in the industry caused by SCC, these have been attributed to susceptible high yield stress materials exposed to aggressive environments, such as lubricants containing molybdenum disulfide. One such case occurred at Ginna Station early in plant life. The bolting which cracked was high-strength (ASTM A 490) RCP embedment anchor studs which had been improperly heat treated, installed with excessive preload, and exposed to borated water leakage. The failure mechanism was determined to be SCC. Replacement A-490 bolting (properly heat-treated and installed with proper preload) has not cracked.

However, a survey of industry experience, technical literature, and laboratory corrosion studies (documented in EPRI Report NP-5769) indicates that SCC should not be a concern for closure bolting in nuclear power plant applications if the specified minimum yield strength is <150 Ksi. For quenched and tempered low-alloy steels typically used for closure bolting (e.g., SA193, Grade B7), susceptibility to SCC is controlled by yield strength. The minimum yield strength specified in SA193 for Grade B7 material is 105 Ksi, which is well below the threshold value of 150 Ksi identified in EPRI Report NP-5769. Furthermore, the selection and use of fastener lubricants for pressure boundary components has been controlled by the Ginna Station Quality Assurance Program since 1983 as part of the response to IE Bulletin 82-02. Limits are also imposed on levels of contaminants such as chlorides and sulfur compounds (including molybdenum disulfide) in lubricants and sealant compounds. Therefore, it is reasonable to conclude that failure by SCC should not be a significant issue for SA193 Grade B7 bolting materials. Ginna Station operating experience supports this conclusion.

3.1.10 Standard Industry Guidance Document Review (Structures and Structural Components)

A technical review of standard industry guidance for aging evaluation of structures and structural components was conducted to evaluate the applicability of aging mechanisms identified for structural materials at Ginna Station. This included a review of original construction contractual requirements, specifications for concrete structures and other materials, site-specific environments, and plant operating experience. Certain aging mechanisms/effects were determined not to be applicable at Ginna Station. Nevertheless, appropriate aging management/monitoring programs are credited for verification that these mechanisms/effects do not, in fact, result in age-related degradation.

3.1.11 Standard Industry Guidance Document Review (Electrical Commodities)

A technical review of standard industry guidance for aging evaluation of electrical commodities was conducted to evaluate the applicability of aging mechanisms identified for electrical components at Ginna Station. This included a review of original construction requirements, specifications for selected electrical components and other insulating materials, site-specific environments, and plant operating experience. Certain aging mechanisms/effects were determined not to be applicable at Ginna Station. Nevertheless, appropriate aging management/monitoring programs are credited for verification that these mechanisms/effects do not, in fact, result in age-related degradation.

3.1.12 Generic Component Assets

It was recognized that certain items/assets such as carbon/low-alloy steel closure bolting or other carbon steel components are present in almost every mechanical system or structure and therefore may be conveniently treated as commodity groups. To facilitate aging management review of such items, generic assets were created in every system and structure to account for the presence of closure bolting and external surfaces of carbon steel components which are subject to the effects of aging. Carbon steel components (CS components) are identified as a specific commodity group to ensure that carbon steel components potentially exposed to borated water leaks are evaluated. The normal external operating environment is evaluated with the specific system-identified components.

Aging effects requiring management for closure bolting were assigned to the generic asset in each system/structure and appropriate aging management programs were identified and credited. For borated water systems or non-borated water systems in close proximity to borated water systems, the potential for boric acid corrosion of carbon/low-alloy steel closure bolting, structural bolting, and external surfaces of equipment and structural members was recognized and accounted for by assigning the applicable aging effects to the generic assets. Appropriate aging management programs were then identified and credited.

Aging management review results for Reactor Coolant Systems are contained in Section 3.2, for Engineered Safety Features Systems in Section 3.3, for Auxiliary Systems in Section 3.4, for Steam and Power Conversion Systems in Section 3.5, for Structures and Component Supports in Section 3.6, and for Electrical and Instrument and Controls Systems in Section 3.7.

3.1.13 Review of NUREG-0933

NUREG-0933 has been reviewed in accordance with the guidance provided in Appendix A.3 of the Standard Review Plan. As a result of this review, the following generic safety issues (GSI) have been evaluated for license renewal and have been addressed in the LRA:

- GSI-168, Environmental Qualification of Electrical Equipment, is addressed in Section 4.4, Environmental Qualification (EQ) of Electric Equipment.
- GSI-190, Fatigue Evaluation of Metal Components for 60-Year Plant Life, is addressed in Section 4.3.7, Environmentally Assisted Fatigue.
- GSI-191, Assessment of Debris Accumulation on PWR Sump Performance, is addressed in Appendix B2.1.24, Protective Coatings Monitoring and Maintenance Program

GSI-191, Assessment of Debris Accumulation on PWR Sump Performance, is addressed in Section B1.2.22, Protective Coatings Monitoring and Maintenance Program.

Table 3.1-1 Internal Service Environments

Environment	Description
Treated Water - Primary, T<140°F	Treated water containing boric acid in the Reactor Coolant System (RCS), T<140°F. The chemistry of this water is monitored and controlled in accordance with the requirements of the Ginna Station Water Chemistry Control Program. Includes consideration of Stagnant, Low Flow <3 fps.
Treated Water - Primary, T>480°F	Treated water containing boric acid in the Reactor Coolant System (RCS), T>480°F. The chemistry of this water is monitored and controlled in accordance with the requirements of the Ginna Station Water Chemistry Control Program. Includes consideration of Stagnant, Low Flow <3 fps.
Treated Water - Primary, 140°F<T<480°F	Treated water containing boric acid in the Reactor Coolant System (RCS), 140°F<T<480°F. The chemistry of this water is monitored and controlled in accordance with the requirements of the Ginna Station Water Chemistry Control Program. Includes consideration of Stagnant, Low Flow <3 fps.
Treated Water - Secondary, T>120°F	Demineralized, deaerated water; secondary water chemistry is monitored and controlled in accordance with the requirements of the Ginna Station Optimized Secondary Water Chemistry Program (included in the Water Chemistry Control Program) and includes High Energy Piping in Main Steam, Feedwater, Blowdown, Auxiliary Feedwater, Condensate, and Sample System - Secondary. Includes steam and consideration of Stagnant, Low Flow, <3 fps.
Treated Water - Secondary, T<120°F (Stagnant, Low Flow <3 fps)	Demineralized, deaerated water; secondary water chemistry is monitored and controlled in accordance with the requirements of the Ginna Station Optimized Secondary Water Chemistry Program (included in the Water Chemistry Control Program). Includes portions of Aux Feedwater, Condensate, and Sample System - Secondary
Treated Water - Borated, T<140°F (Stagnant, Low Flow <3 fps)	Treated water containing boric acid in systems other than the RCS, i.e., Sample System NSSS, CVCS Charging and Letdown, Residual Heat Removal, Safety Injection, Spent Fuel Cooling, Containment Spray, and Waste Disposal Systems. Borated water chemistry is monitored and controlled in accordance with the requirements of the Ginna Station Primary Water Chemistry Control Program.
Treated Water - Borated, T>140°F (Stagnant, Low Flow <3 fps)	Stagnant, low flow (<3 fps) treated water containing boric acid in systems other than the RCS, i.e., Sample System NSSS, CVCS Charging and Letdown, Residual Heat Removal, Safety Injection, Spent Fuel Cooling, Containment Spray, and Waste Disposal Systems. Borated water chemistry is monitored and controlled in accordance with the requirements of the Ginna Station Primary Water Chemistry Control Program.

Table 3.1-1 Internal Service Environments

Environment	Description
Treated Water - Other	Treated water is demineralized water which may be deaerated and include corrosion inhibitors and biocides or some combination of these treatments. The chemistry of this water is monitored and controlled in accordance with the requirements of the Ginna Station Primary Water Chemistry Control Program. Ginna Station treated water systems include Primary Makeup Water, Emergency Diesel Generator Cooling Water, Component Cooling Water, and Chilled Water.
Treated Water - Other (Stagnant, Low Flow <3 fps)	Treated water is demineralized water which may be deaerated and include corrosion inhibitors and biocides or some combination of these treatments. The chemistry of this water is monitored and controlled in accordance with the requirements of the Ginna Station Primary Water Chemistry Control Program. Ginna Station treated water systems include Primary Makeup Water, Emergency Diesel Generator Cooling Water, Component Cooling Water, and Chilled Water.
Treated Water - Other (High velocity, change in flow direction)	Treated water is demineralized water which may be deaerated and include corrosion inhibitors and biocides or some combination of these treatments. The chemistry of this water is monitored and controlled in accordance with the requirements of the Ginna Station Primary Water Chemistry Control Program. Ginna Station treated water systems include Primary Makeup Water, Emergency Diesel Generator Cooling Water, Component Cooling Water, and Chilled Water.
Raw Water (Flowing, >3 fps)	Raw water at Ginna Station includes the lake water used both for Circulating Water (in the main condensers) and for the Service Water System, as well as city water used for the Fire Protection System. The Standby Auxiliary Feedwater System also contains raw water.
Raw Water (Stagnant, low flow <3 fps)	Raw water at Ginna Station includes the lake water used both for Circulating Water (in the main condensers) and for the Service Water System, as well as city water used for the Fire Protection System. The Standby Auxiliary Feedwater System also contains raw water.
Raw Water (High velocity, change in flow direction)	Raw water at Ginna Station includes the lake water used both for Circulating Water (in the main condensers) and for the Service Water System, as well as city water used for the Fire Protection System. The Standby Auxiliary Feedwater System also contains raw water.
Raw Water Drainage	Fluids collected in building drains. These can be treated waters (primary, borated, secondary, or other), raw water (service water, city water), fuel oil or lubricating oil.
Lubricating Oil and Fuel Oil	This category comprises either lubricating oil or diesel fuel oil. Ginna Station systems with this internal environment include the Emergency Diesel Generator (EDG) Fuel Oil and Lube Oil System, and Diesel Fire Pump Fuel Oil and Lube Oil System.

Table 3.1-1 Internal Service Environments

Environment	Description
Lubricating Oil and Fuel Oil - Pooling	This category comprises either lubricating oil or diesel fuel oil with the potential for pooling of water.
Air and Gas	The environments in this category include atmospheric (breathing) air, dry/ filtered instrument air, nitrogen, carbon dioxide, hydrogen, helium and halon. Ginna Station systems exposed to this internal environment include the Instrument Air, Breathing Air, Nitrogen, EDG Air Start System, Control Room HVAC, Computer/Cable Spread Room HVAC, CRDM Cooling, Containment Purge, Emergency Containment Coolers, Emergency Containment Filters, Containment Post Accident evaluation, the Normal Containment Coolers, portions of Waste Disposal, Fire Suppression, and Refrigerated Systems. Note that air operated valves assigned to balance of plant systems are also exposed to this environment.
Air and Gas - Wetted Environment	Moist atmospheric air, unfiltered
Air and Gas - Wetted Environment, T>140°F	Moist atmospheric air, unfiltered, T>140°F

Table 3.1-2 External Service Environments

Category	Description
Outdoor	Moist air, temperature: 0-91°F, 5-100% relative humidity. Exposed to weather including precipitation and wind.
Indoor - No Air Conditioning	Moist air, temperature: 50-104°F, 60% nominal humidity. Not exposed to weather.
Indoor - Air Conditioning	Specific temperature range/humidity dependent upon building/room. Typically, temperature: 70-78°F, 60% relative humidity. Not exposed to weather.
Containment	Moist air, temperature: 60-120°F, 50% nominal humidity, Radiation - total integrated dose 1 rad per hour max. (excluding equipment located inside the reactor cavity). Not exposed to weather.
Buried	Exposed to soil/fill or ground water
Borated Water Leaks	Potentially exposed to borated water leaks
Embedded	Embedded/encased in concrete

Section 3.1 References

1. D. Peckner and I.M. Bernstein, Handbook of Stainless Steels, McGraw Hill, 1977.
2. A.J. Sedricks, Corrosion of Stainless Steels, John Wiley & Sons, 1979, pp. 152-156.

3.2 Aging Management of Reactor Coolant System

The results of the aging management review of the Reactor Coolant System components are provided in this section and summarized in Tables 3.2-1 and 3.2-2. Table 3.2-1 shows the aging management of system components evaluated in NUREG-1801 that are relied on for license renewal of the Reactor Coolant System components at Ginna. Included in the table is a discussion column. The discussion column will provide a conclusion indicating if the aging management evaluation results are consistent with NUREG-1801 along with any clarifications or explanations required to support the stated conclusion if that conclusion is different than those of the NUREG. For a determination to be made that a table line item is “Consistent with NUREG-1801” several criteria must be met. First the plant specific component is reviewed against the GALL to ensure that the component, materials of construction and internal or external service environment are comparable to those described in a particular GALL item. Second, for those that are comparable, the results of the plant aging management review- aging effect evaluation are compared to the aging effects/mechanisms in the GALL. Finally, the programs credited in the GALL for managing those aging effects are compared to the programs invoked in the plant evaluation. If, using good engineering judgment, it could be reasonably concluded that the plant evaluation is in agreement with the GALL evaluation a line item was considered consistent with NUREG-1801. There are cases where components and component material/environment combinations and aging effects are common between a NUREG-1801 line item and the plant evaluation but the aging management program selections differ. In those cases the discussion column will indicate the plant aging management program selection but no conclusion will be made that the line item is consistent with the GALL. Table 3.2-2 contains the Reactor Coolant System components aging management review results that are not addressed in NUREG-1801. A plant component is considered not addressed by the NUREG if the component type is not evaluated in the GALL or has a different material of construction or operating environment than evaluated in the GALL. This table includes the component types, materials, environments, aging effects requiring management, the programs and activities for managing aging, and a discussion column. To avoid confusion, no attempt was made to interrelate material/environment/aging effects from one NUREG-1801 chapter to another. Note that these tables only include those components, materials and environments that are applicable to a PWR.

Materials

The materials of construction of a component have a major influence on the evaluation of aging effects applicable to the component. Sources of information used to identify materials of construction include original equipment specifications, vendor technical manuals and drawings, fabrication drawings, piping line specifications, modification design records and field walkdowns/verifications. The tables below account for the materials of construction for the components requiring an aging management review. Since similar materials are susceptible to the same aging effects/mechanisms, the tables itemize the component types (i.e., groupings) while factoring in the materials of construction.

Environment

As previously described, the environment(s) to which components are exposed are critical in the determination of potential aging mechanisms and effects. A review of plant design documentation was performed to quantify the environmental conditions to which Ginna Station equipment is exposed. This review identified that some equipment is exposed to a variety of environments. This can include normal operating conditions and post accident conditions. Since aging mechanisms and effects will be primarily driven by the environmental conditions to which equipment is exposed on a daily basis, under normal operating conditions, these conditions will differ from the design parameters which are established based upon the worst case scenario (e.g., LOCA conditions). Ginna Station equipment environments may be categorized into basic external and internal environments detailed in Section 3.1.2.

Aging Effects Requiring Management

After the components requiring aging management review were identified and grouped by materials of construction and environment, a review of industry and plant-specific operating experience was performed. The purpose of this review was to assure that all applicable aging effects were identified, and to evaluate the effectiveness of existing aging management programs.

This experience review was performed utilizing various industry and plant-specific programs and databases. Industry operating experience sources included NRC Generic Publications (including Information Notices, Circulars, Bulletins, and Generic Letters), INPO Significant Operating Event Reports (SOER), EPRI Technical Reports, and other information sources, such as the B&W Owners Group Non-Class 1 Mechanical Tools Implementation document, Westinghouse Generic Technical Reports (GTRs), and the Generic Aging Lessons Learned (GALL) report.

Plant specific operating experience sources included Semi-annual and Annual Reports to AEC/NRC, Abnormal Occurrence and Licensee Event Reports (LERs), Non-Conformance Reports (NCRs), Corrective Action Reports (CARs), Refueling, Inspection and Overhaul Reports (RIOs), Inservice Inspection (ISI) Reports, Identified Deficiency Reports (IDRs), and ACTION Reports (ARs) from 1969 to the present. Information from these sources was compiled in various databases. Based upon the material of construction, the applicable environments, and operating experience the potential aging effects requiring management for each of the components was identified as documented in the tables below.

Time-Limited Aging Analysis

In addition to those identified in NUREG-1801, any additional time-limited aging analyses (TLAA) identified as appropriate to the system are identified in Section 4.0.

Confirmation of Topical Report Applicability

Class 1 Piping and Associated Pressure Boundary Components

The Westinghouse Owners' Group Life Cycle Management & License Renewal Program has prepared topical report, WCAP-14575-A, Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components (Reference 1), which has been utilized in the aging management review of the Ginna Class 1 piping and associated pressure boundary components. The scope of the RC components described in the topical report bounds the Ginna Class 1 piping and associated pressure boundary components. A reconciliation of the final SER for WCAP 14575-A applicant action items is provided in Table 3.2.0-1.

Reactor Internals

The Westinghouse Owners' Group Life Cycle Management & License Renewal Program has prepared topical report, WCAP-14577, Rev. 1-A, License Renewal Evaluation: Aging Management for Reactor Internals (Reference 2), which has been utilized in the aging management review of the Ginna Reactor Vessel Internals components. The scope of the Reactor Vessel Internals components described in the topical report bounds the Ginna Reactor Vessel Internals components. A reconciliation of the final SER for WCAP-14577, Rev. 1-A applicant action items is provided in Table 3.2.0-2.

Pressurizer

The Westinghouse Owners' Group Life Cycle Management & License Renewal Program has prepared topical report, WCAP-14574-A, License Renewal Evaluation: Aging Management Evaluation for Pressurizers (Reference 3), which has been utilized in the aging management review of the Ginna Pressurizer components. The Ginna pressurizer is included in WCAP-14574 -A. The scope of the Pressurizer components described in the topical report bounds the Ginna Pressurizer components with the following clarifications:

- For the Ginna pressurizer, the design, fabrication, and installed configuration are the same as specified in the WCAP with the exception of the earthquake lugs and valve support brackets.
- The WCAP identifies stress corrosion cracking (SCC) of the pressurizer sensitized stainless steel nozzle safe ends as a potential aging mechanism. However, the WCAP recognizes that service experience with nozzles and safe ends in Westinghouse pressurizers has been excellent and bases the need for aging management on general industry concerns. The WCAP identifies ASME Section XI inspections as the program to manage SCC of the safe ends. Consistent with the other Class 1 AMRs, SCC of stainless steel materials in the RCS environment can be effectively managed by the Ginna Station Water Chemistry Control Program. Cracking due to flaw growth is considered in the Ginna Pressurizer AMR and the Ginna Station ASME Section XI Inservice Inspection is credited to manage the aging effects. As such, ASME Section XI inspections remain as an aging management program for pressurizer safe-ends.

A reconciliation of the final SER for WCAP-14574 -A applicant action items is provided in Table 3.2.0-3.

Conclusion

The programs and activities selected to manage the aging effects of the Reactor Coolant System are identified in Table 3.2-1 and Table 3.2-2. The results of the applicant action item reviews are also contained in these tables, but in the SRP format. A description of these aging management activities is provided in Appendix B, along with the demonstration that the identified aging effects will be managed for the period of extended operation. Therefore, based on the demonstrations provided in Appendix B, the effects of aging associated with the system components will be adequately managed so that there is reasonable assurance that the intended function(s) will be maintained consistent with the current licensing basis during the period of extended operation.

Table 3.2.0-1 Class 1 Piping and Associated Pressure Boundary Components - WCAP-14575-A Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(1) The license renewal applicant is to verify that its plant is bounded by the topical report. Further, the renewal applicant is to commit to programs described as necessary in the topical report to manage the effects of aging during the period of extended operation on the functionality of the reactor coolant system piping. Applicants for license renewal will be responsible for describing any such commitments and identifying how such commitments will be controlled. Any deviations from the aging management programs within this topical report described as necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the reactor coolant system piping and associated pressure boundary components or other information presented in the report, such as materials of construction, will have to be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1).	The Ginna Station Class 1 piping and reactor coolant pumps are bounded by the topical report with regard to design criteria and features, materials of construction, fabrication techniques, installed configuration, modes of operation and environments/exposures. Aging management programs necessary to manage the effects of aging are consistent with those described in the topical report. Program commitments to manage the effects of aging for Class 1 piping and reactor coolant pumps are described in Appendix B of the License Renewal Application and include the following: • One-Time Inspection Program for Small-Bore Class 1 Piping • Water Chemistry Control Program • ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program • Boric Acid Corrosion Program.
(2) Summary description of the programs and evaluation of Time-Limited Aging Analyses are to be provided in the license renewal FSAR supplement in accordance with 10 CFR 54.21(d).	A summary of the programs identified to manage the effects of aging for Class 1 piping and reactor coolant pumps will be included in the UFSAR. A markup of the UFSAR sections affected by the TLAA evaluations will also be included in the UFSAR revision.

Table 3.2.0-1 Class 1 Piping and Associated Pressure Boundary Components - WCAP-14575-A Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(3) The renewal applicant should complete the updated review of generic communications and capture any additional items not identified by the original review.	The entire set of NRC Generic Communications was reviewed using an automated text search routine developed for WOG. Initial searches were made for the occurrence of terms relating to components within the scope of WCAP-14575-A. Then, the titles of all selected documents were reviewed to eliminate those which did not relate to age-related degradation or which related to equipment not included in WCAP-14575-A. The remaining documents were individually reviewed to determine the applicable aging effect(s). These resulting documents are included in the summary provided in Table 3-1 in WCAP-14575-A An updated review of industry operating experience has been conducted independently by RG&E in support of license renewal activities. This review has included NRC Generic Communications through December 2001.

Table 3.2.0-1 Class 1 Piping and Associated Pressure Boundary Components - WCAP-14575-A Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(4) The license renewal applicant must provide a description of all insulation used on austenitic stainless steel NSSS piping to ensure the piping is not susceptible to stress-corrosion cracking from halogens.	During construction, the Class 1 piping was insulated in accordance with the applicable Westinghouse Equipment Specification. As described in the Ginna Station UFSAR Section 5.2.3.2, "All external insulation of the reactor coolant system components is compatible with the component materials. All other external corrosion resistant surfaces in the reactor coolant system are insulated with a low or halide-free insulating material." Generally, the piping is insulated with a calcium carbonate material covered with a stainless steel sheet covering. Blanket insulation made from a halide-free fabric is also used at locations where periodic inspections and maintenance are required. The Reactor Coolant Pump casings and the SG Channel Heads are insulated with a stainless steel reflective insulation. Since the insulation that is used on the reactor coolant piping is low halide or halide free, the piping is not susceptible to stress corrosion cracking initiated by such halides.
(5) The license renewal applicant should describe how each plant-specific AMP addresses the following 10 elements: (1) scope of the program, (2) preventive actions, (3) parameters monitored or inspected, (4) detection of aging effects, (5) monitoring and trending, (6) acceptance criteria, (7) corrective actions, (8) confirmation process, (9) administrative controls, and (10) operating experience.	Programs necessary to manage the effects of aging for Class 1 piping and reactor coolant pumps address the 10 elements identified. These programs (including the 10 elements) are described in Appendix B of the License Renewal Application.

Table 3.2.0-1 Class 1 Piping and Associated Pressure Boundary Components - WCAP-14575-A Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(6) The license renewal applicant should perform additional inspection of small-bore Reactor Coolant System piping, that is, less than 4-inch-size piping, for license renewal to provide assurance that potential cracking of small-bore Reactor Coolant System piping is adequately managed during the period of extended operation.	A sample of small-bore (< 4-inch NPS) piping welds will be inspected by a volumetric technique prior to the end of the current licensing period. The sample population will be selected on the basis of piping geometry, size, and flow condition. The aging management review and specific program commitments for Class 1 small-bore piping are addressed in Appendix B of the License Renewal Application.

Table 3.2.0-1 Class 1 Piping and Associated Pressure Boundary Components - WCAP-14575-A Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(7) Components that have delta ferrite levels below the susceptibility screening criteria have adequate fracture toughness and do not require supplemental inspection. As a result of thermal embrittlement, components that have delta ferrite levels exceeding the screening criterion may not have adequate fracture toughness and do require additional evaluation or examination. The license renewal applicant should address thermal-aging issues in accordance with the staff’s comments in Section 3.3.3 of this evaluation.	Reduction in fracture toughness for CASS Class 1 piping components and reactor coolant pump casings due to thermal aging embrittlement is addressed by Leak-Before-Break (LBB) analyses. These analyses are identified as TLAAs and are discussed in Section 4.0 of the License Renewal Application. The LBB (fracture mechanics) analyses demonstrate that significant margin exists between detectable flaw sizes and unstable flaws assuming “fully-aged” CASS properties. The Ginna Station methodology is consistent with the staff comments. The following clarification is also provided: The WOG approach to the potential for reduced fracture toughness in CASS components in WCAP-14575-A does not rely on susceptibility screening using delta ferrite; it conservatively assumes that all CASS Class 1 RCS components are potentially susceptible. The WOG report specifies an accepted analytical technique (Leak-Before-Break analysis) as the primary aging management approach to demonstrate adequate fracture toughness at end-of-life. Only if this approach fails are alternative “corrective” actions specified: repair, replacement, or the ASME Section XI inservice examination and flaw evaluation approach. (NOTE: Open item #6 deals with clarifications of these corrective actions.) The staff's comments in Section 3.3.3 of the DSER for valve bodies / pump casings state that existing ASME Section XI inspection requirements are adequate, with the alternative being ASME Code Case N-481 for pump casings. The WOG report specifies demonstration of compliance with the N-481 requirements as the primary aging

Table 3.2.0-1 Class 1 Piping and Associated Pressure Boundary Components - WCAP-14575-A Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(7) (continued)	management approach, with the supplemental visual inspections. If this approach fails, then ASME Section XI volumetric ISI is specified as the alternative. For both LBB and N-481, the WOG report clarifies that “fully-aged” fracture toughness data must be used for the limiting materials for the extended period of operation.
(8) The license renewal applicant should perform additional fatigue evaluations or propose an AMP to address the components labeled I-M and I-RA in Tables 3-2 through 3-16 of WCAP-14575.	An automated cycle counting and Fatigue Monitoring Program (FatiguePro^{TM}) has been implemented at Ginna Station. A review has been conducted of fatigue-sensitive locations in Class 1 piping systems, including components labeled I-M and I-RA in Tables 3-2 through 3-16 of WCAP-14575-A. Locations with highest predicted fatigue usage have been selected for monitoring. These include the surge line, charging nozzle, safety injection nozzle, and RHR tee. Locations subjected to severe thermal transients or fluctuations due to stratification are monitored using a stress-based fatigue methodology. A discussion of the fatigue-monitoring methodology is included in Section 4.0 of the License Renewal Application.

Table 3.2.0-1 Class 1 Piping and Associated Pressure Boundary Components - WCAP-14575-A Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(9) The staff recommendation for the closure of GSI-190 “Fatigue Evaluation of Metal Components for 60-Year Plant Life” is contained in a December 26, 1999, memorandum from Ashok Thadani to William Travers. The license renewal applicant should address the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal. The evaluation of a sample of components with high-fatigue usage factors using the latest available environmental fatigue data is an acceptable method to address the effects of the coolant environment on component fatigue life.	Transient cycle projections to 60 years of plant operation have been made using both a conservative linear cycle projection and a more realistic weighted projection, which assumes that the more recent plant operating history is more representative of future operation than earlier plant history. This assessment of the frequency and severity of actual plant transients demonstrates that there is sufficient conservatism in the original design basis transient set, based on either method of projection (linear or weighted), to adequately bound the period of extended operation. However, a sample of Class 1 piping components with potentially high fatigue usage factors has been selected for monitoring using the FatiguePro™ Automated Cycle-Counting and Fatigue Monitoring Program. Fatigue usage for these locations will be computed by cycle-based or stress-based software modules including the latest available environmental factors. For components with CUFs which are expected to exceed 1.0 during the period of extended operation, corrective actions will include one or more of the following options: • Perform an explicit fatigue analysis (i.e., using sophisticated methods in ASME Section III NB-3200 or NB-3600) including environmental factors to lower the CUF below 1.0 prior to the end of the current license period, or • Repair of the fatigue-sensitive location(s), or • Replacement of the fatigue-sensitive location(s), or • Manage the effects of fatigue by an inspection program that has been reviewed and approved by the NRC (i.e., periodic non-destructive examination of the fatigue-sensitive locations at inspection intervals to be determined by a method accepted by the NRC).

Table 3.2.0-1 Class 1 Piping and Associated Pressure Boundary Components - WCAP-14575-A Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(9) (continued)	A further discussion of the Metal Fatigue TLAA is presented in Section 4.3 of the License Renewal Application.
(10) The license renewal applicant should revise AMP-3.6 to include an assessment of the margin on loads in conformance with the staff guidance provided in Reference 11. In addition, AMP-3.6 should be revised to indicate If the CASS component is repaired or replaced per ASME Code, Section XI IWB4000 or IWB7000, a new LBB analysis based on the material properties of the repaired or replaced component (and accounting for its thermal aging through the period of extended operation, as appropriate), is required to confirm the applicability of LBB. The inservice examination/flaw evaluation option is, per the basis on which the NRC staff has approved LBB in the past, insufficient to reestablish LBB approval.	An LBB analysis has been performed in accordance with NUREG-1061, for the Ginna reactor coolant loop piping applicable to the extended period of operation. The analysis considered loading, pipe geometry and fracture toughness (including the reduction in fracture toughness of CASS components in the RCS, i.e., elbows and RCP casings, due to thermal aging) to assess crack stability in the reactor coolant piping for the period of extended operation. The results demonstrated that significant margin exists between detectable flaw sizes and unstable flaws. Additionally, fatigue crack growth rates including environmental effects were evaluated for primary loop piping materials and shown to be insignificant. The Ginna Station Inservice Inspection Program requires that any repair or replacement of CASS components be performed in accordance with the requirements of ASME Section XI. This would include a new LBB analysis based on the material properties of the repaired or replaced component (and accounting for thermal aging through the period of extended operation).

Table 3.2.0-2 Reactor Internals - WCAP-14577, Rev. 1-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(1) To ensure applicability of the results and conclusions of WCAP-14577 to the applicant's plant(s), the license renewal applicant is to verify that the critical parameters for the plant are bounded by the topical report. Further, the renewal applicant must commit to programs described as necessary in the topical report to manage the effects of aging during the period of extended operation on the functionality of the reactor vessel components. Applicants for license renewal will be responsible for describing any such commitments and proposing the appropriate regulatory controls. Any deviations from the aging management programs described in this topical report as necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the reactor vessel internal components or other information presented in the report, such as materials of construction, must be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1).	The Ginna Station reactor vessel internals are bounded by WCAP-14577 Rev. 1-A with respect to design criteria and features, materials of construction, fabrication techniques, installed configuration, mode of operation and environments/ exposures. Programs necessary to manage the effects of aging are identified in Table 3.2-1 and Table 3.2-2 and summarized in Appendix B of the Application.
(2) A summary description of the programs and activities for managing the effects of aging and the evaluation of TLAAs must be provided in the license renewal FSAR supplement in accordance with 10 CFR 54.21(d).	Programs necessary to manage the effects of aging for the Ginna reactor vessel internals are the ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program, the Reactor Vessel Internals Program, and the Water Chemistry Control Program. Summary descriptions of these programs are provided in Appendix A and Appendix B of the LRA. The only TLAA applicable to the Ginna reactor internals is fatigue. The TLAA for metal fatigue is evaluated in Section 4.3 of the LRA.
(3) For the holddown spring, applicants for license renewal are expected to address intended function, aging management review, and appropriate aging management program(s).	The holddown spring is within the scope of license renewal for the Ginna reactor vessel internals. The intended function, results of the aging management review, and aging management program for the holddown spring are provided in Table 2.3.1-3, Table 3.2-1, Table 3.2-2 and Appendix B of the LRA.

Table 3.2.0-2 Reactor Internals - WCAP-14577, Rev. 1-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(4) The license renewal applicant must address aging management review, and appropriate aging management program(s), for guide tube support pins	In Section 2.6.7.2 of the GTR, it is stated, "As noted above, pin degradation does not lead to a loss of intended function. Generally, pin replacement is considered to be a sound maintenance practice to preclude degradation when industry experience indicates that such degradation has been observed." All 33 guide tube support pins were replaced at Ginna Station during the 1986 Refueling outage. The new pins were fabricated using a Framatome design which had been installed in French nuclear reactors where SCC pin failures had occurred. The original design of the support pin was susceptible to SCC due to an undesirable microstructure caused by solution heat treatment of the pins at a temperature less than 1800°F, followed by age-hardening and application of high preload, resulting in high tensile stresses. The replacement pins were solution heat-treated at 2000°F, followed by age-hardening at 1300°F. Other improvements in machined configuration and surface finish were incorporated in the new design. Final installation torque was reduced to achieve adequate cold preload and still maintain a tight joint. No evidence of cracking of the redesigned guide tube support pins has since been observed at Ginna. The effects of SCC on reactor internals guide tube support pins fabricated from Alloy X-750 with the updated pin designs may therefore be considered insignificant (GTR 3.1.2.2). However, loss of material due to wear is also identified as an aging effect requiring management for the support pins. The ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program (Subsection IWB) is credited with managing loss of material due to wear for the support pins.

Table 3.2.0-2 Reactor Internals - WCAP-14577, Rev. 1-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(5) The license renewal applicant must explicitly identify the materials of fabrication of each of the components within the scope of the topical report. The applicable aging effect should be reviewed for each component based on the materials of fabrication and the environment.	The materials of fabrication were explicitly identified for all subcomponents of the Ginna reactor vessel internals within the scope of license renewal. The list of these materials and source documents are available for review on site. The aging effect evaluations are performed based on these materials and the appropriate environment.
(6) The license renewal applicant must describe its aging management plans for loss of fracture toughness in cast austenitic stainless steel RVI components, considering the synergistic effects of thermal aging and neutron irradiation embrittlement in reducing the fracture toughness of these components.	There are no reactor vessel internals components at Ginna Station within the scope of license renewal which are fabricated from cast austenitic stainless steel.

Table 3.2.0-2 Reactor Internals - WCAP-14577, Rev. 1-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(7) The license renewal applicant must describe its aging management plans for void swelling during the license renewal period.	Recent studies of irradiation-induced swelling and stress relaxation suggest that swelling problems, if they arise in PWR core internals, would be highly localized, occurring in the higher flux and temperature locations. Irradiation-enhanced stress relaxation (or irradiation creep) refers to the accumulation of deformation strain over an extended time period, typically at elevated temperatures. Stress relaxation may mitigate loads resulting from void swelling. TEM studies of thin foils prepared from an intact baffle/former bolt and locking device removed from the Ginna reactor vessel internals in 1999 indicate that voids were present in the threaded end of the bolt but not in the head or the 304 SS locking device. The void volume, 0.004% maximum observed in the 347 SS bolt material, is small and preliminary extrapolation to the end of extended life using a simple square law suggests that void swelling should not be a concern. Ginna Station is participating in industry initiatives to determine the extent of the concerns associated with void swelling and what appropriate changes to the Reactor Vessel Internals Program may be required once an industry position has been established.
(8) Applicants for license renewal must describe how each plant-specific AMP addresses the following elements: (1) scope of the program, (2) preventative actions, (3) parameters monitored or inspected, (4) detection of aging effects, (5) monitoring and trending, (6) acceptance criteria, (7) corrective actions, (8) confirmation process, (9) administrative controls, and (10) operating experience.	The programs necessary to manage the effects of aging for the Ginna reactor vessel internals (RVIs) address the 10 elements identified. These elements are described in Appendix B of the LRA.

Table 3.2.0-2 Reactor Internals - WCAP-14577, Rev. 1-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(9) The license renewal applicant must address plant-specific plans for management of cracking (and loss of fracture toughness) of RVI components, including any plans for augmented inspection activities.	The Ginna Station Reactor Vessel Internals Program is credited for managing cracking and loss of fracture toughness of RVI components. This program is described in Appendix B of the LRA and includes participation in industry initiatives and efforts for development of appropriate enhanced inspection techniques to permit detection and characterizing very small features of interest.
(10) The license renewal applicant must address plant-specific plans for management of age-related degradation of baffle/former and barrel/former bolting, including any plans for augmented inspection activities.	During the 1999 refueling outage, the entire population of 728 Type 347 stainless steel baffle/former bolts were selected for inspection by UT at Ginna Station. Of this number, only 639 bolts could actually be inspected due to limitations on accessibility. A total of 56 bolts were replaced with Type 316 stainless steel bolts during the outage. These were bolts that were found to contain defect-like indications and were part of a pre-qualified minimum bolt pattern for two-loop nuclear plants that was generated by the Westinghouse Owners Group (WCAP-15036). Maintaining the structural integrity of the bolts within this pattern assures compliance with requirements of ASME III, Subsection NG (1989), considering dynamic loads generated by a 10" line break in the reactor coolant system. This LOCA load bounds those that are generated by effects of earthquake, thermal, deadweight, and flow-induced vibration. No further inspections of baffle/former or barrel/former bolts are planned at Ginna Station. However, RG&E will continue to monitor and participate in industry initiatives with regard to baffle/former and barrel/former bolt cracking.
(11) The license renewal applicant must address the TLAA of fatigue on a plant-specific basis	A discussion of fatigue of reactor vessel internals is presented in Section 4.3 of the LRA.

Table 3.2.0-3 Pressurizers - WCAP-14574-A Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(1) **3.3.1.1–1 -** License renewal applicants should identify the TLAAs for the pressurizer components, define the associated CUF and, in accordance with 10 CFR 54.21(c)(1), demonstrate that the TLAAs meet the CLB fatigue design criterion, CUF $\leq$1.0, for the extended period of operation, including the insurge/outsurge and other transient loads not included in the CLB which are appropriate to such an extended TLAA, as described in the WOG report "Mitigation and Evaluation of Thermal Transients Caused by Insurges and Outsurges," MUHP–5060/5061/5062, and considering the effects of the coolant environment on critical fatigue location. The applicant must describe the methodology used for evaluating insurge/outsurge and other off-normal and additional transients in the fatigue TLAAs.	The only TLAA identified for the Ginna pressurizer is thermal fatigue. Transient cycle projections to 60 years of plant operation have been made using both a conservative linear cycle projection and a more realistic weighted projection, which assumes that the more recent plant operating history is more representative of future operation than earlier plant history. This assessment of the frequency and severity of actual plant transients demonstrates that there is sufficient conservatism in the original design basis transient set, based on either method of projection (linear or weighted), to adequately bound the period of extended operation. However, in order to address insurge/outsurge transients and thermal stratification, an automated cycle counting and Fatigue Monitoring Program (FatiguePro^{TM}) has been implemented at Ginna Station. Four fatigue-sensitive pressurizer locations (spray nozzle, surge nozzle, upper shell, and heater well penetration) have been selected for fatigue monitoring using a stress-based method which computes real-time fatigue usage based on actual plant transient data. These locations will be monitored for a sufficient period of time to establish a baseline cyclic history and cumulative fatigue usage. The effects of coolant environment are included in this computation. For locations with CUFs which are expected to exceed 1.0 during the period of extended operation, corrective actions will include one or more of the following options: • Perform an explicit fatigue analysis (i.e., using sophisticated methods in ASME Section III NB-3200 or NB-3600) including environmental factors to lower the CUF below 1.0 prior to the end of the current license period, or • Repair of the fatigue-sensitive location(s), or • Replacement of the fatigue-sensitive location(s), or

Table 3.2.0-3 Pressurizers - WCAP-14574-A Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(1) (continued)	• Manage the effects of fatigue by an inspection program that has been reviewed and approved by the NRC (i.e., periodic non-destructive examination of the fatigue-sensitive locations at inspection intervals to be determined by a method accepted by the NRC).
(2) **3.2.2.1–1 -** In the report, WOG concluded that general corrosion is nonsignificant for the internal surfaces of Westinghouse-designed pressurizers and that no further evaluations of general corrosion are necessary. While the staff concurs that hydrogen overpressure can mitigate the aggressive corrosive effect of oxygen in creviced geometries on the internal pressurizer surfaces, applicants for license renewal will have to provide a basis (statement) in their plant-specific applications about how their water chemistry control programs will provide for a sufficient level of hydrogen overpressure to manage crevice corrosion of the internal surfaces of their pressurizer.	Hydrogen concentration in the reactor coolant system (RCS) primary water at Ginna Station is strictly maintained within specified limits (25 to 50 cc/kg) by measurement of hydrogen concentrations in periodic RCS samples, and adjusting hydrogen overpressure in the volume control tanks accordingly. The hydrogen concentration limits established for the RCS ensure that general corrosion is non-significant for the internal surfaces of the Ginna pressurizer as well as other Class 1 components. Hydrogen concentration limits for the RCS are delineated in the Ginna Station Water Chemistry Control Program described in Appendix B of the License Renewal Application.
(3) **3.2.2.1-2 -** The staff finds that the criteria in GL 88–05 and the Section XI requirements for conducting system leak tests and VT–2 type visual examinations of the pressurizer pressure boundary are acceptable programs for managing boric acid corrosion of the external, ferritic surfaces and components of the pressurizer. However, the report fails to refer to the actual provisions in the ASME Code, Section XI that require mandatory system leak tests of the pressurizer boundary. The applicants must identify the appropriate Code inspection requirements from ASME Code Table IWB-2500-1.	Leak testing of the Ginna pressurizer is required by ASME Section XI, Subsection IWB, Table IWB-2500-1, Category B-P.

Table 3.2.0-3 Pressurizers - WCAP-14574-A Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(4) **3.2.2.3.2–1 -** The staff concurs that the potential to develop SCC in the bolting materials will be minimized if the yield strength of the material is held to less than 150 ksi, or the hardness is less than 32 on the Rockwell C hardness scale; however, the staff concludes that conformance with the minimum yield strength criteria in ASME Specification SA–193 Grade B7 does not in itself preclude a quenched and tempered low-alloy steel from developing SCC, especially if the acceptable yield strength is greater than 150 ksi. To take credit for the criteria in EPRI Report NP–5769, the applicant needs to state that the acceptable yield strengths for the quenched and tempered low-alloy steel bolting materials (e.g., SA–193, Grade B7 materials) are in the range of 105–150 ksi.	Although there have been a few reported cases of cracking of bolting in the industry caused by SCC, these have been attributed to susceptible high yield stress materials exposed to aggressive environments, such as lubricants containing molybdenum disulfide. However, a survey of industry experience, technical literature, and laboratory corrosion studies (documented in EPRI Report NP-5769) indicates that SCC should not be a concern for closure bolting in nuclear power plant applications if the specified minimum yield strength is <150 Ksi. For quenched and tempered low-alloy steels typically used for closure bolting (e.g., SA193, Grade B7), susceptibility to SCC is controlled by yield strength. The minimum yield strength specified in SA193 for Grade B7 material is 105 Ksi, which is well below the threshold value of 150 Ksi identified in EPRI Report NP-5769. Furthermore, the selection and use of fastener lubricants for pressure boundary components has been controlled by the Ginna Station Quality Assurance Program since 1983 as part of the response to IE Bulletin 82-02. Limits are also imposed on levels of contaminants such as chlorides and sulfur compounds (including molybdenum disulfide) in lubricants and sealant compounds. Therefore, it is reasonable to conclude that failure by SCC should not be a significant issue for SA193 Grade B7 bolting materials. Ginna Station operating experience supports this conclusion. Therefore, cracking due to SCC is not considered to be an aging effect requiring management for the Ginna pressurizer bolting.

Table 3.2.0-3 Pressurizers - WCAP-14574-A Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(5) **3.2.5-1 -** The staff considers the discussion in Section 3.5.2 to be extremely confusing in that it appears WOG is making three different conclusions that conflict with one another: a. That fluid flow velocity and particulate conditions are not sufficient in the pressurizer to consider that erosion is a plausible degradation mechanism that could affect the integrity of the subcomponents in the pressurizer. b. That seven components in the pressurizer (refer to the list above) are exposed to fluid flows that have the potential to result in erosion of the components. c. That only one component in the pressurizer (the spray head) is exposed to a fluid flow that has the potential to result in erosion of the component. The applicant should state why erosion is not plausible for the surge nozzle thermal sleeve, spray nozzle thermal sleeve, surge nozzle safe-end, and spray nozzle safe-end. If erosion is plausible, then an AMP is required.	Based on the aging management review of the Ginna pressurizer, loss of material due to erosion is not an aging effect requiring management. Austenitic stainless steels are considered to be resistant to erosion in PWR operating environments. The austenitic stainless steel surge and spray nozzle thermal sleeves and safe ends, and the surge nozzle retaining baskets are not subject to flow rates that are sufficiently high to cause erosion. The spray head couplings and the spray heads do not perform license renewal intended functions and, thus, do not require an aging management review.
(6) **3.3–1 -** Applicants for license renewal must describe how each plant-specific AMP addresses the following 10 elements: (1) scope of the program, (2) preventive action, (3) parameters monitored or inspected, (4) detection of aging effects, (5) monitoring and trending, (6) acceptance criteria, (7) corrective actions, (8) confirmation process, (9) administrative controls, and (10) operating experience.	Programs necessary to manage the effects of aging for the Ginna pressurizer address the 10 elements identified. Detailed program descriptions (including the 10 elements) are provided in Appendix B of the License Renewal Application.

Table 3.2.0-3 Pressurizers - WCAP-14574-A Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(7) **3.3.2.1–1 -** Applicants for license renewal must provide sufficient details in their LRAs about how their GL 88–05 programs and ISI programs will be sufficient to manage the corrosive effects of boric acid leakage on their pressurizer components during the proposed extended operating terms for their facilities, including postulated leakage from the pressurizer nozzles, pressurizer nozzle-to-vessel welds, pressurizer nozzle-to-safe end welds, and pressurizer manway bolting materials.	Loss of material due to boric acid wastage resulting from boric acid leakage is an aging effect requiring management affecting the external surfaces of the Ginna pressurizer, including bolting materials. The Ginna Station Boric Acid Corrosion Program and the ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program are credited with managing this aging effect. Detailed program descriptions provided in Appendix B of the License Renewal Application demonstrate that the effects of aging due to boric acid leakage will be adequately managed during the period of extended operation.

Table 3.2.0-3 Pressurizers - WCAP-14574-A Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(8) **3.3.2.2–1 -** The staff concludes that an AMP is necessary to control and manage the potential for SCC to occur in welded pressurizer penetration nozzles and manway bolting materials, and recommends that a licensee could credit the following programs as the basis for managing the phenomena of PWSCC/IGSCC of the pressurizer components: (1) the primary coolant chemistry control program; (2) the ISI program for the pressurizers; and (3) the plant-specific quality assurance program as it pertains to assuring that previous welding activities on welds in the pressurizer have been controlled in accordance with the pertinent requirements of 10 CFR Part 50, Appendix B, and with the pertinent welding requirements of the ASME Code for Class 1 systems. The staff concludes that applicants need to extend AMP–2–1 to the pressurizer penetration nozzles, to the nozzle-to-vessel welds, and to the manway bolting materials, and to include the appropriate Code requirements among the program attributes listed in Table 4–1 and summarized in the text in Section 4.1 of the report. Applicants for license renewal must provide sufficient details in their LRAs as to how their primary coolant chemistry control programs, ISI programs, and 10 CFR Part 50, Appendix B, quality assurance programs will be sufficient to manage the potential for SCC to occur in the pressurizer nozzle components and bolted manway covers during the proposed extended operating terms for their facilities.	Stress corrosion cracking (SCC), as it applies to the pressurizers, is identified as an aging effect requiring management for pressurizer parts exposed to primary (treated) water. The Ginna Station Water Chemistry Control Program and the ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program are credited for managing cracking due to SCC. The Quality Assurance Program applies to all aging management programs credited for license renewal. The program descriptions provided in Appendix B of the License Renewal Application demonstrate that these programs will adequately manage cracking due to SCC throughout the extended period of operation. As stated previously in the response to Applicant Action Item 3.2.2.3.2-1, cracking due to SCC is not an aging effect requiring management for pressurizer bolting.

Table 3.2.0-3 Pressurizers - WCAP-14574-A Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(9) **3.3.2.2–2 -** Applicants must propose an AMP to verify whether or not thermal fatigue-induced cracking has propagated through the clad into the ferritic base metal or weld metal beneath the clad.	There is no industry experience to suggest that cracks initiating at the clad inner surfaces in the pressurizer will propagate into the underlying base metal or weld material. Observed flaws in other plants were monitored for an extended period of time, and no significant flaw growth was observed. In 1990, several indications were discovered at the Connecticut Yankee Plant. UT inspection confirmed that the indications did not penetrate into the ferritic base metal, and therefore, in accordance with ASME Section XI, the indications were acceptable without repair. A surveillance program was initiated, and after two follow-up inspections that showed no change, the surveillance program was discontinued with NRC approval. In several of the cases of observed cracking, fracture mechanics analyses were performed and demonstrated that the cladding indications would not compromise the integrity of the primary system components. At temperatures >180°F, the cladding has virtually no impact on fracture behavior. This is the low end of the plant operating range. ASME Section XI flaw evaluation rules require that the effects of cladding must be considered in any structural integrity evaluation, especially for postulated flaws that penetrate the cladding into the base metal. The actual impact of the cladding on such an evaluation is negligible. The pressurizer shell design considers fatigue usage throughout the operating lifetime and includes adequate margin. This is expected to preclude the formation of fatigue cracks in the cladding material. The fracture mechanics evaluations performed for actual observed cracks in other plants indicate that the cracks do not grow significantly over the plant lifetime. Therefore, a specific aging management program to manage fatigue cracking of the pressurizer cladding is not required.

Table 3.2.0-3 Pressurizers - WCAP-14574-A Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(9) (continued)	Cracking due to fatigue is identified as a Time-Limited Aging Analysis for the Ginna pressurizer, and is analytically addressed in this TLAA. The conclusion of this analysis is that adequate margin exists in the original design-basis transient set to envelope the period of extended operation. However, a Fatigue Monitoring Program has been implemented as a confirmatory program to ensure that the fatigue analysis remains valid for the license renewal term. Cracking due to flaw growth and stress corrosion is an aging effect requiring management. As noted above, programs credited to manage cracking of pressurizer parts include the Water Chemistry Control Program and the ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program, both of which are described in Appendix B of the License Renewal Application. Based on the aging management review performed for the Ginna pressurizer, no additional aging management program is required.

Table 3.2.0-3 Pressurizers - WCAP-14574-A Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(10) **3.3.2.2–3 -** The staff is concerned that IGSCC in the heat- affected zones of 304 stainless steel supports that are welded to the pressurizer cladding could grow as a result of thermal fatigue into the adjacent pressure boundary during the license renewal term. The staff considers that these welds will not require aging management in the extended operating periods if applicants can provide a reasonable justification that sensitization has not occurred in these welds during the fabrication of these components. Therefore, applicants for license renewal must provide a discussion of how the implementation of their plant-specific procedures and quality assurance requirements, if any, for the welding and testing of these austenitic stainless steel components provides reasonable assurance that sensitization has not occurred in these welds and their associated heat-affected zones. In addition, the staff requests that applicants for license renewal identify whether these welds fall into Item B8.20 of Section XI Examination Category B–H, Integral Attachments for Vessels, and if applicable, whether the applicants have performed the mandatory volumetric or surface examinations of these welds during the ISI intervals referenced in the examination category.	Both the cladding material (308L) used to protect the pressurizer alloy steel shell from primary water, and the weld material (309L) used to join the pressurizer internal supports and the pressurizer cladding were selected to have sufficiently low carbon content to minimize the likelihood of sensitization in these welds.The low carbon (and nitrogen) content of the 304 stainless steel material in the heater support plates and the surge nozzle retaining basket minimize the susceptibility of the material to sensitization as a result of welding. However, in spite of material selection and manufacturing processes which minimize sensitization, the possibility cannot be precluded that sensitized areas exist in the 304 stainless steel supports or their welds. The same Water Chemistry Control Program which precludes SCC in other PWR primary system materials is also effective in preventing SCC in these pressurizer components and welds. Rigorous control of oxygen and chlorides provides an essentially benign environment which has been shown to be effective both in laboratory experiments and years of operating experience. Therefore, the presence of sensitized stainless steel material does not necessarily result in any increase in susceptibility to IGSCC. Note that even in laboratory cases where severely sensitized stainless steels have been deliberately exposed to PWR environments, no intergranular attack has been observed. In summary, the Ginna Station Water Chemistry Control Program is an adequate aging management program to preclude SCC in the pressurizer internal attachment welds for the period of extended operation for the following reasons:

Table 3.2.0-3 Pressurizers - WCAP-14574-A Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(10) (continued)	1. It is possible that some locations of the welded stainless steel attachments in the pressurizer are sensitized, even with the use of 308L weld material and careful control of the welding processes; 2. Studies and operating experience have shown that PWR environments do not lead to stress corrosion cracking in sensitized stainless steel; 3. Service experience has demonstrated that stress corrosion cracking does not occur in stainless steels in a PWR environment, whether or not they are sensitized. In response to the question regarding the applicability of Item B8.20 of Examination Category B-H, this category applies to exterior attachments such as the support skirt, seismic lug and support bracket, and is not applicable to the interior attachment welds.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(1) Reactor coolant pressure boundary components	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	Consistent with NUREG-1801. Cumulative fatigue damage was identified as an aging effect requiring management during the period of extended operation for components listed in this component grouping. Thermal fatigue is addressed as a TLAA in Section 4.3 for those components which contain time-limited assumptions defined by the current operating term and incorporated into the current licensing basis. Secondary-side steam generator pressure boundary components such as the top head, steam nozzle, upper and lower shells, transition cone and feedwater nozzle/impingement plate are included in this grouping although they are not part of the reactor coolant pressure boundary.
(2) Steam generator shell assembly	Loss of material due to pitting and crevice corrosion	Inservice inspection; water chemistry	Yes, detection of aging effects is to be further evaluated	Consistent with NUREG-1801. Loss of material due to general, pitting and crevice corrosion of the steam generator shell assembly (including transition cone) are identified as an aging effect requiring management at Ginna Station. Loss of material from all applicable aging mechanisms on steam generator secondary-side internal surfaces is effectively managed by control of secondary-side water chemistry through the Water Chemistry Control Program and inservice inspections performed in accordance with the ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program. In addition, the Ginna Station Steam Generator Tube Integrity Program which was developed in accordance with NEI Initiative 97-06 provides all-inclusive guidance for the management of steam generator assets. Assessment of secondary-side aging mechanisms is included in the scope of Steam Generator Tube Integrity Program.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(3) Pressure vessel ferritic materials that have a neutron fluence greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of fracture toughness due to neutron irradiation embrittlement	TLAA, evaluated in accordance with Appendix G of 10 CFR 50 and RG 1.99	Yes, TLAA	Consistent with NUREG-1801. Loss of fracture toughness in ferritic reactor pressure vessel materials due to neutron irradiation embrittlement has been identified as an aging effect requiring management during the period of extended operation. Reactor pressure vessel TLAAs, including RT_{PTS}, adjusted reference temperature, and equivalent margins analysis are addressed in Section 4.2. This component group includes the vessel shell and nozzles.
(4) Reactor vessel beltline shell and welds	Loss of fracture toughness due to neutron irradiation embrittlement	Reactor vessel surveillance	Yes, plant specific	Consistent with NUREG-1801. Loss of fracture toughness in reactor vessel beltline shell and weld materials due to neutron irradiation embrittlement has been identified as an aging effect requiring management during the period of extended operation. The upper shell and nozzles are not subject to significant neutron irradiation exposure because of their physical distance from the reactor core. The limiting beltline material in the Ginna Station reactor vessel is the intermediate-to-lower shell beltline circumferential weld. The Ginna Station Reactor Vessel Surveillance Program, in conjunction with TLAA analyses, effectively manages loss of fracture toughness in the beltline materials. The Reactor Vessel Surveillance Program provides adequate material property and neutron dosimetry data to predict fracture toughness in beltline materials at the end of the period of extended operation. In addition, equivalent margins analyses have been performed in accordance with 10 CFR 50 Appendix G methods. These fracture mechanics analyses (see TLAAs, Section 4.2) provide assurance that beltline material toughness values in the Ginna Station reactor vessel will remain at acceptable levels through the period of extended operation.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(5) Westinghouse and B&W baffle/former bolts	Loss of fracture toughness due to neutron irradiation embrittlement and void swelling	Plant specific	Yes, plant specific	Consistent with NUREG-1801. Loss of fracture toughness due to neutron irradiation embrittlement was identified as an aging effect requiring management for the Ginna Station baffle/former bolts. A combination of the ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program and the Reactor Vessel Internals Program (described in Appendix B) will be used to manage this aging effect. During the 1999 refueling outage, all accessible Type 347 stainless steel baffle/former bolts were inspected at Ginna Station and bolts with defect-like indications were replaced with Type 316 stainless steel bolts. The bolt inspection and replacement program assured the structural integrity of bolts within a pre-qualified minimum pattern generated by WOG and thereby assured compliance with ASME Section III, Subsection NB (1989). In addition, destructive metallurgical analysis of intact Type 347 bolts revealed only minor evidence of voids near the threaded end of one bolt, but not in the head end. The void volume (.004%) was small and preliminary extrapolations to the end of life suggest that void swelling should not be a concern. For this reason, change in dimensions due to void swelling is not expected to represent a concern for baffle/former bolts in the Ginna Station reactor vessel internals. These facts notwithstanding, Ginna Station will continue to participate in WOG activities and monitor industry initiatives for the purpose of evaluating the significance of void swelling on selected PWR reactor vessel internals components. As new information and technology becomes available, the plant-specific Reactor Vessel Internals Program will be modified to incorporate enhanced surveillance techniques

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(6) Small-bore reactor coolant system and connected systems piping	Crack initiation and growth due to SCC, intergranular SCC, and thermal and mechanical loading	Inservice inspection; water chemistry; one-time inspection	Yes, parameters monitored/ inspected and detection of aging effects are to be further evaluated	Consistent with NUREG-1801. Included and evaluated with this component grouping are Non-Class 1 RCS small-bore piping, tubing, valves, and other components in connected systems. Crack initiation and growth due to SCC was identified as an aging effect requiring management in small-bore (<NPS 4) reactor coolant system piping and branch lines. Aging management of service-induced cracking will be accomplished by a combination of the Water Chemistry Control Program and the One-Time Inspection Program (described in Appendix B). NUREG-1801 further identifies the ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program as an applicable aging management program, but also notes that volumetric inspections are not required for piping < NPS 4. A sample of small-bore piping welds will be inspected using appropriate volumetric examination techniques near, but prior to, the end of the current license period. This sample will be selected to include various piping sizes, configurations and flow conditions. If a flaw is detected in the sample, the successive examinations described in ASME Code, Section XI, IWB-2420 and additional examinations as described in IWB-2430 would apply as appropriate. The proposed combination of water chemistry controls and volumetric inspections (implemented by the Water Chemistry Control Program and One-Time Inspection Program) is an effective means of managing service-induced cracking in small-bore reactor coolant system piping and connected branch lines during the period of extended operation.
(7) Vessel shell	Crack growth due to cyclic loading	TLAA	Yes, TLAA	Consistent with NUREG-1801. Underclad cracking in carbon/low-alloy steel which has been clad with austenitic stainless steel using weld-overlay processes has been identified as an aging effect requiring management and is addressed as a TLAA. An evaluation of the TLAA for underclad cracking is contained in Section 4.3.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(8) Reactor internals	Changes in dimension due to void swelling	Plant specific	Yes, plant specific	Consistent with NUREG-1801. The specific concerns arising from the effects of void swelling in reactor internals are constriction of flow paths, interference with control rod insertion, and excessive baffle/former bolt loading. Recent destructive examinations of baffle/former bolts removed from the Ginna Station reactor internals suggest that void volumes are very small and changes in dimension in baffle/former bolts due to void swelling should not be a concern during the period of extended operation. In addition, recent studies of irradiation-induced swelling and stress relaxation suggest that swelling problems, if they arise in PWR core internals, would be highly localized, occurring in the higher flux and temperature locations. Irradiation-enhanced stress relaxation (or irradiation creep) may mitigate or limit loads resulting from void swelling. For many reactor internals components, change in dimension does not represent an aging effect requiring management because the intended function of the component(s) is not affected. Additional reactor internal components not identified in NUREG-1801 that are susceptible to changes in dimension due to void swelling are identified in Table 3.2-2 Line Number (7) These facts notwithstanding, the Reactor Vessel Internals Program manages changes in dimension due to void swelling. In addition to inservice inspections performed according to the requirements of ASME Section XI, Subsection IWB, the Reactor Vessel Internals Program provides for augmented visual (VT-1) inspections for certain susceptible (or limiting) components using high resolution techniques yet to be developed. Ginna Station will continue to participate in industry investigations of aging effects applicable to reactor vessel internals as well as initiatives to develop advanced inspection techniques which will permit resolution and measurement of very small features of interest. Ginna Station will incorporate applicable results of industry initiatives related to void swelling in the Reactor Vessel Internals Program.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(9) PWR core support pads, instrument tubes (bottom head penetrations), pressurizer spray heads, and nozzles for the steam generator instruments and drains	Crack initiation and growth due to SCC and/or primary water stress corrosion cracking (PWSCC)	Plant specific	Yes, plant specific	Consistent with NUREG-1801. The reactor vessel leak detection line is fabricated from stainless steel. The portion of the line that is in scope to license renewal is included in the small-bore piping category. Management of service-induced cracking for small-bore piping is addressed in Item 6, and is consistent with NUREG-1801. The pressurizer spray head performs no license renewal intended functions at Ginna Station. The steam generator instrument nozzles are low-alloy steel, not Alloy 600, and therefore are not included in this component group. The core support pads and the bottom head instrument penetrations are fabricated from Alloy 600. Crack initiation and growth of the core support pads and the bottom head penetrations due to SCC/PWSCC is managed at Ginna Station by a combination of the Water Chemistry Control Program and the Reactor Vessel Head Penetration Inspection Program (described in Appendix B). The Reactor Vessel Head Penetration Inspection Program is a plant-specific program which includes participation in industry initiatives related to management of Alloy 600 penetration cracking issues.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(10) Cast austenitic stainless steel (CASS) reactor coolant system piping	Crack initiation and growth due to SCC	Plant specific	Yes, plant specific	Consistent with NUREG-1801. The Ginna Station pressurizer surge line nozzle is cast carbon steel (integral with the pressurizer bottom head) and clad with weld-deposited austenitic stainless steel overlay. The reactor coolant system piping is forged Type 316 stainless steel. However, the fittings (elbows) are CASS (Type CF8M). In addition, the CASS (Type CF8M) reactor coolant pump casings are also included in this component grouping. As in NUREG-1801, crack initiation and growth due to SCC was identified as an aging effect requiring management for reactor coolant system CASS components. The Ginna Station Water Chemistry Control Program monitors and controls primary water chemistry in accordance with the guidelines of EPRI TR-105714 (Rev. 5) and therefore effectively manages crack initiation and growth due to SCC. Additionally, the flaw tolerance evaluations performed by fracture mechanics analysis under the Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program (described in Appendix B) provide assurance that large margins exist for postulated flaw sizes which satisfy leakage detection criteria as compared to unstable flaw sizes.
(11) Pressurizer instrumentation penetrations and heater sheaths and sleeves made of Ni-alloys	Crack initiation and growth due to PWSCC	Inservice inspection; water chemistry	Yes, AMP for PWSCC of Inconel 182 weld is to be evaluated	There are no components fabricated from Alloy 600 in the Ginna Station pressurizer. Instrument penetrations, heater well tubes and adapters are wrought Type 316 stainless steel.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(12) Westinghouse and B&W baffle former bolts	Crack initiation and growth due to SCC and IASCC	Plant specific	Yes, plant specific	Consistent with NUREG-1801. Crack initiation and growth due to SCC and IASCC were identified as aging effects requiring management for Ginna Station baffle/former bolts. A combination of the Water Chemistry Control Program, ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program and the Reactor Vessel Internals Program will be used to manage this aging effect. During the 1999 refueling outage, all accessible Type 347 stainless steel baffle/former bolts were inspected at Ginna Station and bolts with defect-like indications were replaced with Type 316 stainless steel bolts. The bolt inspection and replacement activities assured the structural integrity of bolts within a pre-qualified minimum pattern generated by WOG and thereby assured compliance with ASME Section III, Subsection NB (1989). No further inspections of baffle/former bolts are anticipated at Ginna Station. However, Ginna Station will continue to participate in WOG activities and monitor industry initiatives for the purpose of evaluating the significance of cracking due to IASCC on selected PWR reactor vessel internals components. As new information and technology becomes available, the plant-specific Reactor Vessel Internals Program (described in Appendix B) will be modified to incorporate enhanced surveillance techniques.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(13) Westinghouse and B&W baffle former bolts	Loss of preload due to stress relaxation	Plant specific	Yes, plant specific	Consistent with NUREG-1801. Loss of mechanical closure integrity due to irradiation creep/stress relaxation was identified as an aging effect requiring management for Ginna Station baffle/former bolts. Irradiation-enhanced stress relaxation (or irradiation creep) refers to the accumulation of deformation strain over an extended time period, typically at elevated temperatures. Loss of preload due to stress relaxation will be managed jointly by the ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program and the Reactor Vessel Internals Program. Ginna Station will continue to participate in industry investigations of aging effects applicable to reactor vessel internals as well as initiatives to develop advanced inspection techniques which will permit resolution and measurement of very small features of interest. Aging management activities or surveillance techniques resulting from these initiatives will be incorporated, as required, as enhancements to the Reactor Vessel Internals Program.
(14) Steam generator feedwater impingement plate and support	Loss of section thickness due to erosion	Plant specific	Yes, plant specific	This component group is not applicable to Ginna Station. The feedwater delivery to the steam generators at Ginna Station is through feedrings to Alloy 690 J-tubes. The feedrings and J-tubes perform no license renewal intended function.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(15) (Alloy 600) Steam generator tubes, repair sleeves, and plugs	Crack initiation and growth due to PWSCC, outside diameter stress corrosion cracking (ODSCC), and/or inter-granular attack (IGA) or loss of material due to wastage and pitting corrosion, and fretting and wear: or deformation due to corrosion at tube support plate intersections	Steam generator tubing integrity; water chemistry	Yes, effectiveness of a proposed AMP is to be evaluated	Consistent with NUREG-1801. Cracking due to PWSCC and IGA/IGSCC and loss of material due to pitting and wear were identified as aging effects requiring management for the Ginna Station steam generator tubes and plugs. These aging effects will be jointly managed by the Water Chemistry Control Program (both primary and secondary water chemistry) and the Steam Generator Tube Integrity Program (described in Appendix B). The Steam Generator Tube Integrity Program at Ginna Station was developed to meet the guidelines in NEI 97-06. Consistent with these guidelines, the requirements for SG degradation management are included in Section 3.4.13 of Ginna Station Technical Specifications. These requirements, including tube inspection scope and frequency, plugging, repair and leakage monitoring have been incorporated in plant administrative controls. New, replacement recirculating steam generators (SG) were installed at Ginna Station in 1996. These generators incorporate many enhancements in design and materials of construction. The tubes are fabricated from drawn Alloy 690 TT (thermally-treated) material. The tubes are hydraulically expanded over the full depth of the tubesheet. The tube support design is a lattice-grid structure fabricated from Type 410 stainless steel bars. Anti-vibration supports in the U-bend region of the bundle are also Type 410 stainless steel fan-bars. Sufficient corrosion of the tube support structure to cause tube-denting is not expected based on the resistance of Type 410 stainless steel to the secondary water environment. After four operating cycles (six years), no service-induced defects have been detected in the tubes and no repair plugs or sleeves have been installed other than two factory-installed Alloy 690 TT plugs in each generator. Secondary-side water chemistry control at Ginna Station is based on all-volatile-treatment (AVT), not phosphate treatment.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(16) Tube support lattice bars made of carbon steel	Loss of section thickness due to FAC	Plant specific	Yes, plant specific	Tube support lattice bars are fabricated from Type 410 stainless steel in Ginna Station replacement steam generators. Type 410 stainless steel is not susceptible to FAC. Therefore this component group is not applicable to Ginna Station. A discussion of steam generator components susceptible to FAC is given in Item 21.
(17) Carbon steel tube support plate	Ligament cracking due to corrosion	Plant specific	Yes, effectiveness of a proposed AMP is to be evaluated	There are no carbon steel tube support materials in the Ginna Station steam generators. Therefore this component group is not applicable to Ginna Station. NUREG-1801 does not specify the corrosion mechanisms which might cause ligament cracking. Cracking due to SCC and loss of material due to pitting and crevice corrosion were identified as aging effects requiring management for the lattice grid support bars in the Ginna Station steam generators. These aging effects are managed jointly by the Water Chemistry Control Program and the Steam Generator Tube Integrity Program, which provides for secondary side inspections to verify the effectiveness of water chemistry control. These aging management programs will be identified in Appendix B.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(18) Reactor vessel closure studs and stud assembly	Crack initiation and growth due to SCC and/or IGSCC	Reactor head closure studs	No	Consistent with NUREG-1801. Crack initiation and growth due to SCC or IGSCC were not identified as an aging effect requiring management for the Ginna Station reactor vessel closure studs. Nevertheless, the Reactor Head Closure Studs Program which includes ASME Section XI visual, surface and volumetric inservice inspections capable of detecting cracking due to SCC, is credited for managing aging effects applicable to the reactor head closure studs. NUREG-1801 identifies crack initiation and growth due to SCC is a potential aging effect for reactor vessel closure studs exposed to borated water leaks. Although there have been a few reported cases of cracking of bolting in the industry caused by SCC, these have been attributed to susceptible high yield-stress materials exposed to aggressive environments, such as lubricants containing molybdenum disulfide. A survey of industry experience, technical literature, and laboratory corrosion studies (documented in EPRI Report NP-5769) indicates that SCC should not be a concern for closure bolting in nuclear power plant applications if the specified minimum yield strength is <150 Ksi. The Ginna Station studs are fabricated from SA-320 Gr. L43 material (corresponding to AISI Grade 4340) which is not a high strength steel. The minimum yield strength specified in SA-320 for Grade L43 material is 105 Ksi, which is well below the 150 Ksi threshold. Furthermore, the selection and use of fastener lubricants for pressure boundary components has been controlled by the Ginna Station Quality Assurance Program since 1983 as part of the response to IE Bulletin 82-02. Limits are also imposed on levels of contaminants such as chlorides and sulfur compounds in lubricants and sealant compounds. Therefore, it is reasonable to conclude that failure by SCC should not be a significant issue for SA-320 Gr. L43 bolting materials. Industry and plant-specific operating experience support this conclusion.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(19) CASS pump casing and valve body	Loss of fracture toughness due to thermal aging embrittlement	Inservice inspection	No	Consistent with NUREG-1801. Loss of fracture toughness due to thermal aging embrittlement was identified as an aging effect requiring management for the CASS reactor coolant pump (RCP) casings and Class 1 valve bodies. The ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program (as modified by ASME Code Case N-481) is credited for managing this aging effect for the RCP casings. One of the requirements of Code Case N-481 is a flaw tolerance evaluation performed by fracture mechanics methods for the RCP casings to verify that adequate margin exists for flaw stability after consideration is given to reduction in fracture toughness due to thermal aging embrittlement. This evaluation has been performed and adequate margin was demonstrated throughout the period of extended operation. For Class 1 valve bodies, the ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program is credited for managing loss of fracture toughness due to thermal aging embrittlement.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(20) CASS piping	Loss of fracture toughness due to thermal aging embrittlement	Thermal aging embrittlement of CASS	No	Consistent with NUREG-1801. The Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program is credited with managing loss of fracture toughness due to thermal aging embrittlement. This program invokes ASME Section XI ISI requirements as well as flaw-tolerance analyses using fracture mechanics methods which take into account the effects of thermal aging during the period of extended operation. An updated leak-before-break (LBB) analysis has been performed for the Ginna Station reactor coolant system piping and elbows based on loading, pipe geometry, and end-of-life fracture toughness considering thermal aging effects through the end of the period of extended operation. This analysis is addressed as a TLAA and is discussed in Section 4.0 of the Application. The CRDM pressure housings and all reactor coolant system nozzle safe ends at Ginna Station are wrought stainless steel, not CASS. The pressurizer spray head performs no license renewal intended function at Ginna Station. In addition, the pressurizer nozzle is cast carbon steel (integral with the pressurizer) and clad with austenitic stainless steel weld overlay. The reactor coolant system piping is forged Type 316 stainless steel. However, the fittings (elbows) are CASS (Type CF8M).
(21) BWR piping and fittings; steam generator components	Wall thinning due to flow-accelerated corrosion	Flow-accelerated corrosion	No	Consistent with NUREG-1801. The aging management review for the replacement steam generators did not identify loss of material due to FAC as an aging effect requiring management for the steam outlet nozzle and feedwater inlet nozzle. The steam quality and flows at the steam outlet nozzle are such that FAC would not represent a concern. Furthermore, the design of the feedwater inlet nozzle includes an Alloy 690 thermal sleeve which is extremely resistant to flow-accelerated corrosion damage. Nevertheless, the Flow-Accelerated Corrosion Program is credited for verification that steam generator components are not degraded due to FAC.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(22) Reactor coolant pressure boundary (RCPB) valve closure bolting, manway and holding bolting, and closure bolting in high pressure and high temperature systems	Loss of material due to wear; loss of preload due to stress relaxation; crack initiation and growth due to cyclic loading and/or SCC	Bolting integrity	No	Consistent with NUREG-1801. The closure bolting for reactor coolant system valves, reactor coolant pump, steam generator, and pressurizer is SA-193 Grade B7 material, with specified minimum yield strength of 105 Ksi. Consequently, crack initiation and growth due to SCC is not an applicable aging effect (see discussion for Item 18). Loss of mechanical closure integrity due to boric acid corrosion was also identified as an aging effect requiring management for all RCPB bolting potentially exposed to borated water leaks. The applicable aging management program is the Boric Acid Corrosion Program. For all RCPB bolting other than the reactor vessel closure studs, loss of material due to wear and loss of mechanical closure integrity due to stress relaxation are managed at Ginna Station by the Bolting Integrity Program (described in Appendix B). The Bolting Integrity Program invokes the ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program for assurance that effects of aging for RCPB closure bolting are effectively managed. There are no flanged connections associated with the CRDM penetrations with reactor coolant pressure boundary (RCPB) bolting at Ginna Station.
(23) CRD nozzle	Crack initiation and growth due to PWSCC	Ni-alloy nozzles and penetrations; water chemistry	No	Consistent with NUREG-1801. Crack initiation and growth due to PWSCC was identified as an aging effect requiring management for the Alloy 600 CRDM nozzles and reactor head vent pipe. The aging management programs credited for managing this effect are the Water Chemistry Control Program and the Reactor Vessel Head Penetration Inspection Program.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(24) Reactor vessel nozzles safe ends and CRD housing; reactor coolant system components (except CASS and bolting)	Crack initiation and growth due to cyclic loading, and/or SCC, and PWSCC	Inservice inspection; water chemistry	No	Consistent with NUREG-1801. Crack initiation and growth due to SCC and flaw growth were identified as aging effects requiring management for the reactor vessel nozzle safe ends, CRD housing and RCS components. Aging management programs credited for managing these effects are the Water Chemistry Control Program and ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program. The pressurizer manway and flange are integrally cast with the carbon steel heads and are evaluated with the pressurizer heads. The pressurizer relief tank is not in scope to license renewal at Ginna Station. For small-bore connected systems piping and fittings, aging management programs for managing crack initiation and growth due to SCC are discussed in Item 6.
(25) Reactor vessel internals CASS components	Loss of fracture toughness due to thermal aging, neutron irradiation embrittlement, and void swelling	Thermal aging and neutron irradiation embrittlement	No	The upper and lower internals assemblies in the Ginna Station reactor vessel contain no CASS components. The lower support forging and lower support plate columns are wrought stainless steel. Therefore this component grouping is not applicable to Ginna Station.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(26) External surfaces of carbon steel components in reactor coolant system pressure boundary	Loss of material due to boric acid corrosion	Boric acid corrosion	No	Consistent with NUREG-1801. Loss of material due to boric acid corrosion was identified as an aging effect requiring management for external surfaces of carbon steel components (including closure bolting) in the reactor coolant system pressure boundary. The Boric Acid Corrosion Program is credited for managing this aging effect. Additionally, loss of material due to boric acid corrosion was identified for all borated water systems as well as non-borated water systems in proximity to borated water systems at Ginna Station. The Boric Acid Corrosion Program was also credited for aging management of boric acid corrosion in these additional systems.
(27) Steam generator secondary manways and handholds (CS)	Loss of material due to erosion	Inservice inspection	No	This line item applies to once-through steam generators and is therefore not applicable to Ginna Station.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(28) Reactor internals, reactor vessel closure studs, and core support pads	Loss of material due to wear	Inservice inspection	No	Consistent with items IV.A2.5-f, B2.1-l, B2.5-o, and B2.6-c of NUREG-1801. Loss of material due to wear was identified as an aging effect requiring management for the reactor vessel flange and internals components identified in NUREG-1801. However, loss of material due to wear was also identified for the reactor vessel closure studs and the core support pads for which no specific line items appear in the NUREG-1800 (SRP) table. The ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program was credited for managing loss of material due to wear for all components except the flux thimble tubes and the reactor vessel closure studs. For the flux thimble tubes, Ginna Station credits the Thimble Tubes Inspection Program for managing loss of material due to wear. This program was implemented in response to NRC Bulletin 88-09 which required that an inspection program be established to manage the effects of thimble wear for Westinghouse reactors with bottom-mounted instrumentation. The program provides for eddy current inspections at an appropriate frequency, includes acceptance criteria and corrective actions and has effectively managed thimble tube wear at Ginna Station (see Appendix B). The Reactor Head Closure Studs Program (see Appendix B) is credited for managing loss of material due to wear of reactor vessel closure studs (see Item 35).
(29) Pressurizer integral support	Crack initiation and growth due to cyclic loading	Inservice inspection	No	Consistent with NUREG-1801. Cracking due to flaw growth was identified as an aging effect requiring management for the pressurizer support skirt and flange. Flaw growth occurs as a result of cyclic loading. The ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program was credited for managing this aging effect. Although the aging effect identified by Ginna Station is not described as crack initiation and growth due to cyclic loading, cracking due to any mechanism would be acceptably managed by the ASME Section XI ISI Program.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(30) Upper and lower internals assembly (Westinghouse)	Loss of preload due to stress relaxation	Inservice inspection; loose part and/or neutron noise monitoring	No	Not consistent with NUREG-1801. Loss of mechanical closure integrity due to stress-relaxation was identified as an aging effect requiring management for the holddown spring in the upper internals assembly and for the clevis-insert bolts in the lower internals assembly. However, the ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program was credited for managing this aging effect. Ginna Station does not employ loose-parts or neutron noise monitoring methods for aging management as referenced in NUREG-1801. This item will therefore be included in Table 3.2-2.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(31) Reactor vessel internals in fuel zone region (except Westinghouse and Babcock & Wilcox [B&W] baffle bolts)	Loss of fracture toughness due to neutron irradiation embrittlement, and void swelling	PWR vessel internals; water chemistry	No	Consistent with NUREG-1801. Loss of fracture toughness due to neutron irradiation embrittlement was identified as an aging effect requiring management for reactor vessel internals components in the fuel zone. However, void swelling was not specifically identified as an aging mechanism. The results of recent destructive examinations of one of the Ginna Station baffle/former bolts removed during the 1999 refueling outage suggest that void swelling should not represent a concern during the period of extended operation (see discussion in Item 8). In addition, the lower support forging and the core barrel outlet nozzle were not included among the components subject to significant irradiation embrittlement because of their location remote from the fuel zone. The aging management program referred to in NUREG-1801 is the PWR Vessel Internals Program. However, the SRP references Water Chemistry as well as the PWR Vessel Internals Program. Nevertheless, the Reactor Vessel Internals Program is credited with managing loss of fracture toughness due to neutron irradiation embrittlement and void swelling for the internals components in this component grouping. Ginna Station will incorporate applicable results of industry initiatives related to void swelling in the Reactor Vessel Internals Program as they become available.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(32) Steam generator upper and lower heads; tubesheets; primary nozzles and safe ends	Crack initiation and growth due to SCC, PWSCC and IASCC	Inservice inspection; water chemistry	No	Consistent with NUREG-1801. The only components in this grouping applicable to Ginna Station are the primary nozzles and safe ends (NUREG-1801, Item IV D1.1-i). The steam generator primary head, inlet and outlet nozzles, and manways at Ginna Station are low-alloy steel clad with austenitic stainless steel weld overlay. The primary nozzle safe ends are Type 316 stainless steel. Since the interior (clad) surface of the nozzles, manway, and head are exposed to the same environment, the primary head and manways are included with this component grouping. Crack initiation and growth due to SCC was identified as an aging effect requiring management for the stainless steel-clad primary head, inlet and outlet nozzles, manways, and stainless steel safe ends. The ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program and the Water Chemistry Control Program are credited for managing applicable aging effects for components in this grouping.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(33) Vessel internals (except Westinghouse and B&W baffle former bolts)	Crack initiation and growth due to SCC and IASCC	PWR vessel internals; water chemistry	No	Consistent with NUREG-1801. Ginna Station credited either the Water Chemistry Control Program alone (for components subject to SCC) or in combination with the Reactor Vessel Internals Program (for components subject to IASCC) for management of crack initiation and growth due to SCC/IASCC. Crack initiation and growth due to SCC was identified as an aging effect requiring management for all reactor vessel internals components fabricated from stainless steel. Crack initiation and growth due to IASCC was identified as an aging effect requiring management for those components exposed to neutron fluence >10^{21} n/cm^2 in the core. However, the Ginna Station evaluation determined that not all components listed in NUREG-1801 were considered susceptible to crack initiation and growth due to IASCC. This is a result of fluence exposures being less than the threshold value of 10^{21} n/cm^2. In addition, plant-specific data obtained from destructive evaluation of Type 347 stainless steel baffle/former bolts removed in 1999 indicated very limited evidence of IASCC. Those components determined by evaluation not to be susceptible to IASCC are enumerated in Table 3.2-2.
(34) Reactor internals (B&W screws and bolts)	Loss of preload due to stress relaxation	Inservice inspection; loose part monitoring	No	The components in this grouping are not applicable to Ginna Station.
(35) Reactor vessel closure studs and stud assembly	Loss of material due to wear	Reactor head closure studs	No	Consistent with NUREG-1801. Loss of material due to wear was identified as an aging effect requiring management for the reactor vessel closure studs. The Reactor Head Closure Studs Program is credited with managing this effect.

Table 3.2-1 Reactor Coolant System - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(36) Reactor internals (Westinghouse upper and lower internal assemblies; CE bolts and tie rods)	Loss of preload due to stress relaxation	Inservice inspection; loose part monitoring	No	Loss of mechanical closure integrity due to stress relaxation was identified as an aging effect requiring management for the upper and lower support plate column bolts. The ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program is credited for managing this aging effect. Therefore this is consistent with NUREG-1801. NUREG-1801 cites the Loose Parts Monitoring Program as well as the ASME Section XI ISI Program. However, loose-parts monitoring is not considered to be effective as an aging management program at Ginna Station. This is not consistent with NUREG-1801 and will be further discussed in Table 3.2-2.

Table 3.2-2 Reactor Coolant System - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(1) RPV Closure Head Dome, Closure Head Flange, Vessel Flange, Upper Shell, Primary Inlet Nozzles, Primary Outlet Nozzles, Intermediate Shell (including circumferential Beltline weld), Lower Shell, Bottom Head Torus, Bottom Head Dome BMI Guide Tubes, Seal Table Fittings	Low-Alloy Steel with Stainless Steel Cladding Stainless Steel	Primary Water	Cracking due to SCC	Water Chemistry Control Program	Cracking due to SCC is not identified as an aging effect requiring management for these components in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.2-2 Reactor Coolant System - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(2) RPV Closure Head Dome, Closure Head Flange, Vessel Flange, Upper Shell, Primary Inlet Nozzles, Primary Outlet Nozzles, Intermediate Shell (including circumferential weld), Lower Shell, Bottom Head Torus, Bottom Head Dome	Low-Alloy Steel with Stainless Steel Cladding	Primary Water	Cracking due to Flaw Growth	ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program	Cracking due to flaw growth is not identified as an aging effect requiring management for these components in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.
Primary Nozzle Safe Ends BMI Guide Tubes Core Support Pads	Stainless Steel Weld Butter Stainless Steel Alloy 600			ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program	
CRDM Head Housing Tubes (Head Adapters), Vent Pipe, Instrumentation Tubes and Safe Ends	Alloy 600			Reactor Vessel Head Penetration Inspection Program ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program	

Table 3.2-2 Reactor Coolant System - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(3) RPV Core Support Pads	Alloy 600	Primary Water	Loss of Material due to Wear	ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program	Loss of material due to wear was not identified as an aging effect requiring management in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.
(4) RPV Primary Inlet and Outlet Nozzles and Upper Shell	Low-Alloy Steel with Stainless Steel Cladding	Primary Water	Loss of Fracture Toughness due to Neutron Irradiation Embrittlement	Reactor Vessel Surveillance Program	As discussed in Table 3.2-1 Line Number (4) the RPV upper shell and the lower sides of the primary inlet and outlet nozzles are not subject to significant neutron irradiation embrittlement because of their physical distance from the reactor core. Therefore loss of fracture toughness due to neutron irradiation embrittlement is not identified as an aging effect requiring management at Ginna Station. This is not consistent with NUREG-1801.
(5) RPV Ventilation Shroud Support Ring, Refueling Seal Ledge, Upper Shell, Primary Inlet Nozzles, Primary Outlet Nozzles, Intermediate Shell, Lower Shell, Bottom Head Torus, Bottom Head Dome (CS Components)	Carbon/Low-Alloy Steel	Borated Water Leaks	Loss of Material due to Boric Acid Corrosion	Boric Acid Corrosion Program	Loss of material due to boric acid corrosion of external surfaces of these components was not identified as an aging effect requiring management in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.
(6) Closure Studs, Nuts, Washers	Carbon/Low-Alloy Steel	Containment Air	Loss of Mechanical Closure Integrity due to Stress Relaxation	Reactor Head Closure Studs Program	Loss of mechanical closure integrity due to stress relaxation was not identified in NUREG-1801 as an aging effect requiring management for the reactor vessel closure studs. The aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.2-2 Reactor Coolant System - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(7) Lower Core Plate and Fuel Pins, Lower Support Forging, Lower Support Columns, Core Barrel and Flange, Radial Keys and Clevis Inserts, Baffle and Former Assembly, Core Barrel Outlet Nozzle, Upper Support Plate Assembly, Upper Core Plate and Fuel Alignment Pins, Upper Support Columns, RCCA Guide Tubes and Flow Downcomers, Guide Tube Support Pins, Upper Core Plate Alignment Pins, Holddown Spring, Thermal Shield and Neutron Panels, Bolting for all Bolted Closures	Stainless Steel, Alloy 600 (Clevis Inserts), Alloy X-750 (Guide Tube Support Pins)	Primary Water	Changes in Dimension due to Void Swelling	Reactor Vessel Internals Program	As discussed in Table 3.2-1 Line Number (8) change in dimension due to void swelling was not explicitly identified as an aging effect requiring management for these components at Ginna Station. However, the Reactor Vessel Internals Program manages the effects of void swelling should it become a concern.

Table 3.2-2 Reactor Coolant System - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(8) Lower Support Forging, Core Barrel Outlet Nozzle	Forged Stainless Steel	Primary Water	Loss of Fracture Toughness due to Irradiation Embrittlement	Reactor Vessel Internals Program Water Chemistry Control Program Program	As discussed in Table 3.2-1 Line Number (31) at Ginna Station, loss of fracture toughness due to irradiation embrittlement was not identified as an aging effect requiring management in the AMR for these components because of their location remote from the fuel zone. In addition, void swelling was not explicitly identified as an applicable aging mechanism, although the Reactor Vessel Internals Inspection program manages the effects of void swelling should it become a concern.
(9) Lower Support Forging, Radial Keys and Clevis Supports, Core Barrel Outlet Nozzle, Upper Support Plate Assembly, Upper Core Plate and Fuel Alignment Pins, Upper Support Columns, RCCA Guidetubes and Flow Downcomers, Upper Core Plate Alignment Pins, Holddown Spring, Upper Support Column Bolting, Guide Tube Bolts, Clevis Insert Bolts	Stainless Steel	Primary Water	Cracking due to IASCC	Reactor Vessel Internals Program Water Chemistry Control Program Program	As discussed in Table 3.2-1 Line Number (33) cracking due to IASCC was not identified as an aging effect requiring management in the AMR for these components because neutron fluence exposure is below the threshold value for IASCC susceptibility. That not withstanding, the aging management program(s) referenced are appropriate for the aging effects identified should cracking due to IASCC become a concern.

Table 3.2-2 Reactor Coolant System - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(10) Lower Core Plate and Fuel Pins, Core Barrel Flange, Fuel Alignment Pins, Guide Tube Support Pins, Holddown Spring	Stainless Steel	Primary Water	Loss of Material due to Wear	ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program	Loss of material due to wear was not identified as an aging effect requiring management for these components in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.
(11) Secondary Core Support, Diffuser Plate, Guide Tube Support Pins, Head Vessel Alignment Pins, BMI Columns and Flux Thimbles, Head Cooling Spray Nozzles, Upper Instrumentation Column, Conduits and Supports	Stainless Steel	Primary Water	Cracking due to SCC	Water Chemistry Control Program	Cracking due to SCC was not identified as an aging effect requiring management for these components in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.
(12) Upper and Lower Internals Assembly - Holdown Spring, Upper and Lower Support Column Bolts, Clevis Insert Bolts	Stainless Steel	Primary Water	Loss of Preload due to Stress Relaxation	ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program	As discussed in Table 3.2-1 Line Number (30) and Table 3.2-1 Line Number (36) loss of mechanical closure integrity was identified as an aging effect requiring management for these components. However, loose parts or neutron noise monitoring programs are not used for the purpose of aging management at Ginna Station. This is not consistent with NUREG-1801.
(13) RCS Primary Loop Elbows RCS Valves ≥ 4 in. NPS, Valves < 4 in. NPS	CASS Wrought Stainless Steel	Primary Water	Cracking due to Flaw Growth	ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program	Cracking due to flaw growth is not identified as an aging effect requiring management for these components in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.2-2 Reactor Coolant System - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(14) RCS Valves $\geq$ 4 in. NPS, Valves < 4 in. NPS	Wrought Stainless Steel	Primary Water	Cracking due to SCC	Water Chemistry Control Program	Cracking due to SCC is not identified as an aging effect requiring management in NUREG-1801 for wrought stainless steel valves $\geq$ 4 in. NPS and < 4 in. NPS. The aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.
(15) Reactor Coolant Pump Thermal Barrier Flange, Thermal Barrier Heat Exchanger Tubing, Orifices and Reducers	Wrought Stainless Steel	Primary Water	Cracking due to SCC, Cracking due to Flaw Growth	Water Chemistry Control Program ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program	The Reactor Coolant Pump Thermal Barrier Flange, Heat Exchanger Tubing, and Orifices and Reducers are not identified in NUREG-1801. Although these component types are not included in the NUREG, the aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.
(16) Reactor Coolant Pump Lugs	Stainless Steel	Containment Air	Cracking due to Flaw Growth	ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program	Cracking due to flaw growth for the Reactor Coolant Pump Lugs is not identified as an aging effect requiring management in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.
(17) Pressurizer Safety Nozzle, Pressurizer Relief Nozzle	Carbon Steel with Stainless Steel Cladding	Primary Water	Cracking due to SCC, Cracking due to Flaw Growth	Water Chemistry Control Program ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program	The Pressurizer Safety and Relief Nozzles are not explicitly identified in NUREG-1801. Although these component types are not included in the NUREG, the aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.2-2 Reactor Coolant System - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(18) Pressurizer Manway Cover	Carbon Steel with Stainless Steel Disc Insert	Primary Water	Cracking due to SCC	Water Chemistry Control Program	The Pressurizer Manway Cover is not explicitly identified in NUREG-1801. Although these component types are not included in the NUREG, the aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(19) Pressurizer Spray Nozzle, Relief Nozzle, Safety Nozzle, Surge Nozzle and Manway Cover (CS Components)	Carbon Steel with Stainless Steel Cladding or Disc Insert	Borated Water Leaks	Loss of Material due to Boric Acid Corrosion	Boric Acid Corrosion Program	Loss of material due to boric acid corrosion was not identified as an aging effect requiring management for these components in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.
(20) SG Primary Channel Head, Primary Inlet and Outlet Nozzles, Primary Inlet and Outlet Nozzle Safe Ends, Steam Generator Shell and Transition Cone, Feedwater Nozzle, Steam Outlet Nozzle, Blowdown Piping Nozzle and Secondary-Side Shell Penetrations	Low-Alloy Steel with Stainless Steel Cladding Carbon Steel	Primary Water Secondary Water	Cracking due to Flaw Growth	ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program	Cracking due to Flaw Growth is not identified as an aging effect requiring management for these components in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.2-2 Reactor Coolant System - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(21) SG Tubesheet (Primary Side)	Low-Alloy Steel with Alloy 600 cladding	Primary Water	Cracking due to SCC Cracking due to Flaw Growth	Water Chemistry Control Program ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program	Cracking due to SCC and flaw growth is not identified as an aging effect requiring management for the primary side of the SG tubesheet in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.
(22) SG Tubesheet (Secondary Side)	Low Alloy Steel	Secondary Water	Loss of Material due to General, Pitting and Crevice Corrosion Cracking due to Flaw Growth	Water Chemistry Control Program ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program	Loss of material due to general, pitting and crevice corrosion and cracking due to flaw growth are not identified as aging effects requiring management for the secondary side of the SG tubesheet in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.
(23) SG Primary Channel Head Divider Plate	Alloy 690	Primary Water	Cracking due to SCC	Water Chemistry Control Program	Cracking due to SCC is not identified as an aging effect requiring management for the SG divider plate in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.2-2 Reactor Coolant System - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(24) SG Feedwater Nozzle, Steam Outlet Nozzle, Steam Flow Restrictor, Blowdown Piping Nozzles and Secondary-Side Shell Penetrations, Secondary Closures, and Internal Shroud, Primary and Secondary Decks	Carbon/Low-Alloy Steel	Secondary Water	Loss of Material due to General, Pitting and Crevice Corrosion	Water Chemistry Control Program	Loss of material due to general, pitting and crevice corrosion was not identified in NUREG-1801 as an aging effect requiring management for these components exposed to the secondary-side SG environment. The aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.
(25) SG Lattice Grid Tube Supports, U-Bend Fan Bar Restraints	Stainless Steel	Secondary Water	Cracking due to SCC Loss of Material due to Pitting and Crevice Corrosion	Water Chemistry Control Program Steam Generator Tube Integrity Program	As discussed in Table 3.2-1 Line Number (17) cracking due to SCC and loss of material due to pitting and crevice corrosion were not identified as aging effects requiring management in NUREG-1801 for the lattice grid tube supports and U-bend fan bar restraints. The aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.
(26) SG Primary Inlet and Outlet Nozzles and Support Pads (CS Components)	Carbon/Low-Alloy Steel	Borated Water Leaks	Loss of Material due to Boric Acid Corrosion	Boric Acid Corrosion Program	Loss of material due to boric acid corrosion was not identified as an aging effect requiring management in NUREG-1801 for the external surfaces of the primary inlet and outlet nozzles and support pads. The aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.2-2 Reactor Coolant System - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(27) SG Support Pads, Seismic Lugs	Carbon Steel	Containment Air	Cracking due to Flaw Growth	ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program	Cracking due to flaw growth was not identified in NUREG-1801 as an aging effect requiring management for the support pads and seismic lugs. The aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.
(28) Non-Class 1 RCS Manual Valves, Solenoid-Operated Valves, Strainers, PORV Operators, Accumulators, Nitrogen Surge Tanks, Piping	Carbon Steel, Stainless Steel, CASS, Copper Alloy (Zn<15%) Stainless Steel	Air and Gas Air and Gas, Wetted (<140°F)	No Aging Effects	No AMP Required	Non-Class 1 RCS carbon steel, stainless steel, CASS and copper alloy components exposed to air and gas environments are not identified in NUREG-1801.
(29) Non-Class 1 RCS Piping	Carbon Steel	Air and Gas, Wetted (<140°F)	Loss of Material due to General, Crevice, Pitting, Galvanic Corrosion and MIC	Periodic Surveillance and Preventive Maintenance	Non-Class 1 RCS carbon steel piping exposed to a wetted air and gas (<140°F) environment is not identified in NUREG-1801. Although these component types are not included in the NUREG, the aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.
(30) Reactor Coolant Pump Upper Bearing Cooler	Carbon Steel	Treated Water - Other	Loss of Material due to General, Crevice, and Galvanic Corrosion and MIC	Water Chemistry Control Program	The Reactor Coolant Pump upper bearing cooler is a Non-Class 1 RCS component which is not identified in NUREG-1801. Although these component types are not included in the NUREG, the aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.2-2 Reactor Coolant System - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(31) Reactor Coolant Pump Motor Upper and Lower Bearing Cooling Coil	Copper Alloy (Zn<15%, Zn>15%)	Oil and Fuel Oil	Loss of Material due to MIC	Periodic Surveillance and Preventive Maintenance	The Reactor Coolant Pump Motor upper and lower bearing cooling coils are Non-Class 1 components which are not identified in NUREG-1801. Although these component types are not included in the NUREG, the aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.
		Treated Water - Other	Loss of Material due to MIC, Crevice and Galvanic Corrosion, and Selective Leaching	Water Chemistry Control Program	
		Treated Water - Other	Loss of Heat Transfer due to Particulate and Biological Fouling	Water Chemistry Control Program	
(32) Seal Table	Stainless Steel	Treated Water - Borated (<140°F)	Loss of Material due to Crevice and Pitting Corrosion and MIC	Water Chemistry Control Program One-Time Inspection Program	The seal table is a Non-Class 1 component which is not identified in NUREG-1801 Although these component types are not included in the NUREG, the aging management program(s) referenced are appropriate for the aging effects identified and provide assurance that the aging effects are effectively managed through the period of extended operation.
(33) External Surfaces of Carbon/Low-Alloy Steel Components in Reactor Coolant System	Carbon/Low Alloy Steel	Containment Atmosphere ($\geq$212°F)	No Aging Effects	No AMP Required	External surfaces of carbon/low-alloy steel components in the Reactor Coolant System exposed to the Containment atmosphere are not identified in NUREG-1801. No aging effects are identified for those components which normally operate at temperatures $\geq$ 212°F. For components with service temperatures < 212°F, loss of material due to general and pitting corrosion is an applicable aging effect which is effectively managed by the Systems Monitoring Program.
		Containment Atmosphere (<212°F)	Loss of Material due to General and Pitting Corrosion	Systems Monitoring Program	

Table 3.2-2 Reactor Coolant System - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(34) External Surfaces of Stainless Steel and Nickel-Alloy Components in Reactor Coolant System	Stainless Steel Nickel Alloy	Borated Water Leaks Containment Atmosphere	No Aging Effects	No AMP Required	External surfaces of stainless steel and nickel alloy components in the Reactor Coolant System exposed to the Containment atmosphere or borated water leaks are not identified in NUREG-1801. There are no applicable aging effects for stainless steel or nickel alloy components exposed to borated water leaks or the Containment atmosphere.

Section 3.2 References

1. WCAP-14575-A, Aging Management Evaluation for Class I Piping and Associated Pressure Boundary Components, December, 2000.
2. WCAP-14577, Rev. 1-A, License Renewal Evaluation: Aging Management for Reactor Internals, March, 2001.
3. WCAP-14574-A, License Renewal Evaluation: Aging Management Evaluation for Pressurizers, December, 2000.

3.3 Aging Management of Engineered Safety Features Systems

The results of the aging management review of the Engineered Safety Features Systems components are provided in this section and summarized in Tables 3.3-1 and 3.3-2. Table 3.3-1 shows the aging management of system components evaluated in NUREG-1801 that are relied on for license renewal of the Engineered Safety Features Systems components at Ginna. Included in the table is a discussion column. The discussion column will provide a conclusion indicating if the aging management evaluation results are consistent with NUREG-1801 along with any clarifications or explanations required to support the stated conclusion if that conclusion is different than those of the NUREG. For a determination to be made that a table line item is "Consistent with NUREG-1801" several criteria must be met. First the plant specific component is reviewed against the GALL to ensure that the component, materials of construction and internal or external service environment are comparable to those described in a particular GALL item. Second, for those that are comparable, the results of the plant aging management review- aging effect evaluation are compared to the aging effects/mechanisms in the GALL. Finally, the programs credited in the GALL for managing those aging effects are compared to the programs invoked in the plant evaluation. If, using good engineering judgment, it could be reasonably concluded that the plant evaluation is in agreement with the GALL evaluation a line item was considered consistent with NUREG-1801. There are cases where components and component material/environment combinations and aging effects are common between a NUREG-1801 line item and the plant evaluation but the aging management program selections differ. In those cases the discussion column will indicate the plant aging management program selection but no conclusion will be made that the line item is consistent with the GALL. Table 3.3-2 contains the Engineered Safety Features Systems components aging management review results that are not addressed in NUREG-1801. A plant component is considered not addressed by the NUREG if the component type is not evaluated in the GALL or has a different material of construction or operating environment than evaluated in the GALL. This table includes the component types, materials, environments, aging effects requiring management, the programs and activities for managing aging, and a discussion column. To avoid confusion, no attempt was made to interrelate material/environment/aging effects from one NUREG-1801 chapter to another. Note that these tables only include those components, materials and environments that are applicable to a PWR.

Materials

The materials of construction of a component have a major influence on the evaluation of aging effects applicable to the component. Sources of information used to identify materials of construction include original equipment specifications, vendor technical manuals and drawings, fabrication drawings, piping line specifications, modification design records and field walkdowns/verifications. The tables below account for the materials of construction for the components requiring an aging management review. Since similar materials are susceptible to the same aging effects/mechanisms, the tables itemize the component types (i.e., groupings) while factoring in the materials of construction.

Environment

As previously described, the environment(s) to which components are exposed are critical in the determination of potential aging mechanisms and effects. A review of plant design documentation was performed to quantify the environmental conditions to which Ginna Station equipment is exposed. This review identified that some equipment is exposed to a variety of environments. This can include normal operating conditions and post accident conditions. Since aging mechanisms and effects will be primarily driven by the environmental conditions to which equipment is exposed on a daily basis, under normal operating conditions, these conditions will differ from the design parameters which are established based upon the worst case scenario (e.g., LOCA conditions). Ginna Station equipment environments may be categorized into basic external and internal environments detailed in Section 3.1.2.

Aging Effects Requiring Management

After the components requiring aging management review were identified and grouped by materials of construction and environment, a review of industry and plant-specific operating experience was performed. The purpose of this review was to assure that all applicable aging effects were identified, and to evaluate the effectiveness of existing aging management programs.

This experience review was performed utilizing various industry and plant-specific programs and databases. Industry operating experience sources included NRC Generic Publications (including Information Notices, Circulars, Bulletins, and Generic Letters), INPO Significant Operating Event Reports (SOER), EPRI Technical Reports, and other information sources, such as the B&W Owners Group Non-Class 1 Mechanical Tools Implementation document, Westinghouse Generic Technical Reports (GTRs), and the Generic Aging Lessons Learned (GALL) report.

Plant specific operating experience sources included Semi-annual and Annual Reports to AEC/NRC, Abnormal Occurrence and Licensee Event Reports (LERs), Non-Conformance Reports (NCRs), Corrective Action Reports (CARs), Refueling, Inspection and Overhaul Reports (RIOs), Inservice Inspection (ISI) Reports, Identified Deficiency Reports (IDRs), and ACTION Reports (ARs) from 1969 to the present. Information from these sources was compiled in various databases. Based upon the material of construction, the applicable environments, and operating experience the potential aging effects requiring management for each of the components was identified as documented in the tables below.

Time-Limited Aging Analysis

In addition to those identified in NUREG-1801, any additional time-limited aging analyses (TLAA) identified as appropriate to the system are identified in Section 4.0.

Conclusion

The programs and activities selected to manage the aging effects of the Engineered Safety Features Systems are identified in Table 3.3-1 and Table 3.3-2. A description of these aging management activities is provided in Appendix B, along with the demonstration that the identified aging effects will be managed for the period of extended operation. Therefore, based on the demonstrations provided in Appendix B, the effects of aging associated with the system components will be adequately managed so that there is reasonable assurance that the intended function(s) will be maintained consistent with the current licensing basis during the period of extended operation.

Table 3.3-1 Engineered Safety Features Systems - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(1) Piping, fittings, and valves in emergency core cooling system	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	Consistent with NUREG-1801. Cumulative Fatigue Damage is addressed as a TLAA in Section 4.3.
(2) Components in containment spray (PWR only), standby gas treatment (BWR only), containment isolation, and emergency core cooling systems	Loss of material due to general corrosion	Plant specific	Yes, plant specific	The combination of components, materials and environments identified in Items V.A.2-a and V.A.5-a are not applicable at Ginna Station. Components identified in Item V.C.1-a are included in the containment isolation valves and associated piping entry under line item 4 below in this table.
(3) Components in containment spray (PWR only), standby gas treatment (BWR only), containment isolation, and emergency core cooling systems	Loss of material due to pitting and crevice corrosion	Plant specific	Yes, plant specific	Consistent with NUREG-1801 (containment isolation components and RWST bottom). The One-Time Inspection Program manages these aging effects for RWST bottom. The Systems Monitoring Program is credited for managing all other applicable aging effects.
(4) Containment isolation valves and associated piping	Loss of material due to microbiologically influenced corrosion	Plant specific	Yes, plant specific	Consistent with NUREG-1801 (containment isolation components such as valves and pipe penetrations). The aging effect “loss of material due to microbiologically influenced corrosion (MIC)” is managed by the plant-specific Periodic Surveillance and Preventive Maintenance Program.
(5) High pressure safety injection (charging) pump miniflow orifice	Loss of material due to erosion	Plant specific	Yes, plant specific	The high pressure safety injection pumps are not used for normal charging at Ginna Station. Loss of material due to erosion of miniflow orifices is not applicable at Ginna Station.
(6) Piping and fittings of CASS in emergency core cooling system	Loss of fracture toughness due to thermal aging embrittlement	Thermal aging embrittlement of CASS	No	There are no CASS piping and fittings in the emergency core cooling system at Ginna Station which are subject to loss of fracture toughness due to thermal aging embrittlement.

Table 3.3-1 Engineered Safety Features Systems - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(7) Components serviced by open-cycle cooling system	Local loss of material due to corrosion and/or buildup of deposit due to biofouling	Open-cycle cooling water system	No	The combination of components, materials and environments identified in Items V.A.6-a, V.A.6-b, V.D1.6-b and V.D1.6-c are not applicable at Ginna Station.
(8) Components serviced by closed-cycle cooling system	Loss of material due to general, pitting, and crevice corrosion	Closed-cycle cooling water system	No	Consistent with NUREG-1801. The Closed-Cycle (Component) Cooling Water System Program is credited with managing the aging effect “loss of material due to general, pitting and crevice corrosion.” The program includes maintenance of corrosion inhibitor concentrations to minimize corrosion and periodic surveillance testing and inspections to evaluate system and component performance and condition.
(9) Pumps, valves, piping, and fittings in containment spray and emergency core cooling systems	Crack initiation and growth due to SCC	Water chemistry	No	Consistent with NUREG-1801. Although the NUREG references a temperature gate of < 90°C (200°F) and a single aging effect (cracking due to SCC), materials science supports (1) a temperature gate > 140°F for cracking due to SCC, and (2) loss of material due to pitting (stagnant or low flow conditions) and crevice corrosion for all temperatures. Although the aging effect identified by Ginna (loss of material) for temperatures < 140°F differs from that of the NUREG, the Water Chemistry Control Program credited for managing the aging effects for all temperatures is consistent with the NUREG and will preclude the possibility of crack initiation and growth due to SCC. A One-Time Inspection Program is also credited to verify the adequacy of the Water Chemistry Control program.
(10) Carbon steel components	Loss of material due to boric acid corrosion	Boric acid corrosion	No	Consistent with NUREG-1801. The Boric Acid Corrosion Program is credited with managing the aging effect “loss of material due to boric acid corrosion.”

Table 3.3-1 Engineered Safety Features Systems - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(11) Closure bolting in high pressure or high temperature systems	Loss of material due to general corrosion, loss of preload due to stress relaxation, and crack initiation and growth due to cyclic loading or SCC	Bolting integrity	No	Consistent with NUREG-1801. The Bolting Integrity Program is credited for managing the aging effects "loss of material due to general corrosion and crack initiation and growth due to cyclic loading and SCC." There are no bolts with a specified minimum yield strength > 150 ksi in the ESF Systems. Therefore, SCC is not an applicable aging effect/mechanism.

Table 3.3-2 Engineered Safety Features Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Type	Material Type	Environment Type	AERMs	Program/Activity	Discussion
(1) ACCUMULATOR	Carbon/Low Alloy Steel	Containment	Loss of Material	Systems Monitoring Program	Consistent with Item E.1-b of NUREG-1801. Volume 1, Table 2 includes “External surface of carbon steel components” with a plant specific aging management program. This material and environment grouping is not included in NUREG-1800 Table 3.2-1. The Systems Monitoring Program is credited for managing this aging effect.
(2) BLOWER CASING	Carbon/Low Alloy Steel	Air and Gas (Wetted) < 140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(3) BLOWER CASING	Carbon/Low Alloy Steel	Containment	Loss of Material	Systems Monitoring Program	Consistent with Item E.1-b of NUREG-1801. Volume 1, Table 2 includes “External surface of carbon steel components” with a plant specific aging management program. This material and environment grouping is not included in NUREG-1800 Table 3.2-1. The Systems Monitoring Program is credited for managing this aging effect.
(4) CONTROLLER[1]	Stainless Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(5) CONTROLLER[1]	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(6) CONTROLLER[1]	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.3-2 Engineered Safety Features Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Type	Material Type	Environment Type	AERMs	Program/Activity	Discussion
(7) DELAY COIL	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(8) EDUCTOR	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(9) FASTENERS (BOLTING)	Carbon/Low Alloy Steel	Indoor (No Air Conditioning)	Cracking due to SCC	Bolting Integrity Program	There are no bolts with a specified minimum yield strength > 150 ksi in this system. Therefore, SCC is not an applicable aging effect/mechanism.
(10) FASTENERS (BOLTING)	Carbon/Low Alloy Steel	Indoor (No Air Conditioning)	Loss of Preload due to Stress Relaxation	Bolting Integrity Program	Material and environment grouping are included in NUREG-1801. Aging effect of loss of preload due to stress relaxation is applicable, but is not included in Chapter V - Section E, Chapter VII - Section I, or Chapter VIII - Section H of the NUREG.
(11) FASTENERS (BOLTING)	Stainless Steel	Borated Water Leaks	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(12) FILTER HOUSING	Carbon/Low Alloy Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(13) FILTER HOUSING	Carbon/Low Alloy Steel	Indoor (No Air Conditioning)	Loss of Material	Systems Monitoring Program	Consistent with Item E.1-b of NUREG-1801. Volume 1, Table 2 includes “External surface of carbon steel components” with a plant specific aging management program. This material and environment grouping is not included in NUREG-1800 Table 3.2-1. The Systems Monitoring Program is credited for managing this aging effect.

Table 3.3-2 Engineered Safety Features Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Type	Material Type	Environment Type	AERMs	Program/Activity	Discussion
(14) FLANGE	Carbon/Low Alloy Steel	Containment	Loss of Material	Systems Monitoring Program	Consistent with Item E.1-b of NUREG-1801. Volume 1, Table 2 includes “External surface of carbon steel components” with a plant specific aging management program. This material and environment grouping is not included in NUREG-1800 Table 3.2-1. The Systems Monitoring Program is credited for managing this aging effect.
(15) FLANGE	Carbon/Low Alloy Steel	Outdoor	Loss of Material	Systems Monitoring Program	Consistent with Item E.1-b of NUREG-1801. Volume 1, Table 2 includes “External surface of carbon steel components” with a plant specific aging management program. This material and environment grouping is not included in NUREG-1800 Table 3.2-1. The Systems Monitoring Program is credited for managing this aging effect.
(16) FLOW ELEMENT	Carbon/Low Alloy Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(17) FLOW ELEMENT	Carbon/Low Alloy Steel	Indoor (No Air Conditioning)	Loss of Material	Systems Monitoring Program	Consistent with Item E.1-b of NUREG-1801. Volume 1, Table 2 includes “External surface of carbon steel components” with a plant specific aging management program. This material and environment grouping is not included in NUREG-1800 Table 3.2-1. The Systems Monitoring Program is credited for managing this aging effect.
(18) FLOW ELEMENT	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(19) FLOW ELEMENT	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.3-2 Engineered Safety Features Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Type	Material Type	Environment Type	AERMs	Program/Activity	Discussion
(20) FLOW NOZZLES	Stainless Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(21) FLOW NOZZLES	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(22) HEAT EXCHANGER	Carbon/Low Alloy Steel	Indoor (No Air Conditioning)	Loss of Material	Systems Monitoring Program	Consistent with Item E.1-b of NUREG-1801. Volume 1, Table 2 includes "External surface of carbon steel components" with a plant specific aging management program. This material and environment grouping is not included in NUREG-1800 Table 3.2-1. The Systems Monitoring Program is credited for managing this aging effect.
(23) HEAT EXCHANGER	Cast Iron	Indoor (No Air Conditioning)	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(24) HEAT EXCHANGER	Cast Iron	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(25) HEAT EXCHANGER	Cast Iron	Raw Water	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.3-2 Engineered Safety Features Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Type	Material Type	Environment Type	AERMs	Program/Activity	Discussion
(26) HEAT EXCHANGER	HX-Cast Iron[2]	Oil and Fuel Oil	Loss of Heat Transfer	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(27) HEAT EXCHANGER	HX-Cast Iron[2]	Raw Water	Loss of Heat Transfer	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(28) HEAT EXCHANGER	HX-Nickel Alloy[2]	Treated Water Borated <140	Loss of Heat Transfer	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(29) HEAT EXCHANGER	HX-Nickel Alloy[2]	Treated Water Other	Loss of Heat Transfer	Closed-Cycle (Component) Cooling Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(30) HEAT EXCHANGER	Nickel Alloy	Treated Water Borated <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.

Table 3.3-2 Engineered Safety Features Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Type	Material Type	Environment Type	AERMs	Program/Activity	Discussion
(31) HEAT EXCHANGER	Nickel Alloy	Treated Water Borated <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(32) HEAT EXCHANGER	Nickel Alloy	Treated Water Other	Loss of Material	Closed-Cycle (Component) Cooling Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(33) HEAT EXCHANGER	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(34) INDICATOR[1]	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(35) ORIFICE	Cast Austenitic Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(36) ORIFICE	Cast Austenitic Stainless Steel	Treated Water Borated <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.

Table 3.3-2 Engineered Safety Features Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Type	Material Type	Environment Type	AERMs	Program/Activity	Discussion
(37) ORIFICE	Cast Austenitic Stainless Steel	Treated Water Borated <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(38) ORIFICE	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(39) ORIFICE	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(40) PIPE	Carbon/Low Alloy Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(41) PIPE	Carbon/Low Alloy Steel	Air and Gas (Wetted) <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(42) PIPE	Carbon/Low Alloy Steel	Buried	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(43) PIPE	Copper Alloy (Zn < 15%)	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.3-2 Engineered Safety Features Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Type	Material Type	Environment Type	AERMs	Program/Activity	Discussion
(44) PIPE	Copper Alloy (Zn < 15%)	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(45) PIPE	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(46) PIPE	Stainless Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(47) PIPE	Stainless Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(48) PIPE	Stainless Steel	Concrete	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(49) PIPE	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(50) PIPE	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(51) PIPE	Stainless Steel	Outdoor	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.3-2 Engineered Safety Features Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Type	Material Type	Environment Type	AERMs	Program/Activity	Discussion
(52) PIPE	Stainless Steel	Treated Water Other (Stagnant)	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(53) PIPE	Stainless Steel	Treated Water Other (Stagnant)	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(54) PUMP CASING	Cast Austenitic Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(55) PUMP CASING	Cast Austenitic Stainless Steel	Treated Water Borated <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(56) PUMP CASING	Cast Austenitic Stainless Steel	Treated Water Borated <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.3-2 Engineered Safety Features Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Type	Material Type	Environment Type	AERMs	Program/Activity	Discussion
(57) PUMP CASING	Stainless Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(58) PUMP CASING	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(59) RECOMBINER CASING	Stainless Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(60) RECOMBINER CASING	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(61) SWITCH[1]	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(62) TANK	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(63) TANK	Stainless Steel	Treated Water Other (Stagnant)	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.

Table 3.3-2 Engineered Safety Features Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Type	Material Type	Environment Type	AERMs	Program/Activity	Discussion
(64) TANK	Stainless Steel	Treated Water Other (Stagnant)	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(65) TEMPERATURE ELEMENT[1]	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(66) THERMOWELL	Copper Alloy (Zn < 15%)	Air and Gas (Wetted) <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(67) THERMOWELL	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(68) TRANSMITTER[1]	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(69) VALVE BODY	Carbon/Low Alloy Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(70) VALVE BODY	Carbon/Low Alloy Steel	Air and Gas (Wetted) <140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.3-2 Engineered Safety Features Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Type	Material Type	Environment Type	AERMs	Program/Activity	Discussion
(71) VALVE BODY	Cast Austenitic Stainless Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(72) VALVE BODY	Cast Austenitic Stainless Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(73) VALVE BODY	Cast Austenitic Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(74) VALVE BODY	Cast Austenitic Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(75) VALVE BODY	Cast Austenitic Stainless Steel	Raw Water Drainage	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(76) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Borated <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(77) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Borated <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.3-2 Engineered Safety Features Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Type	Material Type	Environment Type	AERMs	Program/Activity	Discussion
(78) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Borated >140	Cracking due to SCC	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(79) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Borated >140	Cracking due to SCC	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(80) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Borated >140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(81) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Borated >140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.3-2 Engineered Safety Features Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Type	Material Type	Environment Type	AERMs	Program/Activity	Discussion
(82) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Other (Stagnant)	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(83) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Other (Stagnant)	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(84) VALVE BODY	Cast Iron	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(85) VALVE BODY	Cast Iron	Indoor (No Air Conditioning)	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(86) VALVE BODY	Copper Alloy (Zn < 15%)	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(87) VALVE BODY	Copper Alloy (Zn < 15%)	Air and Gas (Wetted) <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.3-2 Engineered Safety Features Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Type	Material Type	Environment Type	AERMs	Program/Activity	Discussion
(88) VALVE BODY	Copper Alloy (Zn < 15%)	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(89) VALVE BODY	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(90) VALVE BODY	Stainless Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(91) VALVE BODY	Stainless Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(92) VALVE BODY	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(93) VALVE BODY	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(94) VALVE BODY	Stainless Steel	Outdoor	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(95) VALVE BODY	Stainless Steel	Treated Water Other (Stagnant)	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.

Table 3.3-2 Engineered Safety Features Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Type	Material Type	Environment Type	AERMs	Program/Activity	Discussion
(96) VALVE BODY	Stainless Steel	Treated Water Other (Stagnant)	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(97) VENTILATION DUCTWORK	Galvanized Carbon Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(98) VENTILATION DUCTWORK	Galvanized Carbon Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

1. Selected instruments were conservatively included within the scope of License Renewal. Consideration was given to the consequences of an instrument housing pressure boundary failure. Where an instrument was unisolable from a pressure source and is of sufficient size that a system function would be degraded should the pressure boundary fail, that instrument is included for License Renewal review.

2. Material prefixes with HX are used to identify heat exchanger materials which perform a heat transfer intended function in addition to the typical material usage function of pressure boundary.

3.4 Aging Management of Auxiliary Systems

The results of the aging management review of the Auxiliary Systems components are provided in this section and summarized in Tables 3.4-1 and 3.4-2. Table 3.4-1 shows the aging management of system components evaluated in NUREG-1801 that are relied on for license renewal of the Auxiliary Systems components at Ginna. Included in the table is a discussion column. The discussion column will provide a conclusion indicating if the aging management evaluation results are consistent with NUREG-1801 along with any clarifications or explanations required to support the stated conclusion if that conclusion is different those of the NUREG. For a determination to be made that a table line item is "Consistent with NUREG-1801" several criteria must be met. First the plant specific component is reviewed against the GALL to ensure that the component, materials of construction and internal or external service environment are comparable to those described in a particular GALL item. Second, for those that are comparable, the results of the plant aging management review- aging effect evaluation are compared to the aging effects/mechanisms in the GALL. Finally, the programs credited in the GALL for managing those aging effects are compared to the programs invoked in the plant evaluation. If, using good engineering judgment, it could be reasonably concluded that the plant evaluation is in agreement with the GALL evaluation a line item was considered consistent with NUREG-1801. There are cases where components and component material/environment combinations and aging effects are common between a NUREG-1801 line item and the plant evaluation but the aging management program selections differ. In those cases the discussion column will indicate the plant aging management program selection but no conclusion will be made that the line item is consistent with the GALL. Table 3.4-2 contains the Auxiliary Systems components aging management review results that are not addressed in NUREG-1801. A plant component is considered not addressed by the NUREG if the component type is not evaluated in the GALL or has a different material of construction or operating environment than evaluated in the GALL. This table includes the component types, materials, environments, aging effects requiring management, the programs and activities for managing aging, and a discussion column. To avoid confusion, no attempt was made to interrelate material/environment/aging effects from one NUREG-1801 chapter to another. Note that these tables only include those components, materials and environments that are applicable to a PWR.

Materials

The materials of construction of a component have a major influence on the evaluation of aging effects applicable to the component. Sources of information used to identify materials of construction include original equipment specifications, vendor technical manuals and drawings, fabrication drawings, piping line specifications, modification design records and field walkdowns/verifications. The tables below account for the materials of construction for the components requiring an aging management review. Since similar materials are susceptible to the same aging effects/mechanisms, the tables itemize the component types (i.e., groupings) while factoring in the materials of construction.

Environment

As previously described, the environment(s) to which components are exposed are critical in the determination of potential aging mechanisms and effects. A review of plant design documentation was performed to quantify the environmental conditions to which Ginna Station equipment is exposed. This review identified that some equipment is exposed to a variety of environments. This can include normal operating conditions and post accident conditions. Since aging mechanisms and effects will be primarily driven by the environmental conditions to which equipment is exposed on a daily basis, under normal operating conditions, these conditions will differ from the design parameters which are established based upon the worst case scenario (e.g., LOCA conditions). Ginna Station equipment environments may be categorized into basic external and internal environments detailed in Section 3.1.2.

Aging Effects Requiring Management

After the components requiring aging management review were identified and grouped by materials of construction and environment, a review of industry and plant-specific operating experience was performed. The purpose of this review was to assure that all applicable aging effects were identified, and to evaluate the effectiveness of existing aging management programs.

This experience review was performed utilizing various industry and plant-specific programs and databases. Industry operating experience sources included NRC Generic Publications (including Information Notices, Circulars, Bulletins, and Generic Letters), INPO Significant Operating Event Reports (SOER), EPRI Technical Reports, and other information sources, such as the B&W Owners Group Non-Class 1 Mechanical Tools Implementation document, Westinghouse Generic Technical Reports (GTRs), and the Generic Aging Lessons Learned (GALL) report.

Plant specific operating experience sources included Semi-annual and Annual Reports to AEC/NRC, Abnormal Occurrence and Licensee Event Reports (LERs), Non-Conformance Reports (NCRs), Corrective Action Reports (CARs), Refueling, Inspection and Overhaul Reports (RIOs), Inservice Inspection (ISI) Reports, Identified Deficiency Reports (IDRs), and ACTION Reports (ARs) from 1969 to the present. Information from these sources was compiled in various databases. Based upon the material of construction, the applicable environments, and operating experience the potential aging effects requiring management for each of the components was identified as documented in the tables below.

Time-Limited Aging Analysis

In addition to those identified in NUREG-1801, any additional time-limited aging analyses (TLAA) identified as appropriate to the system are identified in Section 4.0.

Conclusion

The programs and activities selected to manage the aging effects of the Auxiliary Systems are identified in Table 3.4-1 and Table 3.4-2. A description of these aging management activities is provided in Appendix B, along with the demonstration that the identified aging effects will be managed for the period of extended operation. Therefore, based on the demonstrations provided in Appendix B, the effects of aging associated with the system components will be adequately managed so that there is reasonable assurance that the intended function(s) will be maintained consistent with the current licensing basis during the period of extended operation.

Table 3.4-1 Auxiliary Systems - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(1) Components in spent fuel pool cooling and cleanup	Loss of material due to general, pitting, and crevice corrosion	Water chemistry and one-time inspection	Yes, detection of aging effects is to be further evaluated	Consistent with NUREG-1801. The Water Chemistry Control Program is credited with managing the aging effects of loss of material due to general, pitting, and crevice corrosion. The One-Time Inspection as well as the Periodic Surveillance and Preventive Maintenance Programs will be used to verify the effectiveness of the Water Chemistry Control Program.
(2) Linings in spent fuel pool cooling and cleanup system; seals and collars in ventilation systems	Hardening, cracking and loss of strength due to elastomer degradation; loss of material due to wear	Plant specific	Yes, plant specific	Consistent with NUREG-1801. The Spent Fuel Cooling system (Section A3 of the NUREG) at Ginna Station contains no components that are elastomer lined. For ventilation systems, the One-Time Inspection and Periodic Surveillance and Preventive Maintenance Programs are credited for managing the hardening, cracking and loss of strength aging effects. The Systems Monitoring Program is credited for managing the aging effect of loss of material due to wear.
(3) Components in load handling, chemical and volume control system (PWR), and reactor water cleanup and shutdown cooling systems (older BWR)	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	Consistent with NUREG-1801. Cumulative Fatigue Damage is addressed as a TLAA in Section 4.3.

Table 3.4-1 Auxiliary Systems - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(4) Heat exchangers in reactor water cleanup system (BWR); high pressure pumps in chemical and volume control system (PWR)	Crack initiation and growth due to SCC or cracking	Plant specific	Yes, plant specific	Consistent with NUREG-1801. Although the NUREG references a temperature gate of < 90°C (200°F) and a single aging effect (cracking due to SCC), materials science supports (1) a temperature gate > 140°F for cracking due to SCC, and (2) loss of material due to pitting (stagnant or low flow conditions) and crevice corrosion for all temperatures. The aging effect identified by Ginna (loss of material) for temperatures < 140°F differs from that of the NUREG. That not withstanding, the Water Chemistry Control Program, credited for managing the aging effects for all temperatures, is consistent with the NUREG and will preclude the possibility of crack initiation and growth due to SCC. The One-Time Inspection Program as well as the Periodic Surveillance and Preventive Maintenance Program are credited with verifying the adequacy of the Chemistry program.
(5) Components in ventilation systems, diesel fuel oil system, and emergency diesel generator systems; external surfaces of carbon steel components	Loss of material due to general, pitting, and crevice corrosion, and MIC	Plant specific	Yes, plant specific	Consistent with NUREG-1801. For the internal environments of ventilation systems, the diesel fuel oil systems, and the emergency diesel generator system, the One-Time Inspection , Periodic Surveillance and Preventive Maintenance , Closed-Cycle (Component) Cooling Water System , and the Fuel Oil Chemistry Programs are credited for managing applicable aging effects. For the external surfaces of all carbon steel components, the Systems Monitoring Program will be credited for managing the aging effects of loss of material.
(6) Components in reactor coolant pump oil collect system of fire protection	Loss of material due to galvanic, general, pitting, and crevice corrosion	One-time inspection	Yes, detection of aging effects is to be further evaluated	Consistent with NUREG-1801. The aging effects of components within the Reactor Coolant Pump Oil Collection system will managed by the One-Time Inspection Program. In addition, selected components will be inspected on a periodic basis in conjunction with the Periodic Surveillance and Preventive Maintenance Program.

Table 3.4-1 Auxiliary Systems - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(7) Diesel fuel oil tanks in diesel fuel oil system and emergency diesel generator system	Loss of material due to general, pitting, and crevice corrosion, MIC, and biofouling	Fuel oil chemistry and one-time inspection	Yes, detection of aging effects is to be further evaluated	The Fuel Oil Chemistry Program is credited with managing applicable aging effects. In lieu of the One-Time Inspection Program, Ginna Station has chosen to use the Periodic Surveillance and Preventive Maintenance Program to verify the adequacy of the Fuel Oil Chemistry Program in managing these aging effects.
(8) Heat exchangers in chemical and volume control system	Crack initiation and growth due to SCC and cyclic loading	Water chemistry and a plant-specific verification program	Yes, plant specific	Consistent with NUREG-1801. Although the NUREG references a temperature gate of < 90°C (200°F) and a single aging effect (cracking due to SCC), materials science supports (1) a temperature gate > 140°F for cracking due to SCC, and (2) loss of material due to pitting (stagnant or low flow conditions) and crevice corrosion for all temperatures. The aging effect identified by Ginna (loss of material) for temperatures < 140°F differs from that of the NUREG. That not withstanding, the Water Chemistry Control Program, credited for managing the aging effects for all temperatures, is consistent with the NUREG and will preclude the possibility of crack initiation and growth due to SCC. The One-Time Inspection Program as well as the Periodic Surveillance and Preventive Maintenance Program are credited with verifying the adequacy of the Water Chemistry Control Program.
(9) Neutron absorbing sheets in spent fuel storage racks	Reduction of neutron absorbing capacity and loss of material due to general corrosion (Boral, boron steel)	Plant specific	Yes, plant specific	Consistent with NUREG-1801. The Periodic Surveillance and Preventive Maintenance Program will direct the scheduling of activities that will detect applicable aging effects under the Spent Fuel Pool Neutron Absorber Monitoring Program.
(10) New fuel rack assembly	Loss of material due to general, pitting, and crevice corrosion	Structures monitoring	No	Consistent with NUREG-1801. The Structures Monitoring Program is credited with managing the aging effects of loss of material due to general, pitting, and crevice corrosion.

Table 3.4-1 Auxiliary Systems - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(11) Spent fuel storage racks and valves in spent fuel pool cooling and cleanup	Crack initiation and growth due to stress corrosion cracking	Water chemistry	No	Consistent with NUREG-1801. Although the NUREG references a temperature gate of < 90°C (200°F) and a single aging effect (cracking due to SCC), materials science supports (1) a temperature gate > 140°F for cracking due to SCC, and (2) loss of material due to pitting (stagnant or low flow conditions) and crevice corrosion for all temperatures. The aging effect identified by Ginna (loss of material) for temperatures < 140°F differs from that of the NUREG. That not withstanding, the Water Chemistry Control Program, credited for managing the aging effects for all temperatures, is consistent with the NUREG and will preclude the possibility of crack initiation and growth due to SCC. The One-Time Inspection Program as well as the Periodic Surveillance and Preventive Maintenance Program are credited with verifying the adequacy of the Chemistry program.
(12) Neutron absorbing sheets in spent fuel storage racks	Reduction of neutron absorbing capacity due to Boraflex degradation	Boraflex monitoring	No	Consistent with NUREG-1801. The Spent Fuel Pool Neutron Absorber Monitoring Program is functionally equivalent to the Boraflex Monitoring Program. However, borated stainless steel neutron absorber panels (line item 9 above) are included in the scope of this monitoring program. Existing boraflex neutron absorber panels are not credited in the CLB for reactivity control in the Spent Fuel Pool, and therefore are excluded from the scope of this monitoring program. The Spent Fuel Pool Neutron Absorber Monitoring Program manages the aging effects of reduction of neutron absorbing capacity and loss of material due to general corrosion of the borated stainless steel panels.

Table 3.4-1 Auxiliary Systems - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(13) Closure bolting and external surfaces of carbon steel and low-alloy steel components	Loss of material due to boric acid corrosion	Boric acid corrosion	No	Consistent with NUREG-1801. The Boric Acid Corrosion Program is credited with managing the aging effect of loss of material due to boric acid corrosion on the external surfaces of carbon/low alloy steel components (including closure bolting). Although not addressed in the NUREG, the following additional systems at Ginna Station contain carbon/low alloy steel components and have the potential for exposure to boric acid spillage (located in Containment or the Auxiliary Building) and are included in this evaluation: Component Cooling Water, Service Water, Fire Protection, Containment Ventilation, Essential Ventilation, Waste Disposal, Radiation Monitoring, and Cranes, Hoists and Lifting Devices.
(14) Components in or serviced by closed-cycle cooling water system	Loss of material due to general, pitting, and crevice corrosion, and MIC	Closed-cycle cooling water system	No	Consistent with NUREG-1801. Components within the Chemical and Volume Control, Component Cooling Water, Waste Disposal, and the Emergency Power systems are subject to the Closed-Cycle (Component) Cooling Water System Program. This program is credited with managing the aging effects of loss of material due to general, pitting, and crevice corrosion as well as micro-biologically influenced corrosion (MIC).
(15) Cranes including bridge and trolleys and rail system in load handling system	Loss of material due to general corrosion and wear	Overhead heavy load and light load handling systems	No	The Periodic Surveillance and Preventive Maintenance Program implements the Inspection of Heavy Load and Refueling Handling Systems procedures at Ginna Station. The periodic inspections are credited with managing the aging effects of loss of material due to general corrosion and wear.

Table 3.4-1 Auxiliary Systems - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(16) Components in or serviced by open-cycle cooling water systems	Loss of material due to general, pitting, crevice, and galvanic corrosion, MIC, and biofouling; buildup of deposit due to biofouling	Open-cycle cooling water system	No	Consistent with NUREG-1801. Components within the Service Water, Component Cooling Water, Containment Ventilation, Spent Fuel Cooling, and the Emergency Power systems are subject to the Open-Cycle Cooling (Service) Water System Program as implemented by the Service Water System Reliability Optimization Program (SWSROP). This program is credited with managing the aging effects of loss of material due to general, pitting, crevice, and galvanic corrosion, MIC, and biofouling. The Periodic Surveillance and Preventive Maintenance Program is used at Ginna Station to verify the effectiveness of the Open-Cycle Cooling (Service) Water System Program.
(17) Buried piping and fittings	Loss of material due to general, pitting, and crevice corrosion, and MIC	Buried piping and tanks surveillance or Buried piping and tanks inspection	No Yes, detection of aging effects and operating experience are to be further evaluated	The Buried Piping and Tanks Inspection Program is implemented by the Periodic Surveillance and Preventive Maintenance Program at Ginna Station. Tanks in the Emergency Power system are periodically inspected for signs of applicable aging effects. In addition, a one-time ultrasonic inspection will be performed to verify the effectiveness of the Preventive Maintenance Program. For buried piping, the Fire Water System Program is credited for managing the effects of aging for buried cast iron piping and fittings. External surfaces of buried piping are visually examined during maintenance activities (inspections of opportunity) performed as a result of performance tests. No evidence of age-related degradation has been detected from inspections performed to date. Cast iron fire system and service water piping at Ginna Station is ductile cast iron, not gray cast iron. Ductile irons are not susceptible to loss of structural integrity due to selective leaching mechanisms, and generally display excellent resistance to general corrosion due to exposure to non-aggressive ground water. Ground water/lake water at Ginna Station is analyzed periodically and analyses performed to date confirm that the water is non-aggressive.

Table 3.4-1 Auxiliary Systems - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(18) Components in compressed air system	Loss of material due to general and pitting corrosion	Compressed air monitoring	No	The Instrument and Service Air systems are not in scope to License Renewal and therefore not subject to an aging management review.
(19) Components (doors and barrier penetration seals) and concrete structures in fire protection	Loss of material due to wear; hardening and shrinkage due to weathering	Fire protection	No	Consistent with NUREG-1801. The Fire Protection Program is credited with managing the aging effects of loss of material due to wear and general corrosion and hardening and shrinkage for components/structures that act as fire barriers.
(20) Components in water-based fire protection	Loss of material due to general, pitting, crevice, and galvanic corrosion, MIC, and biofouling	Fire water system	No	Consistent with NUREG-1801. The Fire Water System Program is credited with managing the aging effects of loss of material due to general, pitting, crevice, and galvanic corrosion, MIC, and biofouling. At Ginna Station, the Periodic Surveillance and Preventive Maintenance Program is used to verify the effectiveness of the Fire Water System Program.
(21) Components in diesel fire system	Loss of material due to galvanic, general, pitting, and crevice corrosion	Fire protection and fuel oil chemistry	No	The Fuel Oil Chemistry Program is credited with managing the applicable aging effects. At Ginna Station, the Periodic Surveillance and Preventive Maintenance Program is used to verify the effectiveness of this program.
(22) Tanks in diesel fuel oil system	Loss of material due to general, pitting, and crevice corrosion	Above ground carbon steel tanks	No	There are no aboveground diesel fuel oil tanks in the Emergency Power system at Ginna Station.
(23) Closure bolting	Loss of material due to general corrosion; crack initiation and growth due to cyclic loading and SCC	Bolting integrity	No	The Bolting Integrity Program is credited for managing the aging effects “loss of material due to general corrosion, loss of preload due to stress relaxation, and crack initiation and growth due to cyclic loading and SCC.” There are no bolts with a specified minimum yield strength > 150 ksi in the Auxiliary Systems. Therefore, SCC is not an applicable aging mechanism.

Table 3.4-1 Auxiliary Systems - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(24) Components (aluminum bronze, brass, cast iron, cast steel) in open-cycle and closed-cycle cooling water systems, and ultimate heat sink	Loss of material due to selective leaching	Selective leaching of materials	No	In addition to the Open-Cycle Cooling (Service) Water System and Closed-Cycle (Component) Cooling Water System Programs, the Periodic Surveillance and Preventive Maintenance Program or the One-Time Inspection Program will be credited with managing the aging effect of loss of material for components within the Open and Closed-Cycle Cooling Water systems.

Table 3.4-1 Auxiliary Systems - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(25) Fire barriers, walls, ceilings and floors in fire protection	Concrete cracking and spalling due to freeze-thaw, aggressive chemical attack, and reaction with aggregates; loss of material due to corrosion of embedded steel	Fire protection and structures monitoring	No	The Fire Protection Program in conjunction with the Structures Monitoring Program identifies the evidence that an aging mechanism is present and active and also provides confirmation and verification of the absence of all types of aging effects. Indication of aging effects may be absent if the materials of construction, design specifications, and operational environment preclude an aging mechanism but, due to the long lead time necessary for some effects to manifest themselves, it is prudent to periodically assess the condition of SSCs regardless of the likelihood that a particular aging mechanism is applicable. The degradation of inaccessible concrete can create symptoms of aging effects that are detectable in accessible areas. Conversely, if aging effects are present in accessible areas it is sensible to extrapolate those effects into inaccessible areas and perform additional evaluations. Concrete in indoor and outdoor environments have been evaluated for the following aging mechanisms: Aging Mechanism: Freeze-Thaw Aging Effect: Loss of Material Evaluation: The contract-specified air contents are within the range specified by current revisions of ACI 318, and the contract-specified water-to-cement ratio meets the recommendations of ACI 318-63 ($\leq$ 0.53). Therefore, loss of material and cracking of concrete due to freeze-thaw are not probable aging effects at Ginna Station and have not been observed to date. Aging Mechanism: Aggressive Chemical Attack Aging Effect: Loss of Material, Changes in Material Properties Evaluation: Concrete degradation in air due to aggressive rainwater is insignificant and the below-grade/lake water environment is non-aggressive. Additionally, recent structural inspections revealed no evidence of degradation owing to aggressive chemical attack; therefore, loss of material and change in material properties due to aggressive chemical

Table 3.4-1 Auxiliary Systems - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(25) (continued)				attack are not probable aging effects at Ginna Station and have not been observed to date. The Structures Monitoring Program requires periodic monitoring of ground/lake water to verify chemistry remains non-aggressive. Aging Mechanism: Corrosion of Embedded Steel Aging Effect: Loss of Material, Cracking, Loss of Bond Evaluation: Since the embedded steel is not exposed to an environment which is considered aggressive, loss of material, cracking, and loss of bond due to corrosion of embedded steel are not probable aging effects at Ginna Station and have not been observed to date. Aging Mechanism: Reaction with Aggregates Aging Effect: Cracking, Expansion Evaluation: During construction the aggregates were tested for potential reactivity in accordance with ASTM C227 and ASTM C295, cracking and expansion due to reaction with aggregates are not probable aging effects at Ginna Station and have not been observed to date. Aging Mechanism: Settlement Aging Effect: Cracking, Distortion, Increase in Component Stress Level Evaluation: All structures at Ginna Station are either founded on bedrock, steel foundation piles that are driven to bedrock, or have foundations that consist of caissons extending to bedrock. Structural inspections indicate no visible evidence of settlement since construction of the station. During the Systematic Evaluation Program, the NRC concluded that settlement of foundations and buried equipment is not a safety concern for Ginna Station. Cracking, distortion, and an increase in component stress levels due to settlement are not probable aging effects at Ginna Station and have not been observed to date.

Table 3.4-1 Auxiliary Systems - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(25) (continued)				Aging Mechanism: Leaching of Calcium Hydroxide Aging Effect: Change in Material Properties Evaluation: The original construction specifications met the intent of ACI 201.2R. Change in material properties due to leaching of calcium hydroxide is not a probable aging effect at Ginna Station and has not been observed to date. Additionally, masonry walls are used as fire barriers at Ginna Station. Masonry wall inspections are incorporated into the Structures Monitoring Program . The Structures Monitoring Program effectively manages cracking due to restraint, shrinkage and creep. Operating experience has shown that concrete has not experienced unanticipated aging effects at Ginna Station. That notwithstanding, the identification of the above aging effects by the Structures Monitoring Program , as well as the resistance provided by the materials of construction provide adequate assurance that all types of concrete aging effects will be identified and managed through out the extended period of operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(1) AIR OPERATED DAMPER HOUSING	Cast Iron	Air and Gas (Wetted) <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(2) AIR OPERATED DAMPER HOUSING	Cast Iron	Containment	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(3) AIR OPERATED DAMPER HOUSING	Galvanized Carbon Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(4) AIR OPERATED DAMPER HOUSING	Galvanized Carbon Steel	Containment	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(5) AIR OPERATED DAMPER HOUSING	Galvanized Carbon Steel	Indoor (Air Conditioning)	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(6) AIR OPERATED DAMPER HOUSING	Galvanized Carbon Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(7) BELL[1]	Cast Iron	Indoor (No Air Conditioning)	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(8) BLOWER CASING	Galvanized Carbon Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(9) BLOWER CASING	Galvanized Carbon Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, stainless steel exposed to ventilation air (T<140°F) would not be expected to exhibit loss of material due to pitting and crevice corrosion. Therefore no aging effects are applicable and no aging management program is required.
(10) COMPRESSOR CASING (included for conservatism)	Carbon/Low Alloy Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(11) CONDENSER	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(12) CONDENSER	Stainless Steel	Treated Water Borated >140	Cracking due to SCC	Periodic Surveillance and Preventive Maintenance Program	Structure/component type, material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(13) CONDENSER	Stainless Steel	Treated Water Borated >140	Cracking due to SCC	Water Chemistry Control Program	Structure/component type, material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(14) CONDENSER	Stainless Steel	Treated Water Borated >140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Structure/component type, material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(15) CONDENSER	Stainless Steel	Treated Water Borated >140	Loss of Material	Water Chemistry Control Program	Structure/component type, material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(16) CONDENSER	Stainless Steel	Treated Water Other	Loss of Material	Closed-Cycle (Component) Cooling Water System Program	Structure/component type, material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(17) CONTROLLER[1]	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(18) COOLER	Cast Iron	Indoor (No Air Conditioning)	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(19) COOLER	Cast Iron	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(20) COOLER	Cast Iron	Raw Water	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(21) COOLER	Cast Iron	Treated Water Other	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(22) COOLER	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(23) COOLER	Copper Alloy (Zn < 15%)	Treated Water Other	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(24) COOLER	Copper Alloy (Zn > 15%)	Raw Water	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(25) COOLER	Copper Alloy (Zn > 15%)	Treated Water Other	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(26) COOLER	Stainless Steel	Concrete	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(27) COOLER	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(28) COOLER	Stainless Steel	Treated Water Borated >140	Cracking due to SCC	Periodic Surveillance and Preventive Maintenance Program	Structure/component type, material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(29) COOLER	Stainless Steel	Treated Water Borated >140	Cracking due to SCC	Water Chemistry Control Program	Structure/component type, material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(30) COOLER	Stainless Steel	Treated Water Borated >140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Structure/component type, material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(31) COOLER	Stainless Steel	Treated Water Borated >140	Loss of Material	Water Chemistry Control Program	Structure/component type, material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(32) COOLER	Stainless Steel	Treated Water Other	Loss of Material	Closed-Cycle (Component) Cooling Water System Program	Structure/component type, material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(33) COOLING COIL	Carbon/Low Alloy Steel	Air and Gas (Wetted) <140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(34) COOLING COIL	Copper Alloy (Zn < 15%)	Air and Gas (Wetted) <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(35) COOLING COIL	Copper Alloy (Zn < 15%)	Air and Gas (Wetted) <140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(36) COOLING COIL	Copper Alloy (Zn < 15%)	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(37) COOLING COIL	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(38) COOLING COIL	Galvanized Carbon Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(39) COOLING COIL	Galvanized Carbon Steel	Indoor (Air Conditioning)	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(40) COOLING COIL	Stainless Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(41) COOLING COIL	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(42) COOLING COIL	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(43) CUTTER ASSEMBLY	Aluminum	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(44) CUTTER ASSEMBLY	Aluminum	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(45) DAMPER HOUSING/FRAME	Aluminum	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(46) DAMPER HOUSING/FRAME	Aluminum	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(47) DAMPER HOUSING/FRAME	Galvanized Carbon Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(48) DAMPER HOUSING/FRAME	Galvanized Carbon Steel	Containment	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(49) DAMPER HOUSING/FRAME	Galvanized Carbon Steel	Indoor (Air Conditioning)	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(50) DAMPER HOUSING/FRAME	Galvanized Carbon Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(51) DAMPER HOUSING/FRAME	Galvanized Carbon Steel	Outdoor	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(52) DEMINERALIZER	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(53) DEMINERALIZER	Stainless Steel	Treated Water Borated <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(54) DEMINERALIZER	Stainless Steel	Treated Water Borated <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(55) DIAPHRAGM SEAL	Neoprene	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(56) DIAPHRAGM SEAL	Neoprene	Treated Water Borated <140	Change in Material Properties and Cracking	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(57) ENGINE CASING	Carbon/Low Alloy Steel	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(58) ENGINE CASING	Cast Iron	Indoor (No Air Conditioning)	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(59) EXPANSION JOINT	Flexible Asbestos Cloth	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(60) EXPANSION JOINT	Flexible Asbestos Cloth	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(61) EXPANSION JOINT	Galvanized Carbon Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(62) EXPANSION JOINT	Galvanized Carbon Steel	Air and Gas (Wetted) >140	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(63) EXPANSION JOINT	Galvanized Carbon Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(64) EXPANSION JOINT	Neoprene	Indoor (Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(65) EXPANSION JOINT	Neoprene	Raw Water	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(66) EXPANSION JOINT	Rubber Coated Asbestos	Air and Gas (Wetted) <140	Change in Material Properties and Cracking	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(67) EXPANSION JOINT	Rubber Coated Asbestos	Containment	Change in Material Properties and Cracking	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(68) EXPANSION JOINT	Stainless Steel	Air and Gas (Wetted) >140	Cracking due to SCC	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(69) EXPANSION JOINT	Stainless Steel	Air and Gas (Wetted) >140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(70) EXPANSION JOINT	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(71) FAN CASING	Aluminum	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(72) FAN CASING	Aluminum	Indoor (Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(73) FAN CASING	Aluminum	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(74) FAN CASING	Aluminum	Outdoor	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(75) FAN CASING	Carbon/Low Alloy Steel	Indoor (Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(76) FAN CASING	Cast Iron	Indoor (No Air Conditioning)	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(77) FAN CASING	Galvanized Carbon Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(78) FAN CASING	Galvanized Carbon Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(79) FASTENERS (BOLTING)	Carbon/Low Alloy Steel	Indoor (No Air Conditioning)	Cracking due to SCC	Bolting Integrity Program	There are no bolts with a specified minimum yield strength > 150 ksi in this system. Therefore, SCC is not an applicable aging effect/mechanism.
(80) FASTENERS (BOLTING)	Carbon/Low Alloy Steel	Indoor (No Air Conditioning)	Loss of Preload due to Stress Relaxation	Bolting Integrity Program	Material and environment grouping are included in NUREG-1801. Aging effect of loss of preload due to stress relaxation is applicable, but is not included in Chapter V - Section E, Chapter VII - Section I, or Chapter VIII - Section H of the NUREG.
(81) FASTENERS (BOLTING)	Stainless Steel	Borated Water Leaks	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(82) FILTER HOUSING	Aluminum	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(83) FILTER HOUSING	Aluminum	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(84) FILTER HOUSING	Aluminum	Oil and Fuel Oil	Loss of Material	Fuel Oil Chemistry Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(85) FILTER HOUSING	Aluminum	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(86) FILTER HOUSING	Aluminum	Treated Water Other	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(87) FILTER HOUSING	Carbon/Low Alloy Steel	Indoor (Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(88) FILTER HOUSING	Carbon/Low Alloy Steel	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(89) FILTER HOUSING	Cast Iron	Indoor (No Air Conditioning)	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(90) FILTER HOUSING	Cast Iron	Oil and Fuel Oil	Loss of Material	Fuel Oil Chemistry Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(91) FILTER HOUSING	Cast Iron	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(92) FILTER HOUSING	Fiberglass Reinforced Plastic (FRP)	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(93) FILTER HOUSING	Fiberglass Reinforced Plastic (FRP)	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(94) FILTER HOUSING	Galvanized Carbon Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(95) FILTER HOUSING	Galvanized Carbon Steel	Indoor (Air Conditioning)	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(96) FILTER HOUSING	Galvanized Carbon Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(97) FILTER HOUSING	Stainless Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, stainless steel exposed to ventilation air (T<140°F) would not be expected to exhibit loss of material due to pitting and crevice corrosion. Therefore no aging effects are applicable and no aging management program is required.
(98) FILTER HOUSING	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(99) FILTER HOUSING	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(100) FILTER HOUSING	Stainless Steel	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(101) FILTER HOUSING	Stainless Steel	Treated Water Borated <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(102) FILTER HOUSING	Stainless Steel	Treated Water Borated <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(103) FLAME ARRESTOR	Aluminum	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(104) FLAME ARRESTOR	Aluminum	Oil and Fuel Oil	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(105) FLANGE	Carbon/Low Alloy Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(106) FLOW ELEMENT	Stainless Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(107) FLOW ELEMENT	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(108) FLOW ELEMENT	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(109) FLOW ELEMENT	Stainless Steel	Treated Water Borated <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(110) FLOW ELEMENT	Stainless Steel	Treated Water Borated <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(111) FLOW ELEMENT	Stainless Steel	Treated Water Other	Loss of Material	Closed-Cycle (Component) Cooling Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(112) FLOW METER[1]	Stainless Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(113) FLOW METER[1]	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(114) FLOW NOZZLES	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(115) GAS CYLINDER	Carbon/Low Alloy Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(116) HAND CONTROL STATION	Carbon/Low Alloy Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(117) HAND CONTROL STATION	Carbon/Low Alloy Steel	Indoor (Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(118) HAND CONTROL STATION	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(119) HEAT EXCHANGER	Carbon/Low Alloy Steel	Raw Water	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(120) HEAT EXCHANGER	Carbon/Low Alloy Steel	Raw Water	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Structure/component type, material, and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(121) HEAT EXCHANGER	Cast Iron	Indoor (No Air Conditioning)	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(122) HEAT EXCHANGER	Cast Iron	Raw Water	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(123) HEAT EXCHANGER	Cast Iron	Treated Water Other (Stagnant)	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(124) HEAT EXCHANGER	Copper Alloy (Zn < 15%)	Raw Water	Loss of Material	Open-Cycle Cooling (Service) Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(125) HEAT EXCHANGER	Copper Alloy (Zn < 15%)	Raw Water	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(126) HEAT EXCHANGER	Copper Alloy (Zn < 15%)	Treated Water Other	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(127) HEAT EXCHANGER	Copper Alloy (Zn > 15%)	Air and Gas (Wetted) >140	Loss of Heat Transfer	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(128) HEAT EXCHANGER	Copper Alloy (Zn > 15%)	Air and Gas (Wetted) >140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(129) HEAT EXCHANGER	Copper Alloy (Zn > 15%)	Raw Water	Loss of Heat Transfer	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(130) HEAT EXCHANGER	Copper Alloy (Zn > 15%)	Raw Water	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(131) HEAT EXCHANGER	Copper Alloy (Zn > 15%)	Treated Water Other	Loss of Heat Transfer	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(132) HEAT EXCHANGER	Copper Alloy (Zn > 15%)	Treated Water Other	Loss of Material	Closed-Cycle (Component) Cooling Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(133) HEAT EXCHANGER	Copper Alloy (Zn > 15%)	Treated Water Other	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(134) HEAT EXCHANGER	HX-Copper Alloy (Zn < 15%)[2]	Raw Water	Loss of Heat Transfer	Open-Cycle Cooling (Service) Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(135) HEAT EXCHANGER	HX-Copper Alloy (Zn < 15%)[2]	Raw Water	Loss of Heat Transfer	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(136) HEAT EXCHANGER	HX-Copper Alloy (Zn < 15%)[2]	Treated Water Other	Loss of Heat Transfer	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(137) HEAT EXCHANGER	HX-Copper Alloy (Zn > 15%)[2]	Treated Water Other	Loss of Heat Transfer	Closed-Cycle (Component) Cooling Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(138) HEAT EXCHANGER	HX-Copper Alloy (Zn > 15%)[2]	Treated Water Other	Loss of Heat Transfer	Open-Cycle Cooling (Service) Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(139) HEAT EXCHANGER	HX-Stainless Steel[2]	Raw Water	Loss of Heat Transfer	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(140) HEAT EXCHANGER	HX-Stainless Steel[2]	Treated Water Other	Loss of Heat Transfer	Closed-Cycle (Component) Cooling Water System Program	Structure/component type, material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(141) HEAT EXCHANGER	HX-Stainless Steel[2]	Treated Water Primary <140	Loss of Heat Transfer	Water Chemistry Control Program	Structure/component type, material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(142) HEAT EXCHANGER	HX-Stainless Steel[2]	Treated Water Secondary >120	Loss of Heat Transfer	Water Chemistry Control Program	Structure/component type, material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(143) HEAT EXCHANGER	Stainless Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(144) HEAT EXCHANGER	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(145) HEAT EXCHANGER	Stainless Steel	Raw Water	Loss of Heat Transfer	Open-Cycle Cooling (Service) Water System Program	Structure/component type, material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(146) HEAT EXCHANGER	Stainless Steel	Raw Water	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(147) HEAT EXCHANGER	Stainless Steel	Treated Water Borated <140	Loss of Heat Transfer	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(148) HEAT EXCHANGER	Stainless Steel	Treated Water Borated <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(149) HEAT EXCHANGER	Stainless Steel	Treated Water Borated <140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(150) HEAT EXCHANGER	Stainless Steel	Treated Water Borated <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(151) HEAT EXCHANGER	Stainless Steel	Treated Water Other	Loss of Material	Closed-Cycle (Component) Cooling Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(152) HEAT EXCHANGER	Stainless Steel	Treated Water Primary <140	Loss of Material	Water Chemistry Control Program	Structure/component type, material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(153) HEAT EXCHANGER	Stainless Steel	Treated Water Secondary >120	Cracking due to SCC	Water Chemistry Control Program	Structure/component type, material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(154) HEAT EXCHANGER	Stainless Steel	Treated Water Secondary >120	Loss of Material	Water Chemistry Control Program	Structure/component type, material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(155) HEATER	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(156) HEATER	Copper Alloy (Zn < 15%)	Treated Water Secondary >120	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(157) HEATING COIL	Carbon/Low Alloy Steel	Air and Gas (Wetted) <140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(158) HEATING COIL	Carbon/Low Alloy Steel	Indoor (Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(159) HEATING ELEMENT	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(160) HEATING ELEMENT	Stainless Steel	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(161) HEATING ELEMENT	Stainless Steel	Treated Water Other	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(162) HVAC EQUIPMENT PACKAGE[3]	Carbon/Low Alloy Steel	Air and Gas (Wetted) <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(163) HVAC EQUIPMENT PACKAGE[3]	Galvanized Carbon Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(164) HVAC EQUIPMENT PACKAGE[3]	Galvanized Carbon Steel	Containment	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(165) HVAC EQUIPMENT PACKAGE[3]	Galvanized Carbon Steel	Indoor (Air Conditioning)	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(166) HVAC EQUIPMENT PACKAGE[3]	Galvanized Carbon Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(167) INDICATOR[1]	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(168) INDICATOR[1]	Copper Alloy (Zn < 15%)	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(169) INDICATOR[1]	Copper Alloy (Zn < 15%)	Raw Water	Loss of Material	Open-Cycle Cooling (Service) Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(170) INDICATOR[1]	Copper Alloy (Zn < 15%)	Treated Water Other (Stagnant)	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(171) INDICATOR[1]	Copper Alloy (Zn < 15%)	Treated Water Other (Stagnant)	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(172) INDICATOR[1]	Plastic	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(173) INDICATOR[1]	Plastic	Oil and Fuel Oil	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(174) INDICATOR[1]	Plastic	Raw Water (Stagnant)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(175) INDICATOR[1]	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(176) INDICATOR[1]	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(177) INDICATOR[1]	Stainless Steel	Treated Water Borated <140	Loss of Material	One-Time Inspection Program	Structure/component type, material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(178) INDICATOR[1]	Stainless Steel	Treated Water Borated <140	Loss of Material	Water Chemistry Control Program	Structure/component type, material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(179) INDICATOR[1]	Stainless Steel	Treated Water Other	Loss of Material	Closed-Cycle (Component) Cooling Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(180) LEVEL GLASS	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(181) LEVEL GLASS	Copper Alloy (Zn < 15%)	Oil and Fuel Oil	Loss of Material	Fuel Oil Chemistry Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(182) LEVEL GLASS	Copper Alloy (Zn < 15%)	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Consistent with NUREG-1801. Material/environment grouping and aging effect are included in NUREG-1801. The Periodic Surveillance and Preventive Maintenance Program will be used to verify the effectiveness of the Fuel Oil Chemistry Program.
(183) LEVEL GLASS	Copper Alloy (Zn < 15%)	Treated Water Other (Stagnant)	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(184) LEVEL GLASS	Copper Alloy (Zn < 15%)	Treated Water Other (Stagnant)	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(185) LEVEL GLASS	Glass	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(186) LEVEL GLASS	Glass	Oil and Fuel Oil	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(187) LEVEL GLASS	Plastic	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(188) LEVEL GLASS	Plastic	Treated Water Other	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(189) LEVEL GLASS	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(190) LEVEL GLASS	Stainless Steel	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(191) MOTOR OPERATED DAMPER	Galvanized Carbon Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(192) MOTOR OPERATED DAMPER	Galvanized Carbon Steel	Outdoor	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(193) MUFFLER	Galvanized Carbon Steel	Air and Gas (Wetted) >140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(194) MUFFLER	Galvanized Carbon Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(195) ORIFICE	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(196) ORIFICE	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(197) ORIFICE	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(198) ORIFICE	Stainless Steel	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(199) ORIFICE	Stainless Steel	Raw Water Drainage	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801.
(200) ORIFICE	Stainless Steel	Treated Water Other	Loss of Material	Closed-Cycle (Component) Cooling Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(201) PENETRATION SEAL	Carbon/Low Alloy Steel	Indoor (Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(202) PENETRATION SEAL	Carbon/Low Alloy Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(203) PIPE	Carbon/Low Alloy Steel	Indoor (Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(204) PIPE	Carbon/Low Alloy Steel	Treated Water Other (Stagnant)	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(205) PIPE	Carbon/Low Alloy Steel	Treated Water Other (Stagnant)	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801.
(206) PIPE	Carbon/Low Alloy Steel	Treated Water Secondary >120	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801.
(207) PIPE	Cast Iron	Buried	Loss of Material	Fire Water System Program	Not consistent with NUREG-1801. The Fire Water System Program is credited for managing the effects of aging for buried cast iron piping and fittings. External surfaces of buried piping are visually examined during maintenance activities (inspections of opportunity) performed as a result of performance tests. No evidence of age-related degradation has been detected from inspections performed to date. Cast iron fire system and service water piping at Ginna Station is ductile cast iron, not gray cast iron. Ductile irons are not susceptible to loss of structural integrity due to selective leaching mechanisms, and generally display excellent resistance to general corrosion due to exposure to non-aggressive ground water. Ground water/lake water at Ginna Station is analyzed periodically and analyses performed to date confirm that the water is non-aggressive.
(208) PIPE	Cast Iron	Concrete	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(209) PIPE	Cast Iron	Indoor (No Air Conditioning)	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(210) PIPE	Concrete (Reinforced)	Buried	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(211) PIPE	Concrete (Reinforced)	Raw Water	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(212) PIPE	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(213) PIPE	Copper Alloy (Zn < 15%)	Oil and Fuel Oil	Loss of Material	Fuel Oil Chemistry Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(214) PIPE	Copper Alloy (Zn < 15%)	Raw Water (Stagnant)	Loss of Material	Fire Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(215) PIPE	Copper Alloy (Zn < 15%)	Raw Water (Stagnant)	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Consistent with NUREG-1801. Material/environment grouping and aging effect are included in NUREG-1801. The Periodic Surveillance and Preventive Maintenance Program will be used to verify the effectiveness of the Fire Water System Program.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(216) PIPE	Copper Alloy (Zn < 15%)	Raw Water Drainage	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(217) PIPE	Copper Alloy (Zn < 15%)	Treated Water Other (Stagnant)	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(218) PIPE	Copper Alloy (Zn < 15%)	Treated Water Other (Stagnant)	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(219) PIPE	Neoprene	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(220) PIPE	Neoprene	Containment	Change in Material Properties and Cracking	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(221) PIPE	Neoprene	Containment	Change in Material Properties and Cracking	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(222) PIPE	Neoprene	Containment	Change in Material Properties and Cracking	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(223) PIPE	Neoprene	Indoor (No Air Conditioning)	Change in Material Properties and Cracking	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(224) PIPE	Neoprene	Indoor (No Air Conditioning)	Change in Material Properties and Cracking	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(225) PIPE	Neoprene	Oil and Fuel Oil	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(226) PIPE	Neoprene	Raw Water	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(227) PIPE	Neoprene	Raw Water (Stagnant)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(228) PIPE	Neoprene	Treated Water Other	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(229) PIPE	Plastic	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(230) PIPE	Plastic	Raw Water Drainage	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(231) PIPE	Stainless Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(232) PIPE	Stainless Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(233) PIPE	Stainless Steel	Buried	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(234) PIPE	Stainless Steel	Concrete	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(235) PIPE	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(236) PIPE	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(237) PIPE	Stainless Steel	Raw Water Drainage	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(238) PIPE	Stainless Steel	Treated Water Borated <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(239) PIPE	Stainless Steel	Treated Water Borated <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(240) PIPE	Stainless Steel	Treated Water Borated >140	Cracking due to SCC	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(241) PIPE	Stainless Steel	Treated Water Borated >140	Cracking due to SCC	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(242) PIPE	Stainless Steel	Treated Water Borated >140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(243) PIPE	Stainless Steel	Treated Water Borated >140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(244) PIPE	Stainless Steel	Treated Water Other	Loss of Material	Closed-Cycle (Component) Cooling Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(245) PIPE	Stainless Steel	Treated Water Primary <140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(246) PIPE	Stainless Steel	Treated Water Primary <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(247) PIPE	Stainless Steel	Treated Water Primary, 140<T<480	Cracking due to SCC	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(248) PIPE	Stainless Steel	Treated Water Primary, 140<T<480	Cracking due to SCC	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(249) PIPE	Stainless Steel	Treated Water Primary, 140<T<480	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(250) PIPE	Stainless Steel	Treated Water Primary, 140<T<480	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(251) PROTOMATIC	Cast Iron	Indoor (No Air Conditioning)	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(252) PULSATION DAMPER	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(253) PULSATION DAMPER	Stainless Steel	Treated Water Borated <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(254) PULSATION DAMPER	Stainless Steel	Treated Water Borated <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(255) PUMP CASING	Aluminum	Indoor (Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(256) PUMP CASING	Aluminum	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(257) PUMP CASING	Aluminum	Raw Water Drainage	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(258) PUMP CASING	Aluminum	Raw Water Drainage	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(259) PUMP CASING	Carbon/Low Alloy Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(260) PUMP CASING	Cast Iron	Air and Gas (Wetted) >140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(261) PUMP CASING	Cast Iron	Indoor (Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(262) PUMP CASING	Cast Iron	Indoor (No Air Conditioning)	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(263) PUMP CASING	Cast Iron	Oil and Fuel Oil	Loss of Material	Fuel Oil Chemistry Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(264) PUMP CASING	Cast Iron	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(265) PUMP CASING	Cast Iron	Raw Water Drainage	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(266) PUMP CASING	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(267) PUMP CASING	Copper Alloy (Zn < 15%)	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(268) PUMP CASING	Stainless Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(269) PUMP CASING	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(270) PUMP CASING	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(271) PUMP CASING	Stainless Steel	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(272) PUMP CASING	Stainless Steel	Raw Water Drainage	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(273) PUMP CASING	Stainless Steel	Raw Water Drainage	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(274) PUMP CASING	Stainless Steel	Treated Water Borated <140	Loss of Material	One-Time Inspection Program	Consistent with NUREG-1801. Although the NUREG references a temperature gate of < 90°C (200°F) and a single aging effect (cracking due to SCC), materials science supports (1) a temperature gate > 140°F for cracking due to SCC, and (2) loss of material due to pitting (stagnant or low flow conditions) and crevice corrosion for all temperatures. The aging effect identified by Ginna (loss of material) for temperatures < 140°F differs from that of the NUREG. That not withstanding, the Water Chemistry Control Program, credited for managing the aging effects for all temperatures, is consistent with the NUREG and will preclude the possibility of crack initiation and growth due to SCC. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(275) PUMP CASING	Stainless Steel	Treated Water Borated <140	Loss of Material	Water Chemistry Control Program	Consistent with NUREG-1801. Although the NUREG references a temperature gate of < 90°C (200°F) and a single aging effect (cracking due to SCC), materials science supports (1) a temperature gate > 140°F for cracking due to SCC, and (2) loss of material due to pitting (stagnant or low flow conditions) and crevice corrosion for all temperatures. The aging effect identified by Ginna (loss of material) for temperatures < 140°F differs from that of the NUREG. That not withstanding, the Water Chemistry Control Program, credited for managing the aging effects for all temperatures, is consistent with the NUREG and will preclude the possibility of crack initiation and growth due to SCC. A One-Time Inspection Program is also credited to verify the adequacy of the Chemistry program.
(276) RADIATION DETECTOR	Aluminum	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(277) RADIATION DETECTOR	Aluminum	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(278) RADIATION DETECTOR	Aluminum	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(279) RADIATION DETECTOR	Stainless Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(280) RADIATION DETECTOR	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(281) RADIATION DETECTOR	Stainless Steel	Raw Water	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(282) RADIATION MONITOR SKID	Stainless Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(283) RADIATION MONITOR SKID	Stainless Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(284) RADIATION MONITOR SKID	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(285) RELEASE ASSEMBLY	Aluminum	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(286) RELEASE ASSEMBLY	Aluminum	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(287) RELEASE ASSEMBLY	Carbon/Low Alloy Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(288) RELEASE ASSEMBLY	Copper Alloy (Zn < 15%)	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(289) RELEASE ASSEMBLY	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(290) SCREEN	Cast Iron	Indoor (No Air Conditioning)	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(291) SPECIAL ELEMENT	Aluminum	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(292) SPECIAL ELEMENT	Aluminum	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(293) SPECTACLE FLANGE	Carbon/Low Alloy Steel	Air and Gas (Wetted) <140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(294) SPECTACLE FLANGE	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(295) SPECTACLE FLANGE	Stainless Steel	Buried	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(296) SPECTACLE FLANGE	Stainless Steel	Treated Water Borated <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(297) SPECTACLE FLANGE	Stainless Steel	Treated Water Borated <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(298) SPRINKLER HEAD	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(299) STRAINER HOUSING	Carbon/Low Alloy Steel	Treated Water Secondary >120	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(300) STRAINER HOUSING	Cast Iron	Containment	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(301) STRAINER HOUSING	Cast Iron	Containment	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(302) STRAINER HOUSING	Cast Iron	Indoor (No Air Conditioning)	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(303) STRAINER HOUSING	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(304) STRAINER HOUSING	Copper Alloy (Zn < 15%)	Raw Water (Stagnant)	Loss of Material	Open-Cycle Cooling (Service) Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(305) STRAINER HOUSING	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(306) STRAINER HOUSING	Stainless Steel	Treated Water Borated <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(307) STRAINER HOUSING	Stainless Steel	Treated Water Borated <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(308) STRUCTURE	Aluminum	Indoor (No Air Conditioning)	No Aging Effects	Fire Protection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(309) STRUCTURE	Concrete (Reinforced)	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(310) STRUCTURE	Concrete (Reinforced)	Outdoor	No Aging Effects	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(311) STRUCTURE	Fire Stop Materials	Indoor (No Air Conditioning)	Cracking/Delamination due to Movement	Fire Protection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(312) STRUCTURE	Fire Stop Materials	Indoor (No Air Conditioning)	Cracking/Delamination due to Shrinkage	Fire Protection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(313) STRUCTURE	Fire Stop Materials	Indoor (No Air Conditioning)	Cracking/Delamination due to Vibration	Fire Protection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(314) STRUCTURE	Fire Stop Materials	Indoor (No Air Conditioning)	Hardening and Shrinkage due to Weathering	Fire Protection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(315) STRUCTURE	Fire Stop Materials	Indoor (No Air Conditioning)	Loss of Material	Fire Protection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(316) STRUCTURE	Fire Stop Materials	Indoor (No Air Conditioning)	Separation due to Movement	Fire Protection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(317) STRUCTURE	Fire Stop Materials	Indoor (No Air Conditioning)	Separation due to Shrinkage	Fire Protection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(318) STRUCTURE	Fire Stop Materials	Indoor (No Air Conditioning)	Separation due to Vibration	Fire Protection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(319) STRUCTURE	Fire Wrap Materials	Indoor (No Air Conditioning)	Cracking/ Delamination due to Movement	Fire Protection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(320) STRUCTURE	Fire Wrap Materials	Indoor (No Air Conditioning)	Cracking/ Delamination due to Vibration	Fire Protection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(321) STRUCTURE	Fire Wrap Materials	Indoor (No Air Conditioning)	Loss of Material	Fire Protection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(322) STRUCTURE	Grout	Indoor (No Air Conditioning)	Cracking/ Delamination due to Movement	Fire Protection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(323) STRUCTURE	Grout	Indoor (No Air Conditioning)	Cracking/ Delamination due to Shrinkage	Fire Protection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(324) STRUCTURE	Grout	Indoor (No Air Conditioning)	Cracking/ Delamination due to Vibration	Fire Protection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(325) STRUCTURE	Grout	Indoor (No Air Conditioning)	Hardening and Shrinkage due to Weathering	Fire Protection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(326) STRUCTURE	Grout	Indoor (No Air Conditioning)	Loss of Material	Fire Protection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(327) STRUCTURE	Grout	Indoor (No Air Conditioning)	Separation due to Movement	Fire Protection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(328) STRUCTURE	Grout	Indoor (No Air Conditioning)	Separation due to Shrinkage	Fire Protection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(329) STRUCTURE	Grout	Indoor (No Air Conditioning)	Separation due to Vibration	Fire Protection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(330) STRUCTURE	Structural Steel - Stainless	Indoor (No Air Conditioning)	No Aging Effects	Fire Protection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(331) SWITCH[1]	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(332) SWITCH[1]	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(333) SWITCH[1]	Stainless Steel	Treated Water Other	Loss of Material	Closed-Cycle (Component) Cooling Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(334) TANK	Carbon/Low Alloy Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(335) TANK	Carbon/Low Alloy Steel	Buried	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(336) TANK	Carbon/Low Alloy Steel	Treated Water Other (Stagnant)	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(337) TANK	Carbon/Low Alloy Steel	Treated Water Other (Stagnant)	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(338) TANK	Carbon/Low Alloy Steel	Treated Water Other (Stagnant)	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(339) TANK	Cast Iron	Indoor (No Air Conditioning)	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(340) TANK	Copper Alloy (Zn < 15%)	Air and Gas (Wetted) <140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(341) TANK	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(342) TANK	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(343) TANK	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(344) TANK	Stainless Steel	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(345) TANK	Stainless Steel	Raw Water Drainage	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(346) TANK	Stainless Steel	Treated Water Borated <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(347) TANK	Stainless Steel	Treated Water Borated <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(348) TANK	Stainless Steel	Treated Water Borated >140	Cracking due to SCC	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(349) TANK	Stainless Steel	Treated Water Borated >140	Cracking due to SCC	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(350) TANK	Stainless Steel	Treated Water Borated >140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(351) TANK	Stainless Steel	Treated Water Borated >140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(352) TANK	Stainless Steel	Treated Water Other	Loss of Material	Closed-Cycle (Component) Cooling Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(353) TEMPERATURE ELEMENT[1]	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(354) TEMPERATURE ELEMENT[1]	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(355) TEMPERATURE ELEMENT[1]	Stainless Steel	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(356) TEMPERATURE ELEMENT[1]	Stainless Steel	Treated Water Borated <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(357) TEMPERATURE ELEMENT[1]	Stainless Steel	Treated Water Borated <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(358) TEMPERATURE ELEMENT[1]	Stainless Steel	Treated Water Borated >140	Cracking due to SCC	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(359) TEMPERATURE ELEMENT[1]	Stainless Steel	Treated Water Borated >140	Cracking due to SCC	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(360) TEMPERATURE ELEMENT[1]	Stainless Steel	Treated Water Borated >140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(361) TEMPERATURE ELEMENT[1]	Stainless Steel	Treated Water Borated >140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(362) TEMPERATURE ELEMENT[1]	Stainless Steel	Treated Water Other	Loss of Material	Closed-Cycle (Component) Cooling Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(363) TEMPERATURE ELEMENT[1]	Stainless Steel	Treated Water Other (Stagnant)	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(364) TEMPERATURE ELEMENT[1]	Stainless Steel	Treated Water Other (Stagnant)	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(365) TEMPERATURE ELEMENT[1]	Stainless Steel	Treated Water Primary <140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(366) TEMPERATURE ELEMENT[1]	Stainless Steel	Treated Water Primary <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(367) TEMPERATURE ELEMENT[1]	Stainless Steel	Treated Water Primary, 140<T<480	Cracking due to SCC	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(368) TEMPERATURE ELEMENT[1]	Stainless Steel	Treated Water Primary, 140<T<480	Cracking due to SCC	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(369) TEMPERATURE ELEMENT[1]	Stainless Steel	Treated Water Primary, 140<T<480	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(370) TEMPERATURE ELEMENT[1]	Stainless Steel	Treated Water Primary, 140<T<480	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(371) TRANSMITTER[1]	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(372) TRANSMITTER[1]	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(373) TRANSMITTER[1]	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(374) TRANSMITTER[1]	Stainless Steel	Treated Water Borated <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(375) TRANSMITTER[1]	Stainless Steel	Treated Water Borated <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(376) TRANSMITTER[1]	Stainless Steel	Treated Water Borated >140	Cracking due to SCC	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(377) TRANSMITTER[1]	Stainless Steel	Treated Water Borated >140	Cracking due to SCC	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(378) TRANSMITTER[1]	Stainless Steel	Treated Water Borated >140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(379) TRANSMITTER[1]	Stainless Steel	Treated Water Borated >140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(380) TRANSMITTER[1]	Stainless Steel	Treated Water Other	Loss of Material	Closed-Cycle (Component) Cooling Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(381) TRANSMITTER[1]	Stainless Steel	Treated Water Primary <140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(382) TRANSMITTER[1]	Stainless Steel	Treated Water Primary <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(383) TRAP HOUSING	Carbon/Low Alloy Steel	Treated Water Secondary >120	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(384) VALVE BODY	Aluminum	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(385) VALVE BODY	Aluminum	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(386) VALVE BODY	Carbon/Low Alloy Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(387) VALVE BODY	Carbon/Low Alloy Steel	Air and Gas (Wetted) <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(388) VALVE BODY	Carbon/Low Alloy Steel	Treated Water Other (Stagnant)	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(389) VALVE BODY	Carbon/Low Alloy Steel	Treated Water Other (Stagnant)	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(390) VALVE BODY	Carbon/Low Alloy Steel	Treated Water Other (Stagnant)	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(391) VALVE BODY	Cast Austenitic Stainless Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(392) VALVE BODY	Cast Austenitic Stainless Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(393) VALVE BODY	Cast Austenitic Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(394) VALVE BODY	Cast Austenitic Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(395) VALVE BODY	Cast Austenitic Stainless Steel	Oil and Fuel Oil	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(396) VALVE BODY	Cast Austenitic Stainless Steel	Raw Water	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(397) VALVE BODY	Cast Austenitic Stainless Steel	Raw Water Drainage	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(398) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Borated <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(399) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Borated <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(400) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Borated >140	Cracking due to SCC	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(401) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Borated >140	Cracking due to SCC	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(402) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Borated >140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(403) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Borated >140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(404) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Other	Loss of Material	Closed-Cycle (Component) Cooling Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(405) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Other (Stagnant)	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(406) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Other (Stagnant)	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(407) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Primary <140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(408) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Primary <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(409) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Primary, 140<T<480	Cracking due to SCC	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(410) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Primary, 140<T<480	Cracking due to SCC	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(411) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Primary, 140<T<480	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(412) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Primary, 140<T<480	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(413) VALVE BODY	Cast Iron	Air and Gas (Wetted) <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(414) VALVE BODY	Cast Iron	Air and Gas (Wetted) <140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(415) VALVE BODY	Cast Iron	Buried	Loss of Material	Fire Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(416) VALVE BODY	Cast Iron	Containment	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(417) VALVE BODY	Cast Iron	Indoor (No Air Conditioning)	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(418) VALVE BODY	Cast Iron	Oil and Fuel Oil	Loss of Material	Fuel Oil Chemistry Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(419) VALVE BODY	Cast Iron	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(420) VALVE BODY	Cast Iron	Outdoor	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(421) VALVE BODY	Cast Iron	Raw Water	Loss of Material	Open-Cycle Cooling (Service) Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(422) VALVE BODY	Cast Iron	Raw Water (Stagnant)	Loss of Material	Open-Cycle Cooling (Service) Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(423) VALVE BODY	Cast Iron	Raw Water (Submerged)	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(424) VALVE BODY	Cast Iron	Treated Water Other (Stagnant)	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(425) VALVE BODY	Copper Alloy (Zn < 15%)	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(426) VALVE BODY	Copper Alloy (Zn < 15%)	Air and Gas (Wetted) <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(427) VALVE BODY	Copper Alloy (Zn < 15%)	Air and Gas (Wetted) <140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(428) VALVE BODY	Copper Alloy (Zn < 15%)	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(429) VALVE BODY	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(430) VALVE BODY	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(431) VALVE BODY	Copper Alloy (Zn < 15%)	Oil and Fuel Oil	Loss of Material	Fuel Oil Chemistry Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(432) VALVE BODY	Copper Alloy (Zn < 15%)	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(433) VALVE BODY	Copper Alloy (Zn < 15%)	Outdoor	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(434) VALVE BODY	Copper Alloy (Zn < 15%)	Raw Water Drainage	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(435) VALVE BODY	Copper Alloy (Zn < 15%)	Treated Water Borated <140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(436) VALVE BODY	Copper Alloy (Zn < 15%)	Treated Water Other	Loss of Material	Closed-Cycle (Component) Cooling Water System Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(437) VALVE BODY	Copper Alloy (Zn < 15%)	Treated Water Other	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(438) VALVE BODY	Copper Alloy (Zn < 15%)	Treated Water Other (Stagnant)	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(439) VALVE BODY	Copper Alloy (Zn < 15%)	Treated Water Other (Stagnant)	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(440) VALVE BODY	Copper Alloy (Zn < 15%)	Treated Water Secondary >120	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(441) VALVE BODY	Plastic	Indoor (Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(442) VALVE BODY	Plastic	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(443) VALVE BODY	Plastic	Raw Water Drainage	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(444) VALVE BODY	Stainless Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(445) VALVE BODY	Stainless Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(446) VALVE BODY	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(447) VALVE BODY	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(448) VALVE BODY	Stainless Steel	Oil and Fuel Oil	Loss of Material	Fuel Oil Chemistry Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(449) VALVE BODY	Stainless Steel	Oil and Fuel Oil	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(450) VALVE BODY	Stainless Steel	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(451) VALVE BODY	Stainless Steel	Raw Water Drainage	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(452) VALVE BODY	Stainless Steel	Treated Water Borated <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(453) VALVE BODY	Stainless Steel	Treated Water Borated <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(454) VALVE BODY	Stainless Steel	Treated Water Borated >140	Cracking due to SCC	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(455) VALVE BODY	Stainless Steel	Treated Water Borated >140	Cracking due to SCC	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(456) VALVE BODY	Stainless Steel	Treated Water Borated >140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(457) VALVE BODY	Stainless Steel	Treated Water Borated >140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(458) VALVE BODY	Stainless Steel	Treated Water Other	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(459) VALVE BODY	Stainless Steel	Treated Water Other (Stagnant)	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(460) VALVE BODY	Stainless Steel	Treated Water Other (Stagnant)	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(461) VALVE BODY	Stainless Steel	Treated Water Other (Stagnant)	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(462) VALVE BODY	Stainless Steel	Treated Water Primary <140	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(463) VALVE BODY	Stainless Steel	Treated Water Primary <140	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(464) VALVE BODY	Stainless Steel	Treated Water Primary, 140<T<480	Cracking due to SCC	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(465) VALVE BODY	Stainless Steel	Treated Water Primary, 140<T<480	Cracking due to SCC	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(466) VALVE BODY	Stainless Steel	Treated Water Primary, 140<T<480	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(467) VALVE BODY	Stainless Steel	Treated Water Primary, 140<T<480	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(468) VENTILATION DUCTWORK	Galvanized Carbon Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(469) VENTILATION DUCTWORK	Galvanized Carbon Steel	Containment	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.

Table 3.4-2 Auxiliary Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(470) VENTILATION DUCTWORK	Galvanized Carbon Steel	Indoor (Air Conditioning)	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(471) VENTILATION DUCTWORK	Galvanized Carbon Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.
(472) VENTILATION DUCTWORK	Galvanized Carbon Steel	Outdoor	No Aging Effects	No Aging Management Program Required	Not consistent with NUREG-1801. According to site-specific review of standard industry guidance for aging evaluation of mechanical systems and components, galvanized carbon steel exposed to ventilation air (T<140°F) would be expected to exhibit minimal deterioration of the zinc coating. Therefore no aging effects are applicable and no aging management program is required.

1. Selected instruments were conservatively included within the scope of License Renewal. Consideration was given to the consequences of an instrument housing pressure boundary failure. Where an instrument was unisolable from a pressure source and is of sufficient size that a system function would be degraded should the pressure boundary fail, that instrument is included for License Renewal review.

2. Material prefixes with HX are used to identify heat exchanger materials which perform a heat transfer intended function in addition to the typical material usage function of pressure boundary.

3. HVAC equipment packages include the pressure boundary attributes associated with the package and sub-components such as filter housings, internal damper housings, and fan housings. Both the HVAC package units and their associated sub-components are uniquely identified on plant drawings.

3.5 Aging Management of Steam and Power Conversion Systems

The results of the aging management review of the Steam and Power Conversion Systems components are provided in this section and summarized in Tables 3.5-1 and 3.5-2. Table 3.5-1 shows the aging management of system components evaluated in NUREG-1801 that are relied on for license renewal of the Steam and Power Conversion Systems components at Ginna. Included in the table is a discussion column. The discussion column will provide a conclusion indicating if the aging management evaluation results are consistent with NUREG-1801 along with any clarifications or explanations required to support the stated conclusion if that conclusion is different than those of the NUREG. For a determination to be made that a table line item is "Consistent with NUREG-1801" several criteria must be met. First the plant specific component is reviewed against the GALL to ensure that the component, materials of construction and internal or external service environment are comparable to those described in a particular GALL item. Second, for those that are comparable, the results of the plant aging management review- aging effect evaluation are compared to the aging effects/mechanisms in the GALL. Finally, the programs credited in the GALL for managing those aging effects are compared to the programs invoked in the plant evaluation. If, using good engineering judgment, it could be reasonably concluded that the plant evaluation is in agreement with the GALL evaluation a line item was considered consistent with NUREG-1801. There are cases where components and component material/environment combinations and aging effects are common between a NUREG-1801 line item and the plant evaluation but the aging management program selections differ. In those cases the discussion column will indicate the plant aging management program selection but no conclusion will be made that the line item is consistent with the GALL. Table 3.5-2 contains the Steam and Power Conversion Systems components aging management review results that are not addressed in NUREG-1801. A plant component is considered not addressed by the NUREG if the component type is not evaluated in the GALL or has a different material of construction or operating environment than evaluated in the GALL. This table includes the component types, materials, environments, aging effects requiring management, the programs and activities for managing aging, and a discussion column. To avoid confusion, no attempt was made to interrelate material/environment/aging effects from one NUREG-1801 chapter to another. Note that these tables only include those components, materials and environments that are applicable to a PWR.

Materials

The materials of construction of a component have a major influence on the evaluation of aging effects applicable to the component. Sources of information used to identify materials of construction include original equipment specifications, vendor technical manuals and drawings, fabrication drawings, piping line specifications, modification design records and field walkdowns/verifications. The tables below account for the materials of construction for the components requiring an aging management review. Since similar materials are susceptible to the same aging effects/mechanisms, the tables itemize the component types (i.e., groupings) while factoring in the materials of construction.

Environment

As previously described, the environment(s) to which components are exposed are critical in the determination of potential aging mechanisms and effects. A review of plant design documentation was performed to quantify the environmental conditions to which Ginna Station equipment is exposed. This review identified that some equipment is exposed to a variety of environments. This can include normal operating conditions and post accident conditions. Since aging mechanisms and effects will be primarily driven by the environmental conditions to which equipment is exposed on a daily basis, under normal operating conditions, these conditions will differ from the design parameters which are established based upon the worst case scenario (e.g., LOCA conditions). Ginna Station equipment environments may be categorized into basic external and internal environments detailed in Section 3.1.2.

Aging Effects Requiring Management

After the components requiring aging management review were identified and grouped by materials of construction and environment, a review of industry and plant-specific operating experience was performed. The purpose of this review was to assure that all applicable aging effects were identified, and to evaluate the effectiveness of existing aging management programs.

This experience review was performed utilizing various industry and plant-specific programs and databases. Industry operating experience sources included NRC Generic Publications (including Information Notices, Circulars, Bulletins, and Generic Letters), INPO Significant Operating Event Reports (SOER), EPRI Technical Reports, and other information sources, such as the B&W Owners Group Non-Class 1 Mechanical Tools Implementation document, Westinghouse Generic Technical Reports (GTRs), and the Generic Aging Lessons Learned (GALL) report.

Plant specific operating experience sources included Semi-annual and Annual Reports to AEC/NRC, Abnormal Occurrence and Licensee Event Reports (LERs), Non-Conformance Reports (NCRs), Corrective Action Reports (CARs), Refueling, Inspection and Overhaul Reports (RIOs), Inservice Inspection (ISI) Reports, Identified Deficiency Reports (IDRs), and ACTION Reports (ARs) from 1969 to the present. Information from these sources was compiled in various databases. Based upon the material of construction, the applicable environments, and operating experience the potential aging effects requiring management for each of the components was identified as documented in the tables below.

Time-Limited Aging Analysis

In addition to those identified in NUREG-1801, any additional time-limited aging analyses (TLAA) identified as appropriate to the system are identified in Section 4.0.

Conclusion

The programs and activities selected to manage the aging effects of the Steam and Power Conversion Systems are identified in Table 3.5-1 and Table 3.5-2. A description of these aging management activities is provided in Appendix B, along with the demonstration that the identified aging effects will be managed for the period of extended operation. Therefore, based on the demonstrations provided in Appendix B, the effects of aging associated with the system components will be adequately managed so that there is reasonable assurance that the intended function(s) will be maintained consistent with the current licensing basis during the period of extended operation.

Table 3.5-1 Steam and Power Conversion Systems - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(1) Piping and fittings in main feedwater line, steam line and AFW piping (PWR only)	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	Consistent with NUREG-1801. Cumulative Fatigue Damage is addressed as a TLAA in Section 4.3.
(2) Piping and fittings, valve bodies and bonnets, pump casings, tanks, tubes, tubesheets, channel head and shell (except main steam system)	Loss of material due to general (carbon steel only), pitting, and crevice corrosion	Water chemistry and one-time inspection	Yes, detection of aging effects is to be further evaluated	The Periodic Surveillance and Preventive Maintenance Program will be used to verify the effectiveness of the Water Chemistry Control Program.
(3) Auxiliary feedwater (AFW) piping	Loss of material due to general, pitting, and crevice corrosion, MIC, and biofouling	Plant specific	Yes, plant specific	The combination of components, materials and environments identified in Item VIII.G.1-d are evaluated in the Service Water System. The Service Water System components are reviewed under NUREG-1801 Chapter VII (Auxiliary Systems), Section C1.
(4) Oil coolers in AFW system (lubricating oil side possibly contaminated with water)	Loss of material due to general (carbon steel only), pitting, and crevice corrosion and MIC	Plant specific	Yes, plant specific	Consistent with NUREG-1801. The Periodic Surveillance and Preventive Maintenance Program is credited with managing all applicable aging effects. Other component types such as accumulators, filter housings, orifices, piping, speed increasers, tanks, and valve bodies have been included in this line item at Ginna Station. Although these specific component types were not included in the NUREG section, the material, environment, aging effect/mechanism, and aging management program are consistent.
(5) External surface of carbon steel components	Loss of material due to general corrosion	Plant specific	Yes, plant specific	Consistent with NUREG-1801. The Systems Monitoring Program is credited with managing the aging effect “loss of material due to general corrosion” on the external surfaces of carbon steel components.

Table 3.5-1 Steam and Power Conversion Systems - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(6) Carbon steel piping and valve bodies	Wall thinning due to flow-accelerated corrosion	Flow-accelerated corrosion	No	Consistent with NUREG-1801. The Flow-Accelerated Corrosion Program implements the guidelines of EPRI NSAC-202L and is credited with managing the aging effect “wall thinning due to flow-accelerated corrosion.”
(7) Carbon steel piping and valve bodies in main steam system	Loss of material due to pitting and crevice corrosion	Water chemistry	No	Consistent with NUREG-1801. The Water Chemistry Control Program is credited with managing the aging effects “loss of material due to pitting and crevice corrosion” for components within the Main Steam System.
(8) Closure bolting in high-pressure or high-temperature systems	Loss of material due to general corrosion; crack initiation and growth due to cyclic loading and/or SCC	Bolting integrity	No	Consistent with NUREG-1801. The Bolting Integrity Program is credited for managing the aging effects “loss of material due to general corrosion; crack initiation and growth due to cyclic loading and/or SCC.” There are no bolts with a specified minimum yield strength > 150 ksi in the Steam and Power Conversion Systems. Therefore, SCC is not an applicable aging effect/mechanism.
(9) Heat exchangers and coolers/ condensers serviced by open-cycle cooling water	Loss of material due to general (carbon steel only), pitting, and crevice corrosion, MIC, and biofouling; buildup of deposit due to biofouling	Open-cycle cooling water system	No	The Periodic Surveillance and Preventive Maintenance Program will be credited with managing the applicable aging effects in lieu of the Open-Cycle Cooling (Service) Water System Program.
(10) Heat exchangers and coolers/ condensers serviced by closed-cycle cooling water	Loss of material due to general (carbon steel only), pitting, and crevice corrosion	Closed-cycle cooling water system	No	There are no heat exchangers in the Steam and Power Conversion Systems that are serviced by closed-cycle cooling water at Ginna Station.

Table 3.5-1 Steam and Power Conversion Systems - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(11) External surface of above ground condensate storage tank	Loss of material due to general (carbon steel only), pitting, and crevice corrosion	Above ground carbon steel tanks	No	The above ground condensate storage tank at Ginna Station is not in scope to License Renewal and therefore not subject to an aging management review.
(12) External surface of buried condensate storage tank and AFW piping	Loss of material due to general, pitting, and crevice corrosion and MIC	Buried piping and tanks surveillance or Buried piping and tanks inspection	No Yes, detection of aging effects and operating experience are to be further evaluated	There are no buried tanks or piping in the Steam and Power Conversion Systems at Ginna Station.
(13) External surface of carbon steel components	Loss of material due to boric acid corrosion	Boric acid corrosion	No	Consistent with NURGE 1801. The Boric Acid Corrosion Program is credited with managing the aging effect "loss of material due to boric acid corrosion."

Table 3.5-2 Steam and Power Conversion Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(1) CONTROLLER[1]	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(2) CONTROLLER[1]	Stainless Steel	Treated Water Secondary (Stagnant) <120	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(3) CONTROLLER[1]	Stainless Steel	Treated Water Secondary (Stagnant) <120	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(4) CONVERTER[1]	Stainless Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(5) CONVERTER[1]	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(6) COOLER	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(7) FASTENERS (BOLTING)	Carbon/Low Alloy Steel	Indoor (No Air Conditioning)	Cracking due to SCC	Bolting Integrity Program	There are no bolts with a specified minimum yield strength > 150 ksi in this system. Therefore, SCC is not an applicable aging effect/mechanism.

Table 3.5-2 Steam and Power Conversion Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(8) FASTENERS (BOLTING)	Carbon/Low Alloy Steel	Indoor (No Air Conditioning)	Loss of Preload due to Stress Relaxation	Bolting Integrity Program	Material and environment grouping are included in NUREG-1801. Aging effect of loss of preload due to stress relaxation is applicable, but is not included in Chapter V - Section E, Chapter VII - Section I, or Chapter VIII - Section H of the NUREG.
(9) FASTENERS (BOLTING)	Stainless Steel	Borated Water Leaks	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(10) FILTER HOUSING	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(11) FLOW ELEMENT	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(12) FLOW ELEMENT	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(13) FLOW ELEMENT	Stainless Steel	Treated Water Secondary (Stagnant) <120	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(14) FLOW ELEMENT	Stainless Steel	Treated Water Secondary (Stagnant) <120	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.5-2 Steam and Power Conversion Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(15) FLOW ELEMENT	Stainless Steel	Treated Water Secondary >120	Cracking due to SCC	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(16) FLOW ELEMENT	Stainless Steel	Treated Water Secondary >120	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(17) HEAT EXCHANGER	Cast Iron	Indoor (No Air Conditioning)	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(18) HEAT EXCHANGER	HX-Cast Iron[2]	Oil and Fuel Oil	Loss of Heat Transfer	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(19) HEAT EXCHANGER	Cast Iron	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.5-2 Steam and Power Conversion Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(20) HEAT EXCHANGER	HX-Cast Iron[2]	Raw Water	Loss of Heat Transfer	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(21) HEAT EXCHANGER	Cast Iron	Raw Water	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(22) LEVEL GLASS	Aluminum	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(23) LEVEL GLASS	Aluminum	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(24) OPERATOR	Carbon/Low Alloy Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(25) ORIFICE	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.5-2 Steam and Power Conversion Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(26) ORIFICE	Stainless Steel	Treated Water Secondary (Stagnant) <120	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(27) ORIFICE	Stainless Steel	Treated Water Secondary (Stagnant) <120	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(28) PIPE	Carbon/Low Alloy Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(29) PIPE	Carbon/Low Alloy Steel	Air and Gas (Wetted) <140	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(30) PIPE	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.5-2 Steam and Power Conversion Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(31) PIPE	Copper Alloy (Zn < 15%)	Treated Water Secondary (Stagnant) <120	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(32) PIPE	Copper Alloy (Zn < 15%)	Treated Water Secondary (Stagnant) <120	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(33) PIPE	Stainless Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(34) PIPE	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(35) PIPE	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(36) PIPE	Stainless Steel	Treated Water Secondary (Stagnant) <120	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.

Table 3.5-2 Steam and Power Conversion Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(37) PIPE	Stainless Steel	Treated Water Secondary (Stagnant) <120	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(38) PIPE	Stainless Steel	Treated Water Secondary >120	Cracking due to SCC	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(39) PIPE	Stainless Steel	Treated Water Secondary >120	Cracking due to SCC	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(40) PIPE	Stainless Steel	Treated Water Secondary >120	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.

Table 3.5-2 Steam and Power Conversion Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(41) PIPE	Stainless Steel	Treated Water Secondary >120	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(42) POSITIONER[1]	Stainless Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(43) POSITIONER[1]	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(44) PRESSURE RELAY[1]	Aluminum	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(45) PRESSURE RELAY[1]	Aluminum	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(46) PUMP CASING	Cast Iron	Indoor (No Air Conditioning)	Loss of Material	Systems Monitoring Program	Material and environment grouping are not included in NUREG-1801.
(47) PUMP CASING	Cast Iron	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(48) SCREEN	Stainless Steel	Air and Gas (Wetted) <140	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.5-2 Steam and Power Conversion Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(49) SCREEN	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(50) TANK	Neoprene	Indoor (No Air Conditioning)	Change in Material Properties and Cracking	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(51) TEMPERATURE ELEMENT[1]	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(52) TEMPERATURE ELEMENT[1]	Stainless Steel	Treated Water Secondary >120	Cracking due to SCC	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(53) TEMPERATURE ELEMENT[1]	Stainless Steel	Treated Water Secondary >120	Cracking due to SCC	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.5-2 Steam and Power Conversion Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(54) TEMPERATURE ELEMENT[1]	Stainless Steel	Treated Water Secondary >120	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(55) TEMPERATURE ELEMENT[1]	Stainless Steel	Treated Water Secondary >120	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(56) VALVE BODY	Aluminum	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(57) VALVE BODY	Aluminum	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(58) VALVE BODY	Carbon/Low Alloy Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(59) VALVE BODY	Cast Austenitic Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.

Table 3.5-2 Steam and Power Conversion Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(60) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Secondary (Stagnant) <120	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(61) VALVE BODY	Cast Austenitic Stainless Steel	Treated Water Secondary (Stagnant) <120	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(62) VALVE BODY	Copper Alloy (Zn < 15%)	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(63) VALVE BODY	Copper Alloy (Zn < 15%)	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(64) VALVE BODY	Copper Alloy (Zn < 15%)	Oil and Fuel Oil	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

Table 3.5-2 Steam and Power Conversion Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(65) VALVE BODY	Copper Alloy (Zn < 15%)	Treated Water Secondary (Stagnant) <120	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(66) VALVE BODY	Copper Alloy (Zn < 15%)	Treated Water Secondary (Stagnant) <120	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(67) VALVE BODY	Stainless Steel	Air and Gas	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(68) VALVE BODY	Stainless Steel	Containment	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(69) VALVE BODY	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	No Aging Management Program Required	Material and environment grouping are not included in NUREG-1801.
(70) VALVE BODY	Stainless Steel	Treated Water Secondary (Stagnant) <120	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.

Table 3.5-2 Steam and Power Conversion Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(71) VALVE BODY	Stainless Steel	Treated Water Secondary (Stagnant) <120	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(72) VALVE BODY	Stainless Steel	Treated Water Secondary >120	Cracking due to SCC	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.
(73) VALVE BODY	Stainless Steel	Treated Water Secondary >120	Cracking due to SCC	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.
(74) VALVE BODY	Stainless Steel	Treated Water Secondary >120	Loss of Material	One-Time Inspection Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation. One-time inspections are used to verify the effectiveness of the Water Chemistry Control Program.

Table 3.5-2 Steam and Power Conversion Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(75) VALVE BODY	Stainless Steel	Treated Water Secondary >120	Loss of Material	Water Chemistry Control Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

1. Selected instruments were conservatively included within the scope of License Renewal. Consideration was given to the consequences of an instrument housing pressure boundary failure. Where an instrument was unisolable from a pressure source and is of sufficient size that a system function would be degraded should the pressure boundary fail, that instrument is included for License Renewal review.
2. Material prefixes with HX are used to identify heat exchanger materials which perform a heat transfer intended function in addition to the typical material usage function of pressure boundary.

3.6 Aging Management of Structures and Component Supports

The results of the aging management review of the Structures and Component Supports are provided in this section and summarized in Table 3.6-1 and Table 3.6-2. Table 3.6-1 shows the aging management of system components evaluated in NUREG-1801 that are relied on for license renewal of the Structures and Component Supports at Ginna. Included in the table is a discussion column. The discussion column will provide a conclusion indicating if the aging management evaluation results are consistent with NUREG-1801 along with any clarifications or explanations required to support the stated conclusion if that conclusion is different than those of the NUREG. For a determination to be made that a table line item is "Consistent with NUREG-1801" several criteria must be met. First the plant specific component is reviewed against the GALL to ensure that the component, materials of construction and internal or external service environment are comparable to those described in a particular GALL item. Second, for those that are comparable, the results of the plant aging management review- aging effect evaluation are compared to the aging effects/mechanisms in the GALL. Finally, the programs credited in the GALL for managing those aging effects are compared to the programs invoked in the plant evaluation. If, using good engineering judgment, it could be reasonably concluded that the plant evaluation is in agreement with the GALL evaluation a line item was considered consistent with NUREG-1801. There are cases where components and component material/environment combinations and aging effects are common between a NUREG-1801 line item and the plant evaluation but the aging management program selections differ. In those cases the discussion column will indicate the plant aging management program selection but no conclusion will be made that the line item is consistent with the GALL. Table 3.6-2 contains the Structures and Component Supports aging management review results that are not addressed in NUREG-1801. A plant component is considered not addressed by the NUREG if the component type is not evaluated in the GALL or has a different material of construction or operating environment than evaluated in the GALL. This table includes the component types, materials, environments, aging effects requiring management, the programs and activities for managing aging, and a discussion column. To avoid confusion, no attempt was made to interrelate material/environment/aging effects from one NUREG-1801 chapter to another. Note that these tables only include those components, materials and environments that are applicable to a PWR.

Materials

The materials of construction of a component have a major influence on the evaluation of aging effects applicable to the component. Sources of information used to identify materials of construction include original equipment specifications, vendor technical manuals and drawings, fabrication drawings, piping line specifications, modification design records and field walkdowns/verifications. The tables below account for the materials of construction for the components requiring an aging management review. Since similar materials are susceptible to the same aging effects/mechanisms, the tables itemize the component types (i.e., groupings) while factoring in the materials of construction.

Environment

As previously described, the environment(s) to which components are exposed are critical in the determination of potential aging mechanisms and effects. A review of plant design documentation was performed to quantify the environmental conditions to which Ginna Station equipment is exposed. This review identified that some equipment is exposed to a variety of environments. This can include normal operating conditions and post accident conditions. Since aging mechanisms and effects will be primarily driven by the environmental conditions to which equipment is exposed on a daily basis, under normal operating conditions, these conditions will differ from the design parameters which are established based upon the worst case scenario (e.g., LOCA conditions). Ginna Station equipment environments may be categorized into basic external and internal environments detailed in Section 3.1.2.

Aging Effects Requiring Management

After the components requiring aging management review were identified and grouped by materials of construction and environment, a review of industry and plant-specific operating experience was performed. The purpose of this review was to assure that all applicable aging effects were identified, and to evaluate the effectiveness of existing aging management programs.

This experience review was performed utilizing various industry and plant-specific programs and databases. Industry operating experience sources included NRC Generic Publications (including Information Notices, Circulars, Bulletins, and Generic Letters), INPO Significant Operating Event Reports (SOER), EPRI Technical Reports, and other information sources, such as the B&W Owners Group Non-Class 1 Mechanical Tools Implementation document, Westinghouse Generic Technical Reports (GTRs), and the Generic Aging Lessons Learned (GALL) report.

Plant specific operating experience sources included Semi-annual and Annual Reports to AEC/NRC, Abnormal Occurrence and Licensee Event Reports (LERs), Non-Conformance Reports (NCRs), Corrective Action Reports (CARs), Refueling, Inspection and Overhaul Reports (RIOs), Inservice Inspection (ISI) Reports, Identified Deficiency Reports (IDRs), and ACTION Reports (ARs) from 1969 to the present. Information from these sources was compiled in various databases. Based upon the material of construction, the applicable environments, and operating experience the potential aging effects requiring management for each of the components was identified as documented in the tables below.

Time-Limited Aging Analysis

In addition to those identified in NUREG-1801, any additional time-limited aging analyses (TLAA) identified as appropriate to the system are identified in Section 4.0.

Confirmation of Topical Report Applicability

Containment Structures

The Westinghouse Owners' Group Life Cycle Management & License Renewal Program has prepared topical report, WCAP-14756-A, Aging Management Evaluation for Pressurized Water Reactor Containment Structure (Reference 1), which has been utilized in the aging management review of the Ginna Containment Structures. Therefore, reconciliation of the final SER for WCAP-14756-A applicant action items is provided in Table 3.6.0-1. A description of aging management activities is provided in Appendix B, along with the demonstration that the identified aging effects will be managed for the period of extended operation.

Therefore, based on the demonstrations provided in Appendix B, the effects of aging associated with the Containment Structures will be adequately managed so that there is reasonable assurance that the intended function(s) will be maintained consistent with the current licensing basis during the period of extended operation.

Reactor Coolant System Supports

The Westinghouse Owners' Group Life Cycle Management & License Renewal Program has prepared topical report, WCAP-14422, Rev. 2-A, License Renewal Evaluation: Aging Management for Reactor Coolant System Supports (Reference 2), which has been utilized in the aging management review of the Ginna RC System Supports components. The scope of the RC System supports components described in the topical report bounds the Ginna RC System Supports components.

A reconciliation of the final SER for WCAP-14422 Rev.2-A applicant action items is provided in Table 3.6.0-2. A description of aging management activities is provided in Appendix B, along with the demonstration that the identified aging effects will be managed for the period of extended operation.

Therefore, based on the demonstrations provided in Appendix B, the effects of aging associated with the RC System Supports components will be adequately managed so that there is reasonable assurance that the intended function(s) will be maintained consistent with the current licensing basis during the period of extended operation.

Conclusion

The programs and activities selected to manage the aging effects of the Structures and Component Supports are identified in Table 3.6-1 and Table 3.6-2. The results of the applicant action item reviews are also contained in these tables, but in the SRP format.

A description of these aging management activities is provided in Appendix B, along with the demonstration that the identified aging effects will be managed for the period of extended operation.

Therefore, based on the demonstrations provided in Appendix B, the effects of aging associated with the Structures and Component Supports components will be adequately managed so that there is reasonable assurance that the intended function(s) will be maintained consistent with the current licensing basis during the period of extended operation.

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(1) The license renewal applicant will (i) verify that its plant is bounded by the GTR, (ii) commit to implement programs described as necessary in the GTR to manage the effects of aging during the period of extended operation, and (iii) verify that the programs committed to are conducted in accordance with appropriate regulatory controls (e.g. 10 CFR Part 50, Appendix B). Further, the renewal applicant will identify any deviations from the aging management programs which this GTR describes as necessary to manage the effects of aging during the period of extended operation or to maintain the functionality of the containment structure, and deviations from other information presented in the GTR (e.g., materials of construction). The renewal applicant will evaluate any such deviations in accordance with 10 CFR 54.21(a)(3) and (c)(1) on a plant-specific basis. The following functions, which are specific to containment structures and are understood to be covered by the various intended functions, should be addressed explicitly in the license renewal application: (1) providing structural or functional support of safety-related systems, structures, and components following a design basis accident (DBA); (2) serving as an external missile barrier consistent with the design and licensing basis; and (3) providing passive heat sinks during a DBA or station blackout in addition to the spray system.	The design, configuration, materials of construction, and normal operating service environment of the Ginna Station Containment structure are bounded by the GTR. As part of license renewal, Ginna Station aging management programs will implement the programs described in the GTR as necessary to manage the effects of aging during the period of extended operation. Further, the programs committed to will be conducted in accordance with appropriate regulatory controls (e.g., 10 CFR Part 50, Appendix B). The GTR evaluated aging of the pressurized water reactor containment structure to ensure that the intended functions will be maintained during the extended period of operation. Four intended functions performed by the PWR containment structure are identified in the GTR. • Ensuring the integrity of the reactor coolant pressure boundary (RCPB) • Ensuring the capability to contain a shut down of the reactor and maintain it in a safe shutdown condition • Ensuring the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the 10 CFR 100 guidelines • Ensuring compliance with the USNRC regulations for environmental qualification (10 CFR 50.49). Three additional intended functions are identified in Action Item Number 1. These additional intended functions are indirectly addressed in the GTR aging management

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(1) (continued)	evaluation and aging management options since they are a subset of the four given in the GTR. Specifically, each of the additional intended functions are subsets of the following: • The additional intended function (1), providing structural or functional support of safety-related systems, structures, and components following a design basis accident (DBA), is a subset of the GTR intended function 3. • The additional intended function (2), serving as an external missile barrier consistent with the design and licensing basis, is a subset of GTR intended functions 3 and 4. • The additional intended function (3), providing passive heat sinks during a DBA or station blackout in addition to the spray system, is a subset of GTR intended functions 2 and 3. Therefore, there is no need to further address the added functions in the license renewal application.
(2) A summary description of the programs and activities for managing the effects of aging and the evaluation of TLAAs is to be provided in the license renewal FSAR supplement, in accordance with 10 CFR 54.21(d).	A summary description of aging management programs credited for managing the effects of aging and evaluation of the TLAA's for the Ginna Station Containment Structure is provided in Appendix A of the LRA.
(3) Individual plant applicants will need to provide a comprehensive list of structures and components subject to an aging management review and the methodology used to develop this list as part of their license renewal applications. Any components determined by the applicant to be subject to an aging management review for license renewal but not within the scope of the GTR are required to be addressed in the license renewal application.	A comprehensive list of the structures and components subject to aging management review was developed in accordance with the methodology described in Engineering Procedure EP-3-S-0713, “Scoping and Screening for License Renewal.” This list is available for on-site review.

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(4) Provide cross-section drawings for the containment structures, and detailed drawings of the sand pocket region and other plant-specific features, if applicable.	Drawings showing the cross-section of the Ginna Station Containment structure and other specific features, including the sand-box region, are available at the plant site for review.
(5) Provide legible drawings of equipment and penetration details as part of the description of the containment structure components	Drawings of equipment and penetration details of the Ginna Station Containment structure are available at the plant site for review.
(6) For prestressed concrete containments, indicate whether the tendon access gallery is included as a containment structure component subject to an aging management review. If it is, provide the details of the aging management review and the credited aging management program. If not, provide a technical basis for its exclusion, addressing the potential for degradation of the lower vertical tendon anchors resulting from the environmental conditions in the tendon access gallery.	The wall tendons in the Ginna Station Containment structure are permanently coupled at the base of the cylinder wall to rock anchors. The rock anchors consist of 90-wire tendons which were inserted into 6-in. dia. holes drilled 43 feet into base rock, then grouted and pre-stressed. The rock anchors were grouted up to the bearing plate for corrosion protection. The tendon anchorage at the rock-anchor coupling as well as the wall tendons are encased in a steel sheath which is filled with protective grease (paraffin-based mineral oil blended with a micro-crystalline wax). Therefore, there is no lower tendon access gallery at Ginna Station.

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(7) Discuss plant-specific operating experience relevant to age-related degradation of containment structure components and how this experience has been considered in the aging management review.	A thorough review of Ginna Station operating experience was conducted. Sources included Semi-annual and Annual Reports to AEC/NRC, Abnormal Occurrence and Licensee Event Reports (LERs), Non-Conformance Reports (NCRs), Corrective Action Reports (CARs), Refueling, Inspection and Overhaul Reports (RIOs), Inservice Inspection (ISI) Reports, Identified Deficiency Reports (IDRs), and ACTION Reports (ARs) from 1969 to the present. In addition, plant-specific response to any NRC generic communication was reviewed for applicability. The results of unique inspections of opportunity and NRC required ASME Section XI, Subsections IWE/IWL inspections were also reviewed. No additional or unique aging effects requiring management were identified from this review beyond those identified in Table 3.6-1 of the LRA.

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(8) For concrete containments, verify that the original plant design and construction specifications satisfy the criteria which are relied upon to exclude leaching of calcium hydroxide and reaction with aggregates as significant aging mechanisms. If these mechanisms are not excluded, describe the aging management program (AMP) which is credited to manage the aging effects associated with these aging mechanisms.	Leaching of Calcium Hydroxide According to NUREG-1801, leaching of calcium hydroxide from reinforced concrete becomes significant only if the concrete is exposed to flowing water. Even if reinforced concrete is exposed to flowing water, such leaching is not significant if the concrete is constructed to ensure that it is dense, well cured, has low permeability, and that cracking is well controlled. Cracking is controlled through proper arrangement and distribution of reinforcing bars. All of the above characteristics are assured if the concrete was constructed with the guidance of ACI 201.2R-77. The reinforced concrete of the Containment structure (base mat, ring beam and cylinder walls) is not exposed to flowing water. The only reinforced concrete that is exposed to flowing water is at the Screen House and Discharge Canal. In addition, concrete at Ginna Station was specified in accordance with the guidance of ACI 201.2R. Reaction with Aggregates Construction of concrete structures at Ginna Station was performed under three contracts: Gilbert specification, GC-3799, “Technical Specification for Structural Concrete”, Gilbert specification GC-3526, “General Construction Specification Service Building”, and Rochester Gas and Electric (RGE) Corporation specification, “Construction of Screen House, Heating Boiler Room, and Service Water Building Foundations, Walls, Floors to Elevation 253'-6”. Specification GC-3799 states that aggregate shall conform to ACI 301-66 and to the State of New York Department of Public Works Specification latest edition. ACI 301-66 states that aggregate shall conform to ASTM C33.

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(8) (continued)	At the time of these contracts, ASTM C33 required that aggregates be tested for potential reactivity in accordance with ASTM C227 and ASTM C295. Additionally, the New York State Department of Transportation requires aggregates to be tested in accordance with ASTM C227 and C295. Specification GC-3526 states that concrete and concrete work shall conform to ACI 318-63. ACI 318-63 requires aggregate to conform to ASTM C-33. ASTM C33 requires aggregate to be tested in accordance with ASTM C227 and C295. The RGE Corporation specification required aggregates to conform to ASTM C33. Since the aggregates were tested for potential reactivity in accordance with ASTM C227 and ASTM C295, cracking and expansion or concrete due to reaction with aggregates is not an aging effect requiring management at Ginna Station. These facts notwithstanding, the current mandated inspections performed in accordance with the requirements of 10 CFR 50.55a and ASME Code Section XI, Subsection IWL, Examination Category L-A, and the Structures Monitoring Program manage the concrete aging effects associated with leaching of calcium hydroxide and reaction with aggregates. Therefore, the potential aging effects associated with leaching of calcium hydroxide and reaction with aggregates will be adequately managed during the period of extended operation in the unlikely event that these mechanisms should become active.

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(9) For concrete containments, discuss whether local heating of containment concrete at the main steam and/or any other penetrations results in sustained concrete temperatures exceeding 200°F. If this condition exists, provide an aging management review and describe the credited aging management program.	Elevated temperature was evaluated as an aging mechanism for Containment structure concrete components. The temperature of the concrete around hot-piping penetrations such as main steam and feedwater is maintained at Ginna Station below the limits in ASME Section III, Div. 2, Subsection CC-3340, i.e., 200°F for long term periods during normal operation. These penetrations were designed with a forced air cooling system connected to cooling coils integrated with the penetration sleeves. The cooling coils are in the form of an embossing welded directly to the inner surface of the penetration sleeves. The cooling air exit temperature is monitored and can be related to the concrete -to-sleeve interface temperature (UFSAR Section 3.8.1.5.4). The Penetration Cooling System consists of two full capacity fans located in the Auxiliary Bldg., cooling coils, piping to the appropriate penetration cooling coils, associated valves and instrumentation. The Penetration Cooling System is within the scope of license renewal and is included in the aging management review for the Essential Ventilation System. The primary shield wall concrete is also subject to extended local heatup at Ginna Station. The purpose of the reactor compartment cooling system is to remove the heat generated by gamma rays in the primary shield and the thermal radiation from the reactor vessel and out-of-core detectors electrical load. Removal of this heat maintains the concrete temperature in the primary shield walls below degradation threshold and localized temperature limits of ACI standards (i.e., 150°F). The reactor compartment cooling is also within the scope of license renewal and is included in the aging management review for the Containment Ventilation system.

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(9) (continued)	No other concrete components are exposed to elevated temperature. Therefore, elevated temperature is not an aging mechanism that can lead to loss of material for Containment internal structural concrete components.

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(10) Identify the codes, edition and/or date of codes and standards which govern plant containment design, inspection and repair.	The design, materials, fabrication, inspection, and proof testing of the containment complied with the applicable parts of the following codes and standards: 1. ASME Boiler and Pressure Vessel Code, Section III - Nuclear Vessels; Section VIII - Unfired Pressure Vessels; Section IX - Welding Qualifications. 2. Building Code Requirements for Reinforced Concrete (ACI 318-63). 3. American Institute of Steel Construction Specifications: a. Specifications for the Design, Fabrication, and Erection of Structural Steel for Buildings, adopted April 17, 1963. b. Code of Standard Practice for Steel Buildings and Bridges, revised February 20, 1963. 4. USAS N 6.2 - 1965, Safety Standard for Design, Fabrication, and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors. 5. ACI 306-66, Specifications for Structural Concrete for Buildings. 6. ASTM C 150-64, Specifications for Portland Cement. 7. State of New York Department of Public Works Specification. 8. ASTM C 260-63T, Specifications for Air-Entrained Admixtures for Concrete. 9. ASTM A 15-64T, Specifications for Billet-Steel Bars for Concrete Reinforcement.

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(10) (continued)	10. ASTM A 305-56T, Specifications for Minimum Requirements for Deformation of Deformed Bars for Concrete Reinforcement. 11. ASTM A408-64T, Specifications for Special Large Size Deformed Billet-Steel Bars for Concrete Reinforcement. Additional codes and standards governing design, materials, inspection and repair for the Ginna Station Containment Structure are contained in Section 3.8.1.2.5 of the UFSAR.

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(11) Specify whether freeze-thaw is an applicable aging mechanism which will be managed by AMP 5.1 or AMP 5.2, as applicable. If not, provide the technical basis for exclusion.	According to NUREG-1801, freeze-thaw does not cause loss of material from reinforced concrete in foundations and in above- and below-grade exterior concrete for plants located in a geographic region of negligible weathering conditions (weathering index <100 day-inch/yr.). Loss of material from such concrete is not significant at plants located in areas where weathering conditions are severe (weathering index >500 day-inch/yr.) or moderate (100-500 day-inch/yr.) provided that the concrete mix design meets the air content (entrained air 3-6%) and water-to-cement ratio (0.35-0.45) specified in ACI 318-63 or ACI 349-85. Construction of concrete structures at Ginna Station was performed under three contracts: Gilbert specification, GC-3799, “Technical Specification for Structural Concrete”, Gilbert specification GC-3526, “General Construction Specification Service Building”, and Rochester Gas and Electric (RGE) Corporation specification, “Construction of Screen House, Heating Boiler Room, and Service Water Building Foundations, Walls, Floors to Elevation 253'-6". Since the contract-specified air contents are within the range specified by current revisions of ACI 318, and the contract-specified water-to-cement ratio meets the recommendations of ACI 318-63 (≤ 0.53), loss of material and cracking of concrete due to freeze-thaw is not an applicable aging mechanism at Ginna Station. Nevertheless, the current mandated inspections performed in accordance with the requirements of 10 CFR 50.55a and the ASME Section XI, Subsections IWE & IWL Inservice Inspection Program and the Structures Monitoring Program manage the

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(11) (continued)	concrete aging effects associated with freeze-thaw. Therefore, the potential aging effects associated with freeze-thaw will be adequately managed during the period of extended operation in the unlikely event that this mechanisms should become active.

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(12) Specify whether aggressive chemical attack is an applicable aging mechanism which will be managed by AMP 5.3 or AMP 5.4, as applicable. If not, provide the technical basis for exclusion.	NUREG-1801 provides guidance for determining whether aggressive chemical attack is an applicable aging mechanism for concrete structures. This guidance includes threshold values for the pH of the environment, and chloride and sulfate concentrations. The environments of concern are air (as influenced by rainwater), below-grade (as influenced by soil or groundwater) and lake water. These three environments are in direct contact with structures and components within the scope of license renewal. According to NUREG-1801, aggressive chemical attack is not significant if the concrete is not exposed to an aggressive environment. An aggressive environment is defined as pH < 5.5, or >500 ppm chlorides, or >1500 ppm sulfates. The below-grade and lake water environments at Ginna Station are therefore classified as non-aggressive. Chemical analyses of rock and groundwater indicate pH values of 6.9-7.7, and chloride and sulfate concentrations < 50 ppm. Data collected by the National Atmospheric Deposition Program/National Trends Network (NADP/NTN) indicate that Ginna Station is exposed to an aggressive air environment based on analysis of rainwater pH. However, according to the Class I Structures License Renewal Industry Report, concrete used in Class I structures is not likely to be significantly affected by acid rain because of the properties of the concrete. In addition, if the concrete is exposed to an aggressive environment for intermittent periods only, degradation caused by aggressive chemicals is not significant. Rain is an intermittent event. Therefore, degradation due to aggressive chemical attack by rainwater is not an aging effect requiring management. The PWR Containment Structures License Renewal Industry Report also states that exposure to acid rain and intermittent

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(12) (continued)	exposure to aggressive chemicals will not cause significant degradation to PWR containments. These facts notwithstanding, the current mandated inspections performed in accordance with the requirements of 10 CFR 50.55a and ASME Code Section XI, Subsection IWL, Examination Category L-A, and the Structures Monitoring Program manage the concrete aging effects associated with aggressive chemical attack. Therefore, the potential aging effects associated with aggressive chemical attack will be adequately managed during the period of extended operation in the unlikely event that these mechanisms should become active.
(13) Provide details of the groundwater monitoring program and discuss potential seasonal variation in ground water chemistry.	A groundwater monitoring process is being implemented at Ginna Station and will address potential seasonal variation in groundwater chemistry. There is no permanent dewatering system at Ginna Station.

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(14) For prestressed concrete containments, discuss plant experience with respect to tendon grease leakage and, if applicable, how the leakage will be managed; also discuss the potential effects of grease leakage on the shear load capacity of the containment structure.	Minor leakage of grease from tendon conduits was observed from the tendon fill-port piping in 2000. The tendon fill ports are located around the exterior of the Containment wall at the base mat level. The ports are nominal 3" diameter galvanized carbon steel pipes which are embedded in concrete. Each pipe is connected to the bottom of a tendon conduit at one end, and the other (fill) end projects vertically above the base mat for a short distance (about 12 inches). Each fill-port pipe is capped with a bronze valve. During original construction, liquefied grease was pumped into the bottom of each tendon conduit through the fill port and into a carbon steel reservoir (grease can) located at the top of each tendon. After a tendon was filled, the valve was shut and plugged with a threaded bronze plug. Ingress of ground water has occurred periodically over the years around the base of the Containment wall, at times submerging the fill ports and causing general corrosion of the exposed fill-port piping. The presence of the bronze valve created a galvanic couple which acted to accelerate the corrosion of the carbon steel pipe. Wall thinning progressed through-wall in some fill-ports, causing grease to leak out of the pipe stubs. After engineering assessment, repair of the fill-port pipe stubs was accomplished by encapsulating the short exposed length of pipe with a high-strength epoxy plastic sleeve, secured to the base mat surface with anchor bolts. The ingress of water has been brought under control. A visual inspection of the tendon top anchorage assemblies was undertaken in 2000-2001 to evaluate the grease levels in the upper reservoirs (grease cans) and to assess the condition of the wires and other anchorage hardware. Grease levels varied from tendon to tendon, but generally the grease levels were lower than desired and, in some cases, the tendon wires were exposed. Water

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(14) (continued)	intrusion was observed in one grease can. For those tendons with exposed wires, it was determined after careful inspection that the wires were covered with a film of grease and no evidence of corrosion was found. Gasket seals at the top and bottom of each can were replaced, and grease was added to these cans in the summer of 2001. Management of grease leakage is accomplished by the Periodic Surveillance and Preventive Maintenance Program, the Structures Monitoring Program and the ASME Section XI, Subsections IWE & IWL Inservice Inspection Program. These programs include accepted elements that manage grease leakage. Further, since the aging management program follows NRC accepted inspection and maintenance procedures for prestressing systems, that also include the management of tendon grease leakage, there will be no detrimental effect on the shear load capacity of the containment structure due to grease leakage effects.

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(15) Each license renewal applicant needs to describe its plant-specific program to address the stress corrosion cracking (SCC) for dissimilar metal welds, and for stainless steel bellows assemblies, if the material is not shielded from a corrosive environment. For the period of extended operation, ASME Section XI, IWE examination Categories E-B and E-F and augmented VT-1 visual examination of bellows assemblies and dissimilar metal welds are required or a suitable alternative proposed.	Dissimilar metal welds and stainless steel bellows assemblies at Ginna Station are exposed to Containment air (external surfaces) and nitrogen or dry instrument air (internal surfaces). These environments are not corrosive. Stainless steel components exposed to these environments are not susceptible to stress corrosion cracking; therefore cracking due to SCC is not an aging effect requiring management for dissimilar metal welds and stainless steel bellows assemblies. Plant-specific operating experience at Ginna Station confirms this conclusion. Nevertheless, the ASME Section XI, Subsections IWE & IWL Inservice Inspection Program which includes augmented VT-1 examinations in certain cases is implemented at Ginna Station and would adequately manage cracking due to SCC in the unlikely event that such an aging effect were to develop.

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(16) Discuss the plant-specific coatings monitoring and maintenance program and specify whether it is credited as an AMP for containment steel elements.	While it is recognized that coatings provide protection to metal surfaces, coatings themselves are not credited at Ginna Station for management of aging effects since they perform no license renewal intended function. Degradation of coatings typically is a result of aggressive chemical or thermal environments. As coatings degrade, the metal surfaces to which they are applied also degrade. Aging management is accomplished by assessing the integrity of the metal surfaces. If degradation of a coated steel component is identified, corrective action includes repair or reapplication of the coating. Aging effects associated with aggressive environments are managed by the ASME Section XI, Subsections IWE & IWL Inservice Inspection Program, the Structures Monitoring Program and the Boric Acid Corrosion Program. These programs have been demonstrated to effectively manage age-related degradation of Containment steel surfaces at Ginna Station.
(17) For prestressed concrete containments, specify whether post-tensioning system degradation will be managed by AMP 5.6 (Section XI, Subsection IWL, Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants, Examination Category L-B, Unbonded Post-Tensioning System, 1992 Code Edition with 1992 Addenda of the ASME Code) and the additional requirements delineated in 10 CFR 50.55a(b)2(ix). If not, provide the technical basis for exclusion.	Degradation of the post-tensioning system at Ginna Station will be managed in accordance with the requirements of ASME Section XI, Subsection IWL, Category L-B, 1992 Edition with 1992 Addenda along with the additional requirements delineated in the amendment to 10 CFR § 50.55a (see SECY-96-080).

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(18) Specify whether settlement of the containment foundation is an applicable aging mechanism which will be managed by AMP 5.7. If not, provide the technical basis for exclusion.	Settlement of structure is directly related to the physical properties of the foundation material. For structures located on rock strata or a suitable foundation, any settlement should have occurred during or immediately following construction. For structures not located on rock strata or a suitable foundation, inspections may be performed to verify that excessive settlement has not occurred. The majority of the settlement should have occurred prior to the end of the current licensing period. There is no permanent dewatering system at Ginna Station. The Cable Tunnel is founded on steel foundation piles driven to bedrock or selected backfill. The Standby Auxiliary Feedwater Building is supported by 12" caissons that are socketed into competent rock. All other structures are founded on competent bedrock, which is fine-grained sandstone with nearly horizontal bedding planes and joints of limited vertical extent. The Containment cylinder is founded on bedrock which acts as an integral part of the containment structure by the use of post-tensioned rock anchors. Structural inspections indicate no visible evidence of settlement since construction of the station. During the Systematic Evaluation Program, the NRC concluded that settlement of foundations and buried equipment is not a safety concern for Ginna Station. Therefore, settlement is not an applicable aging mechanism for Containment structure concrete components. This notwithstanding, the ASME Section XI, Subsections IWE & IWL Inservice Inspection Program and the Structures Monitoring Program effectively manage aging effects due to settlement of the Containment structure during the period of extended operation.

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(19) Identify whether erosion of the porous concrete sub-foundation layer is an applicable aging mechanism; if applicable, provide an aging management review and describe the credited aging management program.	The NRC has issued Information Notices (IN) 97-11 and 98-26 informing nuclear facilities of the possibility of degradation to structure foundations due to the erosion of porous concrete sub-foundations. According to NUREG-1801, erosion of porous concrete sub-foundations must be analyzed. According to the Ginna Station review of NRC INs 97-11 and 98-26, the structure foundations at Ginna Station are constructed of normal concrete and not the subject porous type. Ginna Station is not one of the nine nuclear plants identified by the NRC in October 1996 as having the potential of degradation due to porous concrete. Since porous concrete is not used at Ginna Station, erosion of porous concrete sub-foundations is not applicable aging mechanism.
(20) The GTR listed only six (6) attributes to form the basis for each aging management program. However, the “Draft Standard Review Plan for the Review of License Renewal applications for Nuclear Power Plans,” dated April 21, 2000, identifies ten (10) elements (attributes) as appropriate for an acceptable AMP. The GTR predates the Draft standard Review Plan for the review of Licensing Renewal applications for Nuclear Power Plans, and states in Section 4.0 that the report only presents program attributes for the AMPs, and that plant-specific details of the AMPs will be developed during the preparation of license renewal applications. Therefore, applicants for license renewal will be responsible for developing and describing the plant-specific AMPs and addressing each of the ten elements specified in the Draft Standard Review Plan.	Aging management programs credited for managing effects of aging for concrete and steel Containment components and the post-tensioning system contain the ten (10) attributes identified in the SRP. These programs are described in Appendix B of the LRA.

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(21) The WOG GTR indicates that the license renewal applicant may update an existing design fatigue analysis to account for the additional years of plant operation or manage the effects of the aging mechanism through aging management programs. The GTR uses AMP 5.5 for managing the effects of fatigue during the renewal license period, and basically endorses the ASME Code Section XI surveillance and testing program. For components where CLB fatigue TLAAs exist, this option would allow the CLB fatigue Section III cumulative usage factors (CUF) to be exceeded during the period of extended operation. The staff has not endorsed this option on a generic basis at this time. An applicant wishing to pursue this option would have to obtain staff review and approval on a case-by-case basis. For components where CLB fatigue TLAAs do not exist (are not addressed in 10 CFR 54.21), aging effects due to fatigue can be addressed by either a Section III fatigue analysis (including the additional years for the period of extended operation) or by adequately managing these effects for the period of extended operation.	Fatigue is the progressive degradation of materials subjected to application of cyclic loads that are less than the maximum allowable static loads. Concrete components at Ginna Station were designed in accordance with ACI 318 and therefore have excellent low-cycle fatigue properties. ACI standards limit the maximum design stress to less than 50% of the static stress of the concrete. The concrete fatigue strength is about 55% of its static strength at extremely high cycles (>10^7 cycles) of loading. Therefore, fatigue is not an aging mechanism that can lead to cracking for Containment structure concrete components at Ginna Station. For steel components, fatigue is the cumulative effect of microstructural localized plastic deformation in the material section that occurs with each cycle of applied stress of sufficient magnitude. Class I steel structures were designed in accordance with American Institute of Steel Construction (AISC) Code where consideration was given to the number of stress cycles, the expected range of stress, and type and location of structural members/detail. The maximum stress and the maximum range of stress are specified in the code. The stress permitted in structural steel Class I structure components provide an adequate safety margin against fatigue failure. Additionally, more margin is available since the actual cyclic loading is lower than that assumed in the analysis, which typically uses bounding conditions. Typically, loads applied to structural members are constant or static; non-constant loading (i.e., due to wind, etc.) is infrequent. Fatigue has been evaluated with respect to its effects on the ability of Class I steel structures to perform their intended safety function during the period of extended operation. Since the Class I structures were designed in

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(21) (continued)	accordance with the AISC Code, the stress ranges in steel components and connections will be limited and therefore cracking due to fatigue is not an aging effect requiring management. Cracking due to metal fatigue is treated as a Time-Limited Aging Analysis (TLAA) at Ginna Station if included in the current licensing basis (CLB).
(22) Specify the containment structure components and provide plant-specific details of the TLAAs for prediction of cumulative fatigue usage through the period of extended operation.	There are no TLAA's for prediction of cumulative fatigue usage of Containment structure components at Ginna Station.

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(23) Specify those containment structure components for which fatigue is an applicable aging mechanism, but no CLB fatigue analysis based on a 40-year plant life exists. In addition to implementation of AMP 5.5, the requirements of 10 CFR 50.55a should be met.	Containment Liner and Penetrations The Containment liner, liner penetrations, and liner steel components were originally designed to comply with ASME Code Section III-1965 for pressure boundary and the AISC Code for structural steel. The Containment liner and penetrations, including the personnel and equipment hatch penetrations, were designed as Class B vessels. The winter 1965 Addenda of ASME Section III, Subsection B, N-1314(a) requires that the Containment vessel satisfy the provisions of Subsection A, N-415.1, “Vessels Not Requiring Analysis for Cyclic Operation,” in order that the Subsection B rules may be applied. ASME Section III, N-415.1 states that a fatigue analysis is not required provided the service loading of the vessel or component meets six (6) conditions. An analysis has been performed which demonstrates that all six conditions are met for the liner and penetrations and that the ASME Section III Code rules for fatigue are met for the extended period of operation. Containment Liner Channel Anchors A fatigue analysis of the fillet weld attaching the channel anchors to the liner was performed as part of the original design. The liner anchorage was originally designed for 100,000 stress cycles. This corresponds to more than four full stress cycles on a daily basis for 60 years. Fluctuations of temperature and pressure in the Containment on a daily basis are not significant enough in magnitude to cause four cycles of design basis stress at the liner anchorage weld every day. Therefore the original fatigue analysis remains valid for the period of extended operation.

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(23) (continued)	Containment Stainless Steel Tendon Bellows The fatigue usage factor for the Containment stainless steel tendon bellows was calculated for the period of extended operation using the allowable radial and vertical displacements. The CUF was calculated to be .004 at the end of 60 years. Therefore the structural integrity of the bellows will be maintained throughout the period of extended operation. Tendon Wire Fatigue Analysis Fatigue tests were conducted by the Swiss Federal Testing Station (EMPA) in 1960 on individual 7-mm tendon wires and on 18-wire tendons. The tendons and wires were cycled between 0.7 UTS and 0.8 UTS (ultimate tensile strength) for over 2 million cycles without the failure of a single wire. An analysis has been performed which demonstrates that the original seismic fatigue evaluation for the tendon wires remains valid throughout the period of extended operation.

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(24) For prestressed concrete containments, provide plant-specific details of the TLAA for prediction of tendon prestress losses through the period of extended operation.	In accordance with ACI 318-63 and the Ginna Station UFSAR the design of the Containment Structure post-tensioning system accounts for prestress losses caused by the following mechanisms/processes:· • Elastic shortening of concrete· • Creep of concrete· • Shrinkage of concrete· • Stress-relaxation in steel tendon wires· • Frictional loss due to curvature in the tendons and contact with tendon conduits. No allowance is made for seating of the BBRV anchor since no slippage occurs in the anchor during transfer of the tendon load to the structure. Loss of prestress of the Containment structure post-tensioning system is a Time-Limited Aging Analysis and is discussed in Section 4.5 of the LRA.
(25) The GTR identified structural connections as containment structure components that require aging management in Table 2-1. However, there is no definition or description of structural connections in GTR Section 2.0. A definition and a description of the AMP for structural connections are needed.	There are no Containment structural connections unique to Ginna Station. The parts or subcomponents that could be considered structural connections in GTR Table 2-1 are addressed under penetrations, equipment and personnel hatches. Therefore, a description of the AMP for structural connections need not be provided.
(26) The GTR identified embedments as containment structure components that require aging management in Table 2-1. However, there is no definition or description of embedments in GTR Section 2. A definition and a description of the AMP for embedments are needed.	The embedments are part of subcomponents given in GTR Table 2-1 (e.g., equipment and personnel hatches). Additional embedments are associated with the liner and other miscellaneous equipment attached to the containment structure. This equipment is identified elsewhere in the LRA, along with the aging management program. Therefore, a description of the AMP for embedments need not be provided here.

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(27) The GTR does not commit to inspection of inaccessible areas when there is no indication of degradation of adjacent accessible areas, except when the potential for degradation is “event driven”; i.e., some unusual event has occurred which has the potential to degrade inaccessible areas of the containment structures. Therefore, the GTR cannot be referenced by license renewal applicants for managing aging of inaccessible areas. Individual license renewal applicants are required to describe a program for inspection of inaccessible areas or adopt a program endorsed by the staff in similar applications.	Structural components inaccessible for inspection were evaluated for potential aging effects based on their environment as part of the aging management review. Several structural components that are inaccessible for visual inspection require aging management at Ginna Station. Examples include buried concrete, embedded steel, and structural components blocked by installed equipment or structures. Structural components inaccessible for inspection are managed by inspecting accessible structures with similar materials and environments for aging effects that may be indicative of age-related degradation of inaccessible structural components. The programs credited for managing aging effects of inaccessible structural components are the Structures Monitoring Program and the ASME Section XI, Subsections IWE & IWL Inservice Inspection Program. These programs are described in Appendix B. These aging management programs have been implemented at Ginna Station to meet the USNRC requirements as given in SECY-96-080 to improve the quality and effectiveness of containment inspections, and to inspect, and as needed to take corrective action for defects, in critical areas that include inaccessible areas. These programs will be continued into the extended period of operation as defined by the licensee renewal program. Therefore, the programs used at Ginna Station to inspect inaccessible areas follows the program endorsed by the staff in the amendment to 10 CFR § 50.55a (see via SECY-96-080). It is also noted that these aging management programs address inaccessible areas as follows:§ • via the requirements of 10 CFR § 50.55a amendment (SECY-96-080);

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(27) (continued)	• when “serious degradation in inaccessible areas” is identified from other utility plant inspections that may be an area of concern for this plant; • when indicators exist that degradation may be occurring in an inaccessible area; • when an event occurs that could affect an inaccessible area making it susceptible to degradation.

Table 3.6.0-1 Containment Structure - WCAP-14756-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(28) The aging effects in concrete due to leaching of calcium hydroxide and alkali aggregate reaction are identified in the GTR as not requiring aging management. This is unacceptable because plant-specific evaluation of their applicability is needed. Therefore, if these aging mechanisms (leaching of calcium hydroxide and alkali aggregate reaction) are applicable, applicants would be required to propose a plant specific aging management program. Alternatively, applicant can credit the ASME Code, Section XI, Examination Category L-A as an adequate aging management program.	Leaching of calcium hydroxide is typically observed in concrete that is exposed to flowing water. Containment concrete structures and components are not exposed to flowing water at Ginna Station. In addition, concrete structures and components at Ginna Station are constructed of dense, well-cured concrete with an amount of cement suitable for strength development, and a water-to-cement ratio that is characteristic of low-permeability concrete. This is consistent with the guidance provided by the ACI 201.2R. Therefore, degradation caused by leaching of calcium hydroxide is not significant and leaching is not an applicable aging mechanism for Containment structure concrete components at Ginna Station. Concrete components at Ginna Station were constructed using non-reactive aggregates that were tested for potential reactivity in accordance with ASTM C227 and ASTM C295, which are established industry standards. Therefore, reaction with aggregates is not an aging mechanism that can lead to degradation of Containment structure concrete components. These facts notwithstanding, the ASME Section XI, Subsections IWE & IWL Inservice Inspection Program and the Structures Monitoring Program are aging management programs implemented at Ginna Station which manage aging effects due to leaching of calcium hydroxide and reaction with aggregates. These programs provide assurance that all potential concrete aging effects will be adequately managed throughout the period of extended operation.

Table 3.6.0-2 Reactor Coolant System Supports - WCAP-14422, Rev. 2-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(1) Definition of "local" and "adjacent" (Section 3.1) The Westinghouse Owners Group did not clearly define the term "local" in its report. However, the aging management programs could be the same for all concrete structures and structural components, therefore, the license renewal applicants must describe the aging management program for adjacent concrete structures and any differences from the aging management program for the local concrete structures.	Concrete adjacent to the support is addressed in the Containment Structure aging management review. Concrete local to the embedment is addressed with the RCS supports.
(2) Detailed description of the Reactor Coolant System supports (Section 3.1) A license renewal applicant will have to justify any differences between its Reactor Coolant System support system and the figures and descriptions of the supports systems contained in the Westinghouse Owners Group report.	The configurations of the RCS component supports at Ginna Station are identical to those described in the GTR, namely: • Reactor Vessel Supports - Configuration 1 • Steam Generator Supports - Configuration 3 • Pressurizer Supports - Configuration 2 • Reactor Coolant Pump Supports - Configuration 2 • RCS Pressurizer Surge Line Support - Variable spring hanger
(3) Discrepancies and Omissions (Section 3.1) The Westinghouse Owners Group report contains many discrepancies and omissions. A license renewal applicant needs to resolve these discrepancies and omissions in its application. 1. Wear plates and bearing pads are included as support components and are within the scope of this Westinghouse Owners Group report but are not identified in Table 2-1 as parts and sub-components requiring an aging management review.	 1. Table 2.4.2-12 includes wear plates and bearing pads

Table 3.6.0-2 Reactor Coolant System Supports - WCAP-14422, Rev. 2-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
2. Sketches of Reactor Coolant Pump support configuration 4 and Pressurizer support configuration 2 are not provided in the Westinghouse Owners Group report.	2. The Ginna RCP supports are represented by Configuration 2, not Configuration 4. Figure 2-11 in the GTR depicts Configuration 2 by eliminating the upper supports with the PZR bolted into the concrete floor.
3. Section 3.2.9 of the Westinghouse Owners Group report indicates that ASTM A36 steel is used in Steam Generator and Reactor Coolant Pump supports, however, ASTM A36 steel is not included in the list of material for the primary component supports (Table 2-4).	3. ASTM A36 structural carbon steel material is included in the scope of the aging management review as it is utilized in SG supports. RCP supports do not utilize A36 steel.
4. The 1963 AISC manual (Ref. 3) states that the following steel materials are commonly used for steel construction but they are not listed in Table 2-4 of the Westinghouse Owners Group report. They are ASTM A7, A36, A242, A373, A440, and A441 structural steel and ASTM A325 bolts.	4. Of the materials listed, only ASTM A36 is used in Ginna RCS supports. This material has been included in the material matrix table in the aging management review report.
5. There are no specific descriptions and sketches for the pressurizer surge line supports.	5. A description and sketch of the Ginna PZR surge line supports has been included in the aging management review report. This support (RCU-1) is a variable spring hanger and the only support in the line.
(4) Strain Aging Embrittlement (Section 3.3.1.4) Temper embrittlement and strain aging embrittlement are the most common forms of thermal embrittlement that are seen in ferritic materials as stated in Section 3.2.4 of the Westinghouse Owners Group report. The Westinghouse Owners Group report has determined that temper embrittlement is not a concern for the ferritic materials of Reactor Coolant System supports. However, the Westinghouse Owners Group report does not address the aging effects from strain aging embrittlement but states that thermal embrittlement is not applicable. The license renewal applicants will address the applicability of the aging effects due to strain energy embrittlement to their plants.	RCS supports at Ginna Station: a. do not contain cast austenitic stainless or low carbon steels; b. are not loaded beyond the elastic limit during normal operation, and c. operate at temperatures below 450°F. Consequently, it can be concluded that thermal and strain-aging embrittlement are not applicable aging mechanisms at Ginna Station and therefore no aging management of age-related degradation caused by these mechanisms is required.

Table 3.6.0-2 Reactor Coolant System Supports - WCAP-14422, Rev. 2-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(5) Low Fracture Toughness (Section 3.3.1.6) Appendix C of NUREG-0577 addresses this item and groups many Westinghouse Owners Group member plants as Group I “plants requiring further evaluation.” Although Table B9 of NUREG-1557 indicated that “low fracture toughness is not significant for containment internal structures,” in general, these two documents only addressed the containment internal structures as a whole and did not specifically address the Reactor Coolant System support components. Westinghouse Owners Group recognizes this concern and states in Section 3.2.9 of its report that “Utilities with potential problems were required to demonstrate that the suspect structures have adequate fracture toughness to comply with the criteria defined in NUREG-0577.” However, it further states that “low fracture toughness does not cause detrimental aging effects that must be addressed by maintenance programs.” The staff does not believe that the Westinghouse Owners Group report provides sufficient information to support this conclusion. A license renewal applicant will address, if its plant is listed as Group 1 in Appendix C of NUREG-0577, that its plant had performed an analysis and the steel components of its RCS supports have adequate fracture toughness that no maintenance program is necessary.	Plant-specific evaluation of low fracture toughness and lamellar tearing as applicable to Ginna RCS supports was performed and submitted to the NRC in 1978. The primary conclusion of this analysis was that low fracture toughness and lamellar tearing are not a concern for the design and installation of the RCS supports at Ginna Station.Furthermore, Ginna Station is not listed as a Group I plant in Appendix C of NUREG-0577.Therefore, no aging management program is required for the effects of low fracture toughness and lamellar tearing during the extended period of operation.

Table 3.6.0-2 Reactor Coolant System Supports - WCAP-14422, Rev. 2-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(6) Fatigue (Section 3.3.1.7) A license renewal applicant will have to justify any differences between the materials used for its Reactor Coolant System supports and the values listed in Table 2-4 of the Westinghouse Owners Group report.	The only difference between materials used for Ginna RCS supports and those shown in Table 2-4 of the GTR is ASTM A36 carbon steel which is used in the steam generator supports. However, A36 is used in a design configuration where the loading is not cyclic and therefore fatigue is not a concern.The rigid bumpers in the upper support system of the steam generators also utilized materials that are not listed in Table 2-4 of the GTR. These are listed below: • Clevis Pin - A193 • Tangential Plate - A537 • Guide Shaft - A193 • Tie Rod - A193 • Rod Head - A668 • Boss - A537 • Tangential End and Rod End Bearing Plate - A36 • Stop Nut - A193 These materials are also not subjected to cyclic or fluctuating loads and fatigue is therefore not a concern.

Table 3.6.0-2 Reactor Coolant System Supports - WCAP-14422, Rev. 2-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(7) Irradiation of Concrete (Section 3.3.2.3) The Westinghouse Owners Group report states that concrete degradation from irradiation will be addressed by plant-specific evaluation. The staff agrees with this suggestion and the license renewal applicant must develop plant-specific program(s) to evaluate this concern.	Fast neutron fluence and gamma ray dose at the cavity liner/concrete interface 6.0 feet above the core midplane (at the location of the RPV supports) have been evaluated. The maximum neutron fluence at 54 EFPY is 9.49 x 10^{17} n/cm^2. The maximum gamma ray dose at the same location at 54 EFPY is 4.94 x 10^9 rad. These values are below the 10^{18} n/cm^2 fast neutron fluence and 10^{10} rad gamma ray thresholds for concrete damage due to irradiation (cracking and change in properties). Therefore, degradation of the concrete material associated with the RPV supports due to neutron irradiation damage is not a concern. The Ginna RCS supports are loaded in compression during normal operation. Furthermore, these supports (other than the RPV supports) are located sufficiently far from the reactor vessel core that effect of neutron embrittlement is not a concern. Therefore, no aging management program is required for neutron irradiation embrittlement.

Table 3.6.0-2 Reactor Coolant System Supports - WCAP-14422, Rev. 2-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(8) Elevated Temperature of Concrete (Section 3.2.4) The Westinghouse Owners Group report states that concrete operating temperature should not exceed 150°F and local area temperature should be kept under 200°F. The Westinghouse Owners Group report further states that reactor pressure vessel supports could be subjected to high temperatures that could potentially result in a local temperature above 200°F if supplemental cooling is not provided. For those support configurations where the local temperature at concrete surfaces could exceed 200°F, special design features are incorporated based on air or water cooling to keep local temperature below 200°F. These temperatures are specified in the ASME Code. Therefore, elevated temperature is not a concern for concrete. Because the operating temperature of concrete components are kept below the limits specified by the code by means of supplemented cooling, the staff considers that the aging effects of elevated temperature are applicable to the Reactor Coolant System supports and are being managed by supplemented cooling features. The license renewal applicants will address the concern that the aging effects associated with elevated temperature are applicable and demonstrate that the existing design features in the plants are capable of preventing any unacceptable degradation during the period of extended operation.	Cracking due to elevated temperature is not an aging effect requiring management at Ginna Station since concrete temperatures are maintained below American Concrete Institute (ACI) code thresholds due to normal containment and supplemental cooling. Cooling systems that limit long term concrete temperatures to less than 150°F with local areas limited to 200°F include the Reactor Compartment Fan Cooling System and the RPV Support Pad Cooling System. The air temperature exiting the reactor compartment is monitored to assure compliance with these design requirements. Component cooling water circulates continuously through the cooling channels in the RPV support pads. These cooling systems are within the scope of license renewal and will continue to be operated through the period of extended operation, providing assurance that concrete operating temperature limits will be maintained at Ginna Station.

Table 3.6.0-2 Reactor Coolant System Supports - WCAP-14422, Rev. 2-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(9) SRP-LR (Section 3.4) The attributes of the aging management programs provided in the Westinghouse Owners Group report do not address all elements as listed in Table A1-1 of Appendix A of the SRP-LR. The applicants should address the missing review elements and describe the plant-specific experience, if any, related to aging degradation of the Reactor Coolant System supports in their applications.	The aging management program descriptions contained in Appendix B of the License Renewal Application (LRA) address all required attributes. The program descriptions also contain any relevant Ginna-specific operating experience.
(10) Details of leakage walk-ons and leakage monitoring program (Section 3.4.2) A license renewal applicant must provide the necessary details to perform leakage identification walkdowns and the details of the leakage monitoring program(s), especially the frequencies, for Aging Management Program 1-1 and Aging Management Program 1-2.	Details of Ginna leakage identification and monitoring programs are contained in plant operating procedures. The Ginna Station Boric Acid Corrosion Program, as implemented by plant procedures, contains details of the leakage identification walkdowns and monitoring requirements, including frequencies, associated with identifying boric acid leakage. A description of the Boric Acid Corrosion Program is contained in Appendix B of the LRA. Plant procedures are available for on-site review.
(11) Baseline Inspection (Section 3.4.2) All structures and structural components need a baseline inspection to document the condition of the structures and structural components. Therefore, the renewal applicants must have plant-specific baseline inspection results for all structures and structural components, or a planned inspection to obtain such results and validate the aging management programs prior to entering the period of extended operation.	Although not characterized as “baseline inspections” at the time they were performed, inspections that serve as baseline inspections have been performed and documented for the RCS Supports under the Ginna Station ASME Section XI, Subsection IWF Inservice Inspection Program.

Table 3.6.0-2 Reactor Coolant System Supports - WCAP-14422, Rev. 2-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(12) Inspection of inaccessible areas (Section 3.4.2) For RCS supports located in inaccessible areas, a license renewal applicant must provide an inspection program to inspect these RCS supports or provide technical justification for not performing inspection.	Structural components inaccessible for inspection were evaluated for potential aging effects based on their environment as part of the aging management review. Several structural components that are inaccessible for visual inspection require aging management at Ginna Station. Examples include buried concrete, embedded steel, and structural components blocked by installed equipment or structures. Structural components inaccessible for inspection are managed by inspecting accessible structures with similar materials and environments for aging effects that may be indicative of age-related degradation of inaccessible structural components. The programs credited for managing aging effects of inaccessible structural components are the ASME Section XI, Subsections IWE & IWL Inservice Inspection Program and the Structures Monitoring Program . These programs are described in Appendix B.

Table 3.6.0-2 Reactor Coolant System Supports - WCAP-14422, Rev. 2-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(13) Surveillance Frequency for AMP-1.2 (Section 3.4.3) AMP-1.2 specifies inspection frequency in accordance with the requirements of Subsection IWF-2410 (Inspection Program) and Table IWB-2412-1, each 10-year interval following the first interval, 10-year inspection program, with IWB-2412. The staff considers the frequency proposed by Westinghouse Owners Group not to be adequate. The proposed frequency is in accordance with ASME standards, but the inspections are to the requirements of ACI Standards, therefore, the frequency of inspection should also follow the recommendations of the ACI standards. Inspection frequencies recommended by ACI 349.3R-96 are every 10 years for below grade structures and controlled interiors and every 5 years for all other structures. Section 4.2.4.1 of NUREG/CR-6424 has the same recommendation for inspection frequencies. A license renewal applicant must address this concern in its application.	Two aging management programs were identified for concrete embedments. The Boric Acid Corrosion Program is credited for managing degradation from boric acid leaks and the Structures Monitoring Program for managing general loss of material and change in material properties. Both aging management programs, described in Appendix B, meet or exceed the frequencies recommended in ACI 349.3R.
(14) Acceptance criteria for leakage walkdowns (Section 3.4.4) In accordance with the Westinghouse Owners Group report, leakage walkdowns and monitoring are plant-specific. Therefore, a license renewal applicant will have to provide the necessary qualitative or quantitative acceptance criteria for leakage walkdowns and monitoring.	Acceptance criteria for leakage walkdowns and monitoring are included in the Boric Acid Corrosion Program, which is described in Appendix B of the LRA.

Table 3.6.0-2 Reactor Coolant System Supports - WCAP-14422, Rev. 2-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(15) Acceptance Criteria for AMP-1.2 (Section 3.4.4) AMP-1.2 specifies acceptance criteria in accordance with several ACI standards. These ACI standards are ACI 201.2R-77, ACI224.1R-89, and ACI 224R-89. The staff has reviewed these ACI standards and concluded that, except for ACI 224.1R, they are mainly for design and construction rather than aging effects management because those concrete properties are built-in by design and construction. However, they do contain attributes that can be used to develop inspection acceptance criteria for AMP-1.2. For leakage walkdowns and leakage monitoring, the acceptance criteria are the same as that listed for AMP-1.1. The staff has also reviewed ACI 349.3R-96, which is referenced in the Westinghouse Owners Group report for surveillance technique, and concluded it has acceptance criteria that can be modified and used as the inspection acceptance criteria for AMP-1.2. These criteria include acceptance without further evaluation, acceptance after review, and conditions requiring further evaluation. The license renewal applicants will provide a description of the inspection acceptance criteria in their application for the staff to review.	The inspection criteria for the aging management program to control effects of aggressive chemical attack and corrosion of concrete embedments at Ginna Station are described in the Structures Monitoring Program . These criteria are based on ACI and ASME Section XI guidance. ACI 201.2R-77, ACI224.1R-89, ACI 224R-89, and ACI 349.3R-96 are recommended as guides in the GTR for establishing acceptance criteria. ACI 201.2R-7 and ACI 224.1R-89 are mainly for design and construction rather than aging management. However, acceptance criteria contained in ACI 349.3R-96 that can be used as inspection acceptance criteria for the concrete embedment area associated with RCS supports are described below: Acceptance criteria without further evaluation a. Absence of leaching and chemical attack b. Popouts and voids less than 20 mm (3/4 in.) in diameter or equivalent surface area c. Scaling less than 5 mm (3/16 in.) in depth d. Spalling less than 10 mm (3/8 in.) in depth and 100 mm (4 1/4 in.) in any dimension e. Absence of any signs of corrosion in reinforcing steel system, anchorage components, exposed embedded metal surfaces and corrosion stains around the embedded metal f. Passive cracks (i.e., absence of recent growth or other degradation mechanisms at the crack) less than 0.4 mm (0.015 in.) in maximum width below any surface enhanced widening

Table 3.6.0-2 Reactor Coolant System Supports - WCAP-14422, Rev. 2-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(15) (continued)	g. Absence of excessive deflections, settlements or other physical movements that may affect structural performance h. Absence of detached embedments or loose bolts i. Absence of indications of degradation due to vibratory loads from piping and equipment Acceptance criteria requiring review a. Appearance of leaching or chemical attack b. Popouts and voids less than 50 mm (2 in.) in diameter or equivalent surface area c. Scaling less than 30 mm (1 1/8 in.) in depth d. Spalling less than 20 mm (3/4 in.) in depth and 200 mm (8 in.) in any dimension e. Corrosion staining of undefined source of concrete surfaces f. Passive cracks less than 1 mm (0.04 in.) in maximum width g. Passive settlements or deflections within the original design limits Criteria defining further evaluation a. Observed concrete surface conditions which exceed the acceptance criteria limits given for the cases requiring review b. Conditions found to be detrimental to the structural or functional integrity as a result of review

Table 3.6.0-2 Reactor Coolant System Supports - WCAP-14422, Rev. 2-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
(16) Plant-Specific Programs. Recommendations from Section 5 of the Westinghouse Owners Group report (Section 3.6)	Responses to specific items are arranged in the same order as in the preceding column.
• Identification and evaluation of any plant-specific Time-Limited Aging Analyses applicable to their Reactor Coolant System supports.	• Fatigue is the only TLAA associated with the RCS supports. A discussion of TLAA's is presented in Section 4.0 of the LRA. No explicit fatigue evaluation was performed during the design of RCS supports for Ginna Station. Review of the CLB and other analyses related to RCS supports did not identify any other TLAA.
• Identification and evaluation of current-term programs implemented within the current licensing term to address technical issues from industry practices and United States Nuclear Regulatory Commission (NRC) directives [that] should be continued into the license renewal term. Modifications to or elimination of these programs have to be justified.	• Current-term programs implemented within the current licensing term at Ginna Station to address technical issues that were identified from industry practices and NRC directives will be continued into the license renewal term, without modifications.
• Identification and justification of plant-specific programs that deviate from the recommended aging management programs.	• Plant-specific programs for aging management of Ginna RCS supports do not deviate from those recommended by the GTR such that the effectiveness of the program is reduced.
• Identification of any specific program necessary to ensure that proper preload is retained for the component supports within the scope of this report.	• Loss of preload due to creep and stress-relaxation is not an aging effect requiring management at Ginna Station. Nevertheless, inspections performed under the ASME Section XI, Subsection IWF Inservice Inspection Program and the Structures Monitoring Program (described in Appendix B of the LRA) provide assurance that proper pre-load is retained for RCS supports within the scope of this AMR report.
• Identification of any evidence of aging degradation in inaccessible areas during the current licensing term, that is considered to potentially affect system intended functions. A plan of action to address any identified potential degradation should be provided.	• The Structures Monitoring Program covers the identification of aging degradation in inaccessible areas during the current licensing term. A plan of action to resolve or address identified problems is also provided.

Table 3.6.0-2 Reactor Coolant System Supports - WCAP-14422, Rev. 2-A, Final Safety Evaluation Report Response to Applicant Action Items

Renewal Applicant Action Item	Plant-Specific Response
• Verification that the plant is bounded by this GTR. The actions applicants must take to verify that this plant is bounded will be provided in an implementation procedure.	• Based on review of the installed configuration, operating environment, and materials of construction of the RCS supports, Ginna Station is bounded by the Reactor Coolant System Supports GTR Plant-specific data are given in this report show that indeed this is true.
• Plant-specific evaluation of potential degradation due to irradiation of the components within the scope of this report.	• It has been demonstrated that potential degradation due to irradiation of the RCS supports components within the scope of this AMR is very low. Consequently, no aging management program is needed.

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(1) Penetration sleeves, penetration bellows, and dissimilar metal welds	Cumulative fatigue damage (CLB fatigue analysis exists)	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	A fatigue analysis of penetration sleeves is contained in the Ginna Station CLB and has been evaluated as a TLAA (see Section 4.6). However, penetration bellows and dissimilar metal welds are not incorporated into Ginna Station's current licensing basis as TLAAs.
(2) Penetration sleeves, bellows, and dissimilar metal welds.	Cracking due to cyclic loading, or crack initiation and growth due to SCC	Containment ISI and Containment leak rate test	Yes, detection of aging effects is to be evaluated	The Containment Program implements and formally adopts the requirements of the ASME Section XI, Subsections IWE & IWL Inservice Inspection Program as part of the Ginna Station Inservice Inspection Program. Included in the scope of the IWE program are the exposed portions of the containment liner, the liner for the fuel transfer penetration, all other penetrations, associated bolting, moisture barriers, and all airlocks, seals, gaskets and penetration bellows previously included in the scope of Appendix J. The ASME Section XI, Subsections IWE & IWL Inservice Inspection Program includes inspections and leak rate tests which would indicate the presence of significant degradation from to cracking due to cyclic loading or crack initiation and growth due to SCC. That notwithstanding, SCC is not an applicable aging mechanism for penetration sleeves, bellows and dissimilar metal welds. The carbon steel components within penetrations are not susceptible to SCC. The stainless steel components require both a high temperature (>140°F) and exposure to an aggressive chemical environment (e.g. exposure to chlorides). The bellows at Ginna Station are not exposed to aggressive chemical environments. A review of plant specific operating experience did not identify any occurrences of bellows failures due to SCC. Furthermore a review of industry operating experience indicated that SCC of bellows was typically caused by poor design controls leading to the inadvertent introduction of contaminants.

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(3) Penetration sleeves, penetration bellows, and dissimilar metal welds	Loss of material due to corrosion	Containment ISI and Containment leak rate test	No	Consistent with NUREG-1801. The ASME Section XI, Subsections IWE & IWL Inservice Inspection Program ensures that the containment ISI and leak rate tests adopt and implement the requirements of ASME Section XI, Subsections IWE & IWL. Included in the scope of the IWE program are the exposed portions of the containment liner, the liner for the fuel transfer penetration, all other penetrations, associated bolting, moisture barriers, and all airlocks, seals, gaskets and penetration bellows previously included in the scope of Appendix J. Inspections and leak rate tests performed in accordance with the containment inspection procedures verify the containment pressure boundary integrity and may be credited for detecting loss of material due to corrosion.
(4) Personnel airlock and equipment hatch	Loss of material due to corrosion	Containment ISI and Containment leak rate test	No	Consistent with NUREG-1801. The ASME Section XI, Subsections IWE & IWL Inservice Inspection Program ensures that the containment ISI and leak rate tests adopt and implement the requirements of ASME Section XI, Subsections IWE & IWL. Included in the scope of the IWE program are the exposed portions of the containment liner, the liner for the fuel transfer penetration, all other penetrations, associated bolting, moisture barriers, and all airlocks, seals, gaskets and penetration bellows previously included in the scope of Appendix J. This program may be credited for managing the aging effects of loss of material due to corrosion. Additionally, the Periodic Surveillance and Preventive Maintenance Program also requires visual inspections of hatches, hinges, locks, and closure mechanisms as well as elastomeric seals associated with the containment air locks and is also credited for managing the aging effect loss of material due to corrosion. The program may also be credited with managing the aging effects of loss of leak tightness in the closed position due to mechanical wear of locks, hinges and closure mechanisms.

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(5) Personnel airlock and equipment hatch	Loss of leak tightness in closed position due to mechanical wear of locks, hinges and closure mechanism	Containment leak rate test and Plant Technical Specifications	No	Consistent with NUREG-1801. The ASME Section XI, Subsections IWE & IWL Inservice Inspection Program ensures that the containment ISI and leak rate tests adopt and implement the requirements of ASME Section XI, Subsections IWE & IWL. Included in the scope of the IWE program are the exposed portions of the containment liner, the liner for the fuel transfer penetration, all other penetrations, associated bolting, moisture barriers, and all airlocks, seals, gaskets and penetration bellows previously included in the scope of Appendix J. This program may be credited for managing the aging effects of loss of material due to corrosion. Additionally, the Periodic Surveillance and Preventive Maintenance Program also requires visual inspections of hatches, hinges, locks, and closure mechanisms as well as elastomeric seals associated with the containment air locks and is also credited for managing the aging effects of loss of material due to corrosion. The program may also be credited with managing the aging effects of loss of leak tightness in the closed position due to mechanical wear of locks, hinges and closure mechanisms.

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(6) Seals, gaskets, and moisture barriers	Loss of sealant and leakage through containment due to deterioration of joint seals, gaskets, and moisture barriers	Containment ISI and Containment leak rate test	No	Consistent with NUREG-1801. The ASME Section XI, Subsections IWE & IWL Inservice Inspection Program ensures that the containment ISI and leak rate tests adopt and implement the requirements of ASME Section XI, Subsections IWE & IWL. Included in the scope of the IWE program are the exposed portions of the containment liner, the liner for the fuel transfer penetration, all other penetrations, associated bolting, moisture barriers, and all airlocks, seals, gaskets and penetration bellows previously included in the scope of Appendix J. This program may be credited for managing the aging effects of loss of sealant and leakage through containment due to deterioration of joint seal, gaskets, and moisture barriers. The Periodic Surveillance and Preventive Maintenance Program also requires visual inspections of hatches, hinges, locks, and closure mechanisms as well as elastomeric seals associated with the containment air locks and may also be credited for managing the aging effects of loss of sealant and leakage through containment due to deterioration of joint seal, gaskets, and moisture barriers.

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(7) Concrete elements: foundation, walls, dome.	Aging of accessible and inaccessible concrete areas due to leaching of calcium hydroxide, aggressive chemical attack, and corrosion of embedded steel	Containment ISI	Yes, if aging mechanism is significant for inaccessible areas	Consistent with NUREG-1801. The ASME Section XI, Subsections IWE & IWL Inservice Inspection Program identifies the evidence that an aging mechanism is present and active and also provides confirmation and verification of the absence of all types of aging effects. Indication of aging effects may be absent if the materials of construction, design specifications, and operational environment preclude an aging mechanism but, due to the long lead time necessary for some effects to manifest themselves, it is prudent to periodically assess the condition of SSCs regardless of the likelihood that a particular aging mechanism is applicable. The degradation of inaccessible concrete can create symptoms of aging effects that are detectable in accessible areas. Conversely, if aging effects are present in accessible areas it is sensible to extrapolate those effects into inaccessible areas and perform additional evaluations. Containment accessible and inaccessible concrete has been evaluated for the following aging mechanisms: Aging Mechanism: Aggressive Chemical Attack Aging Effect: Loss of Material, Changes in Material Properties Evaluation: Concrete degradation in air due to aggressive rainwater is insignificant and the below-grade/lake water environment is non-aggressive. Additionally, recent structural inspections revealed no evidence of degradation owing to aggressive chemical attack; therefore, loss of material and change in material properties due to aggressive chemical attack are not probable aging effects at Ginna Station and have not been observed to date. The Structures Monitoring Program requires periodic monitoring of ground/lake water to verify chemistry remains non-aggressive.

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(7) (continued)				Aging Mechanism: Corrosion of Embedded Steel Aging Effect: Loss of Material, Cracking, Loss of Bond Evaluation: Since the embedded steel is not exposed to an environment which is considered aggressive, loss of material, cracking, and loss of bond due to corrosion of embedded steel are not probable aging effects at Ginna Station and have not been observed to date. Aging Mechanism: Leaching of Calcium Hydroxide Aging Effect: Change in Material Properties Evaluation: The original construction specifications met the intent of ACI 201.2R. Change in material properties due to leaching of calcium hydroxide is not a probable aging effect at Ginna Station and has not been observed to date. Operating experience has shown that concrete has not experienced unanticipated aging effects at Ginna Station. That notwithstanding, the identification of the above aging effects by the ASME Section XI, Subsections IWE & IWL Inservice Inspection Program, as well as the resistance provided by the materials of construction provide adequate assurance that all types of concrete aging effects will be identified and managed through out the extended period of operation.

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(8) Concrete elements: foundation	Cracks, distortion, and increases in component stress level due to settlement	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program	Consistent with NUREG-1801. Cracks, distortion, and increase in component stresses due to settlement of concrete foundations are considered in the Structures Monitoring Program . All structures at Ginna Station are either founded on bedrock, steel foundation piles that are driven to bedrock, or have foundations that consist of caissons extending to bedrock. Structural inspections indicate no visible evidence of settlement since construction of the station. During the Systematic Evaluation Program, the NRC concluded that settlement of foundations and buried equipment is not a safety concern for Ginna Station. Cracking, distortion, and an increase in component stress levels due to settlement are not probable aging effects at Ginna Station and have not been observed to date. That notwithstanding, the Structures Monitoring Program monitors for cracks and distortion and contains inspection criteria to verify these aging effects are not developing.
(9) Concrete elements: foundation	Reduction in foundation strength due to erosion of porous concrete subfoundation	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program	Consistent with NUREG-1801. Reduction in foundation strength due to erosion of porous concrete subfoundations is not an aging effect requiring management at Ginna. Ginna Station's structure foundations are constructed of normal concrete and not the subject porous type, nor are foundations subject to flowing water. That notwithstanding, the Structures Monitoring Program monitors for settlement and cracking. The identification of indications of settlement by the Structures Monitoring Program , as well as the resistance provided by the materials of construction, provide adequate assurance that reductions in foundation strength for any reason will be identified and managed through out the extended period of operation.

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(10) Concrete elements: foundation, dome, and wall	Reduction of strength and modulus due to elevated temperature	Plant specific	Yes, for any portions of concrete containment that exceed specified temperature limits	Consistent with NUREG-1801. For plant areas of concern, temperatures are normally maintained below the specified limits; therefore, loss of material, cracking, and change in material properties due to elevated temperature at Ginna Station have not been observed to date. (Note: The SSCs relied upon to maintain the concrete surrounding containment penetrations and the reactor vessel support pad within specified temperature limits are within the scope of the License Renewal Rule, i.e. penetration cooling and component cooling water.) That notwithstanding, the ASME Section XI, Subsections IWE & IWL Inservice Inspection Program monitors for loss of material, cracks, and changes in material properties and contains inspection criteria to verify these aging effects are not developing.
(11) Prestressed containment: tendons and anchorage components	Loss of prestress due to relaxation, shrinkage, creep, and elevated temperature	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	Consistent with NUREG-1801. Loss of prestress is addressed as a TLAA in Section 4.3. Additionally, the ASME Section XI, Subsections IWE & IWL Inservice Inspection Program implements and formally adopts the requirements of the ASME Section XI, Subsections IWE & IWL Inservice Inspection Program as part of the Ginna Station Inservice Inspection Program. Included in the scope of the IWL program are all exterior exposed accessible areas and exterior suspect areas of the concrete containment, and the post-tensioning system. Tendon Surveillance Program procedure PT-27.2 performs periodic liftoff tests, grease analysis, and visual inspection of the top tendon anchorage hardware and thus provides reasonable assurance that loss of prestress due to stress relaxation, shrinkage, creep and elevated temperature and loss of material due to corrosion are effectively managed.

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(12) Steel elements: liner plate, containment shell	Loss of material due to corrosion in accessible and inaccessible areas	Containment ISI and Containment leak rate test	Yes, if corrosion is significant for inaccessible areas	Consistent with NUREG-1801. The ASME Section XI, Subsections IWE & IWL Inservice Inspection Program includes inspections and leak rate tests which would indicate the presence of significant degradation due to loss of material from all applicable corrosion mechanisms. Additionally, plant operating experience has shown that borated water spills in containment have the potential to impact the containment liner. Accordingly, the Boric Acid Corrosion Program is also credited with assessing and managing loss of material in the containment liner. Additional (non-pressure retaining) structural steel evaluations not accounted for by other items in this table are found in Table 3.6-2 Line Number (15) and Table 3.6-2 Line Number (17)
(13) Steel elements: protected by coating	Loss of material due to corrosion in accessible areas only	Protective coating monitoring and maintenance	No	Protective coatings are not credited with managing the effects of aging at Ginna Station. Ginna recognizes the benefits derived from protective coatings. However coatings, in and of themselves, do not perform License Renewal intended functions. That notwithstanding, steel elements in containment are inspected for corrosion by both the Structures Monitoring Program and the ASME Section XI, Subsections IWE & IWL Inservice Inspection ISI Program. When a steel coating is found degraded it is evaluated and repaired in accordance with station procedures.

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(14) Prestressed containment: tendons and anchorage components	Loss of material due to corrosion of prestressing tendons and anchorage components	Containment ISI	No	Consistent with NUREG-1801. The Ginna Station ISI program has been revised to implement and formally adopt ASME Section XI, Subsections IWE & IWL Inservice Inspection . Included in the scope of the IWL program are all exterior exposed accessible areas and exterior suspect areas of the concrete containment, and the post-tensioning system. Tendon Surveillance Program procedure PT-27.2 performs periodic liftoff tests, grease analysis, and visual inspection of the top tendon anchorage hardware and thus provides reasonable assurance that loss of prestress due to stress relaxation, shrinkage, creep and elevated temperature as well as loss of material due to corrosion, are effectively managed. Ginna Station has a tendon anchor system that includes the use of rock anchors. Rock anchors basically consist of high strength steel wires and button heads grouted directly into the bedrock under containment. The Structures Monitoring Program requires periodic measurement of the galvanic potential between the anchor and the environment. Should the rock anchor/ tendon assemblies indicate electrical current flow of sufficient magnitude to support corrosion, mitigative measures can be taken to protect the rock anchors by application of a cathodic potential. Additionally, due to operating experience (grease loss, water intrusion), the Periodic Surveillance and Preventive Maintenance Program requires visual inspection of the containment tendon grease cans for evidence of loss of seal (water intrusion) due to changes in gasket elastomeric properties, loss of grease due to leakage, and structural soundness of the exposed fill port pipe segments encapsulated in epoxy plastic. Thus, by ensuring the tendons are not exposed to moisture, the Periodic Surveillance and Preventive Maintenance Program also ensures loss of material due to corrosion of prestressing tendons and anchorage components is effectively managed.

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(15) Concrete elements: foundation, dome, and wall	Scaling, cracking, and spalling due to freeze-thaw; expansion and cracking due to reaction with aggregate	Containment ISI	No	Consistent with NUREG-1801. The ASME Section XI, Subsections IWE & IWL Inservice Inspection Program identifies the evidence that an aging mechanism is present and active and also provides confirmation and verification of the absence of all types of aging effects. Indication of aging effects may be absent if the materials of construction, design specifications, and operational environment preclude an aging mechanism but, due to the long lead time necessary for some effects to manifest themselves, it is prudent to periodically assess the condition of SSCs regardless of the likelihood that a particular aging mechanism is applicable. The degradation of inaccessible concrete can create symptoms of aging effects that are detectable in accessible areas. Conversely, if aging effects are present in accessible areas it is sensible to extrapolate those effects into inaccessible areas and perform additional evaluations. Containment concrete elements have been evaluated for the following aging mechanisms: Aging Mechanism: Freeze-Thaw Aging Effect: Loss of Material Evaluation: The contract-specified air contents are within the range specified by current revisions of ACI 318, and the contract-specified water-to-cement ratio meets the recommendations of ACI 318-63 ($\leq$ 0.53). Therefore, loss of material and cracking of concrete due to freeze-thaw are not probable aging effects at Ginna Station and have not been observed to date. Aging Mechanism: Reaction with Aggregates Aging Effect: Cracking, Expansion Evaluation: During construction the aggregates were tested for potential reactivity in accordance with ASTM C227 and ASTM C295, cracking and expansion due to reaction with aggregates are not probable aging effects at Ginna Station and have not been observed to date.

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(15) (continued)				Operating experience has shown that concrete has not experienced unanticipated aging effects at Ginna Station. That notwithstanding, the identification of the above aging effects by the ASME Section XI, Subsections IWE & IWL Inservice Inspection Program, as well as the resistance provided by the materials of construction provide adequate assurance that all types of concrete aging effects will be identified and managed through out the extended period of operation.

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(16) All Groups except Group 6: accessible interior/exterior concrete & steel components	All types of aging effects	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program	Consistent with NUREG-1801. The Structures Monitoring Program identifies the evidence that an aging mechanism is present and active and also provides confirmation and verification of the absence of all types of aging effects. Indication of aging effects may be absent if the materials of construction, design specifications, and operational environment preclude an aging mechanism but, due to the long lead time necessary for some effects to manifest themselves, it is prudent to periodically assess the condition of SSCs regardless of the likelihood that a particular aging mechanism is applicable. The degradation of inaccessible concrete can create symptoms of aging effects that are detectable in accessible areas. Conversely, if aging effects are present in accessible areas it is sensible to extrapolate those effects into inaccessible areas and perform additional evaluations. Accessible interior and exterior concrete have been evaluated for the following aging mechanisms: Aging Mechanism: Freeze-Thaw Aging Effect: Loss of Material Evaluation: The contract-specified air contents are within the range specified by current revisions of ACI 318, and the contract-specified water-to-cement ratio meets the recommendations of ACI 318-63 ($\leq$ 0.53). Therefore, loss of material and cracking of concrete due to freeze-thaw are not probable aging effects at Ginna Station and have not been observed to date. Aging Mechanism: Elevated Temperature Aging Effect: Loss of Material, Cracking, Changes in Material Properties Evaluation: For plant areas of concern, temperatures are normally maintained below the specified limits; therefore, loss of material, cracking, and change in material properties due to elevated temperature are not probable aging effects at Ginna Station and have not been observed to date. (Note: The SSCs

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(16) (continued)				relied upon to maintain the concrete surrounding containment penetrations and the reactor vessel support pad within specified temperature limits are within the scope of the License Renewal Rule, i.e. penetration cooling and component cooling water.) Aging Mechanism: Aggressive Chemical Attack Aging Effect: Loss of Material, Changes in Material Properties Evaluation: Concrete degradation in air due to aggressive rainwater is insignificant and the below-grade/lake water environment is non-aggressive. Additionally, recent structural inspections revealed no evidence of degradation owing to aggressive chemical attack; therefore, loss of material and change in material properties due to aggressive chemical attack are not probable aging effects at Ginna Station and have not been observed to date. The Structures Monitoring Program requires periodic monitoring of ground/lake water to verify chemistry remains non-aggressive. Aging Mechanism: Corrosion of Embedded Steel Aging Effect: Loss of Material, Cracking, Loss of Bond Evaluation: Since the embedded steel is not exposed to an environment which is considered aggressive, loss of material, cracking, and loss of bond due to corrosion of embedded steel are not probable aging effects at Ginna Station and have not been observed to date. Aging Mechanism: Reaction with Aggregates Aging Effect: Cracking, Expansion Evaluation: During construction the aggregates were tested for potential reactivity in accordance with ASTM C227 and ASTM C295, cracking and expansion due to reaction with aggregates are not probable aging effects at Ginna Station and have not been observed to date.

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(16) (continued)				Aging Mechanism: Settlement Aging Effect: Cracking, Distortion, Increase in Component Stress Level Evaluation: All structures at Ginna Station are either founded on bedrock, steel foundation piles that are driven to bedrock, or have foundations that consist of caissons extending to bedrock. Structural inspections indicate no visible evidence of settlement since construction of the station. During the Systematic Evaluation Program, the NRC concluded that settlement of foundations and buried equipment is not a safety concern for Ginna Station. Cracking, distortion, and an increase in component stress levels due to settlement are not probable aging effects at Ginna Station and have not been observed to date. Aging Mechanism: Leaching of Calcium Hydroxide Aging Effect: Change in Material Properties Evaluation: The original construction specifications met the intent of ACI 201.2R. Change in material properties due to leaching of calcium hydroxide is not a probable aging effect at Ginna Station and has not been observed to date. Operating experience has shown that concrete has not experienced unanticipated aging effects at Ginna Station. That notwithstanding, the identification of the above aging effects by the Structures Monitoring Program , as well as the resistance provided by the materials of construction provide adequate assurance that all types of concrete aging effects will be identified and managed through out the extended period of operation. Accessible interior and exterior steel components have been evaluated for the following aging mechanisms: Aging Mechanism: General Corrosion Aging Effect: Loss of Material Evaluation: Carbon and low-alloy steel surfaces, which are exposed to typical plant environments, can experience general

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(16) (continued)				corrosion. Additionally, structural steel can be subject to boric acid corrosion. Corrosion of accessible interior and exterior structural steels is an aging effect that requires management at Ginna Station. Operating experience has shown that corrosion can be initiated and/or accelerated by unique factors. Although Ginna Station is not located in an area subject to high ambient chloride or sulfate ion concentrations, localized factors such as the deposition of bird excrement on exterior structural steel have been shown to create adverse conditions that promulgate corrosion. Accordingly, the Structures Monitoring Program identifies and evaluates corrosion of interior and exterior structural steel. Additionally, accessible carbon low alloy structural steel located in areas that contain borated water systems are subject to the requirements of Boric Acid Corrosion Program.

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(17) Groups 1-3, 5, 7-9: inaccessible concrete components, such as exterior walls below grade and foundation	Aging of inaccessible concrete areas due to aggressive chemical attack, and corrosion of embedded steel	Plant-specific	Yes, if an aggressive below-grade environment exists	Consistent with NUREG-1801. Inaccessible wall and concrete foundations are considered in the Structures Monitoring Program . Results of inspections for accessible concrete are evaluated and, if aging effects are noted, the Structures Monitoring Program evaluates the symptom and possible causes with respect inaccessible areas. The Structures Monitoring Program requires periodic monitoring of ground water to verify chemistry remains non-aggressive. Concrete degradation in air due to aggressive rainwater is insignificant and the below-grade/lake water environment is non-aggressive. Additionally, recent structural inspections revealed no evidence of degradation owing to aggressive chemical attack; therefore, degradation due to chemical attack is not a probable aging effect at Ginna Station. The concrete at Ginna Station was designed in accordance with ACI 301-66 or ACI 318-63. ACI 301-66 refers to ACI 318 for concrete reinforcement. Designing concrete to ACI 318 also provides for sufficient concrete cover over embedded steel to provide ample corrosion protection. Chemical analyses performed on the rock and groundwater indicate these environments are non-aggressive. Since the embedded steel is not exposed to an environment which is considered aggressive, corrosion of embedded steel is not a probable aging effect at Ginna Station and has not been observed to date.

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(18) Group 6: all accessible/ inaccessible concrete, steel, and earthen components	All types of aging effects, including loss of material due to abrasion, cavitation, and corrosion	Inspection of Water-Control Structures or FERC/US Army Corps of Engineers dam inspections and maintenance	No	Ginna Station water control structures include the circulating water system discharge canal, the canal's interface with the pump screen house, and a stone revetment which protects the site from surge flooding. The water control structure inspections are performed in accordance with the Periodic Surveillance and Preventive Maintenance Program and the Structures Monitoring Program . Although the components and aging attributes monitored are consistent with NUREG-1801, the program assignment is not. Accordingly the components that comprise water control structures (Group 6 items) are detailed and evaluated in Table 3.6-2 Line Number (7) .
(19) Group 5: liners	Crack initiation and growth from SCC and loss of material due to crevice corrosion	Water Chemistry Program and Monitoring of spent fuel pool water level	No	Consistent with NUREG-1801. The Water Chemistry Control Program is credited with managing the aging effects of crack initiation and growth from SSC and loss of material due to crevice corrosion for the spent fuel pool liner and refueling transfer canal liner. Plant Technical Specification, 3.7.11 Spent Fuel Pool (SFP) Water Level, as well as plant operating procedures provide monitoring and control of the spent fuel pool water level.
(20) Groups 1-3, 5, 6: all masonry block walls	Cracking due to restraint, shrinkage, creep, and aggressive environment	Masonry Wall	No	Consistent with NUREG-1801. Masonry wall inspections are incorporated into the Structures Monitoring Program . The Structures Monitoring Program effectively manages cracking due to restraint, shrinkage and creep. Concrete degradation in air due to aggressive rainwater is insignificant and the below-grade/lake water environment is non-aggressive. Additionally, recent structural inspections revealed no evidence of degradation owing to aggressive chemical attack; therefore, degradation due to chemical attack is not a probable aging effect for concrete and masonry block walls. That notwithstanding, the Structures Monitoring Program monitors for indications of chemical attack on masonry walls.

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(21) Groups 1-3, 5, 7-9: foundation	Cracks, distortion, and increases in component stress level due to settlement	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program	Consistent with NUREG-1801. Cracks, distortion, and increase in component stresses due to settlement of concrete foundations are considered in the Structures Monitoring Program . All structures at Ginna Station are either founded on bedrock, steel foundation piles that are driven to bedrock, or have foundations that consist of caissons extending to bedrock. Structural inspections indicate no visible evidence of settlement since construction of the station. During the Systematic Evaluation Program, the NRC concluded that settlement of foundations and buried equipment is not a safety concern for Ginna Station. Cracking, distortion, and an increase in component stress levels due to settlement are not probable aging effects at Ginna Station and have not been observed to date. That notwithstanding, the Structures Monitoring Program monitors for cracks and distortion and contains inspection criteria to verify these aging effects are not developing.
(22) Groups 1-3, 5-9: foundation	Reduction in foundation strength due to erosion of porous concrete subfoundation	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program	Consistent with NUREG-1801. Reduction in foundation strength due to erosion of porous concrete subfoundations is not an aging effect requiring management at Ginna. Ginna Station's structure foundations are constructed of normal concrete and not the subject porous type, nor are foundations subject to flowing water. That notwithstanding, the Structures Monitoring Program monitors for settlement and cracking. The identification of indications of settlement indication by the Structures Monitoring Program , as well as the resistance provided by the materials of construction, provide adequate assurance that reductions in foundation strength for any reason will be identified and managed through out the extended period of operation.

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(23) Groups 1-5: concrete	Reduction of strength and modulus due to elevated temperature	Plant-specific	Yes, for any portions of concrete that exceed specified temperature limits	Consistent with NUREG-1801. For plant areas of concern, temperatures are normally maintained below the specified limits; therefore, loss of material, cracking, and change in material properties due to elevated temperature at Ginna Station have not been observed to date. (Note: The SSCs relied upon to maintain the concrete surrounding containment penetrations and the reactor vessel support pad within specified temperature limits are within the scope of the License Renewal Rule, i.e. penetration cooling and component cooling water.) That notwithstanding, the Structures Monitoring Program monitors for loss of material, cracks, and changes in material properties and contains inspection criteria to verify these aging effects are not developing.
(24) Groups 7, 8: liners	Crack Initiation and growth due to SCC; Loss of material due to crevice corrosion	Plant-specific	Yes	All tanks within the scope of License Renewal receive their aging management evaluation with the system they serve. Thus, this line item is not applicable to class 1 structures at Ginna Station.

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(25) All Groups: support members: anchor bolts, concrete surrounding anchor bolts, welds, grout pad, bolted connections, etc.	Aging of component supports	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program	Consistent with NUREG-1801. (Note: Equipment included in component support groups B1.1and B1.2 (ASME class 1,2 and 3 supports), as well as the effects of boric acid corrosion on all groups, are also discussed as separate items in the following sections. Group B1.3 is applicable to BWRs.) The aging effects associated with component supports are considered in the Structures Monitoring Program . Additionally, component supports submerged in raw water are considered in the Periodic Surveillance and Preventive Maintenance Program. Component supports include those structural elements that are connected to civil structures and which extend to a system or system components for the purpose of providing support or restraint. Inclusive in this boundary definition are any vibration dampeners or other passive connective appurtenances intrinsic to the functioning of the support. The group also includes spray or drip shields attached to equipment as well as electrical system rack, panels and enclosures. Component supports are located throughout the plant. Included in the evaluation of the component supports are supports for both safety-related components and non-safety related components whose failure could affect a safety function (typically referred to as seismic II/I). Component supports including support members; anchor bolts, concrete surrounding anchor bolts, welds, grout pad, bolted connections, etc. have been evaluated for the following aging mechanisms: Aging Mechanism: Environmental Corrosion Aging Effect: Loss of material Evaluation: Carbon and low-alloy steels, which are exposed to typical plant environments, can experience general corrosion. The Structures Monitoring Program and Periodic Surveillance and Preventive Maintenance Program identify and evaluate corrosion of component supports.

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(25) (continued)				Aging Mechanism: Service-induced cracking or other concrete aging mechanisms Aging Effect: Reduction in concrete anchor capacity due to local concrete degradation Evaluation: Operating experience has shown that service induced cracking can occur in grouted foundations. The Structures Monitoring Program and Periodic Surveillance and Preventive Maintenance Program identify and evaluate cracking and other concrete aging mechanisms for component supports. Aging Mechanism: Degradation of vibration isolation elements Aging Effect: Reduction/loss of isolation function Evaluation: Operating experience has shown that elastomer materials can degrade over time. The Structures Monitoring Program identifies and evaluates the degradation of vibration isolation elements. Aging Mechanism: Metal Fatigue Aging Effect: Cracking Evaluation: The metals and welds used in component supports and support fasteners are subjected to both service induced and, potentially, unanticipated or upset loads. Cracking due to fatigue in component supports is not a probable aging effect at Ginna Station and has not been observed to date. That notwithstanding, the Structures Monitoring Program and Periodic Surveillance and Preventive Maintenance Programs look for metal discontinuities and cracks that may be evidence of fatigue.
(26) Groups B1.1, B1.2, and B1.3: support members: anchor bolts, welds	Cumulative fatigue damage (CLB fatigue analysis exists)	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	A fatigue analysis for structures and component supports is not incorporated into Ginna Station's current licensing basis. Consequently, this line item is not applicable to Ginna Station.

Table 3.6-1 Structures and Component Supports - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(27) All Groups: support members: anchor bolts, welds	Loss of material due to boric acid corrosion	Boric acid corrosion	No	Consistent with NUREG-1801. The Boric Acid Corrosion Program monitors for loss of material due to boric acid corrosion in all plant areas that contain systems that use boric acid. In addition to support members, the program also monitors and evaluates structural members, fasteners and welds that could be potentially exposed to borated water leaks.
(28) Groups B1.1, B1.2, and B1.3: support members: anchor bolts, welds, spring hangers, guides, stops, and vibration isolators	Loss of material due to environmental corrosion; loss of mechanical function due to corrosion, distortion, dirt, overload, etc.	ISI	No	Consistent with NUREG-1801. The ASME Section XI, Subsection IWF Inservice Inspection Program monitors all elements of safety related supports for degradation and fouling. Visual examinations inspect for corrosion, deformation, misalignment, improper clearances, damage to sliding surfaces, and missing detached, or loose support items. Additionally, some non-ASME supports are also included in the scope of the Inservice Inspection Program (e.g. selected high energy line pipe supports).
(29) Group B1.1: high strength low-alloy bolts	Crack initiation and growth due to SCC	Bolting integrity	No	Consistent with NUREG-1801. The Bolting Integrity Program includes the use of ISI to evaluate and monitor crack initiation and growth due to SSC in high strength low-alloy steel bolts used in NSSS component supports.

Table 3.6-2 Structures and Component Supports - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(1) Aluminum-Indoor This generic asset includes aluminum used in flood barriers and aluminum conduit protected from the weather.	Aluminum	Indoor (No Air Conditioning)	No Aging Effects	Structures Monitoring Program	The aging effects of flood barriers and electrical conduit in areas containing safety related equipment are considered in the Structures Monitoring Program. Aluminum components in indoor air have no aging effects. That notwithstanding, the Structures Monitoring Program verifies that no unforeseen aging mechanisms are causing aluminum degradation.
(2) ARCH-EXT The generic asset ARCH-EXT represents the non-safety weather barrier system that provides shelter for safety related equipment from the elements and allows for building habitability control. Load bearing frame members are evaluated in the building SS(CS)-INT asset.	ARCH-EXT includes the non-load bearing building elements not relied upon in the safety analysis which provide normal habitability control and weather proofing, e.g., building siding, built up roof systems, windows, etc. Materials vary but generally include: Carbon Steel siding, Elastomer roof membranes, Build-up roof materials (insulating materials, stone, etc.) Glass Windows, Aluminum Flashing, Scuppers, downspouts, etc.	Outdoor (exposed to the weather)	Hardening and Shrinkage due to Weathering, Loss of Material due to General Corrosion, Cracking due to Restraint, Shrinkage and Creep	Structures Monitoring Program	The aging effects of architectural materials used in areas containing vital equipment are considered in the Structures Monitoring Program. Architectural materials provide weather resistance and habitability control. By verifying the material condition of building roofing and siding systems the Structures Monitoring Program ensures adverse weather conditions will not introduce unanticipated failures in vital plant systems.

Table 3.6-2 Structures and Component Supports - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(3) Cast Iron - Indoor (Control Building de-watering Flapper Valve)	Cast Iron	Indoor (No Air Conditioning)	Loss of Material due various mechanisms.	Structures Monitoring Program	The aging effects of indoor cast iron components are considered in the Structures Monitoring Program. The Structures Monitoring Program verifies the integrity of the Control Building HELB pressurization wall. The wall includes the flapper valve that is credited in the CLB with ensuring a service or fire water line break in the control building mechanical equipment room will not cause a water buildup that has an adverse effect on the adjoining battery room.
(4) Cast Iron - Outdoor (Duct Bank manhole covers, roof drain pipes)	Cast Iron	Outdoor (exposed to the weather)	Loss of Material due various mechanisms.	Structures Monitoring Program	The aging effects of outdoor cast iron components are considered in the Structures Monitoring Program. The Structures Monitoring Program evaluates essential yard components to ensure that structural integrity is not degraded.
(5) Elastomer- Indoor (Door seals, flood barrier seals, refueling cavity seal, seismic joint (gap) filler, racks/panels/electrical enclosure gaskets and seals, etc.)This generic asset includes all elastomer (e.g., vibration isolator equipment mounts) that is indoor (i.e., protected from the weather). Also included in this evaluation are cabinet door seals, gaskets, and other seals. Fire barrier sealing material is evaluated as a separate commodity group.	Elastomers (Butyl rubber, Neoprene, Nitrile Rubber, Silicone Rubber, etc.)	Indoor (No Air Conditioning)	Cracking due to Thermal Stress, Cracking due to Ultraviolet Radiation and Ozone, Change in Material Properties due to Thermal Stress	Structures Monitoring Program Systems Monitoring	The aging effects of indoor elastomers used in civil features and non-NSSS system component supports are considered in the Structures Monitoring Program. The Structures Monitoring Program mandates inspection of the civil features and SSCs relied upon in the CLB to protect the pubic. These inspections include verification of the integrity of elastomer materials used in the associated SSC. The aging effects of indoor elastomers used in cabinet seals and spray shields for electrical enclosures are considered in the Systems Monitoring Program. The condition of the refueling cavity seal is evaluated prior to installation.

Table 3.6-2 Structures and Component Supports - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(6) Elastomer- Outdoor (Flood barrier door seals)	Elastomers (Butyl rubber, Neoprene, Nitrile Rubber, Silicone Rubber, etc.)	Outdoor (exposed to the weather)	Cracking due to Thermal Stress, Cracking due to Ultraviolet Radiation and Ozone, Change in Material Properties due to Thermal Stress	Structures Monitoring Program	The aging effects of outdoor elastomers used in civil features is considered in the Structures Monitoring Program. The Structures Monitoring Program mandates inspection of the civil features and SSCs relied upon in the CLB to protect the public. These inspections include verification of the integrity of elastomer materials used in the associated SSC.

Table 3.6-2 Structures and Component Supports - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(7) Structures and Component Supports A6. Group 6 Structures (Water-Control Structures) All accessible/ inaccessible concrete, steel and earthen components. (Embedded steel, reinforcement, and the embedded portion of anchor bolts are included.)	Concrete-Submerged, Concrete-Outdoor (exposed to the weather)	Submerged in raw water, outdoor (exposed to the weather)	All types of aging effects, including loss of material due to abrasion, cavitation, and corrosion.	Periodic Surveillance and Preventive Maintenance Structures Monitoring Program	Concrete - Exterior above and below grade and exposed to flowing water: The Structures Monitoring Program and Periodic Surveillance and Preventive Maintenance Program identifies the evidence that an aging mechanism is present and active and also provides conformation and verification of the absence of all types of aging effects. Indication of aging effects may be absent if the materials of construction, design specifications, and operational environment preclude an aging mechanism but, due to the long lead time necessary for some effects to manifest themselves, it is prudent to periodically assess the condition of SSCs regardless of the likelihood that a particular aging mechanism is present. The degradation of inaccessible concrete can create symptoms of aging effects that are detectable in accessible areas. Conversely, if aging effects are present in accessible areas it is sensible to extrapolate those effects into inaccessible areas and perform additional evaluations. Concrete used in water control structures has been evaluated for the following aging mechanisms: Aging Mechanism: Freeze-Thaw Aging Effect: Loss of Material Evaluation: The contract-specified air contents are within the range specified by current revisions of ACI 318, and the contract-specified water-to-cement ratio meets the recommendations of ACI 318-63 ($\leq$ 0.53). Therefore, loss of material and cracking of concrete due to freeze-thaw are not probable aging effects at Ginna Station and have not been observed to date.

Table 3.6-2 Structures and Component Supports - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(7) (continued)					Aging Mechanism: Leaching of Calcium Hydroxide Aging Effect: Change in Material Properties (Increase in porosity and permeability, loss of strength) Evaluation: The original construction specifications met the intent of ACI 201.2R. Change in material properties due to leaching of calcium hydroxide is not a probable aging effect at Ginna Station and has not been observed to date. Aging Mechanism: Reaction with Aggregates, Aging Effect: Cracking, Expansion Evaluation: During construction the aggregates were tested for potential reactivity in accordance with ASTM C227 and ASTM C295, cracking and expansion due to reaction with aggregates are not probable aging effects at Ginna Station and have not been observed to date. Aging Mechanism: Corrosion of Embedded Steel Aging Effect: Loss of Material, Cracking, Loss of Bond Evaluation: Since the embedded steel is not exposed to an environment which is considered aggressive, loss of material, cracking, and loss of bond due to corrosion of embedded steel are not probable aging effects at Ginna Station and have not been observed to date. The concrete at Ginna Station was designed in accordance with ACI 301-66 or ACI 318-63. ACI 301-66 refers to ACI 318 for concrete reinforcement. Designing concrete to ACI 318 also provides for sufficient concrete cover over embedded steel to provide ample corrosion protection. Chemical analyses performed on the rock and groundwater indicate these environments are non-aggressive. Since the embedded steel is not exposed to an environment that is considered aggressive, corrosion of embedded steel is not a probable aging effect at

Table 3.6-2 Structures and Component Supports - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(7) (continued)					Ginna Station and has not been observed to date. Aging Mechanism: Aggressive Chemical Attack Aging Effect: Loss of Material (spalling and scaling), Changes in Material Properties (Increase in porosity and permeability, cracking) Evaluation: Concrete degradation in air due to aggressive rainwater is insignificant and the below-grade/lake water environment is non-aggressive. Additionally, recent structural inspections revealed no evidence of degradation owing to aggressive chemical attack; therefore, loss of material and change in material properties due to aggressive chemical attack are not probable aging effects at Ginna Station and have not been observed to date.The Structures Monitoring Program requires periodic monitoring of ground/lake water to verify chemistry remains non-aggressive. Aging Mechanism: Settlement Aging Effect: Cracking, Distortion, Increase in Component Stress Level Evaluation: All structures at Ginna Station are either founded on bedrock, steel foundation piles that are driven to bedrock, or have foundations that consist of caissons extending to bedrock. Structural inspections indicate no visible evidence of settlement since construction of the station. During the Systematic Evaluation Program, the NRC concluded that settlement of foundations and buried equipment is not a safety concern for Ginna Station. Cracking, distortion, and an increase in component stress levels due to settlement are not probable aging effects at Ginna Station and have not been observed to date. That notwithstanding, the Structures Monitoring Program monitors for cracks and distortion and contains inspection criteria to verify these aging effects are not developing.

Table 3.6-2 Structures and Component Supports - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(7) (continued)					Aging Mechanism: Erosion of porous concrete subfoundations Aging Effect: Reduction in foundation strength, cracking, differential settlement Porous concrete - Material not used at Ginna Station Evaluation: Reduction in foundation strength due to erosion of porous concrete subfoundations is not an aging effect requiring management at Ginna. Ginna Station's structure foundations are constructed of normal concrete and not the subject porous type, nor are foundations subject to water flowing under them. That notwithstanding, the Structures Monitoring Program monitors for settlement and cracking. The identification of indications of settlement by the Structures Monitoring Program, as well as the resistance provided by the materials of construction, provide adequate assurance that reductions in foundation strength for any reason will be identified and managed through out the extended period of operation. Aging Mechanism: Cavitation Aging Effect: Loss of Material/Abrasion Evaluation: Flow velocities at the Screen House and Discharge canal are less than the values at which cavitation occur. Additionally, recent under water inspections of water control structures indicate no unusual concrete degradation due to abrasion or cavitation. Under water inspections are performed as a repetitive task as part of the Periodic Surveillance and Preventive Maintenance Program. The Periodic Surveillance and Preventive Maintenance Program also inspects for silting and fouling of water control structures.Divers and submarine mounted cameras are used to inspect

Table 3.6-2 Structures and Component Supports - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(7) (continued)					the under water surfaces of the Screen House, discharge canal, canal valves and weir gates, and the intake tunnels and structure. Results of these inspections are reviewed by qualified engineers as part of the Structures Monitoring Program. Operating experience has shown that Water Control Structures have not experienced unanticipated aging effects at Ginna Station. That notwithstanding, the identification of the above aging effects by the Structures Monitoring Program and Periodic Surveillance and Preventive Maintenance Program, as well as the resistance provided by the materials of construction provide adequate assurance that all types of concrete aging effects will be identified and managed through out the extended period of operation.
	Carbon Steel Components	Submerged in raw water, outdoor (exposed to the weather)	Loss of Material due to General Corrosion	Periodic Surveillance and Preventive Maintenance Structures Monitoring Program	Aging Mechanism: General Corrosion Aging Effect: Loss of Material Evaluation: Carbon and low-alloy steel surfaces that are exposed to outdoor and submerged environments can experience loss of material from general, pitting and crevice corrosion. The Structures Monitoring Program and Periodic Surveillance and Preventive Maintenance Programs evaluate all carbon steel surfaces used for water control structures to ensure the aging effect is not progressing at an unanticipated rate.
	Rock	Outdoor (exposed to the weather)	Erosion, Movement	Structures Monitoring Program	Large armor stones are used in a revetment that protects the plant from storm surges. The revetment received a site specific review from the Army Corp of Engineers for NRC use in the review of Systematic Evaluation Program topics: II-3.A, II-3.B and II-3.C; “Hydrology, Flooding, and Ultimate Heat Sink.” The Structures Monitoring Program and Periodic Surveillance and Preventive Maintenance Programs execute the

Table 3.6-2 Structures and Component Supports - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(7) (continued)					recommendations made by the Corp by performing surveys and inspections of the armor stone and cap rocks to ensure erosion and stone movement do not compromise the effectiveness of the water control structure. Thus, although Ginna Station does not utilize Reg. Guide 1.127, “Inspections of Water-Control Structures Associated with Nuclear Power Plants” or utilize the Army Corp of Engineers for inspections and maintenance, the activities performed by the Structures Monitoring Program and Periodic Surveillance and Preventive Maintenance Program satisfy all the appropriate criteria and provide assurance that the intended function of water control structures will be maintained through the period of extended operation. No masonry walls or earthen water control structures are used at Ginna Station.
(8) IB-LEAD-INT This generic asset represents the lead in the shielded enclosure constructed over the primary sample containment isolation valves. The lead is held in place by a steel frame.	Lead	Indoor (No Air Conditioning)	No Aging Effects	Structures Monitoring Program	Lead components in indoor air have no aging effects. That notwithstanding, the Structures Monitoring Program verifies that no unforeseen aging mechanisms are causing Lead degradation.

Table 3.6-2 Structures and Component Supports - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(9) TUNNEL-SS(CS) PILES-BURIED This generic asset represents the carbon steel piles the Cable Tunnel is founded on.	Carbon Steel	Buried (Below Grade)	Loss of Material due to General Corrosion, Loss of Material due to Pitting Corrosion, Loss of Material due to MIC	Structures Monitoring Program	Buried carbon steel components can experience loss of material from corrosion. The Cable Tunnel is founded on steel piles driven to bedrock. These piles are inaccessible. The Structures Monitoring Program evaluates the effects of pile aging by monitoring the tunnel for signs of settlement which would indicate foundation degradation. Site operating experience on sheet piles, below grade on one side and exposed to air on the other, has shown that only minimal loss of material has occurred since construction. Additionally, inspections of opportunity performed on other buried carbon steel components provide valuable information that may be used to infer the condition of inaccessible carbon steel piles. Thus, it can be concluded that the Structures Monitoring Program provides reasonable assurance that the aging effects of carbon steel piles will be managed through the period of extend operation.
(10) Pipe and valve This asset represents the Cable Tunnel escape hatch cofferdam drain.	Carbon Steel	Buried (Below Grade)	Loss of Material due to General Corrosion, Loss of Material due to Pitting Corrosion, Loss of Material due to MIC	Structures Monitoring Program	Buried carbon steel components can experience loss of material from corrosion. The Cable Tunnel has a cofferdam around its escape hatch. The hatch cover has a rain collection gutter that drains through an open valve to the transformer yard. Immediately preceding potential floods the valve is closed by operations. The pipe and valve are loosely covered in gravel. The Structures Monitoring Program is used evaluate the material condition of the valve and pipe and to verify valve operation.

Table 3.6-2 Structures and Component Supports - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(11) Wood This asset represents wood used for component supports.	Wood	Indoor (No Air Conditioning)	No Aging Effects	Structures Monitoring Program	Wood is used as a platform base for a non-safety fan assembly. An angle iron support frame retains the wood. Wood is also used as an electrical cable spacer. Research has shown that dry wood, maintained under cover and not exposed to parasites, will not rot or decay. Plant operating experience confirms this conclusion. That notwithstanding, wood used indoors is evaluated by the Structures Monitoring Program to ensure no unforeseen aging effects are causing degradation.
(12) FAST(HSLAS)-INT The generic asset FAST(HSLAS)-INT represents the exposed portion of high strength carbon steel fasteners used in the construction of the Control Building and Diesel Building high energy line break pressure resistant wall. (High strength fasteners used in NSSS component supports receive a separate evaluation.)	High Strength Low Alloy Steel	Indoor (No Air Conditioning)	Loss of Material due to General Corrosion	Structures Monitoring Program	The non-nuclear safety pressurization walls are comprised of carbon steel sheet piles welded together to form a diaphragm.To provide stiffness structural steel members are fastened with bolts horizontally across the diaphragm. The diaphragm is welded to the to the building structural steel frame to provide the required barrier. These walls are located between the Turbine Building and the Control and Diesel Generator Buildings. Inspection of the wall for signs of degradation is included in the Structures Monitoring Program.

Table 3.6-2 Structures and Component Supports - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(13) CV-BLOCK This generic asset includes all masonry block walls used in the Containment Vessel protected from the weather. The containment elevator shaft is block. Mortar is included in this asset evaluation.	Masonry Block	Indoor (No Air Conditioning)	Cracking due to restraint, shrinkage and creep	Structures Monitoring Program	Masonry wall inspections are incorporated into the Structures Monitoring Program. The Structures Monitoring Program effectively manages cracking due to restraint, shrinkage and creep.
(14) CV-INSULATION This generic asset includes the Containment Vessel thermal insulation panels.	Plastic (PVC)	Indoor (No Air Conditioning)	No Aging Effects	None Required	The liner insulation is Vinylcel as manufactured by Johns-Manville. This material is a closed-cell polyvinyl chloride foam insulation with low conductivity, low water absorption, and high strength. UFSAR Section 3.8.1.6.8, Liner Insulation, and Section 6.1.2.8.4, Vinylcel Insulation, provide an extensive discussion concerning the selection criteria and properties of the containment liner insulation. Insulation panels are occasionally removed to gain access to the containment liner. Operating experience confirms that no aging management is required. That notwithstanding containment insulation will continue to be evaluated during inspections of opportunity in accordance with the guidance given in the Structures Monitoring Program.

Table 3.6-2 Structures and Component Supports - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(15) CV-SS(CS)-INT This generic asset includes all carbon structural steel of the Containment Vessel that is protected from the weather. Columns, posts, beams, baseplates, bracing, crane support girders, crane rails, and the exposed faces of plates and structural members are included. The containment liner is evaluated as a separate asset. This evaluation does not include carbon structural steel used as component supports.	Carbon Low Alloy Steel	Indoor (No Air Conditioning)	Loss of Material due to General Corrosion, Loss of Material due to Pitting Corrosion	Structures Monitoring Program Boric Acid Corrosion	Carbon and low-alloy steel surfaces, which are exposed to typical plant environments, can experience general corrosion. Additionally, structural steel can be subject to boric acid corrosion. Corrosion of structural steels is an aging effect that requires management at Ginna Station. Operating experience has shown that corrosion can be initiated and/or accelerated by unique factors. Accordingly, the Structures Monitoring Program identifies and evaluates corrosion of structural steels used in containment. Additionally, accessible carbon low alloy structural steel located in areas that contain borated water systems are subject to the requirements of Boric Acid Corrosion Program.
(16) CV-FAST(CS)-INT This generic asset includes the exposed portion of carbon steel threaded fasteners for the Containment Vessel that are protected from the weather. The exposed portion of high strength low alloy steel fasteners are evaluated in the component supports commodity group	Carbon Low Alloy Steel	Indoor (No Air Conditioning)	Loss of Material due to General Corrosion, Loss of Material due to Pitting Corrosion	Structures Monitoring Program Boric Acid Corrosion	Carbon and low-alloy steel surfaces, which are exposed to typical plant environments, can experience general corrosion. Additionally, structural steel can be subject to boric acid corrosion. Corrosion of structural steels is an aging effect that requires management at Ginna Station. Operating experience has shown that corrosion can be initiated and/or accelerated by unique factors. Accordingly, the Structures Monitoring Program identifies and evaluates corrosion of structural steels used in containment. Additionally, accessible carbon low alloy structural steel located in areas that contain borated water systems are subject to the requirements of Boric Acid Corrosion Program

Table 3.6-2 Structures and Component Supports - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(17) CV-SS(SS)-INT This generic asset includes all stainless structural steel of the Containment Vessel that is protected from the weather. Included in this evaluation is the refueling cavity and fuel transfer liners (including attachments). Stainless steel expansion bellows used with containment penetrations are addressed with the penetration.	Stainless Steel	Indoor (No Air Conditioning)	No Aging Effects	Periodic Surveillance and Preventive Maintenance	The stainless steel refueling cavity liners are normally maintained dry and have no aging effects that require management. That notwithstanding plant operating experience shows that when the cavity is flooded for refueling, borated water weeps from welded seams and equipment attachment points. The corrosion consequences for carbon steel components in containment from these leaks warrant inspections. The Periodic Surveillance and Preventive Maintenance Program works to implement the Boric Acid Corrosion Program and effectively manages the consequences of leaks from the cavity liners.

Section 3.6 References

1. WCAP-14756-A, Aging Management Evaluation for Pressurized Water Reactor Containment Structure, May 2001.
2. WCAP-14422, Rev. 2-A, License Renewal Evaluation: Aging Management for Reactor Coolant System Supports, December 2000.

3.7 Aging Management of Electrical and Instrument and Controls Systems

The results of the aging management review of the Electrical and Instrument and Control System components are provided in this section and summarized in Table 3.7-1 and Table 3.7-2. Table 3.7-1 shows the aging management of system components evaluated in NUREG-1801 that are relied on for license renewal of the Electrical and Instrument and Control System components at Ginna. Included in the table is a discussion column. The discussion column will provide a conclusion indicating if the aging management evaluation results are consistent with NUREG-1801 along with any clarifications or explanations required to support the stated conclusion if that conclusion is different than those of the NUREG. For a determination to be made that a table line item is "Consistent with NUREG-1801" several criteria must be met. First the plant specific component is reviewed against the GALL to ensure that the component, materials of construction and internal or external service environment are comparable to those described in a particular GALL item. Second, for those that are comparable, the results of the plant aging management review- aging effect evaluation are compared to the aging effects/mechanisms in the GALL. Finally, the programs credited in the GALL for managing those aging effects are compared to the programs invoked in the plant evaluation. If, using good engineering judgment, it could be reasonably concluded that the plant evaluation is in agreement with the GALL evaluation a line item was considered consistent with NUREG-1801. There are cases where components and component material/environment combinations and aging effects are common between a NUREG-1801 line item and the plant evaluation but the aging management program selections differ. In those cases the discussion column will indicate the plant aging management program selection but no conclusion will be made that the line item is consistent with the GALL. Table 3.7-2 contains the Electrical and Instrument and Control System components aging management review results that are not addressed in NUREG-1801. A plant component is considered not addressed by the NUREG if the component type is not evaluated in the GALL or has a different material of construction or operating environment than evaluated in the GALL. This table includes the component types, materials, environments, aging effects requiring management, the programs and activities for managing aging, and a discussion column. To avoid confusion, no attempt was made to interrelate material/environment/aging effects from one NUREG-1801 chapter to another. Note that these tables only include those components, materials and environments that are applicable to a PWR.

Materials

The materials of construction of a component have a major influence on the evaluation of aging effects applicable to the component. Sources of information used to identify materials of construction include original equipment specifications, vendor technical manuals and drawings, fabrication drawings, cable circuit schedules, design records and field walkdowns/verifications. The tables below account for the materials of construction for the components requiring an aging management review. Since similar materials are susceptible to the same aging effects/mechanisms, the tables itemize the component types (i.e., groupings) while factoring in the materials of construction. Specific materials of construction were not used to determine the scope of components in the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program. The program includes all in-scope, electrical cables and connections within specified plant spaces. Therefore further review of materials of construction was not required for electrical cables and connections.

Environment

As previously described, the environment(s) to which components are exposed are critical in the determination of potential aging mechanisms and effects. A review of plant design documentation was performed to quantify the environmental conditions to which Ginna Station equipment is exposed. This review identified that some equipment is exposed to a variety of environments. This can include normal operating conditions and post accident conditions. Since aging mechanisms and effects will be primarily driven by the environmental conditions to which equipment is exposed on a daily basis, under normal operating conditions, these conditions will differ from the design parameters which are established based upon the worst case scenario (e.g., LOCA conditions). Ginna Station equipment environments may be categorized into basic external and internal environments detailed in Section 3.1.2.

Since passive electrical components do not have internal environments in the same sense that mechanical components do, the term self-heating is used to describe the effect of resistive heating that may occur in electrical components. This heating can result in the component having a service temperature that is hotter than the ambient conditions. Self heating is only applicable to components that carry significant current, and therefore instrumentation circuits, such as Resistance Temperature Detectors (RTDs), Thermocouples, and related loop wiring is considered not to be subject to self-heating. For those components that are subject to self-heating (i.e., power cables, phase bus) the ultimate temperature is a result of the ambient temperature and the square of the ratio of the actual current to the ampacity of the conductor. Plant design guidelines do not normally permit a cable to be loaded to more than 80% of ampacity. Ginna Station has identified specific plant spaces that may lead to cables exceeding 80% of ampacity due to cable tray fill deratings. These areas have been included in the scope of the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program.

For the review of passive electrical commodities, the environments discussed in Section 3.1.2, were summarized for analysis as follows:

Environment	Temperature	Radiation
Inside Containment Sheltered	60-120°F	1 R/hr outside of loop areas, 10 R/hr inside loop areas (assumed)
Outside Containment Sheltered	60-104°F	10 mR/hr
Underground Bus Duct	32-104°F	Not Applicable
Outdoor Exposed	0-104°F	Not Applicable

Most organic materials used as electrical insulators were first evaluated against the environment "Inside Containment Sheltered" to determine if aging effects required management. This environment is considered conservative for normal ambient temperature and radiation dose.

Ginna includes limited installations of underground passive electrical components (cables). Of those, Ginna has four medium voltage power cables installed in underground duct banks. The functions of these cables were reviewed and determined that a failure of the cable would not prevent the satisfactory accomplishment of any intended function. Therefore a further review of this environment was not required.

The switchyard components and a limited number of cables are addressed in the environment of "Outdoor Exposed." These components are subject to normal environmental conditions, including precipitation.

Aging Effects Requiring Management

After the components requiring aging management review were identified and grouped by materials of construction and environment, a review of industry and plant-specific operating experience was performed. The purpose of this review was to assure that all applicable aging effects were identified, and to evaluate the effectiveness of existing aging management programs. This experience review was performed utilizing various industry and plant-specific programs and databases. Industry operating experience sources included NRC Generic Publications (including Information Notices, Circulars, Bulletins, and Generic Letters), INPO Significant Operating Event Reports (SOER), EPRI Technical Reports, and other information sources, such as the Sandia Aging Management Guidelines for Electrical Cable and Terminations, Westinghouse Generic Technical Reports (GTRs), and the Generic Aging Lessons Learned (GALL) report. Plant specific operating experience sources included Semi-annual and Annual Reports to AEC/NRC, Abnormal Occurrence and Licensee Event Reports (LERs), Non-Conformance Reports (NCRs), Corrective Action Reports (CARs), Refueling, Inspection and Overhaul Reports (RIOs), Inservice Inspection (ISI) Reports, Identified Deficiency Reports (IDRs), and ACTION Reports (ARs) from 1969 to the present. Information from these sources was compiled in various databases. Based upon the material of construction, the applicable environments, and operating experience the potential aging effects requiring management for each of the components was identified as documented in Table 3.7-1 and Table 3.7-2.

The most common aging effect for passive electrical components is electrical failure due to thermal/thermoxidative degradation of organics. Thermal life was evaluated using methodology similar to Appendix G of the Sandia Aging Management Guideline for Electrical Cables and Connections (Reference 1). In many cases, conservative assumptions were used to simplify the analysis. Thermal life was not used to determine the scope of components in the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program. The program includes all in-scope, electrical cables and connections within specified plant spaces, and adequately addresses aging effects due to thermal conditions. Therefore further review of the thermal aging results was not required for electrical cables and connections.

Electrical failure due to radiolysis and radiation induced oxidation is considered a significant aging effect only for those passive electrical components installed in containment. For these components, the moderate damage threshold for the materials were reviewed against expected radiation environments. Although the review identified very few susceptible materials, the results of the review were not used to determine the scope of the components in the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program. The program includes all in-scope, electrical cables and connections within specified plant spaces, and adequately addresses aging effects due to radiation. Therefore further review of the radiation induced aging effects was not required for electrical cables and connections.

Moisture induced electrical failure for in-scope passive electrical components is not considered to be a significant aging effect at Ginna Station. While medium voltage cables are known to experience water-treeing in a wet environment, the only environment that may qualify as wet is the underground duct banks. As discussed previously, there are no in-scope medium voltage cables in the underground duct banks. Industry and plant operating experience does not support moisture as a significant stressor for other passive electrical components. The results of this review were not used to determine the scope of the components in the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program. The program includes all in-scope, electrical cables and connections within specified plant spaces, and adequately addresses aging effects due to moisture. Therefore further review of moisture induced aging effects was not required for electrical cables and connections.

Aging effects for components not included in NUREG-1801 were identified and included on Table 3.7-2. Using basic materials properties and operating experience as a basis, these aging effects were not determined to require aging management for the period of extended operation. Nonetheless, the relevant aging effects were listed in Table 3.7-2 in the column, "AERMs" indicating that they are aging effects requiring management. Therefore the aging effects listed in the AERM column have been provided for conservatism to indicate those aging effects that have been evaluated for the components listed in the table.

Time-Limited Aging Analysis

In addition to those identified in NUREG-1801, any additional time-limited aging analyses (TLAA) identified as appropriate to the system are identified in Section 4.0.

A description of the aging management activities for this area are provided in Appendix B, along with the demonstration that the identified aging effects will be managed for the period of extended operation.

Therefore, based on the demonstrations provided in Appendix B, the effects of aging associated with the Electrical and Instrumentation and Control System components will be adequately managed so that there is reasonable assurance that the intended function(s) will be maintained consistent with the current licensing basis during the period of extended operation.

Conclusion

The programs and activities selected to manage the aging effects of the Electrical and Instrument and Control Systems are identified in Table 3.7-1 and Table 3.7-2. A description of these aging management activities is provided in Appendix B, along with the demonstration that the identified aging effects will be managed for the period of extended operation.

Therefore, based on the demonstrations provided in Appendix B, the effects of aging associated with the Electrical and Instrument and Control Systems components will be adequately managed so that there is reasonable assurance that the intended function(s) will be maintained consistent with the current licensing basis during the period of extended operation.

Table 3.7-1 Electrical and Instrumentation and Controls Systems - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(1) Electrical equipment subject to 10 CFR 50.49 environmental qualification (EQ) requirements	Degradation due to various aging mechanisms	Environmental qualification of electric components	Yes, TLAA	Evaluation of Time-Limited Aging Analyses for EQ equipment is provided in application Section 4.4.
(2) Electrical cables and connections not subject to 10 CFR 50.49 EQ requirements	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance (IR); electrical failure caused by thermal/ thermoxidative degradation of organics; radiolysis and photolysis (ultraviolet [UV] sensitive materials only) of organics; radiation-induced oxidation; moisture intrusion	Aging management program for electrical cables and connections not subject to 10 CFR 50.49 EQ requirements	No	Consistent with NUREG-1801. The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will adequately manage the potential aging effects for this component. An analysis of material/environment combinations for normal plant environments indicates that a large majority of components have no aging effects that require management throughout the period of extended operation. Exceptions to this include PVC cables in containment subject to heating above ambient temperatures (self-heating), and cables installed in adverse localized equipment environments. However, due to plant specific operating experience, all material/environment combinations will be included in the scope of the program using an encompassing approach.

Table 3.7-1 Electrical and Instrumentation and Controls Systems - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(3) Electrical cables used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements that are sensitive to reduction in conductor insulation resistance (IR)	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced IR; electrical failure caused by thermal/ thermoxidative degradation of organics; radiation-induced oxidation; moisture intrusion	Aging management program for electrical cables used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements	No	Not consistent with NUREG-1801. Rochester Gas and Electric (RG&E) believes that invoking the NUREG-1801 XI.E1, Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program to manage the effects of aging in accessible non-EQ cable and connectors provides reasonable assurance that these SC's will perform their intended function during the period of extended operation. The aging effects of cable and connector insulation may have very long incubation periods. In essence, routine maintenance, calibration and repair activities on the active components in an instrument loop initially work to mask indications of cable and cable and connector insulation degradation. Only after the active portions of a loop can no longer be adjusted to compensate for cable and connector degradation would the passive portions of the instrument loop become suspect. Surveillance provides meaningful information, but that information is primarily used to cause changes to the active portions of an instrument loop. The predominate cause of non-event driven degradation in cable and connector insulation is thermal aging. External inspection of cables and connectors and their host environments identifies the possibility of thermal aging long before instrument loop adjustments can't compensate for current leakage. Because of this, RG&E feels that the only legitimate way to ensure the continued functioning of the long-lived passive components are those inspection activities performed under the XI.E1 program, Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements .

Table 3.7-1 Electrical and Instrumentation and Controls Systems - Aging Management Programs Evaluated in NUREG-1801 that are Relied on for License Renewal

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(4) Inaccessible medium-voltage (2 kV to 15 kV) cables (e.g., installed in conduit or direct buried) not subject to 10 CFR 50.49 EQ requirements	Formation of water trees, localized damage leading to electrical failure (breakdown of insulation); water trees caused by moisture intrusion	Aging management program for inaccessible medium-voltage cables not subject to 10 CFR 50.49 EQ requirements	No	Not Applicable. Medium voltage cables and connections subject to an aging management review are not installed in environments that lead to the formation of water trees therefore no aging management program is required. That notwithstanding, should aging effects be observed in accessible medium voltage cables the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will require that inaccessible cables be evaluated to ensure no adverse aging effects are developing
(5) Electrical connectors not subject to 10 CFR 50.49 EQ requirements that are exposed to borated water leakage	Corrosion of connector contact surfaces caused by intrusion of borated water	Boric acid corrosion	No	Consistent with NUREG-1801. Corrosion of connectors due to Boric Acid Corrosion is an aging effect requiring management. The Boric Acid Corrosion Program effectively manages corrosion of contact surfaces caused by the intrusion of borated water.

Table 3.7-2 Electrical and Instrumentation and Controls Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(1) Electrical Phase Bus	Aluminum, Copper (bus, solid and flexible connectors and straps), Steel (bolts, washers, nuts, etc.), Rigid Bus parts (porcelain insulators, etc)	Indoor (No Air Conditioning), Outdoor	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance (IR); electrical failure caused by thermal/ thermoxidative degradation of organics; moisture intrusion.	One-Time Inspection Program	Based on a materials analysis, the high reliability of the phase bus at Ginna, and partial upgrades performed on the 4KV system for redundancy purposes, there is no reason to believe that there are aging effects requiring management for electrical phase bus. A review of industry operating experience indicated that plants have experienced failures of electrical phase bus. The review concluded that most failures were not due to aging mechanisms. However as a confirmatory measure, an inspection was performed on selected portions (both indoors and outdoors) of the 4KV bus duct at Ginna Station during the 2002 RFO. The results showed that the internal components were in "like new" condition. Because no aging effects requiring management were identified, Ginna Station considers the One-Time Inspection results adequate to demonstrate that no additional aging management programs are required through the period of extended operation.

Table 3.7-2 Electrical and Instrumentation and Controls Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(2) Switchyard Bus	Copper, Copper Alloy, Steel (bolts, washers, nuts, etc.)	Outdoor	Loss of material due to corrosion leading to increased resistance.	Not Applicable	Rochester Gas and Electric's Energy Delivery Department performs inspection and maintenance of the Switchyard Bus components. Switchyard Bus components subject to aging management review contain materials that when exposed to plant operating environments could potentially lead to aging effects requiring management. Plant Operating Experience reviews have not identified any case where aging effects requiring management have developed however, evidence of aging effects may in fact be removed (masked) by ongoing routine Energy Deliver Department maintenance activities. That notwithstanding, the Energy Delivery Department inspections identify if the evidence of an aging mechanism is present and active and also provides the confirmation and verification of the absence of all types of aging effects. Indication of aging effects may be absent if the materials of construction and operational environment preclude an aging effect but, due to the long lead time necessary for some effects to manifest themselves, it is prudent to periodically assess the condition of SSCs regardless of the likelihood that a particular aging mechanism is applicable. Plant operating experience reviews show that the activities performed by the Energy Delivery Department on the Switchyard Buses are effective in managing Switchyard Bus components. The Maintenance Rule activities monitor the effectiveness of the Energy Delivery Department Activities by tracking system level performance indicators.

Table 3.7-2 Electrical and Instrumentation and Controls Systems - Component Types Subject to Aging Management not Evaluated in NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(3) High Voltage Insulators	Porcelain, Cement, Steel	Outdoor	Cracks; Loss of material due to corrosion; loss of dielectric strength leading to reduced insulation resistance (IR)	Not Applicable	Rochester Gas and Electric's Energy Delivery Department performs inspection and maintenance of the High Voltage Insulators. High Voltage Insulator components subject to aging management review contain materials that when exposed to plant operating environments could potentially lead to aging effects requiring management. Plant Operating Experience reviews have not identified any case where aging effects requiring management have developed however, evidence of aging effects may in fact be removed (masked) by ongoing routine Energy Deliver Department maintenance activities. That notwithstanding, the Energy Delivery Department inspections identify if the evidence of an aging mechanism is present and active and also provides the confirmation and verification of the absence of all types of aging effects. Indication of aging effects may be absent if the materials of construction and operational environment preclude an aging effect but, due to the long lead time necessary for some effects to manifest themselves, it is prudent to periodically assess the condition of SSCs regardless of the likelihood that a particular aging mechanism is applicable. Plant operating experience reviews show that the activities performed by the Energy Delivery Department on the High Voltage Insulators are effective in managing Phase Bus components. The Maintenance Rule activities monitor the effectiveness of the Energy Delivery Department Activities by tracking system level performance indicators.

Section 3.7 References

1. SAND 96-0344, "Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable and Terminations," Sandia National Laboratories for the U. S. Department of Energy, September 1996.
2. NUREG-1801, "Generic Aging Lessons Learned (GALL)," U. S. Nuclear Regulatory Commission, April 2001.
3. NUREG-1800, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," U. S. Nuclear Regulatory Commission, April 2001.

4.0 TIME-LIMITED AGING ANALYSES

Two areas of plant technical assessment are required to support an application for a renewed operating license. The first area of technical review is the Integrated Plant Assessment, which is described in Sections 2 and 3 of this License Renewal Application. The second area of technical review required is the identification and evaluation of plant-specific time-limited aging analyses and exemptions. The identifications and evaluations included in this section meet the requirements contained in 10 CFR 54.21(c) and provide the information necessary for the NRC to make the finding contained in 10 CFR 54.29(a)(2).

4.1 Identification of Time-Limited Aging Analyses

Title 10 of the Code of Federal Regulations, Part 54 (10 CFR 54) sets forth the requirements for License Renewal of Operating Nuclear Power Plants. Part 54.21(c)(1) of Title 10 requires a listing and an evaluation of Time-Limited Aging Analyses (TLAAs). Part 54.21(c)(2) requires a listing and evaluation of active plant-specific exemptions granted under 10 CFR 50.12 that are based on TLAAs as defined in Part 54.3(a).

4.1.1 Identification Process of Time-Limited Aging Analyses

This section documents the identification and disposition of Time-Limited Aging Analyses (TLAAs), including TLAA related exemptions granted in accordance with 10 CFR 50.12, which are applicable to Ginna Station for the period of extended operation.

Time-limited aging analyses are defined in 10 CFR 54.3 as those licensee calculations and analyses that meet the following criteria:

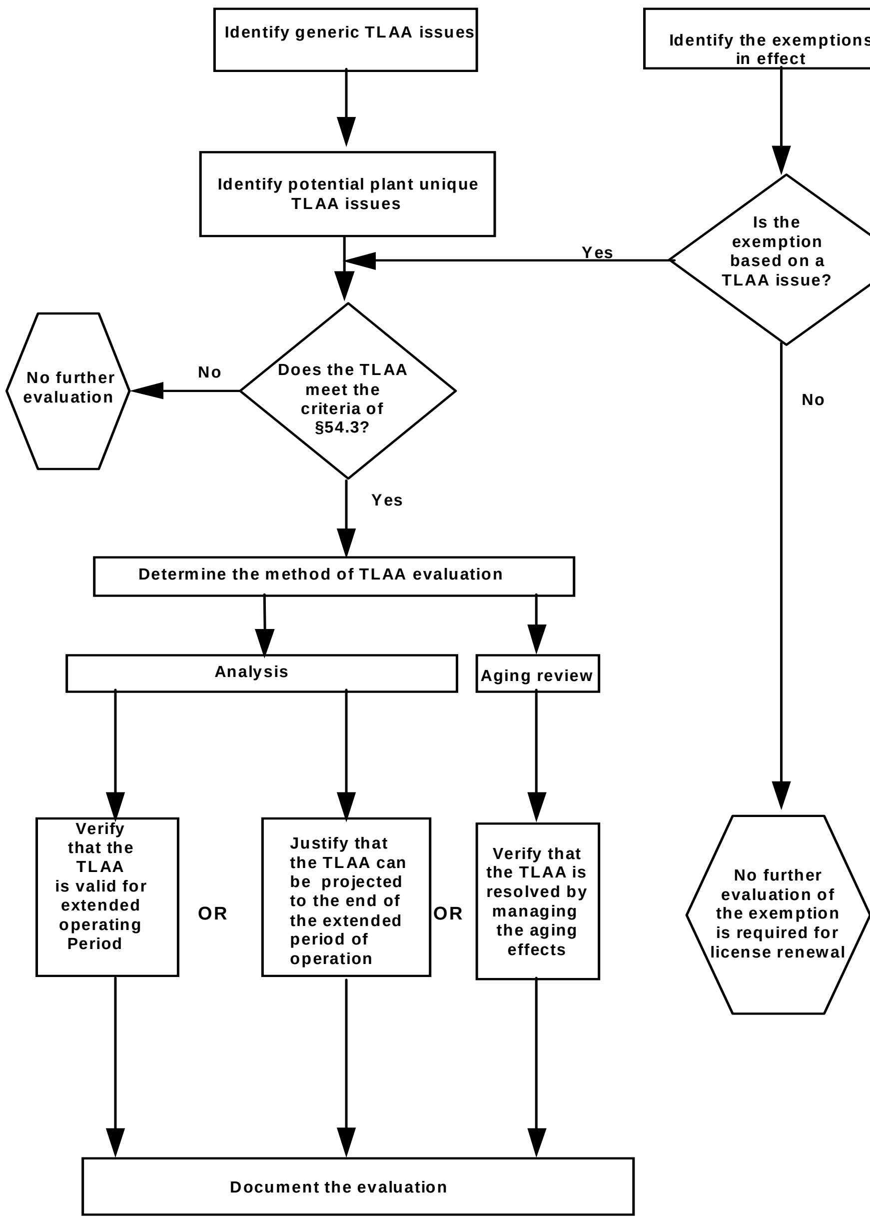
- Involve systems, structures, and components within the scope of license renewal, as delineated in 10 CFR 54.4(a);
- Consider the effects of aging;
- Involve time-limited assumptions defined by the current operating term, for example, 40 years;
- Were determined by the licensee to be relevant in making a safety determination;
- Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in10 CFR 54.4(b); and
- Are contained or incorporated by reference in the current licensing basis.

Some evaluations originally included in the Ginna Station Licensing Basis as TLAAs were re-evaluated within the current licensing period and were subsequently shown not to be time-limited for the 60 year operating period. Those evaluations have been included in this section to provide reviewers with a clear understanding of how these potential TLAAs were resolved. Affected TLAAs include Reactor Vessel Nozzle-to-Vessel Weld Defect (Section 4.3.5), Pressurizer Fracture Mechanics Analysis (Section 4.3.6) and RCP Flywheel (Section 4.7.6).

4.1.2 TLAA Methodology

This section discusses the methodology used to complete this process. Ginna Station engineering procedure EP-3-S-0717 provides the instructions for performing this identification and evaluation. The methodology provided in the procedure is consistent with NEI 95-10 (Reference 4) and NUREG-1800 (Reference 3). The overall methodology is provided in (Figure 4.1-1).

Figure 4.1-1 TLAA Methodology



Potential TLAAs are identified in two ways:

- Reviewing lists of previously identified TLAAs and choosing those generically applicable to Ginna Station for further evaluation. Primary sources of information include the following:
  - NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants (Reference 3)
  - NEI 95-10 [Revision 3], “Industry Guideline for Implementing the Requirements of 10 CFR 54 -the License Renewal Rule” (Reference 4)
  - Previous License Renewal Applications (up to October 2001)
- Searching the Ginna Station CLB for calculations/analyses which meet the definition of a TLAA. The primary sources of information include the following:
  - UFSAR
  - Technical Specifications
  - NRC Correspondence (includes SERs)
  - Gilbert and Westinghouse Correspondence
  - Configuration Management Information System (CMIS)
  - Commitment Action Tracking System (historical database)

Key Word Searches

The Ginna Station CLB documents are available in electronic format or have been electronically indexed and were keyword searched. To ensure a complete review was been performed, Engineering subject matter experts reviewed the results to provide additional assurance that all potential TLAAs were identified.

Life	Lifetime
EFPY	License
Erosion Allowance	Corrosion Allowance
Forty Years	40 Years
Cycle	Fatigue Analysis (within 10 words of) year

Documenting the search process:

For each potential TLAA, a form is initiated. This includes the assignment of a tracking number, the document under review, the document date (if applicable), the subject, and some specific information which indicates why the document was “flagged” for review.

4.1.3 Identification and Evaluation of Active Plant-specific Exemptions

All exemptions are contained within the Ginna Station Operating License. All active NRC exemptions granted under 10 CFR 50.12 were identified, documented as potential TLAAs and evaluated. No TLAA-related exemptions were found.

4.1.4 Screening of Potential Time-Limited Aging Analyses

The process of screening was performed to make a final determination of which issues/documents would be evaluated per §54.21(c)(1). This was done by applying the six criteria delineated in §54.3. The guidance used for applying the six criteria is provided below.

1. Involve systems, structures, and components within the scope of license renewal as delineated in §54.4(a). The system, structure, and component scoping step of the IPA should be performed prior to or concurrent with the TLAA identification. Alternatively, the LRE may use engineering judgement to consider a SSC within the scope of the rule using a bounding approach without a specific evaluation against §54.4(a).

2. Consider the effects of aging. The effects of aging include but are not limited to: loss of material, loss of toughness, loss of prestress, settlement, cracking, and loss of dielectric properties.

3. Involve time-limited assumptions defined by the current operating term, for example 40 years. The defined operating term should be explicit in the analysis. Simply asserting that a component is designed for a service life or plant life is not sufficient. A calculation or analysis that explicitly includes a time limit must support the assertion.

4. Were determined relevant in making a safety determination. Relevancy is a determination that is made based on a review of the information available. A calculation or analysis is relevant if it can be shown to have direct bearing on the action taken as a result of the analysis performed. Analyses are also relevant if they provide the basis for a safety determination, and in the absence of the analyses, a different conclusion may have resulted.

5. Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, or component to perform its intended functions as delineated in §54.4(b). As stated in the first criterion, the intended functions must be identified prior to or concurrent with the TLAA identification. Analyses that do not affect the intended functions of the system, structure, or components are not TLAAs.

6. Are contained or incorporated by reference in the CLB. Plant specific documents contained or incorporated by reference in the CLB include the UFSAR, SERs, Technical Specifications, the fire protection plan/hazards analyses, correspondence to and from the NRC, QA plan, or topical reports included as reference to the UFSAR. Calculations and analyses that are not in the CLB or not incorporated by reference are not TLAAs.

4.1.5 Evaluation Process of Time-Limited Aging Analyses

Once a TLAA was identified, an evaluation was performed, as required by 10 CFR 54.21(c)(1), to demonstrate that at least one of the following criteria is applicable:

i. The analyses remain valid for the period of extended operation.

ii. The analyses have been projected to the end of the period of extended operation.

iii. The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

The results of these evaluations are provided in Table 4.1-1 and are discussed in Sections 4.2 through 4.7.

Table 4.1-1 Time-Limited Aging Analysis Categories

TLAA Category	Analysis Description	Disposition	Report Section
Reactor Vessel Neutron Embittlement	Upper-Shelf Energy	(ii)	Section 4.2.1
	Pressurized Thermal Shock	(ii)	Section 4.2.2
	Pressure-Temperature (P-T) Limits	(ii)	Section 4.2.2
Metal Fatigue	ASME Boiler and Pressure Vessel Code Section III, Class 1	(i)	Section 4.3.1
	Reactor Vessel Underclad Cracking	(ii)	Section 4.3.3
	ANSI B31.1	(i)	Section 4.3.2
	Accumulator Check Valve	(i)	Section 4.3.4
	Reactor Vessel Nozzle-to-Shell Weld Defect	(i)	Section 4.3.5
	Pressurizer Fracture Mechanics Analysis	(i)	Section 4.3.6
	Environmentally Assisted Fatigue	Not a TLAA	Section 4.3.7
Environmental Qualification of Electrical Equipment	EEQ Evaluations	(i), (ii), or (iii) depending on the TLAA related components	Section 4.4
Concrete Containment Tendon Prestress	Concrete Containment Tendon Prestress	(iii)	Section 4.5
Containment Liner Plate and Penetration Fatigue	Containment Liner Plate and Penetration Fatigue	(ii)	Section 4.6

Table 4.1-1 Time-Limited Aging Analysis Categories

Other Plant Specific TLAAs	Containment Liner Stress	(iii)	Section 4.7.1
	Containment Tendon Fatigue	(i)	Section 4.7.2
	Containment Liner Anchorage Fatigue	(i)	Section 4.7.3
	Containment Tendon Bellows Fatigue	(i)	Section 4.7.4
	Crane Load Cycle Limit	(i)	Section 4.7.5
	Reactor Coolant Pump Flywheel	(iii)	Section 4.7.6
	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)	(ii)	Section 4.7.7

4.2 Reactor Vessel Neutron Embrittlement

The reactor vessel is subjected to neutron irradiation from the core. This irradiation results in the embrittlement of the reactor vessel materials. A neutron fluence calculation has been performed as part of the RCS pressure-temperature operating limits analysis and subsequently been used as a basis for fluence values used in other Reactor Vessel Neutron Embrittlement Analyses. The methodology used to perform neutron fluence calculations in consistent with Regulatory Guideline 1.190. Analyses have been performed that address the following:

- Upper shelf energy
- Pressurized thermal shock
- RCS pressure-temperature operating limits

4.2.1 Upper Shelf Energy

Introduction

The Charpy upper shelf energy is associated with the determination of acceptable Reactor Vessel toughness during the license renewal period. 10 CFR Part 50 Appendix G paragraph IV.A.1 requires that the reactor vessel beltline materials must have a Charpy upper shelf energy of no less than 68 J (50 ft-lb) throughout the life of the reactor vessel, unless otherwise approved by the NRC. Surveillance capsules are attached to the inside of the RPV at locations designed to provide a higher irradiation rate, thus providing an irradiation “lead” factor that allows for prediction of future vessel irradiation damage.

In the event that the 50 ft-lb requirement cannot be satisfied as stated in 10 CFR 50 Appendix G, or by alternative procedures acceptable to the NRC, the reactor vessel may continue to operate provided requirement 1 of Appendix G is satisfied. This requirement states that an analysis must conservatively demonstrate, making appropriate allowances for uncertainties, the existence of equivalent margins of safety for continued operation.

Conclusion

The upper shelf energy for the reactor vessel beltline weld material at the end of the extended period of operation is expected to decrease to less than 50 ft-lbs based on predictions using RG 1.99. A low upper-shelf fracture mechanics analysis has been performed (Reference 22) to evaluate the weld material for ASME Levels A, B, C, and D Service Loadings, based on the acceptance criteria of the ASME Code, Section XI, Appendix K.This analysis follows the same approach previously approved by the NRC in BAW-2192PA and BAW-2178PA (Reference 23 and Reference 24). The analysis demonstrates that the limiting reactor vessel beltline weld satisfies the ASME Code

Requirements of Appendix K for ductile flaw extensions and tensile stability using projected low upper-shelf Charpy impact energy levels for the weld material at 52 effective full power years of plant operation. Therefore the analysis associated with upper-shelf energy has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

4.2.2 Pressurized Thermal Shock

Introduction

The PTS rule, 10 CFR 50.61 established screening criteria that are a measure of a limiting level of reactor vessel material embrittlement beyond which operation cannot continue without further plant-specific evaluation. The screening criteria are given in terms of reference temperature RT_{PTS}. The screening criteria are 270°F for plates and axial welds, and 300°F for circumferential welds. The RT_{PTS} is defined as:

$RT_{PTS} = I + \Delta RT_{NDT} + M$

Where: I = Initial reference temperature

ΔRT_{NDT} = Mean value of adjustment in reference temperature

M = Margin

The initial reference temperature is the measured unirradiated value as defined in the ASME Code, Paragraph NB-2331. If measured values are unavailable for the heat of the material of interest, generic values may be used. The generic values are based on the data for materials of all heats that were made by the same vendor using similar processes. The generic values of initial reference temperature for welds are defined in the PTS rule and used in this analysis for conservatism.

The ΔRT_{NDT} depends upon the amount of neutron irradiation and the amounts of copper and nickel in the material. It is calculated as the product of a fluence factor and a chemistry factor. The fluence factor is calculated from the best-estimate neutron fluence at the interface of cladding, weld, and metal on the inside surface of the vessel at a location where the material receives the highest fluence at the end of the period of evaluation. The fluence value used in this analysis is based on the results of calculations performed following the guidance in Regulatory Guide 1.190. The chemistry factor may be determined using credible surveillance data or from the chemistry factor tables in the PTS rule; these tables are used in this analysis for conservatism.

The margin term is intended to account for variability in initial reference temperature and the adjustment in reference temperature caused by irradiation. The value of the margin term is dependent of whether the initial reference temperature was a measured or generic value and whether the adjustment in reference temperature was determined from credible surveillance data or from the chemistry factor tables in the PTS rule. For the purpose of this analysis, the margin term will be based on a generic value for welds fabricated using Linde 80 flux as documented in B&W Owners Group report BAW-1803, Rev. 1.

Conclusion

The results of the revised PTS analysis for the limiting material have been reviewed for compliance with 10 CFR 50.61. The methodology used in PTS analysis is based on the projected neutron fluence at the end of the period of extended operation (52 EFPY) and does not rely on plant-specific surveillance data to calculate RT_{PTS}. Although plant-specific surveillance data could have been used, generic data calculated in accordance with Regulatory Guide 1.99, Rev. 2, Position 1.1 proved to be more conservative.

Table 4.2-1 Values of RT_{PTS} at 52 EFPY - Ginna RPV Beltline Materials

Material	Heat Number	Inner Surface Fluence E19 n/cm^2	Initial RT_{NDT} °F	Margin °F	Chemistry Factor °F	Inside Surface Fluence Factor	ΔRT_{NDT} °F	RT_{PTS} °F
Intermediate Shell	125S255VA1	4.85	20	34[1]	44[1]	1.396	61.4	115.4
Lower Shell	125P666VA1	4.85	40	34[1]	31[1]	1.396	43.3	117.3
Circumferential Weld	61782/ SA-847	4.85	-4.8	56[1]	170.4[1]	1.396	237.9	289.1

[1]Regulatory Guide 1.99, Rev. 2, Position 1.1

The RT_{PTS} values for the intermediate and lower shell forgings remain below the NRC screening criterion of 270^{o}F and the RT_{PTS} value for the beltline circumferential weld (SA-847) remains below the NRC screening criterion of 300^{o}F at 52 EFPY. The analysis associated with PTS has been projected to the end of the period of extended operation and is consistent with 10 CFR 54.21(c)(1)(ii)

4.2.2 Pressure-Temperature (P-T) Limits

Introduction

10 CFR Part 50 Appendix G requires that the reactor pressure vessel (RPV) be maintained within established pressure-temperature (P-T) limits including during heatup and cooldown. These limits specify the maximum allowable pressure as a function of reactor coolant temperature. As the reactor pressure vessel becomes embrittled and its fracture toughness is reduced, the allowable pressure (given the required minimum temperature) is reduced.

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT} and adding a margin.

Conclusion

The reactor vessel neutron fluence values corresponding to the end of the period of extended operation and the reactor vessel beltline material properties have been calculated consistent with Regulatory Guide 1.190. The revised fluence values have been used to determine the limiting value of RT_{NDT} using the methods of Regulatory Guide 1.99. The limiting value of RT_{NDT} was used to calculate reactor coolant system (RCS) pressure-temperature (P-T) operating limits that are valid through the end of the period of extended operation (Reference 25). P-T Curves were developed using ASME Code Case N-641, which allows for the use of the KIC methodology (ASME Code Case N-640) and the relaxed “Circ Flaw” methodology (ASME Code Case N-588). Consistent with NUREG-1800 section 4.2.2.1.3.3, it is not necessary to implement P-T limits to carry the reactor vessel through 60 years at the time of application. The updated limits must be contained in a pressure-temperature limit report (PTLR) or in the Technical Specification (TS) prior to the period of extended operation.

The analyses associated with reactor vessel pressure-temperature limits will be available prior to entering the period of extended operation, consistent with 10 CFR 54.21(c)(1)(ii).

4.3 Metal Fatigue

Although fatigue is not necessarily time-limited in the same manner as other TLAAs (since design limits are based on cycles and not an explicit time period), it has been identified by the NRC and previous license renewal applicants as a TLAA. There are two aspects to fatigue life evaluation. The first is fatigue design, which is based on transient cycles and is a TLAA and part of the plant CLB. The second is the consideration of reactor water environmental effects on fatigue life, as described in GSI-190, and is not part of the CLB. The TLAAs on fatigue design have been resolved by projecting that the original transient design cycles remain valid for the 60-year operating period. Reactor water environmental effects on fatigue life are evaluated using the most recent data from laboratory simulation of the reactor coolant environment. Environmentally-assisted fatigue effects are addressed by the Fatigue Monitoring Program.

4.3.1 ASME Boiler and Pressure Vessel Code, Section III, Class 1

Introduction

The reactor vessel, pressurizer, steam generators, and reactor coolant pumps have been designed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class 1. The reactor vessel internals were designed in accordance with Westinghouse criteria which were later incorporated into the ASME Boiler and Pressure Vessel Code. The ASME Boiler and Pressure Vessel Code, Section III, Class 1 requires a design analysis to address fatigue and establish limits such that initiation of fatigue cracks is precluded.

Fatigue usage factors for critical locations in the NSSS components were determined using design cycles that were specified in the plant design process. These design cycles were intended to be conservative and bounding for all foreseeable plant operational conditions. The design cycles were subsequently utilized in the design stress reports for various NSSS components satisfying ASME fatigue usage design requirements, and became part of the plant Technical Specifications.

Experience has shown that actual plant operation is often very conservatively represented by these design cycles. The use of actual operating history data allows the quantification of these conservatisms in the existing fatigue analyses. To demonstrate that the Class 1 component fatigue analyses remain valid for the period of extended operation, the design cycles applicable to the Class 1 components were assembled. The actual frequency of occurrence for the design basis cycles was determined and compared to the design cycle set. The severity of the actual plant transients was compared to the severity of the design cycles. This comparison was performed in order to demonstrate that on an event-by-event basis the design-cycle profiles envelope actual plant operation. In addition, a review of the applicable administrative and operating procedures was performed to verify the effectiveness of the current design cycle counting program.

Conclusion

This review concluded that the existing design cycles and cycle frequencies are conservative and bounding for the period of extended operation. The analyses associated with verifying the structural integrity of the reactor vessel, reactor vessel internals, pressurizers, steam generators, and reactor coolant pumps have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

Prior to the expiration of the current operating license, a Fatigue Monitoring Program will be implemented as a confirmatory program.

4.3.2 ANSI B31.1 Piping

Introduction

The Reactor Coolant System primary loop piping and balance-of-plant piping were originally designed to the requirements of USAS B31.1, Power Piping Code. The pressurizer surge line was reanalyzed in 1991 and is treated separately in Section 4.3.7.

Design requirements in ANSI B31.1 assume a stress range reduction factor to provide conservatism in the piping design to account for fatigue due to thermal cyclic operation. This reduction factor is 1.0 provided the number of anticipated cycles is limited to 7000 equivalent full temperature cycles. This represents a condition where a piping system would have to be cycled approximately once every 3 days over the extended plant life of 60 years. Considering this limit, a review of the ANSI B31.1 piping within the scope of license renewal was performed in order to identify those systems that operate at elevated temperature and to establish their cyclic operating practices. Under current plant operating practices, piping systems within the scope of license renewal (except for the NSSS Sampling System) are only occasionally subject to cyclic operation. Typically these systems are subject to continuous steady-state operation and operating temperatures only vary during plant heatup and cooldown, during plant transients, or during periodic testing.

Conclusion

The results of the evaluation for ANSI B31.1 piping systems (except for the NSSS Sampling System) demonstrate that the number of assumed thermal cycles will not be exceeded in 60 years of plant operation. The analyses associated with ANSI B31.1 piping fatigue have been evaluated and determined to remain valid for the period of extended operation, in accordance with CFR 54.21(c)(1)(i).

For the NSSS Sampling System, an engineering analysis will be performed prior to the end of the current license period to verify that the allowable piping stresses (accounting for a stress range reduction factor less than 1.0) will not be exceeded during the period of extended operation. If the results of this analysis are not acceptable, an approach will be developed which will include one or more of the following options:

- Further refinement of the fatigue analysis;
- Manage the effects of fatigue by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC); or
- Replacement of the affected components.

4.3.3 Reactor Vessel Underclad Cracking

Introduction

Underclad cracking is associated with reactor pressure vessel. The industry has reported cracking in the low-alloy base metal heat-affected zone (HAZ) beneath the austenitic stainless steel weld overlay that is deposited to protect the ferritic material from corrosion.

A detailed analysis of underclad cracks was provided by Westinghouse in topical report WCAP-7733 (Reference 1). This report justified the continued operation of Westinghouse plants for 32 effective full power years with underclad cracks in the vessel.

A re-evaluation of the underclad cracking issue for 60 years of plant operation was performed in WCAP-15338 (Reference 2) and concluded that “underclad cracks in a reactor vessel are of no concern relative to the structural integrity of the vessel for continued plant operation, even through 60 years of operation.” The NRC reviewed WCAP-15338 and included two applicant action items to verify that a plant is bounded by the report evaluation and that the TLAA be described in the plant FSAR supplement.

Conclusion

WCAP-15338 is bounding for all Westinghouse plants and the Underclad Cracking TLAA is described in the UFSAR supplement. Therefore, the two applicant action items have been resolved.

The analysis associated with reactor vessel underclad crack growth has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

4.3.4 Accumulator Check Valves

Introduction

Anamet Laboratories report 172.1 describes analyses performed on the 10-C48Z Self-Actuating Swing Check Valves manufactured by the Darling Valve Co. and used in conjunction with the Ginna Station accumulators. Fatigue of components is recognized as time-dependent and therefore the analysis was reviewed for fatigue related to these valves. Fatigue failure is based upon the criteria of the cumulative usage factor (CUF). The Anamet report concludes that the maximum CUF is 0.74. The analysis is based on load condition occurrence limits provided by Westinghouse Electric Corp.

Conclusion

The 2000 Transient Monitoring Report was reviewed to confirm transient limits and total transient counts to date. The load condition occurrences used in the Anamet report bound the transient limits monitored by plant procedures. In accordance with 10 CFR 54.21(c)(1)(i), the existing CUF calculations remain valid because the number of assumed transients would not be exceeded during the period of extended operation.

4.3.5 Reactor Vessel Nozzle-to-Vessel Weld Defect

Introduction

In 1979, during the first-interval ISI of the reactor vessel, a flaw indication was discovered by UT examination in a primary inlet nozzle-to-vessel weld (Nozzle N2B). The size of this indication (5.27" long X .93" deep) was determined to be in excess of the size permitted by the acceptance criteria for the examination method in ASME Section XI, 1974 Edition. As a result, the flaw indication was evaluated by Teledyne Engineering Services in accordance with the Section XI (Appendix G) requirements for acceptance by evaluation and found to be acceptable. A review of original construction radiographs confirmed the presence of slag at the same location as the indication.

The same flaw indication was again recorded in Nozzle N2B by UT examination during the second-interval reactor vessel ISI performed by Southwest Research Institute (SwRI) in 1989. The size was again found to exceed the acceptance criteria in Section XI. However, using a 15° focused beam search unit, the indication was resolved into two separate indications which met the criteria for acceptance by examination in ASME Section XI, 1974 edition with Summer 1975 Addenda. A fracture mechanics analysis performed by Structural Integrity Associates also confirmed that the indication was acceptable by evaluation according to the requirements of ASME Section XI (Appendix G).

Conclusion

A safety evaluation performed by the NRC (Reference 26) concluded that the flaw indication was probably a volumetric reflector resulting from the fabrication process that had remained unchanged since construction and that augmented ISI required by the ASME Code was not warranted. Therefore, no further evaluation of this defect is required.

4.3.6 Pressurizer Fracture Mechanics Analysis

Introduction

During the preservice UT examination of the pressurizer, a “defect-like” indication was reported in the lower shell-to-head circumferential weld (C-3). The indication was reported as a linear reflector approximately 11 1/2” long X 1/2” width embedded partially in the circumferential weldment and the base metal of the pressurizer shell. Based on a fracture mechanics analysis performed by Westinghouse, it was concluded that the “defect” would not cause failure of the pressurizer during the design life (40 years) of the component. The analysis was based on several conservative assumptions, including transposing the defect from the embedded position to the internal surface of the pressurizer wall.

This indication was subsequently examined by UT in 1971, 1972, 1974, 1980, 1991, and 2002 during ASME Section XI inservice inspections. The examinations in 1974, 1980, and 1991 characterized the indication as consisting of several intermittent, low-amplitude indications located in the center 1/3 of the weld thickness. These indications were evaluated and found to meet the acceptance criteria by examination of ASME Code, Section XI. The most recent inspection was performed using both automated and manual UT examinations. Intermittent, low-amplitude indications were recorded in the center 1/3 of the weld thickness. These indications were also evaluated and found to meet the acceptance criteria by examination in ASME Code, Section XI, 1995 Edition (1996 Addenda).

Conclusion

Since it has been demonstrated that the initial indication is actually a number of small, discrete indications which meet the ASME Code, Section XI acceptance criteria by examination, the fracture mechanics analysis is no longer applicable or relevant.

4.3.7 Environmentally Assisted Fatigue

Generic Safety Issue (GSI) 190 was initiated by the NRC staff because of concerns about the potential effects of reactor water environments on reactor coolant system component fatigue life during the period of extended operation. The NRC closed GSI-190 in December 1999 and concluded that environmental effects have a negligible impact on core damage frequency, and as such, no generic regulatory action is required. However, as part of the closure of GSI-190, NRC concluded that licensees who apply for license renewal should address the effects of coolant environment on component fatigue life as part of their aging management programs (Reference 5). See (Figure 4.3-1) for a graphic depiction of the approach used for addressing the fatigue TLAAs and the Environmentally-Assisted Fatigue evaluation

An analysis must satisfy all six criteria defined in 10 CFR 54.3 to qualify as a TLAA. Failure to satisfy any one of these criteria eliminates the analysis from further consideration as a TLAA. Fatigue design analysis for Ginna Station has been determined to be a TLAA, even though the design limits are based on cycles rather than an explicit time period. However, reactor water environmental effects, as described in GSI-190, are not included in the Ginna Station current licensing basis (CLB), such that the criterion specified in 10 CFR 54.3(a)(6) is not satisfied.

Nevertheless, environmental effects on Class 1 component fatigue have been evaluated separately for Ginna Station to determine if any additional actions are required for the period of extended operation.

The Fatigue Monitoring Program addresses the effects of the reactor coolant environment on component fatigue life by assessing the impact of reactor coolant environment on a selected set of critical component locations that includes those components identified in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components" (Reference 6) for the older vintage Westinghouse plant. This evaluation can take one of two approaches; (1) demonstration that the fatigue analysis of design transients, when compared to an evaluation based on actual transients, will bound any environmental effects during the extended operating period, or (2) assessment of actual expected fatigue usage factor, specifically including the influence of environmental effects.

The second approach evaluates the selected plant component location fatigue usage utilizing the environmental life correction factor formulae contained in NUREG/CR-6583

"Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels" (Reference 7) for carbon and low-alloy steels and NUREG/CR-5704 "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels" (Reference 8) for austenitic stainless steels. These formulae are then used in an F_{en} approach, originally developed by EPRI, in which an environmental fatigue multiplier (F_{en}) is computed when certain conditions of dissolved oxygen, temperature, strain rate, strain range, sulfur content and flow rate are satisfied. The Fatigue Monitoring Program uses the second approach for the selected set of component locations

Approach for Addressing the Fatigue TLAAs and the Environmentally-Assisted Fatigue Evaluation

The approach for addressing fatigue TLAAs and reactor water environmental effects at Ginna Station accomplishes two objectives, as illustrated in (Figure 4.3-1). First, the TLAAs on fatigue design have been resolved by projecting that the original transient design cycles remain valid for the 60-year operating period. Confirmation by the Fatigue Monitoring Program will ensure that these transient design cycles are not exceeded. Second, reactor water environmental effects on fatigue life are examined using the most recent data from laboratory simulation of the reactor coolant environment. These two aspects of fatigue design are kept separate, because fatigue design for Ginna is part of the plant CLB and a TLAA, while the consideration of reactor water environmental effects on fatigue life, as described in GSI-190, is not considered part of the Ginna CLB.

It is important to note that there are three areas of margin included in the Ginna Fatigue Monitoring Program that are worthy of consideration. These areas include margins resulting from actual cycle experience, cycle severity, and moderate environmental effects.

Margin Due to Actual Cycles: It has been concluded that the original 40-year design cycle set for Class 1 components is valid for the 60-year extended operating period. Conservative projections conclude that the design cycle limits will not be exceeded. Additional margin is available in the current Class 1 component fatigue analyses since the cumulative fatigue usage factors for all Class 1 components remain below the acceptance criteria of 1.0.

Margin Due to Transient Severity: Much of the conservatism in the fatigue analysis methodology is due to design cycle definitions. It has been concluded that the severity of the original Ginna design cycles bound actual plant operation. Additional industry fatigue studies conclude that the fatigue impact of conservative design basis cycle definitions by themselves overwhelms the contributing impact of reactor water environmental effects.

Margin Due to Moderate Environmental Effects: A portion of the safety factors applied to the ASME Code Section III fatigue design curves includes moderate environmental effects. While there is debate over the amount of margin this represents, it is noteworthy to recognize this safety factor in this qualitative discussion of margin.

Considering the three margins above, the Ginna Fatigue Monitoring Program is conservative from an overall perspective. Nevertheless, specific assessments of potential environmental effects have been addressed. Idaho National Engineering Laboratories (INEL) evaluated in NUREG/CR-6260 (Reference 6) fatigue-sensitive component locations at plants designed by all four U. S. nuclear steam supply system (NSSS) vendors, as a part of the industry effort to address environmental effects for operating nuclear power plants during the current 40-year licensing term. The pressurized water reactor (PWR) calculations included in NUREG/CR-6260, especially the “Older Vintage Westinghouse Plant”, bound Ginna Station with respect to the design codes used. Additionally, the evaluated design cycles considered in the evaluation bound the Ginna Station design.

The fatigue-sensitive component locations chosen in NUREG/CR-6260 for the older vintage Westinghouse plant were:

- Reactor vessel shell and lower head (lower shell at the core support pads)
- Reactor vessel inlet and outlet nozzles
- Pressurizer surge line (including hot leg and pressurizer nozzles)
- Reactor coolant piping charging system nozzle
- Reactor coolant piping safety injection nozzle
- Residual Heat Removal system Class 1 piping

NUREG/CR-6260 calculated fatigue usage factors for these locations utilizing the interim fatigue curves provided in NUREG/CR-5999 (Reference 10). However, the data included in more recent industry studies (Reference 7 and Reference 8) need to be considered in the evaluations of environmental effects.

Environmental fatigue calculations have been performed for Ginna Station for those component locations included in NUREG/CR-6260 using the appropriate methods contained in NUREG/CR-6583 for carbon/low alloy steel material, or NUREG/CR-5704 for stainless steel material. Based on these results, all component locations were determined to be acceptable for the period of extended operation, with the exception of the pressurizer surge line.

4.3.7.1 Reactor Vessel Locations

Appropriate F_{en} factors were computed for each individual load pair in the governing fatigue calculation so that an overall multiplier on the fatigue usage factor for environmental effects was determined for each component. Application of these factors to the design fatigue usage resulted in acceptable values for the period of extended operation. The locations analyzed are the RPV inlet nozzles, RPV outlet nozzles and the RPV Shell at the core support pads.

4.3.7.2 Surge Line Locations

A structural evaluation of the Ginna surge line considering the effects of thermal stratification was performed by Westinghouse in 1991 (Reference 12). WCAP-12928 describes the stress and cumulative usage factor analysis performed for the surge line in accordance with NRC Bulletin 88-11. The highest CUF was calculated at the surge line nozzle connection to the RCS hot leg.

The F_{en} approach was not able to produce acceptable results for the period of extended operation due to the significant thermal cycling duty and high environmental fatigue multipliers. The environmentally-adjusted fatigue usage value for the limiting pressurizer surge line location is calculated to exceed 1.0 before the end of the period of extended operation.

An aging management program will be used to address environmentally-assisted fatigue for the Ginna pressurizer surge line during the period of extended operation. Prior to the end of the current license period, critical weld locations (i.e. pressurizer surge line nozzle weld, surge line reducer-to-pipe weld below the pressurizer, and the RCS hot leg surge line nozzle weld) on the surge line will be inspected in accordance with the appropriate requirements of IWB under the ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program. Prior to the period of extended operation, the results of these inspections and research planned by the EPRI-sponsored Materials Reliability Program (MRP) will be used to determine the appropriate approach for addressing environmentally-assisted fatigue of the surge line. The approach developed will include one or more of the following options:

- Further refinement of the fatigue analysis to lower the CUF to below 1.0;
- Manage the effects of fatigue by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC);
- Repair of the affected location(s); or
- Replacement of the affected location(s).

Should RG&E select the inspection option to manage environmentally-assisted fatigue for the surge line during the period of extended operation, inspection details such as scope, qualification, method, and frequency will be provided to the NRC for review and approval prior to entering the period of extended operation. This position is consistent with previous applicants' positions.

Various pressurizer surge line welds at locations of high fatigue usage have been examined at Ginna Station in the past. No reportable indications were found by these examinations. The absence of any reportable indications on the surge line to-date supports the use of the aging management program outlined above.

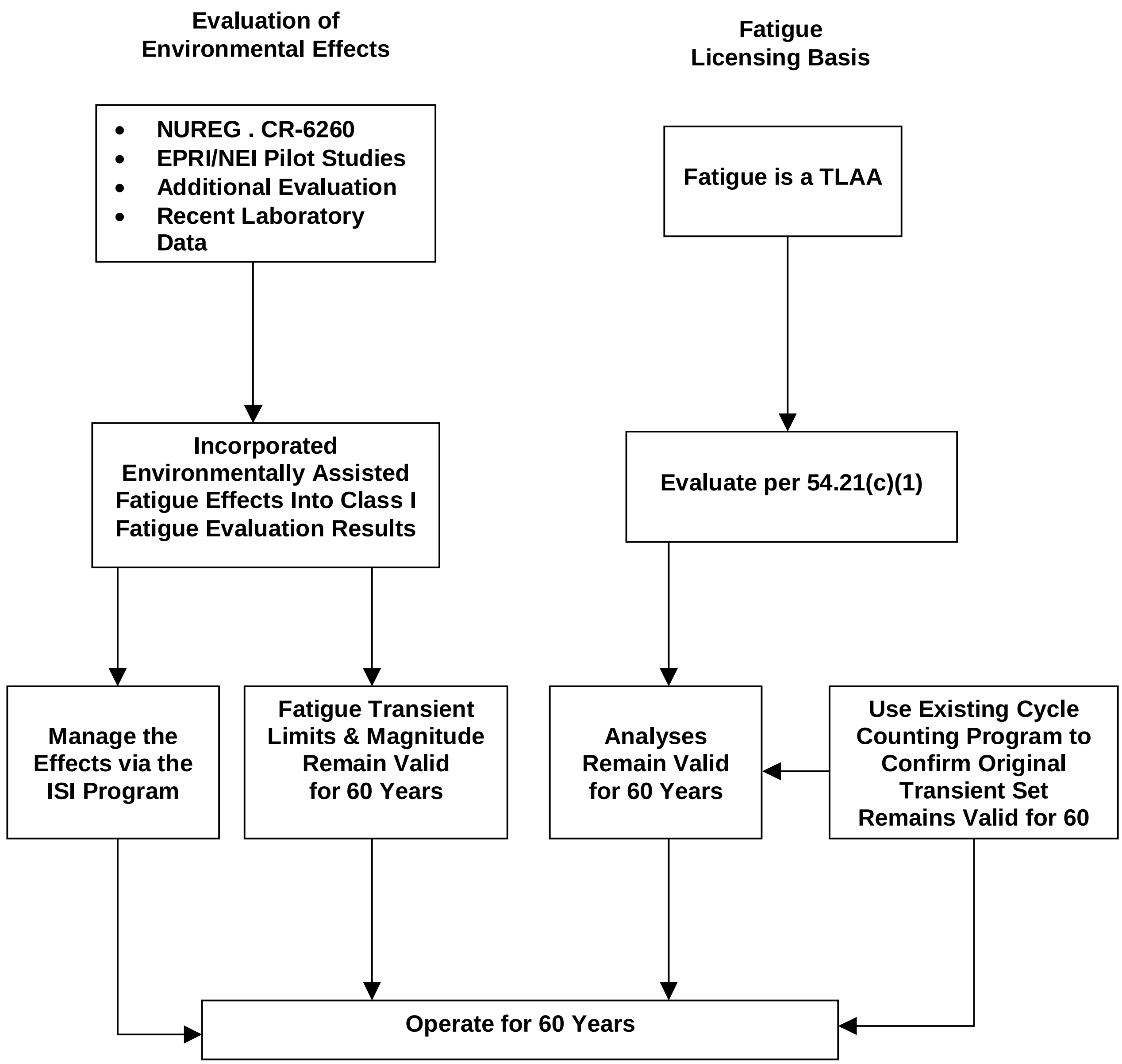
4.3.7.3 USAS B31.1 Locations

As with the older vintage Westinghouse plants evaluated in NUREG/CR-6260, detailed fatigue usage calculations do not exist for the Ginna Station RHR tee, charging nozzle and safety injection (accumulator) nozzle, because the design basis for the Ginna Station piping is USAS B31.1, which does not require specific fatigue analysis. In response to this circumstance, the authors of NUREG/CR-6260 developed a detailed ASME Class 1 fatigue analysis for each of these three components based on typical inputs These analyses were used to develop Ginna-specific environmental fatigue calculations. The design transients and cycle counts in NUREG/CR-6260 bound the design transients and cycle counts for Ginna Station for these three component locations. The design inputs for Ginna Station (e.g. material, geometry) were compared to those summarized in NUREG/CR-6260 for these three components. The Ginna Station charging nozzle and safety injection nozzle were determined to be identical in terms of materials and geometry as those presented in NUREG/CR-6260. The Ginna Station RHR tee is identical in terms of material, but is larger than the RHR tee analyzed in NUREG/CR-6260. The NUREG/CR-6260 RHR tee was determined to bound the stress ranges for the larger Ginna Station RHR tee. Adjustments to the design basis fatigue usage (without environmental effects) were made based on the design input comparison. Plant-specific F_{en} factors were computed and were applied to the Ginna-specific design basis fatigue usage to yield Ginna-specific environmental fatigue values. This process resulted in acceptable values for the period of extended operation.

Conclusion for Environmentally Assisted Fatigue

The Fatigue Monitoring Program utilizes 10 CFR 54.21(c)(1)(i) for each of the plant components within the scope of the program. The Fatigue Monitoring Program is a confirmatory program that monitors loading cycles due to thermal and pressure transients for selected critical components. The program provides an analytical basis for confirming that the number of cycles established by the analysis of record will not be exceeded before the end of the period of extended operation. The effects of reactor coolant environment are considered through the evaluation of the seven component locations identified in NUREG/CR-6260 using the appropriate environmental fatigue factors. Formulae for calculating the environmental life correction factors are contained in NUREG/CR-6583 (Reference 7) for carbon or low-alloy steels and in NUREG/CR-5704 (Reference 8) for austenitic stainless steels.

Figure 4.3-1 TLAA & GSI-190 Environmentally Assisted Fatigue Evaluation Process



4.4 Environmental Qualification (EQ) of Electric Equipment

10 CFR 50.49, Environmental qualification of electric equipment important to safety for nuclear power plants, requires that selected electrical equipment that is relied upon to remain functional during and following a design basis event be environmentally qualified to perform its intended function. Equipment within the scope of the EQ rule has been identified in accordance with 10 CFR 50.49 paragraph (d) and is listed in the Ginna Station EQ Master List. Only the equipment qualification packages which indicate a qualified life of greater than 40 years will be reviewed as a Time-Limited Aging Analysis (TLAA). Equipment qualification packages that indicate a qualified life of less than 40 years are not a TLAA as defined in 10 CFR 54.3 and therefore will not be discussed in the context of license renewal.

To establish reasonable assurance that the safety related electrical equipment will perform its safety function when exposed to postulated harsh environmental conditions, licensees are required to develop an environmental qualification program. The program must demonstrate that the safety related electrical equipment required to perform the various safety related functions, identified in 10 CFR 50.49, is qualified to perform as intended. The program must maintain the environmental qualification of the equipment for its installed life. Periodic replacement and/or refurbishment of equipment is performed in order to maintain the qualified life of the device. The qualified life of an equipment type is that period of time the equipment can be installed, under normal and abnormal plant operating conditions (thermal and radiation exposure), and can still perform its intended function following a postulated design basis event. The qualified life of an equipment type is determined using the ambient environmental conditions to which it is exposed for the predicted installation period as well as any internal heat rise and cyclic stresses.

Many of the EQ analyses may have been adequate under existing conditions and therefore could have been dispositioned per 10 CFR 54.21(c)(1)(i), however it was felt to be conservative to perform a confirmatory evaluation to verify that the assumptions in the existing analysis were adequate for the period of extended operation. Confirmatory analyses do not alter any conservatisms, use data reduction methods or use different analysis methodology.

Existing EQ analyses have radiation levels for accident conditions based on a power level of 1520 MWt. However the UFSAR was revised to account for instrument uncertainty and therefore provided environmental tables were revised for 102% power or 1550.4 MWt. All EQ packages have been reviewed to verify that the margin between the required radiation qualification and the actual radiation qualification is adequate to cover the increase.

Table 4.4-1 contains a list of re-analysis attributes and methodology descriptions used for EQ TLAAs. Unless otherwise specified, each TLAA reviewed in accordance with 10 CFR

54.21(c)(1)(i) or 10 CFR 54.21(c)(1)(ii) (and 10 CFR 54.21(c)(1)(iii) in some cases) have been or will be performed using the methodology described in this table.

Table 4.4-1 Environmental Qualification Reanalysis Attributes

Reanalysis Attributes	Description
Analytical methods	The analytical models used in the reanalysis of an aging evaluation should be the same as those previously applied during the prior evaluation. The Arrhenius methodology is an acceptable thermal model for performing a thermal aging evaluation. The analytical method used for a radiation aging evaluation is to demonstrate qualification for the total integrated dose (that is, normal radiation dose for the projected installed life plus accident radiation dose). For license renewal, one acceptable method of establishing the 60-year normal radiation dose is to multiply the 40 year normal radiation dose by 1.5 (that is, 60 years/40 years). The result is added to the accident radiation dose to obtain the total integrated dose for the component. For cyclical aging, a similar approach may be used. Other models may be justified on a case-by-case basis.
Data collection and reduction methods	Reducing excess conservatisms in the component service conditions (for example, temperature, radiation, cycles) used in the prior aging evaluation is the chief method used for a reanalysis. Temperature data used in an aging evaluation should be conservative and based on plant design temperatures or on actual plant temperature data. When used, plant temperature data can be obtained in several ways, including monitors used for technical specification compliance, other installed monitors, measurements made by plant operators during rounds, and temperature sensors on large motors (while the motor is not running). A representative number of temperature measurements are conservatively evaluated to establish the temperatures used in an aging evaluation. Plant temperature data may be used in an aging evaluation in different ways, such as (a) directly applying the plant temperature data in the evaluation, or (b) using the plant temperature data to demonstrate conservatism when using plant design temperatures for an evaluation. Any changes to material activation energy values as part of a reanalysis should be justified. Similar methods of reducing excess conservatisms in the component service conditions used in prior aging evaluations can be used for radiation and cyclical aging.

Table 4.4-1 Environmental Qualification Reanalysis Attributes

Underlying assumptions	Environmental qualification component aging evaluations contain sufficient conservatisms to account for most environmental changes occurring due to plant modifications and events. When unexpected adverse conditions are identified during operational or maintenance activities that affect the environment of a qualified component, the affected environmental qualification component is evaluated, and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions.
Acceptance criteria and corrective actions	The reanalysis of an aging evaluation should extend the qualification of the component. If the qualification cannot be extended by reanalysis, the component must be refurbished, replaced, or requalified prior to exceeding the current qualification. A reanalysis should be performed in a timely manner (such that sufficient time is available to refurbish, replace, or requalify the component if the reanalysis is unsuccessful).

4.4.1 **Solenoid Operated Valves**

4.4.1.1 **ASCO Solenoid Valve Model X-HAV210**

Introduction

ASCO Solenoid Valve, Model X-HAV210 is used inside the Containment Building at Ginna Station in a single EQ solenoid operated valve (SOV) application. This SOV controls air supply to an air driven pump used to obtain post-accident samples at Sump A. Plant documents identify the SOV as installed in 1982. The critical components of the ASCO SOV that determine qualified life are the gaskets. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The thermal, radiation, and cyclical wear analyses support a qualified life in excess of 60 years. Thermal aging calculations consider the effects of elevated operating temperature during valve operation. A review of plant temperatures in the reanalysis provides reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models, and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.2 Valcor Solenoid Operated Valve, Model V526-5440-2

Introduction

Valcor SOV, Model V526-5440-2 is used in the Intermediate Building at Ginna Station to control instrument air to main steam isolation valves (MSIVs). There are eight EQ SOVs, four providing redundant isolation and venting to each of two MSIVs. These SOVs are considered to be a part of the Main Steam system. They are located on the Mezzanine Level of the Intermediate Building in the steam header A and B areas. Plant documents identify these SOVs as installed in 1984 with one replacement in 1987. Valcor SOV, Model V526-5440-2, is a pilot assisted latching valve type with separate opening and closing coils. Valcor supplied rectifiers, Valcor Part No. S1140-8-1 associated with these valves are installed in terminal boxes separate from the valve housings, but are considered for purposes of qualification to be part of the valve due to the failure effects. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The thermal, radiation, and cyclical wear analyses support a qualified life in excess of 60 years. The SOV coils are not continuously energized and are therefore not subject to significant self heating. A review of plant temperatures provides reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.3 Valcor Solenoid Operated Valve, Model V526-6130-2

Introduction

One Valcor SOV, Model V526-6130-2 is used in the Containment Building at Ginna Station to control containment Sump A sampling pump discharge flow. This direct lift, normally open two way SOV is considered to be a part of the Plant Sampling System (PSS). The SOV is located on the 235 foot level, basement Accumulator B area. Plant documents identify this SOV as installed in 1983. Valcor supplied rectifiers and diodes, Valcor Part No. S1140-8-1 associated with this valve are installed inside the valve electrical terminal housing.

Qualification for Valcor SOV, Model V526-6130-2 is performed in accordance with IEEE 323-1974 and includes the installed rectifier and diode configuration. The vendor provided qualification report states that all O-rings meet a 40 year qualified life. In anticipation of lack of control of service conditions for installed valves, Valcor recommends that elastomeric solenoid seals (O-rings) be replaced at shorter intervals. At Ginna Station SOV service conditions have been monitored and analyzed. Therefore, Valcor SOVs will be considered to be installed under controlled service conditions. However, due to thermal setting properties of EPR, O-rings will be replaced whenever the valve is disassembled after operation.

Conclusion

The thermal, radiation, and cyclical wear analyses support a qualified life in excess of 60 years. Thermal aging calculations consider the effects of elevated operating temperature during valve operation. A review of plant temperatures in the reanalysis provides reasonable assurance that the reduced temperature used to determine qualified life is conservative.The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.4 Head Vent Solenoid Operated Valves

Introduction

Two Valcor model V526-6042-3 and two Valcor model V526-6042-17 solenoid operated valves (SOVs) are used to vent noncondensable gases from the reactor.These valves use the original electrical solenoid assembly, however two of the four valves have replacement valve bodies and therefore carry a different model number. Plant documents identify that these valves were installed in 1984 and qualified in accordance with IEEE 323-1974.

Conclusion

At this time, there are no plans to extend the qualification of the existing valves, and therefore the qualified portions of the valves will remain scheduled for replacement prior to the year 2022. A review of plant temperatures for the existing analyis provides reasonable assurance that the temperature used to determine qualified life remains conservative. In the event that reanalysis is performed, it will be completed consistent with the guidance provided in NUREG-1800 table 4.4-1. Therefore the environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21 (c)(1)(iii).

4.4.2 Motors

4.4.2.1 Westinghouse Containment Recirculation Fan Motor

Introduction

The containment recirculation fan motors are part of the containment recirculation fan cooling (CRFC) system. Five of these motors are rotated through the four CRFC units as necessitated by motor maintenance requirements. Four motors are original plant equipment, identified by Westinghouse shop order number 68F13557 and serial numbers 1S-69, 2S-68, 3S-68, and 4S-68. The fifth motor was initially installed in April of 1995 and is identified by Westinghouse shop Number 17570LN. All five are 300HP, 440VAC motors.

Qualification life for these motors is based on three critical components. These are winding insulation, bearings, and bearing lubrication. A motor bearing and its lubrication are treated as a single replaceable part, therefore winding insulation is the limiting component for evaluation of motor qualified life. These windings are insulated with Thermalastic Epoxy. The existing qualification is based on DOR guidelines.

Conclusions

The thermal and radiation analyses support a qualified life in excess of 60 years. Thermal aging calculations consider the effects of elevated operating temperature during motor operation. A review of plant temperatures in the reanalysis provides reasonable assurance that the reduced temperature used to demonstrate qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.2.2 Limitorque Actuators, Outside Containment

Introduction

Nine EQ Limitorque Actuators are located in the Auxiliary Building basement. Four additional EQ units are located in the Auxiliary Building subbasement RHR pump pit. These actuators were put in service November 1969 and were originally qualified to DOR Guidelines. In the early 1990's, in response to NRC Generic Letter 89-10 all actuators were completely refurbished including replacement torque and limit switches qualified to IEEE 323-1974. All motors are Class B insulated with six units retaining their original motors. All internal wiring is replacement Anaconda by RG&E with separate qualification to IEEE 323-1974.

Conclusion

The thermal, radiation, and cyclical wear analyses support a qualified life in excess of 60 years. Equipment operation is a negligible portion of service life and self heating due to temperature rise is not a significant contributor to age related degradation. A review of plant temperatures provides reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.2.3 Limitorque Actuators, Inside Containment

Introduction

Two EQ Limitorque Actuators are used on the RHR discharge valves located in the Containment Building basement. These actuators were placed in service in November of 1969 and were originally qualified to DOR guidelines. In the early 1990's, in response to NRC Generic Letter 89-10 all actuators were completely refurbished including replacement torque and limit switches qualified to IEEE 323-1974. All motors are replacement Class RH insulated and qualified to IEEE 323-1974. All internal wiring is Anaconda and Brand-Rex, replaced by RG&E with separate qualification to IEEE 323-1974.

Conclusion

The thermal, radiation, and cyclical wear analyses support a qualified life in excess of 60 years. Equipment operation is a negligible portion of service life and self heating due to temperature rise is not a significant contributor to age related degradation. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. The analytical models, temperature, and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.2.4 Limitorque Actuators, PORV Block Valves

Introduction

Limitorque Model SMB-00-15 Motor Operated Valves (MOVs) with Reliance motors are used as block valves to prevent inadvertent depressurization of the primary system should the pressurizer power operated relief valves fail to close. These components are located in the pressurizer cubicle in the containment building. During the refueling outage of 1989, these MOVs were replaced because the seat rings were approching the maximum allowable limits for remachining The age degradable sub-components considered include the Fiberite torque switch, the type RH motor insulation, and the Viton shaft seal.

Conclusion

The previous thermal and radiation analyses support a qualified life of 40 years. The motors are only cycled for testing, and are not normally energized. Self-heating is not considered significant for these motor operated valves. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Due to the date of installation and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.2.5 Westinghouse Safety Injection Pump 1A/1C Motor

Introduction

Westinghouse Model Lifeline A ABDP motors are installed in the Auxiliary Building and used to provide motive power to rotate the shaft of safety injection pumps 1A and 1C during the worst case design basis accident. The motors are original plant equipment and qualification is performed consistent with DOR guidelines.

Conclusion

Reanalysis of the motor qualified life is scheduled to be performed prior to the end of the current license period. Although the existing analysis indicates a qualified life of significantly greater than 60 years, the formal reanalysis will verify that the conditions and assumptions used in the original analysis remain valid for the period of extended operation. If the qualification cannot be extended by reananlysis, the motor will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.2.6 Westinghouse/Reliance Safety Injection Pump 1B Motor

Introduction

A Westinghouse Model ABDP motor was refurbished and qualified by the Reliance Electric Company. The motor is used to provide motive power to rotate the shaft of the safety injection pump during the worst case design basis accident. It was refurbished in 1994 and qualified to IEEE 323-1974.

Conclusion

The previous thermal and radiation analyses support a qualified life of 44 years. Thermal aging calculations consider the effects of elevated operating temperature during motor operations. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Due to the date of installation and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.2.7 Westinghouse Containment Spray Pump Motors

Introduction

Westinghouse motor model Lifeline T TBDP is used to provide motive power to rotate the shaft of the containment spray pumps during the worst case design basis accident. The motors are installed in the Auxiliary Building basement and are normally de-energized. For qualified life calculations, it is conservatively assumed that these motors operate for 5 hours per year. This motor is original plant equipment qualified consistent with DOR guidelines.

Conclusion

Reanalysis of the motor qualified life will be performed prior to the end of the current license period. Although the existing analysis indicates a qualified life of significantly greater than 60 years, the formal reanalysis will verify that the conditions and assumptions used in the original analysis remain valid for the period of extended operation. If the qualification cannot be extended by reananlysis, the motor will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.2.8 Westinghouse RHR Pump 1A Motor

Introduction

Westinghouse motor model Lifeline T TBDP is used to provide motive power to rotate the shaft of the residual heat removal pump 1A during the worst case design basis accident. The motor is installed in the Auxiliary Building sub-basement. This motor is original plant equipment qualified consistent with DOR guidelines. The existing qualified life for this motor is 47 years. This life is calculated by conservatively assuming the operating time is 10% of total life. Since plant operating cycle time has been increased to 18 months, it is expected that this proportion will decrease.

Conclusion

Reanalysis of the motor qualified life will be performed prior to the end of the current license period. If the qualification cannot be extended by reanalysis, the motor will be refurbished, replaced, or requalified prior to exceeding the current qualification period. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.2.9 Westinghouse RHR Pump 1B Motor

Introduction

Westinghouse motor model Lifeline T TBDP is used to provide motive power to rotate the shaft of the residual heat removal pump 1B during the worst case design basis accident. This component is installed in the Auxiliary Building sub-basement. The motor was refurbished by Westinghouse in 1989 and qualified to IEEE 323-1974.

Conclusion

The previous thermal and radiation analyses support a qualified life of 40 years. Thermal aging calculations consider the effects of elevated operating temperature during motor operations. Due to the date of installation and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.2.10 Westinghouse Hydrogen Recombiner Blower Motor

Introduction

Westinghouse motor model TBFC is used to provide motive power to rotate the shaft of the hydrogen recombiner blowers. This component is installed in the Containment Building and is normally de-energized. Although the original analysis calculates a possible operation time of 8 hours per month, this motor is not normally operated and therefore an assumption of 8 hours per year would be conservative. The motor is original plant equipment, qualified consistent with DOR Guidelines.

Conclusion

Reanalysis of the motor qualified life will be performed prior to the end of the current license period. Although the existing analysis indicates a qualified life of significantly greater than 60 years, the formal reanalysis will verify that the conditions and assumptions used in the original analysis remain valid for the period of extended operation. If the qualification cannot be extended by reanalysis, the motor will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.2.11 Reliance AC Random Wound Motor Model 145TCV

Introduction

The Reliance Model 145TCV motor is used to dewater and prevent flooding of the Auxiliary Building Sub-Basement sump in the event of an RHR pump seal failure during the recirculation phase of a LOCA. The motor was purchased from Reliance in 1995 and qualified to IEEE 323-1974.

Conclusion

The previous thermal and radiation analyses support a qualified life of 40 years. Thermal aging calculations consider the effects of elevated operating temperature during motor operations. Due to the date of installation and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.3 Electrical Penetration Assemblies

4.4.3.1 Crouse-Hinds Electrical Penetration Assemblies

Introduction

Crouse-Hinds Electrical Penetration Assemblies are used for power, control, and instrumentation applications at Ginna station. These penetrations are original plant equipment and qualification is based on DOR Guidelines. The qualification is applicable to the penetration seals, canister, and internal connections. Critical penetration sealing functions are performed by non-organic materials such as ceramic and glass. Electrical conductors internal to Crouse-Hinds penetrations are soldered into ceramic header insulators with splicing to stranded conductor leads accessible at inboard and outboard headers. Additional internal and external electrical insulation is provided by fiberglass and silicone rubber (Varglas), and GE RTV 615.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Self-heating temperature rise was included in the calculation of thermal life. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.3.2 Westinghouse Electrical Penetration Assembly, Model WX32714

Introduction

A single Westinghouse WX32714 electrical penetration assembly was installed at Ginna station in 1975. This penetration is designated for use in instrumentation applications with low voltage and low power requirements. Applications include instrument transmitters, RTDs, and LVDTs. Critical penetration sealing functions are performed by O-rings and Westinghouse Q epoxy. A secondary Scotch epoxy sealant protects the lead connections to the wire electrode feedthroughs. This epoxy surrounds the spliced connections and may be considered the only sealant that is exposed to any effects of beta radiation. This sealant may be considered a sacrificial layer and does not have an adverse impact on the intended function. The primary inner seal is enclosed within a metal tube which protects against beta radiation effects. Qualification is based on DOR guidelines.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Due to the low power circuits, self-heating is not a concern for this application. A review of plant temperatures provides reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.3.3 Westinghouse Modular Electrical Penetration Assemblies

Introduction

Four Westinghouse modular electrical penetration assemblies were installed at Ginna station in 1985. The circuit applications include core exit thermocouples, transmitter circuits, high range radiation monitor, resistance temperature detectors, solenoid operated valves and non-Class 1E welding circuits. The solenoid valves and welding circuits are normally de-energized. All four penetrations are different in internal electrical design, but all use similar materials of construction. These penetrations have a primary inner seal which is enclosed within a metal tube which protects against beta radiation (during accident conditions). A secondary epoxy surrounds the spliced connections, however it may be considered a sacrificial layer and does not have an adverse impact on the intended function. Qualification for these penetrations is performed in accordance with IEEE 323-1976.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Due to the low power circuits, self-heating is not a concern for this application. A review of plant temperatures demonstrated that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.4 Heat Shrink Tubing

4.4.4.1 Raychem WCSF-N Splice Sleeves for Containment Penetration Fan Cooler Motors

Introduction

Raychem WCSF-N Power Cable Sleeves are used in a single EQ penetration application in the Containment Building at Ginna Station. Two sizes of these sleeves are installed as qualified by special test for this unique Ginna Station configuration. These sleeve sizes are Raychem WCSF 600-1250, 300-500 nominal and WCSF1000/3000, 650-1250 MCM nominal. This unique configuration is used to insulate bolted connections between Crouse-Hinds electrical penetration ceramic bushing terminals and power cable circuits supplying Recirculation Fan Cooler motors. The Recirculation Fan Cooler motor circuits are made up of 500 MCM Kerite 600 Volt HTK Insulated, FR Jacketed power cable.

These Raychem WCSF-N Power Cable Sleeves were a one time purchase at Ginna Station made in 1978. Qualification is based on DOR Guidelines.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Self-heating temperature rise was included in the calculation of thermal life. A review of plant temperatures provides reasonable assurance that the reduced temperature used to determine the qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.4.2 Raychem Nuclear Splice Kits - NMCK, NPKC, NPKP, and NPKS

Introduction

Raychem Nuclear Splice Kits are used in applications both inside and outside of containment. Splice kits provide an insulation function for electrical conductors. The kits are used for specific applications and are composed of several insulation sleeves. Variations in models and part numbers indicate different sizes and shapes, however all components are constructed of the same Raychem extrusion and molding materials. These Raychem Nuclear Splice Kits were first purchased in 1985. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life of 44.8 years at full rated temperature. Due to the date of purchase and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.4.3 **Raychem WCSF-050-N Shim Stock Cable Sleeves**

Introduction

Raychem WCSF-050-N is used in applications both inside and outside of containment. Cable sleeves provide an insulation function for electrical conductors. This specific type of insulation sleeve is thin walled and is normally used to build up small diameter cables for use with larger Raychem cable sleeves. These Raychem cable sleeves were first purchased in 1985. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The existing thermal and radiation analyses support a qualified life of 42.8 years at full rated temperature. Reanalysis of the cable sleeve qualified life will be performed prior to the end of the current license period. If the qualification cannot be extended by reanalysis, the cable sleeves will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.4.4 **Raychem WCSF-N Cable Sleeves**

Introduction

Raychem WCSF-N cable sleeves are used in applications both inside and Outside of containment. Cable sleeves provide an insulation function for electrical conductors. The type of cable sleeves covered by this evaluation are general purpose use to environmentally seal cable splices in electrical safety related circuits. These Raychem cable sleeves were first purchased in 1980. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The existing thermal and radiation analyses support a qualified life of 40 years at full rated temperature. Reanalysis of the cable sleeve qualified life will be performed prior to the end of the current license period. If the qualification cannot be extended by reanalysis, the cable sleeves will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.4.5 Raychem Nuclear Splice Kits, Model NESK

Introduction

Raychem Nuclear Splice Kits, model NESK are used in applications both inside and Outside of containment. Splice kits provide an insulation function for electrical conductors. These kits are considered cable breakout and end sealing kits used for specific applications. These Raychem Nuclear Splice Kits were first purchased in 1985. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The existing thermal and radiation analyses support a qualified life of 40 years at full rated temperature. Reanalysis of the splice kit qualified life will be performed prior to the end of the current license period. If the qualification cannot be extended by reanalysis, the splice kit will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.4.6 Raychem Nuclear Splice Kits, Model NPKV

Introduction

Raychem Nuclear Splice Kits, model NPKV are used in applications both inside and Outside of containment. Splice kits provide an insulation function for electrical conductors. These kits are considered stub connection kits (with end sealing) used for specific applications. These Raychem Nuclear Splice Kits were first purchased in 1982. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The existing thermal and radiation analyses support a qualified life of 42 years at full rated temperature. Reanalysis of the splice kit qualified life will be performed prior to the end of the current license period. If the qualification cannot be extended by reanalysis, the splice kit will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.5 Wire and Cable

4.4.5.1 Kerite 600Volt HTK Insulated, FR Jacketed Power Cable

Introduction

Kerite 600Volt HTK Insulated, FR Jacketed Power Cable is used throughout Ginna station, both inside and outside of the Containment Building. EQ applications include the Recirculation Fan Cooler motors, RHR pump motors, Hydrogen Recombiner Fan motors, and high voltage portions of the Hydrogen Recombiner igniter circuits. The electrical loads supplied by these cables do not run continuously. The CRFC and RHR systems use 500MCM and 350MCM cable respectively and are the only EQ applications with a normal run-time of greater than or equal to 50% of service life. Qualification is based on DOR guidelines.

Conclusions

The thermal and radiation reanalyses support a qualified life in excess of 60 years. Thermal aging calculations consider the effects of cable self-heating. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.2 Kerite 600 Volt FR Insulated, FR Jacketed Control Cable

Introduction

Kerite 600 Volt FR Insulated, FR Jacketed control cable is used throughout Ginna station both inside and outside of the containment building. EQ applications include both inboard and outboard low voltage portions of the Hydrogen Recombiner igniter circuits. Also, inboard control circuits for the residual heat removal discharge motor operated valve (MOV) use this type of cable. Qualification is based on DOR guidelines.

Qualification life for this cable is based on maintaining the electrical function of the insulation. Therefore the insulation is the limiting component for evaluation of cable life.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Normal load current for these control circuits is much less than the ampacity and therefore self-heating is not considered significant. A review of plant temperatures provides reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.3 Conax Electric Conductor Seal Assembly

Introduction

Conax Electrical Conductor Seal Assemblies (ECSAs) are used in the Auxiliary, Containment, and Intermediate Buildings at Ginna Station to seal electrical conductor entrance into electrical terminal housings. An ECSA is a passive electrical device consisting of Kapton polymide film insulated solid copper conductors sealed into a stainless steel tube using polysulfone plastic. The stainless steel tube is swaged into threaded metal fittings. External leads are covered with polyolefin heat shrinkable tubing to provide mechanical protection for the conductor insulation and are routed to terminal boxes via flexible metal conduit.The ECSAs are used to seal solenoid operated valves (SOVs) and resistance temperature detectors (RTDs). Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Thermal aging calculations consider the effects of elevated operating temperature during SOV operation. A review of plant temperatures provides reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.4 Anaconda FR-EP and FR-EP/CPE Cable

Introduction

Anaconda FR-EP and FR-EP/CPE (CPE Jacket) instrument and control cable is used throughout Ginna station in motor operated vale (MOV), solenoid operated valve (SOV), position switch/indicator, thermocouple, pressure transmitter, and resistance temperature detector applications. The cable is installed both inside and outside the Containment Building. The MOV power applications use #10 AWG field wiring extending from penetration splice boxes to the pressurizer cubicle. The stroke time for these valves is less than two minutes. Therefore operation is a negligible portion of service life and self-heating due to power loads is not a significant contributor to thermal aging. The MOV and position switch applications use #14 AWG cable for internal DC control wiring and #12 AWG for position switch wiring, respectively. Qualification is based on IEEE 323-1974. A unique (and limiting) application of this cable is in the pressurizer cubicle. The temperature inside of the cubicle is significantly higher than ambient.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Normal load current for these circuits is much less than the ampacity and therefore self-heating is not considered significant. A review of plant temperatures provides reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.5 General PVC Insulated and Jacketed Control Cable

Introduction

General PVC insulated and jacketed control cable is used outside of the containment building for two EQ solenoid operated valves that are associated with the Post Accident Sampling System. The SOV applications use #12 multi-conductor cables installed between a control panel and a penetration outboard splice box. Qualification is consistent with DOR guidelines.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Normal loading on this type of cable consists of position indication lamps which present a load current that is much less than the cable ampacity. Therefore, self-heating is not considered significant. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation, and no reductions in excess conservatisms were necessary. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.6 PVC Instrument Cable Outside Containment (Rome Cable Corp.)

Introduction

Rome Cable Corporation PVC Insulated and Jacketed Instrument cable is used outside the Containment Building in Hydrogen Recombiner circuits running between the control panels and Intermediate Building outboard penetration splice boxes. These circuits carry flow transmitter and thermocouple signals. This cable is also used in pressure transmitter circuits located in both the Auxiliary and Intermediate Buildings. These Auxiliary and Intermediate Building circuits are subject to HELB environments. The conductors are PVC insulated, twisted, shielded and jacketed with glass braid. Qualification is based on a combination of tests which are acceptable for DOR Guideline equipment.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. All of the EQ cables carry low voltage level analog instrument signals and operate with milliampere circuit loadings and therefore have negligible cable self-heating. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. The analytical models, temperature, and activation energies used in the reanalysis are the same as those used in the prior evaluation, and no reductions in excess conservatisms were necessary. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.7 BIW Tefzel ETFE Coaxial and Triaxial Cable

Introduction

BIW Tefzel ETFE Coaxial and Triaxial Cable are used in two High Range Radiation Monitor installations at Ginna Station. The High Range Radiation Monitor detectors are located in Containment. There are two circuits for each detector; these are separate high voltage supply and signal cables. For each circuit BIW Tefzel ETFE coaxial cable is run in Containment and the Intermediate Building. BIW Tefzel ETFE Coaxial and Triaxial Cable installed at Ginna Station is qualified to IEEE 323-1974. The BIW coaxial cable was purchased from Victoreen with the High Range Radiation Monitors in 1979. The triaxial cable was purchased from BIW in 1986. Both the cable insulation and jacket material is Dupont Tefzel fluoropolymer (ETFE). These cables operate with milliampere circuit loadings and therefore have negligible cable self-heating.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 50 years. These cables operate with milliampere circuit loadings and therefore have negligible cable self-heating. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Since the purchase order date for the HRRMs including coaxial cable is 1979 and the triaxial cable was purchased at a later date, this qualified life provides adequate margin for extended operation to 2029. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.8 BIW EPR Insulated and CSPE Jacketed Cable

Introduction

BIW EPR Insulated and CSPE Jacketed cable is used inside the Containment Building and throughout the Ginna StationGinna Station. EQ applications in the Containment Building include motor operated valve (MOV) control, solenoid operated valve (SOV), position indication, pressure transmitter, and resistance temperature detector circuits. MOV control circuits extend from containment electrical penetrations to the pressurizer cubicle.. A bounding analysis was performed by applying the pressurizer cubicle temperatures to the SOV circuits.

Cable loading for the SOVs is based on twice the normal operating requirements of once per week or 52 cycles per year at one hour per cycle. This results in an average of 104 operating hours per year. This cable was first purchased for installation in EQ applications at Ginna Station in 1985. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Thermal aging calculations consider the effects of cable self-heating. A review of plant temperatures in the reanalysis provide reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.9 Raychem Flamtrol Shielded Cable

Introduction

Raychem Flamtrol Shielded cable is used in the Containment Building at Ginna Station in EQ position indication, and pressure transmitter applications. Power Operated Relief Valve position indicator circuits extend in conduit from the Containment Building basement up the pressurizer cubicle inside walls to the top level. This cable was a one time purchase at Ginna Station made in 1975. The elevated ambient temperature in the pressurizer cubicle is considered the worst case for this installation. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. All EQ Raychem Flamtrol Shielded cable circuits are instrument applications, operate with milliampere circuit loadings, and therefore have negligible cable self-heating. A review of plant temperatures demonstrates that the temperature used to determine the qualified life is conservative. The analytical models, temperatures, and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.10 General PVC Insulated and Jacketed Control Cable

Introduction

General PVC Insulated and Jacketed Control cable is used inside the Containment Building at Ginna Station in a single EQ solenoid operated valve (SOV) application . This SOV controls a Post Accident Sample System sample valve located in the Containment Building basement, Loop B area. This SOV application uses both single conductor #14 AWG and ten conductor #12 AWG cable. The operating function requires energization 1.5 times per day for a total energized time of 45 minutes per day. This cable is original plant equipment. The elevated ambient temperature in the pressurizer cubicle is considered the worst case for this installation. Qualification is based DOR Guidelines.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Normal load current for this control cable is much less than the ampacity and therefore self-heating is not considered significant. A review of plant temperatures in the reanalysis provide reasonable assurance that the temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.11 Containment Electrical Penetration Pigtail Extension Cables

Introduction

The penetration extension cable is single conductor silicone rubber insulated with a braided jacket. This cable is noted as installed in containment. Qualification is based on DOR guidelines.

Conclusion

The existing thermal and radiation analyses support a qualified life of 52 years. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life remains conservative. At this time, there are no plans to extend the qualified life of the existing cable, and therefore the cable will remain scheduled for replacement prior to the year 2021. In the event that reanalysis is performed, it will be completed consistent with the guidance provided in NUREG-1800 table 4-4. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.5.12 Coleman Silicone Rubber Instrument Cable Inside Containment

Introduction

The Coleman cable is twisted shielded instrumentation cable with silicone rubber insulation and a glass braid jacket. This cable is noted as installed in containment. Qualification is based on DOR guidelines.

Conclusion

The previous therman and radiation analyses support a qualified life of 52 years. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. At this time, there are no plans to extend the qualified life of the existing cable, and therefore the cable will remain scheduled for replacement prior to the year 2021. In the event that reanalysis is performed, it will be completed consistent with the guidance provided in NUREG-1800 table 4-4. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.5.13 Okonite Control Cable

Introduction

The Okonite cable which is part of the Reactor Head Vent cable-connector assemblies is 5/C #12 with 0.30 Okonite-FMR (EP) insulation and 0.045 Okolon (SCPE) jacket. This cable is an integral part of the Reactor Head Vent cable connector assembly and therefore the qualified life is considered to be limited by the connector qualified life (30 years). Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life of 30 years. A review of plant temperatures for the existing analyses provide reasonable assurance that the temperature used to determine qualified life remains conservative. At this time, there are no plans to extend the qualified life of the existing cable-connector assemblies, and therefore these assemblies will remain scheduled for replacement prior to the year 2017. In the event that reanalysis is performed, it will be completed consistent with the guidance provided in NUREG-1800 table 4-4. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.5.14 Conax Core Exit Thermocouple Connector/Cable Assemblies

Introduction

Conax Core Exit Thermocouple (CET) Connector/Cable assemblies are installed in the Containment Building, in open trays above and to the side of the reactor head. These Connector/Cable assemblies provide signals for display of Reactor Coolant System temperature in the reactor vessel. The Conax CET Connector/Cable assemblies are circular two pin connectors consisting of a mechanically swaged stainless steel, polysulfone and Kapton wire insulation interfaces. The oldest of these assemblies was purchased in 1984 with installation that year.

For the evaluation of qualified life, the polysulfone insulating material used in the connector is the limiting component for the thermal analysis, and the Kapton wire insulation in the connector is the limiting component for the radiation analysis.

Conclusion

The thermal, radiation and cyclical wear analyses support a qualified life in excess of 60 years. Normal load current for these Connector/Cable assemblies is much less than the ampacity and therefore self-heating is not considered significant. A review of plant temperatures in the reanalysis provides reasonable assurance that the temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.15 Namco Limit Switch Connector/Cable Assemblies

Introduction

The Namco limit switch connector/cable assemblies are used for pressurizer PORV position indication. The connector/cable assemblies were purchased in 1994 and qualified to IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life of 40 years. The normal load current through the pressurizer PORV limit switches is much less than the ampacity and therefore self-heating of the connector/cable assemblies is not considered significant. A review of plant temperatures in the existing analysis provides reasonable assurance that the temperature used to determine qualified life is conservative. Due to the date of purchase and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.5.16 Brand Rex Electrical Cable

Introduction

The Brand Rex cables are used to provide instrumentation and control power for class 1E circuits. These cables were purchased from 1985 to 1997 and qualified to IEEE 323-1974.

Conclusion

The existing thermal and radiation analyses support a qualified life of 40 years. Reanalysis of the cable qualified life will be performed prior to the end of the current license period. If the qualification cannot be extended by reanalysis, the cable will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.6 Electrical Connectors

4.4.6.1 Amphenol Triaxial Cable Connector

Introduction

Amphenol Triaxial Cable Connectors (Plug#53175-1004 and Jack#52957-1001) are used in the high range radiation monitor circuits. The connectors were purchased in 1989 and qualified to IEEE 323-1974. The insulator used in the connector is made of cross linked polyolefin.

Conclusion

The thermal and radiation analyses support a qualified life of 40 years. The installed application is for instrumentation circuits and therefore self-heating temperature rise is not considered to be significant. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Due to the date of purchase and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.6.2 EGS Quick Disconnect Electrical Connectors, ITT Cannon GB-1

Introduction

ITT Cannon GB-1 connectors supplied by EGS are used to provide environmentally sealed connections for Class 1E circuits. The connector is a single conductor, separable connector with an elastomer body. The connectors were purchased in 1992 and qualified to IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 40 years. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Due to the date of purchase and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.6.3 Quick Disconnect Electrical Connectors, EGS

Introduction

EGS quick disconnect connectors are used to provide environmentally sealed connections for Class 1E power and control circuits. These are multiple conductor, bayonet type, quick-disconnect connectors. The connectors were purchased in 1996 and qualified to IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life of 40 years. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Due to the date of purchase and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.6.4 States Terminal Blocks Model M-25012

Introduction

States terminal blocks are used for an EQ application in the Intermediate Building for main steam line isolation valve instrument air solenoid valves. The terminal block is a passive device consisting of phenolic as the most age sensitive material. Plant documents identify the States terminal blocks were installed in 1983. The installed location is not normally considered a harsh environment, however the equipment supported must maintain the intended function for a main steam line break scenario. Therefore the components are considered to be exposed to elevated temperatures for a portion of the service life which will encompass the design basis event. Qualification is based on DOR guidelines.

Conclusion

The thermal and radiation analysis support a qualifies life of 40 years. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Reanalysis of the terminal block qualified life will be performed prior to the end of the current license period. If the qualification cannot be extended by reanalysis, the terminal blocks will be refurbished, replaced, or requalified prior to exceeding the current qualification.

Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.6.5 AMP Butt Splices, Models 53549-1 and 53550-1

Introduction

AMP Nuclear PIDG Window Indent Butt Splices Models 53549-1 and 53550-1, are used in containment and throughout the plant for electrical connections. The butt splices are a passive component, consisting of KYNAR Polyvinylidene Fluoride (PVDF) as the most age sensitive material. Plant documents identify that the AMP butt splices were first purchased for EQ applications in 1985 and qualified to IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life of 40 years. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Reanalysis of the splice qualified life will be performed prior to the end of the current license period. If the qualification cannot be extended by reanalysis, the splice will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.6.6 AMP Butt Splices, Models 52979 and 52980

Introduction

AMP Nuclear pre-insulated environmentally sealed butt splices Models 52979 and 52980, are used in containment and throughout the plant for electrical connections. The butt splices are a passive component, consisting of KYNAR Polyvinylidene Fluoride (PVDF) as the most age sensitive material. Plant documents identify that the AMP butt splices were first purchased for EQ applications in 1985 and qualified to IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life of 40 years. The qualified life of these components considers the self-heating temperature rise consistent with rated load current. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Reanalysis of the splice qualified life will be performed prior to the end of the current license period. If the qualification cannot be extended by reanalysis, the splice will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.7 Resistance Temperature Detectors

4.4.7.1 Conax Resistance Temperature Detector Models 7N92-10000 and 7A22-10000

Introduction

Conax RTD Models 7N92-10000 and 7A22-10000 are used in the reactor coolant system to measure process fluid temperatures.These units are direct immersion RTDs located in the A and B hot and cold leg piping, inside the shield wall with cold end lead configurations external to process piping insulation. The two models account for two different cold end lead configurations.

These RTDs consist of dual three wire platinum coils connecting to Kapton polyimide film insulated solid copper leads all in a mineral insulated stainless steel sheath with polysulfone plastic sealant at the cold end lead wire exit. Mineral electrical insulation is used where hot end components are exposed to process heating.

Plant documents identify the oldest of the installed RTDs to have been purchased in 1982. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Self heating for instrument cable is considered to be insignificant. A review of plant temperatures in the reanalysis provides reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.7.2 Conax Resistance Temperature Detector Model 7DB9-10000

Introduction

Conax RTD Model 7DB9-10000 is used in the reactor vessel level monitoring system to measure process fluid temperatures. These submersible RTDs are installed in three locations outside the shield wall. These RTDs consist of dual three wire platinum coils connecting to Kapton polyimide film insulated solid copper leads all in a mineral insulated stainless steel sheath with polysulfone plastic sealant at the cold end lead wire exit. Self heating for instrument cable is considered to be insignificant. Mineral electrical insulation is used where hot end components are exposed to process heating. The effects of elevated ambient temperatures have been considered for the calculation of thermal life.

Plant documents identify the installed RTDs were purchased and installed in 1989. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Self heating for instrument cable is considered to be insignificant. A review of plant temperatures in the reanalysis provides reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.7.3 Pyromation Resistance Temperature Detector

Introduction

Pyromation model RT186S28123-00-36Z RTDs are used to provide Containment Building atmosphere temperature monitoring. This is an indicating function based on Regulatory Guide 1.97 Category II. These RTDs consist of a single three wire platinum coil connecting to Kapton polyimide film insulated solid copper leads. The limiting age degradable material is the United Resins potting compound used for lead wire terminations. The RTDs were first purchased in 1989 and qualified to IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life of 40.5 years. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Due to the date of purchase and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.8 Victoreen High Range Radiation Monitor

Introduction

Two Victoreen supplied Detector and Cable/Connector assemblies are installed in the Containment Building near the outside wall. . These components are used with Victoreen High Range Radiation Monitors which are located in the Control Room. The assemblies were installed with their current connector configuration in 1982 and were originally qualified to the DOR Guidelines.

Detector materials are metals and ceramics, with no applicable aging mechanisms. Qualification life is therefore based on the cable/connector assembly. For the evaluation of qualified life, the cable insulation (Tefzel) is the limiting component for the thermal analysis, and the Raychem connector encapsulation seal is the limiting component for the radiation analysis.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. There is no significant self-heating associated with instrument and control wiring. A review of plant temperatures provides reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.9 Rosemount Conduit Seal

Introduction

Rosemount conduit seals (model 353C) are used to seal transmitter housings and transmit process signals for steam generator wide range level indication. The seals were purchased in 1991 and qualified to IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life of 40 years. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Due to the date of purchase and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.10 Transamerica Delaval Level Switch

Introduction

The Transamerica Delaval, Gem Level Sensor , Model LS57761 is a float switch used in the Containment Building basement to provide Sump B level indication following a design basis event. The level sensor is a magnetic reed type liquid switch. A spherical float surrounds a stationary stem and moved up and down with the liquid level. All metal parts are stainless steel and the stem and junction box are completely filled with Dow-Corning 710 silicone fluid, with splices submerged. The junction box is connected to conduit in a configuration arranged to avoid field wiring submergence prior to switch activation. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. A review of plant temperatures in the reanalysis provides reasonable assurance that the reduced temperature used to determine qualified life is conservative.The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation, and no reductions in excess conservatisims were necessary. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.11 Schaevitz Linear Variable Differential Transformer

Introduction

Schaevitz Linear Variable Differential Transformers (LVDTs) Model 500 XS-ZTR are used for indication of pressurizer safety valve position. The LVDTs were purchased in 1986 and qualified based on DOR guidelines. They are designed to operate continuously under 2500 psi, 650EF conditions, with a total integrated dose of 2.5E11 RADs. These design conditions are much more severe than the postulated accident conditions at Ginna.

Conclusion

A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. The LVDTs are constructed of completely inorganic insulating materials (i.e. ceramics and metals). Therefore, there are no aging effects in the period of extended operation, and reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.12 Hydrogen Recombiner Exciter, Ignitor, and Thermocouples

Introduction

The Hydrogen Recombiner system consists of two full-rated subsystems, each capable of maintaining the post-LOCA containment hydrogen concentration below 4 volume percent. The specific subsystem components covered by this evaluation include the exciter, ignitor, and thermcouples. The exciter consists of a transformer, capacitors, inductor, rectifiers, electron discharge tube and resistors enclosed in a seal welded metal case. The input and output connections are made with a gasket sealed junction box welded to the side of the metal case. The junction box is filled with GE RTV-7403 to preclude the entry of moisture. The ignitor is a component that produces a spark from the 2300V DC pulse generated by the exciter. This component does not contain any sub-components which are susceptible to age, radiation, temperature or chemical spray degradation. To ensure proper lite-off and temperature control of the hydrogen recombiners, they incorporate two thermocouples that sense the combustor outlet temperature. Qualification for this equipment is based on DOR guidelines.

Conclusion

Reanalysis of the hydrogen recombiner equipment qualified life will be performed prior to the end of the current license period. If the qualification is still required by 10 CFR 50.44, and cannot be extended by reanalysis, the components will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.5 Concrete Containment Tendon Prestress

Introduction

The Ginna Station containment structure is a reinforced concrete, vertical right cylinder with a flat base and hemispherical dome. The cylinder wall is prestressed vertically and reinforced circumferentially with mild steel deformed bars. Vertical prestress is provided by 160 unbonded post-tensioned tendons. The base slab and dome are constructed of reinforced concrete.

The design of the containment provides for prestressing the cylinder walls in the vertical direction with sufficient compressive force to ensure that for all design load combinations there are no membrane tensile forces in the concrete. The design requires that all bending and shear forces be resisted by mild steel reinforcement, which controls potential crack width, spacing, and depth.

The prestressing force of containment tendons may decrease over time due to creep, shrinkage and elastic shortening of the concrete, and stress-relaxation of the prestressing tendon wires. Prestressing tendon integrity is monitored and confirmed by the ASME Section XI, Subsection IWE/IWL Inservice Inspection Program.

An analysis was performed to evaluate the trend in the loss of prestress for each of the 160 tendons at Ginna Station. A review of the historical lift-off force measurements for the tendons was conducted. It was appropriate to review the results as two separate groups, i.e., the 23 tendons which were retensioned in 1969, and the 137 tendons which were retensioned in 1980. Of the 23 tendons that were retensioned in 1969, eleven have been tested during the surveillances since 1980. Of the 137 tendons that were retensioned in 1980, forty-seven have been tested during the subsequent surveillances. The number of tendons sampled during the surveillance tests exceeds the requirements of Regulatory Guide 1.35.

Using the guidance in RG 1.35.1, tolerance bands were calculated and the lift-off forces measured during surveillance tests were expressed in terms of margins. It was concluded that the group of 23 tendons originally retensioned in 1969 should be retensioned as documented in the Evaluation of Loss of Prestress in Containment Tendons TLAA . These tendons have exhibited loss of prestress as determined during previous surveillance tests. Retensioning should preclude further loss of prestress.

Conclusion

It is concluded that retensioning the group of 23 tendons in 2005 will provide additional assurance that the minimum design tendon prestress force will be maintained through the period of extended operation.

Based on this review, the ASME Section XI, Subsections IWE & IWL Inservice Inspection Program will manage the effects of aging for the Containment post-tensioning system during the extended period of operation in accordance with 10 CFR 54.21(c)(1)(iii).

4.6 Containment Liner Plate and Penetration Fatigue

Introduction

The interior surface of the Containment Structure is lined with welded steel plate to provide an essentially leak-tight barrier. At all penetrations, the liner plate is thickened to reduce stress concentrations.

The containment liner, liner penetrations and liner steel components of the Ginna Station Containment Structure comply with the ASME Code Section III-1965 for pressure boundary and the AISC Code for structural steel. The containment liner and penetrations, including the equipment and personnel hatch penetrations, were designed as Class B Vessels. The Winter 1965 Addenda of ASME Section III, Subsection B, N-1314(a) requires that the containment vessel satisfy the provisions of Subsection A, N-415.1, "Vessels Not Requiring Analysis for Cyclic Operation," in order that Subsection B rules be applicable.

ASME Section III, N-415.1 states that a fatigue analysis is not required, and it may be assumed that the peak stress intensity limit has been satisfied for a vessel or component by compliance with the applicable requirements for materials, design, fabrication, testing, and inspection, provided the service loading of the vessel or component meets all of six (6) conditions. These conditions address the following types of loads:

- · Atmospheric to Service Pressure Cycles
- · Normal Service Pressure Fluctuation
- · Temperature Difference - Startup and Shutdown
- · Temperature Difference - Normal Service
- · Temperature Difference - Dissimilar Materials
- · Mechanical Loads

The pressure boundary components analyzed include the liner adjacent to the penetration, the penetration sleeve, and the annular plate connecting the pressure piping to the sleeve.

An analysis was performed which verifies that each of the six conditions described above are satisfied for the period of extended operation. This analysis demonstrates that the liner and penetrations comply with the ASME Section III - 1965 Code Rules for fatigue through the period of extended operation.

Conclusion

The six criteria of ASME Code Section III, N-415.1 (Reference 17) have been reevaluated and shown to be satisfied for the Containment liner plate, penetrations and penetration sleeves for 60 years of plant operation. Based on this evaluation, the fatigue analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.7 Other Plant Specific TLAAs

4.7.1 Containment Liner Stress

Introduction

The containment liner is carbon steel plate conforming to ASTM A442-60T Grade 60 with a minimum yield of 32,000 psi, and a buckling stress of 16,600 psi at operating conditions. The liner plate thickness is 1/4 in. for the base and 3/8 in. for the cylinder and dome. The liner stresses (meridional directions) were calculated to be 4500 psi compression based upon a prestress force of 0.70 fs. The concrete strain due to creep and shrinkage was established as being 320 x 10^{-6} in/in. This increases the liner stress to 14,100 psi at the end of 40 years.

Conclusion

The creep and shrinkage strain occurring over a 60-year plant life was calculated, and the resulting compressive liner stress due to both time-dependent and non-time dependent loads is determined to be 14,870 psi. This liner stress is less than the liner buckling stress of 16,600 psi and therefore the analysis has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.7.2 Containment Tendon Fatigue

Introduction

A discussion of seismic considerations for tendons is provided in the Ginna UFSAR. Fatigue tests were conducted on tendon wire materials in 1960 by an independent testing lab. The tests indicated that the tendons were capable of withstanding over 2 million cycles at stress levels between 135 and 158 ksi. The test results were used to conclude that dynamic loads, considering especially pulsating loads resulting from an earthquake, do not jeopardize buttonhead anchorage.

This discussion may not meet the definition of a TLAA as described in 10 CFR 54.3, however it has been included for conservatism.

Conclusion

The tendon fatigue test results described in the UFSAR were cited to address tendon integrity during cyclic loading of the containment caused by a design basis seismic event. The tendons were tested to 2 million cycles, which exceeds, by many orders of magnitude, the total cycles that could accumulate through multiple seismic events over 60 years. The seismic fatigue evaluation remains valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.7.3 Containment Liner Anchorage Fatigue

Introduction

A fatigue analysis of the fillet weld attaching the channel anchors to the liner was performed as part of the original design. The allowable fatigue stress of the attachment weld was set equal to the stress caused by static loading. This stress equals 13,600 psi and corresponds to 100,000 stress cycles.

This discussion may not meet the definition of a TLAA as described in 10 CFR 54.3, however it has been included for conservatism.

Conclusion

A total of 100,000 stress cycles corresponds to more than 4 full stress cycles per day for 60 years. Fluctuations of temperature and pressure in containment on a daily basis are not significant enough in magnitude to cause four cycles of design basis stress at the liner anchorage weld each day. Therefore, the original fatigue analysis remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.7.4 Containment Tendon Bellows Fatigue

Introduction

The allowable radial and vertical displacements of the containment stainless steel tendon bellows are given in the UFSAR and are limited to two cycles per year for the 40-year life of the plant. This limits the total number of allowable displacement cycles to 80. Since the completion of construction, displacements at the tendon bellows have occurred due to pressure testing and temperature changes in the cylindrical shell wall due to summer/winter conditions and reactor shutdown during refueling outages.

This discussion may not meet the definition of a TLAA as described in 10 CFR 54.3, however it has been included for conservatism.

Conclusion

Assuming that 80 full cycles of allowable displacement results in a fatigue usage factor of 1.0, the actual fatigue usage factor over a 60-year period has been calculated to be much less than 0.01. Therefore, the original fatigue analysis remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.7.5 Crane Load Cycle Limit

Introduction

Each of the crane estimated cycle numbers were compared to the Design Load Cycles. They are all well below the upper Design Loading Cycle limit. In addition, the average percent of the rated load lifted was well below the 50% level, relative to the design load cycles, as set forth in the design criteria.

Conclusion

Since the number of operating load cycles for the cranes will be less than the design cycles and the average percent of rated load lifted is less than 50% for the design load cycles, the crane designs will remain valid for the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

4.7.6 RCP Flywheel

Introduction

During normal operation, the reactor coolant pump (RCP) flywheel possesses sufficient kinetic energy to produce high-energy missiles in the event of failure. Conditions which may result in over-speed of the RCP increase both the potential for failure and the kinetic energy. The aging effect of concern is fatigue crack growth in the flywheel bore keyway.

Westinghouse Topical Report WCAP-14535, “Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination” (Reference 18), presents an evaluation of the probability of failure over an extended operating period of 60 years. This report demonstrates that the flywheel design has a high structural reliability with very high flaw tolerance and negligible flaw crack extension over a 60-year service life. The Westinghouse Topical Report provides technical justification for elimination of the RCP flywheel inspection. This Topical report was reviewed and approved by the NRC (Reference 19), for referencing in licensing, with certain restrictions and limitations specified. This was re-published as WCAP-14535A in November 1996.

Regulatory Position C.4.b(1) of NRC Regulatory Guide 1.14, Revision 1, August 1975 (Reference 20), recommended performance of an in-place ultrasonic volumetric examination of the areas of higher stress concentration at the Reactor Coolant Pump flywheel bore and keyway at approximately 3-year intervals. Regulatory Position C.4.b(2) of this regulatory guide (Reference 20), recommended a surface examination of all exposed surfaces and complete ultrasonic examination at approximately 10-year intervals, coinciding with the ASME Section XI Inservice Inspection program schedule. This recommendation was incorporated in the Ginna Station Inservice Inspection (ISI) program. Based on

WCAP14535A, and in accordance with NRC recommendations, RG&E requested and received a releif request from the NRC revising the ISI frequency of examination to perform the flywheel inspection once every 10 years. The method of examination includes either an ultrasonic examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or an ultrasonic and a surface examination of exposed surfaces defined by the volume of the disassembled flywheel.

Conclusion

In accordance with the requirements of the ISI Program, the RCP flywheel inspection program will continue to ensure that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

4.7.7 Thermal Aging of Cast Austenitic Stainless Steel (CASS)

Introduction

Thermal aging refers to changes in the microstructure and properties of a susceptible material due to prolonged exposure to elevated temperatures above 482^{o}F. Typical RCS temperatures exceed this threshold. The effect of thermal aging on Class 1 components is loss of fracture toughness (embrittlement) of the duplex ferritic-austenitic stainless steel elbow castings in the reactor coolant piping and the cast reactor coolant pump casings. Since the embrittlement reaction is time-dependent, the associated aging effect is therefore treated as a TLAA. A Leak-Before-Break (LBB) (flaw tolerance) analysis is typically performed to demonstrate that any leaks from through-wall cracks that develop in RCS piping would be detected by plant monitoring systems before the cracks could grow to unstable proportions. For RCS piping, this analysis must consider the reduction in fracture toughness of CASS as a result of thermal aging.

A fracture mechanics analysis (Reference 27) has been performed which considers loading, pipe geometry and fracture toughness to assess crack stability in the reactor coolant piping for the period of extended operation. This analysis, which considered the reduction of fracture toughness in CASS elbows in the RCS piping for the period of extended operation, again demonstrated that significant margin exists between detectable flaw sizes and unstable flaws. Additionally, fatigue crack growth rates including environmental effects were evaluated for primary loop piping and shown to be insignificant.

Similarly, in lieu of performing volumetric inspections of the cast austenitic stainless steel (CF8M) RCP casings, a fracture mechanics analysis, according to the requirements of ASME Code Case N-481 has been performed for the period of extended operation (Reference 28). The results of this analysis demonstrated that the fracture toughness of the pump casing materials in the fully-aged condition is sufficient to meet the stability criteria for a postulated flaw.

Conclusions

Flaw-tolerance analyses have been performed to evaluate the reduction in fracture toughness due to thermal aging of CASS reactor coolant system elbows and the RCP casings through the extended period of operation. The results demonstrate that large margins exist for postulated flaw sizes against flaw instability. Therefore, the analyses have been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

Section 4.0 References

1. WCAP 7733, Reactor Vessel Weld Cladding - Base Metal Interaction, July 1971
2. WCAP-15338, A review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants, March 2000
3. NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, Published July 2001
4. NEI 95-10, [Revision 3], “Industry Guideline for Implementing the Requirements of 10 CFR 54 -the License Renewal Rule” March 2001
5. Memorandum, Ashok C. Thadani, Director, Office of Nuclear Regulatory Research, to William D. Travers, Executive Director of Operations – “Closeout of Generic Safety Issue 190, Fatigue Evaluation of Metal Components for 60 Year Plant Life," U. S. Nuclear Regulatory Commission, December 26, 1999
6. NUREG/CR-6260 (INEL-95/0045), “Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components,” U.S. Nuclear Regulatory Commission, March 1995
7. NUREG/CR-6583, “Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels,” U.S. Nuclear Regulatory Commission, March 1998
8. NUREG/CR-5704, “Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels,” U.S. Nuclear Regulatory Commission, April 1999.
9. O. Chopra, "Status of Fatigue Issues at Argonne National Laboratory," Presented at EPRI Conference on Operating Nuclear Power Plant Fatigue Issues & Resolutions, Snowbird, UT, August 22-23, 1996.
10. NUREG/CR-5999 (ANL-93/3), “Interim Fatigue Design Curves for Carbon, Low-Alloy, and Austenitic Stainless Steels in LWR Environments,” U. S. Nuclear Regulatory Commission, August 1993
11. Structural Integrity Associates Report Number SIR-02-005, “Plant-Specific Fatigue, Environmental Fatigue Calculations for Point Beach Nuclear Plant, Units 1 and 2,” Revision 1, May 2002
12. WCAP-12928, “Structural Evaluation of the Robert E. Ginna Pressurizer Surge Line, Considering the Effect of Thermal Stratification”, May 1991

13. Ginna Station Fourth Interval Inservice Inspection (ISI) Program, January 1, 2000

14. ASME Boiler and Pressure Vessel Code, Section XI, Appendix L “Operating Plant Fatigue Assessment,” 1995 Edition

15. WCAP 7410-L, Environmental Testing of Engineered Safety Feature Related Equipment (NSSS - Non-Standard Scope)

16. IEEE 323-1976, Environmental Qualification of Electrical Equipment Important to Safety - Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.

17. ASME Code Section III, Paragraph N-415.1

18. Westinghouse Topical Report WCAP-14535A, “Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination” SER published, September 1996.

19. Westinghouse Topical Report WCAP-14535A, “Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination” Republication November 1996

20. NRC Regulatory Guide 1.14, Revision 1, Reactor Coolant Pump Flywheel Integrity (for Comment), August 1975

21. Safety Evaluation, Topical Report on Reactor Coolant Pump Flywheel inspection Elimination, Beaver Valley 1 and 2

22. BAW-2425, Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessel of R. E. Ginna for Extended Life of 54 Effective Full Power Years, May 2002

23. BAW-2192-PA, Low Upper-Shelf Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level A and B Service Loads, April 1994

24. BAW-2178-PA, Low Upper-Shelf Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level C and D Service Loads, April 1994

25. WCAP-15885 Revision 0, R. E. Ginna Heatup and Cooldown Limit Curves for Normal Operation, May 2002

26. USNRC Letter Johnson to Mecredy, “Ginna Flaw Indication in the Reactor Vessel Inlet Nozzle Weld - 1989 Reactor Vessel Examination (TAC No. 71906), July 7, 1989

27. WCAP-15837, Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the R. E. Ginna Nuclear Power Plant for the License Renewal Program, April 2002

28. WCAP-15873, A Demonstration of the Applicability of ASME Code Case N-481 to the Primary Loop Casings of R. E. Ginna Nuclear Power Plant for the License Renewal Program, April 2002

APPENDIX A

UFSAR SUPPLEMENT

APPENDIX A
Contents

Contents

A1.0 APPENDIX A - INTRODUCTION

The application for a renewed operating license is required by 10 CFR 54.21(d) to include a FSAR Supplement. This appendix provides that supplement for the Ginna UFSAR. Section A2.0 of this appendix contains a summarized description of the programs for managing the effects of aging. Section A3.0 of this appendix contains a summary of the evaluation of time-limited aging analyses (TLAAs) for the period of extended operation. TLAA supporting activity summaries are contained in Section A4.0

A2.0 PROGRAMS THAT MANAGE THE EFFECTS OF AGING

This section provides summaries of the programs and activities credited for managing the effects of aging, in alphabetical order. The Ginna Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Section A.2 of NUREG-1800, Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants, published July 2001. The Quality Assurance Program includes the elements of corrective action, confirmation process, and administrative controls, and is applicable to the safety-related and non- safety-related structures, systems, and components that are within the scope of license renewal.

A2.1 Aging Management Programs

The description of the Ginna Aging Management Programs are consistent with their status as configured to apply to the period of extended operation.

A2.1.1 **Aboveground Carbon Steel Tanks**

The functional intent of this program is implemented by the Systems Monitoring and One-Time Inspection Programs. The programs provide for periodic system walkdowns and inspections to monitor the condition of selected above ground carbon steel storage tanks, including an assessment of tank surfaces protected by paints or coatings, although the coatings themselves are not credited to perform a preventive function. For inaccessible surfaces such as concrete foundation interfaces, an inspection of the tank bottom wall thickness is performed.

A2.1.2 **ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection**

The program consists of periodic volumetric, surface, and/or visual examinations and leakage tests of all Class 1, 2, and 3 pressure-retaining components to identify evidence of degradation. Surface and visual examinations of integral attachments are also performed. This program is in accordance with ASME Section XI, 1995 edition through the 1996 addenda. The program also provides for evaluation of inspection results and appropriate corrective actions.

A2.1.3 ASME Section XI, Subsections IWE & IWL Inservice Inspection

The program consists of periodic visual inspection of concrete surfaces for reinforced and prestressed concrete containments, and periodic visual inspection and sample tendon testing of unbonded post-tensioning systems for prestressed concrete containments, for evidence of degradation, assessment of damage and corrective actions. Measured tendon lift-off forces are compared to predicted tendon forces calculated in accordance with Regulatory Guide 1.35. The ASME Section XI, Subsection IWE program consists of periodic visual, surface, and volumetric inspection of pressure retaining components of steel and concrete containments for evidence of degradation. The program also provides for assessment of damage and appropriate corrective actions. This program is in accordance with ASME Section XI, Subsections IWE and IWL, 1992 edition including 1992 addenda.

A2.1.4 ASME Section XI, Subsection IWF Inservice Inspection

This program consists of periodic visual examinations of component supports for evidence of degradation. The program provides for evaluation of inspection results and appropriate corrective actions. This program is in accordance with ASME Section XI, Subsection IWF, 1995 edition, including 1996 addenda.

A2.1.5 Bolting Integrity

The functional intent of this program is implemented by the following programs: 1) ASME Section XI, Subsections IWB, IWC, and IWD ISI Program, 2) ASME Section Subsection IWF ISI Program, 3) Periodic Surveillance and Preventive Maintenance Program, 4) Boric Acid Corrosion Program, 5) Systems Monitoring Program, and 6) Structures Monitoring Program. The program consists of periodic inspections of pressure retaining bolting as delineated in NUREG-1339, and other industry recommendations in EPRI NP-5679 (with exceptions noted in NUREG-1339) for safety related bolting, and EPRI TR-104213 for pressure retaining and structural bolting. The program provides for periodic inspection of closure bolting for indication of loss of preload, cracking, and loss of material.

A2.1.6 Boric Acid Corrosion

The program consists of: (1) visual inspection of external surfaces that are potentially exposed to borated water leakage, (2) timely discovery of leak path and removal of the boric acid residues, (3) assessment of any damage, and (4) follow-up inspections to assure effectiveness of corrective actions. The program scope includes RCS components in accordance with Generic Letter 88-05 as well as non-RCS mechanical, electrical and structural components susceptible to boric acid corrosion which are potentially exposed to borated water leaks.

A2.1.7 Buried Piping and Tanks Inspection

The functional intent of this program is implemented by the One-Time Inspection Program. The Program includes provisions for: (1) preventive measures to mitigate corrosion, and (2) periodic inspections to manage the effects of corrosion on the pressure retaining capacity of buried carbon steel piping and tanks during inspections of opportunity. Preventive measures are in accordance with standard industry practice for maintaining external coatings and wrappings. Buried piping and tanks are inspected visually for any evidence of degradation when they are uncovered for any reason.

A2.1.8 Closed-Cycle (Component) Cooling Water System

The program includes: 1) preventive measures to minimize corrosion by maintaining corrosion inhibitor concentrations within specified limits, 2) surveillance tests and inspections, and 3) nondestructive evaluations of internal surfaces of system components. Evaluations to verify the effectiveness of water chemistry controls are based on the guidelines of EPRI TR-107396 for closed-cycle cooling water systems.

A2.1.9 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

The program requires that cables and connections in accessible areas exposed to adverse localized environments caused by heat, radiation, or moisture are inspected on a periodic basis. Visual inspections for cable and connector jacket surface anomalies such as embrittlement, discoloration, cracking, and surface contamination are performed at least once every ten years.

A2.1.10 Fire Protection

The program includes inspection of fire barriers and functional testing of fire pumps. The Fire Protection Program requires periodic visual inspections of fire barrier penetration seals, fire barrier walls, ceilings, and floors, and periodic visual inspections and functional tests of fire rated doors and dampers to ensure that their operability is maintained. The Fire Protection Program requires that the diesel driven fire pump be periodically tested to ensure that the fuel supply line can perform the intended function. The program also includes periodic inspection and testing of the halon fire suppression system.

A2.1.11 Fire Water System

The program consists of inspections and functional tests of fire suppression components such as sprinklers, hydrants, valves and piping. Periodic full flow flush tests and system performance tests are conducted to prevent corrosion due to silting and biofouling of components. In addition, the system is normally maintained at required operating pressure and is monitored such that loss of system pressure is immediately detected and corrective actions initiated. Internal portions of the fire water system are visually inspected when disassembled for maintenance. Volumetric NDE inspections using appropriate techniques are performed to detect wall loss and fouling. Replacement or representative sample testing of sprinklers with a service life of 50 years is specified.

A2.1.12 Flow-Accelerated Corrosion

The program consists of: (1) conducting appropriate analyses to determine critical locations susceptible to FAC, (2) conducting baseline inspections to determine the extent of thinning at these locations, and (3) performing follow-up inspections to confirm predicted degradation rates. Corrective actions such as repair or replacement are evaluated based on inspection results and predicted rates of wall loss. The program implements the EPRI guidelines in the Nuclear Safety Analysis Center (NSAC) 202L-R2 and utilizes the CHECWORKS predictive code.

A2.1.13 Fuel Oil Chemistry

The program consists of a combination of surveillance and maintenance activities. Monitoring and control of fuel oil contamination in accordance with the guidelines in ASTM Standards D975, D1796, D2709 and D4057 maintains fuel oil quality. Exposure to fuel oil contaminants such as water and microbiological organisms is minimized by periodic cleaning/draining of storage tanks and verifying the quality of new fuel oil before introduction into the tanks.

A2.1.14 Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems

The program evaluates the effectiveness of testing and monitoring activities as well as the effects of past and future usage on the structural reliability of the cranes, hoists and lifting devices that were evaluated in Ginna Station’s response to NUREG-0612. The number and magnitude of lifts made by the hoist or crane are also reviewed. Rails and girders are visually inspected on a periodic basis for evidence of degradation. Functional tests are also performed to assure structural integrity.

A2.1.15 **One-Time Inspection**

The intent of this program is to verify the effectiveness of existing aging management programs by confirming the absence of an aging effect or verifying that the aging effect is developing so slowly that the intended function is expected to be maintained through the period of extended operation. The program methodology includes selection of appropriate inspection techniques and sample size to ensure that the specified age-related degradation will be discovered in a timely manner. The program provides for evaluation of inspection results and appropriate corrective actions.

A2.1.16 **Open-Cycle Cooling (Service) Water System**

The program is based on implementation of the recommendations of Generic Letter 89-13 to ensure that the effects of aging on open cycle cooling water system components will be managed for the extended period of operation. The program includes surveillance and control techniques to manage aging effects caused by biofouling, corrosion, erosion, protective coating failures, and silting in the open cycle cooling water system or structures and components serviced by the open cycle cooling water system.

A2.1.17 **Periodic Surveillance and Preventive Maintenance**

The Periodic Surveillance and Preventive Maintenance Program is used to maintain plant equipment and structure condition and ensure reliability at Ginna Station. The program provides for visual inspection and examination of surfaces of selected equipment items and components, including fasteners, for evidence age-related degradation such as corrosion, wear, cracking, fouling, and loss of mechanical closure integrity on a specified frequency based on operating experience and previous inspection results. The program provides for evaluation of inspection results and appropriate corrective actions. The program also provides for replacement or refurbishment of certain components on a specified frequency based on operational experience.

A2.1.18 **Reactor Vessel Head Penetration Inspection**

The Reactor Vessel Head Penetration Inspection program includes: 1) susceptibility assessment of head components (including alloy 690TT subcomponents) to primary water stress corrosion cracking (PWSCC), 2) monitoring and control of reactor coolant water chemistry to mitigate PWSCC, and 3) inservice inspection (ISI) of reactor vessel head penetrations and bottom-mounted instrument tube penetrations, in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWB, Table IWB 2500-1 (1995 edition through the 1996 addenda). The program provides for evaluation of inspection results and appropriate corrective actions.

A2.1.19 **Reactor Vessel Internals**

The program includes: 1) augmented VT-1 examinations of reactor vessel internals components by techniques yet to be developed, and 2) monitoring and control of reactor coolant water chemistry in accordance with the EPRI guidelines in TR-105714 to ensure the long-term integrity and safe operation of pressurized water reactor (PWR) vessel internal components. The program includes monitoring industry initiatives to develop enhanced visual examination techniques capable of detecting features on the order of .0005 inches in dimension.

A2.1.20 **Reactor Vessel Surveillance**

The program provides for periodic testing of metallurgical surveillance samples to monitor the progress of neutron embrittlement of reactor pressure vessel materials as a function of neutron fluence in accordance with Regulatory Guide 1.99, Rev. 2.

A2.1.21 **Spent Fuel Pool Neutron Absorber Monitoring**

The program monitors long-term performance of borated stainless steel (BSS) panels, credited as a neutron absorber in portions of the spent fuel pool (soluble boron is credited in the rest of the pool). Borated stainless steel surveillance coupons are periodically removed and examined to evaluate coupon thickness and weight loss. The program provides for evaluation of inspection results and appropriate corrective actions.

A2.1.22 **Steam Generator Tube Integrity**

The program incorporates the guidance of NEI 97-06 and EPRI TR-107569 for maintaining the integrity of steam generator tubes. The effects of aging are managed by a balance of prevention, inspection, assessment, repair, and leakage monitoring measures. Plant Technical Specifications assure timely assessment of tube integrity and compliance with primary to secondary leakage limits.

A2.1.23 **Structures Monitoring Program**

The Structures Monitoring Program consists of periodic inspection and monitoring of the condition of structures and structural elements as well as selected non-safety component supports to ensure that aging degradation will be detected and corrected prior to loss of intended function. The program is implemented in accordance with 10 CFR 50.65, NUMARC 93-01, Rev. 2, and Regulatory Guide 1.160, Rev. 2. The program provides for evaluation of inspection results and appropriate corrective actions.

A2.1.24 Systems Monitoring

The program identifies the evidence of age-related degradation on normally accessible exterior surfaces of piping, components and equipment in systems which are within the scope of license renewal. As part of the implementation of 10 CFR 50.65 (Maintenance Rule), specific guidelines for assessing the material condition of systems, structures, and components during system engineer walkdowns were developed. The effects of aging are managed through visual inspection and monitoring of external surfaces for leakage and evidence of material degradation, such as corrosion, cracking, degradation of coatings, sealants and caulking, deformation, debris and corrosion product buildup. The program provides for evaluation of inspection results and appropriate corrective actions.

A2.1.25 Thimble Tubes Inspection

The program manages the integrity of the incore neutron monitoring thimble tubes, which serve as a portion of the reactor coolant pressure boundary. The program provides for periodic inspections to detect thimble tube wall thinning due to wear caused by flow induced vibration and preventive maintenance such as flushing, cleaning and replacement. Thimble tube wear is detected at locations associated with geometric discontinuities or area changes along the reactor coolant flow path. The program provides for evaluation of inspection results and appropriate corrective actions.

A2.1.26 Water Chemistry Control

The program mitigates the effects of aging by controlling the internal environment of components in the primary, borated, and secondary water systems. Chemical species known to accelerate corrosion (e.g., chloride, fluoride, and sulfate) are controlled within specified limits. The program implements the guidelines in EPRI TR-105714 for primary water chemistry, and TR-102134 for secondary water chemistry. The program provides for assessment and trending of water chemistry and implements corrective action strategies.

A3.0 EVALUATION OF TIME-LIMITED AGING ANALYSES

As part of a License Renewal Application, 10 CFR 54.21(c) requires that an evaluation of time-limited aging analyses (TLAAs) for the period of extended operation be provided. The following TLAAs have been identified and evaluated to meet this requirement.

A3.1 Reactor Vessel Neutron Embrittlement

The following analyses affected by neutron irradiation caused embrittlement that have been identified as TLAAs:

- Upper shelf energy Appendix A3.1.1
- Pressurized thermal shock Appendix A3.1.2
- RCS pressure-temperature operating limits Appendix A3.1.3

A3.1.1 Upper Shelf Energy

The Charpy upper shelf energy (USE) is associated with the determination of acceptable Reactor Vessel toughness during operation. 10 CFR Part 50 Appendix G requires that the reactor vessel beltline materials must have a USE of no less than 50 ft-lb) throughout the life of the reactor vessel, unless otherwise approved by the NRC. If the USE of a reactor vessel beltline material is predicted to not meet Appendix G requirements, then licensees must submit an analysis that demonstrates an equivalent margin of safety at least three years prior to the time the material is predicted to not meet those requirements.

In the event that the 50 ft-lb requirement cannot be satisfied as stated in 10 CFR 50 Appendix G, or by alternative procedures acceptable to the NRC, the reactor vessel may continue to operate provided requirement IV.A.1. of Appendix G is satisfied. This requirement states that an analysis must conservatively demonstrate, making appropriate allowances for uncertainties, the existence of equivalent margins of safety for continued operation. Procedures for the analysis are provided in NUREG-0744. Acceptance criteria are included in ASME Section XI, Appendix K.

The upper shelf energy of the limiting circumferential beltline weld in the Ginna reactor vessel is expected to decrease below 50 ft-lbs during the period of extended operation. In order to demonstrate equivalent margins of safety for continued operation, a low upper-shelf toughness fracture mechanics analysis has been performed (BAW-2425) to evaluate the limiting circumferential beltline weld (SA-847) for ASME Levels A, B, C, and D Service Loadings. The analysis demonstrates that the limiting beltline weld satisfies the Appendix K requirements for ductile flaw extensions and tensile stability using projected low upper-shelf Charpy impact energy levels for the weld material at 54 EFPY.

A3.1.2 Pressurized Thermal Shock

The PTS rule, 10 CFR 50.61 provides screening criteria that are a measure of a limiting level of reactor vessel material embrittlement beyond which operation may not continue without further plant-specific evaluation. The pressurized thermal shock screening criteria are given in terms of reference temperature RT_{PTS}. The screening criteria are 270°F for plates and axial welds, and 300°F for circumferential welds.

The results of the revised PTS analysis for the limiting material have been reviewed for compliance with 10 CFR 50.61. The methodology used in PTS analysis is based on the projected neutron fluence at the end of the period of extended operation and does not rely on plant-specific surveillance data to calculate ΔRT_{PTS}. Although plant-specific surveillance data could have been used, generic data proved to be more conservative. The analysis associated with PTS has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii) and found to be acceptable.

A3.1.3 Pressure-Temperature Limits

10 CFR Part 50 Appendix G requires that the reactor pressure vessel (RPV) be maintained within established pressure-temperature (P-T) limits including during heatup and cooldown. These limits specify the maximum allowable pressure as a function of reactor coolant temperature. As the reactor pressure vessel becomes embrittled and its fracture toughness is reduced, the allowable pressure (given the required minimum temperature) is reduced.

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT} and adding a margin.

The reactor vessel neutron fluence values corresponding to the end of the period of extended operation and the reactor vessel beltline material properties have been calculated consistent with Regulatory Guide 1.190. The revised fluence values have been used to determine the limiting value of RT_{NDT} using the methods of Regulatory Guide 1.99. The limiting value of RT_{NDT} was used to calculate reactor coolant system (RCS) pressure-temperature (P-T) operating limits that are valid through the end of the period of extended operation. Consistent with NUREG-1800 section 4.2.2.1.3.3, it is not necessary to implement P-T limits to carry the reactor vessel through 60 years at the time of application. The updated limits will be contained in a pressure-temperature limit report (PTLR) or in the Technical Specification (TS) prior to the period of extended operation.

The analysis associated with P-T operating limits has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

A3.2 Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)

The potential exists for a loss of fracture toughness due to thermal aging of cast austenitic stainless steel (CASS) components. An evaluation of the susceptibility of CASS components at Ginna Station was made, based on the casting method, molybdenum content, and percent ferrite. It was determined that the CASS RCS elbows were susceptible to a loss of fracture toughness due to thermal aging. A plant-specific flaw tolerance evaluation was conducted, and documented in WCAP-15837, “Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the R.E. Ginna Nuclear Power Plant for the License Renewal Program”, April 2002. The evaluation concluded that adequate fracture toughness exists for the RCS loop, including the cast elbows, for the period of extended operation (60 years).

A separate evaluation was made for the reactor coolant pump casings. In WCAP-15873, “A Demonstration of the Applicability of ASME Code Case N-481 to the Primary Loop Pump Casings of R.E. Ginna Nuclear Power Plant for the License Renewal Program,” May 2002, it was concluded that the primary loop pump casings are qualified to item (d) of ASME Code Case N-481 for the period of extended operation (60 years).

The evaluation associated with thermal aging embrittlement has been found to be acceptable to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

A3.3 Metal Fatigue

The following issues are considered separately under the TLAA for Metal Fatigue:

- ASME Boiler and Pressure Vessel Code, Section III, Class 1 Appendix A3.3.1
- Reactor Vessel Underclad Cracking Appendix A3.3.2
- ANSI B31.1 Piping Appendix A3.3.3
- Accumulator Check Valves Appendix A3.3.4
- Environmentally Assisted Fatigue Appendix A3.3.5

Fatigue is the gradual deterioration of a material that is subjected to repeated cyclic loads. Components have been designed or evaluated for fatigue according to the requirements of applicable codes.

A3.3.1 **ASME Boiler and Pressure Vessel Code, Section III, Class 1**

The reactor vessel, pressurizer, steam generators, and reactor coolant pumps have been designed in accordance with the requirements of the ASME Boiler and Pressure Vessel

Code, Section III, Class 1. The reactor vessel internals were designed according to Westinghouse criteria which were later incorporated into ASME Boiler and Pressure Vessel Code. Design codes for the above components are identified in UFSAR Table 5.2-3. The ASME Boiler and Pressure Vessel Code, Section III, Class 1 requires a design analysis to address fatigue and establish limits such that initiation of fatigue cracks is precluded.

Fatigue usage factors for critical locations in the NSSS components were determined using design cycles that were specified in the plant design process. These design cycles were intended to be conservative and bounding for all foreseeable plant operational conditions. The design cycles were subsequently utilized in the design stress reports for various NSSS components satisfying ASME fatigue usage design requirements, and became part of the plant Technical Specifications.

Experience has shown that actual plant operation is often very conservatively represented by these design cycles. The use of actual operating history data allows the quantification of these conservatisms in the existing fatigue analyses. To demonstrate that the Class 1 component fatigue analyses remain valid for the period of extended operation, the design cycles applicable to the Class 1 components were assembled. The actual frequency of occurrence for the design basis cycles was determined and compared to the design cycle set. The severity of the actual plant transients was compared to the severity of the design cycles. This comparison was performed in order to demonstrate that on an event-by-event basis the design cycle profiles envelope actual plant operation. In addition, a review of the applicable administrative and operating procedures was performed to verify the effectiveness of the current design cycle counting program.

This review concluded that the existing design cycles and cycle frequencies are conservative and bounding for the period of extended operation. The analyses associated with verifying the structural integrity of the reactor vessel, reactor vessel internals, pressurizer, steam generators, and reactor coolant pumps have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i). A confirmatory Fatigue Monitoring Program (described in Appendix B of the LRA) has also been implemented at Ginna Station to provide additional assurance that the fatigue analyses remain valid during the period of extended operation.

A3.3.2 Reactor Vessel Underclad Cracking

Underclad cracking has been reported in the low alloy base metal heat-affected zone (HAZ) beneath the austenitic stainless steel weld overlay that is deposited to protect the ferritic material from corrosion.

A re-evaluation (WCAP 15338) of the generic Westinghouse fracture mechanics evaluation (WCAP 7733) concerning the underclad cracking issue has been performed for 60 years of plant operation. It was concluded that “underclad cracks are of no concern to the structural integrity of the vessel for continued plant operation, even through 60 years of operation.” WCAP 15338 is bounding for all Westinghouse plants.

The analysis associated with reactor vessel underclad crack growth has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

A3.3.3 ANSI B31.1 Piping

Design requirements in ANSI B31.1 assume a stress range reduction factor to provide conservatism in the piping design to account for fatigue due to thermal cyclic operation. This reduction factor is 1.0 provided the number of anticipated cycles is limited to 7000 equivalent full temperature cycles. This represents a condition where a piping system would have to be cycled approximately once every 3 days over the extended plant life of 60 years. Considering this limit, a review of the ANSI B31.1 piping within the scope of license renewal was performed in order to identify those systems that operate at elevated temperature and to establish their cyclic operating practices. Under current plant operating practices, piping systems within the scope of license renewal are only occasionally subject to cyclic operation. Typically these systems are subject to continuous steady-state operation and vary operating temperatures only during plant heatup and cooldown, during plant transients, or during periodic testing.

The results of the evaluation for ANSI B31.1 piping systems demonstrated that the number of assumed thermal cycles will not be exceeded in 60 years of plant operation except for the Nuclear Sampling System. For all systems except the Nuclear Sampling System, it has been determined that operation can be projected to the end of the period of license renewal, in accordance with 19 CFR 54.21 (c)(1)(ii). For the sampling system, detailed evaluation of operating cycles will be conducted, and reanalysis, repair, or replacement performed prior to the period of extended operation.

A3.3.4 Accumulator Check Valves

Fatigue of components is recognized as time dependent and therefore the analysis was reviewed for fatigue related to these valves. Fatigue failure is based upon the criteria of the cumulative usage factor (CUF). An analysis was performed on the accumulator check valves at Ginna Station. The analysis concludes that the maximum CUF is 0.74 based on specified load conditions.

Plant transients were reviewed to confirm transient limits and total transient counts to date. The load condition occurrences used in the above analysis bound the transient limits monitored by plant procedures. In accordance with 10 CFR 54.21(c)(1)(i), the existing CUF calculations remain valid because the number of assumed transients would not be exceeded during the period of extended operation.

A3.3.5 Environmentally Assisted Fatigue

Generic Safety Issue (GSI)-190, Fatigue Evaluation of Metal Components for 60 Year Plant Life, identifies a concern of the NRC staff about the potential effects of reactor water environments on reactor coolant system component fatigue life during the period of extended operation. GSI-190, which was closed in December 1999, has concluded that environmental effects have a negligible impact on core damage frequency, and as such, no generic regulatory action is required. However, as part of the closure of GSI-190, the NRC has concluded that licensees who apply for license renewal should address the effects of coolant environment on component fatigue life as part of their aging management programs.

Fatigue-sensitive component locations were evaluated in NUREG/CR-6260 for the older vintage Westinghouse plant. These locations were:

1. Reactor vessel shell and lower head (lower shell at the core support pads)
2. Reactor vessel inlet and outlet nozzles
3. Pressurizer surge line (including hot leg and pressurizer nozzles)
4. Reactor coolant piping charging system nozzle
5. Reactor coolant piping safety injection nozzle
6. Residual Heat Removal system Class 1 piping

Environmental fatigue calculations have been performed for Ginna for those component locations included in NUREG/CR-6260 using the appropriate environmental life correction factor formulae contained in NUREG/CR-6583 for carbon/low alloy steel material, or NUREG/CR-5704 for stainless steel material, as appropriate.

Based on these results, all component locations were determined to be acceptable for the period of extended operation, with the exception of the pressurizer surge line.

A3.3.6 Pressurizer Surge Line

A structural evaluation of the Ginna surge line considering the effects of thermal stratification was performed by Westinghouse in 1991. WCAP-12928 describes the stress and cumulative usage factor analysis performed for the surge line in accordance with NRC Bulletin 88-11. The highest CUF was calculated at the surge line nozzle connection to the RCS hot leg. By accounting for the environmental effects of fatigue, it is recognized that the cumulative fatigue usage factor (CUF) for the pressurizer surge line may exceed the Code allowable value of 1.0 during the period of extended operation. Locations on the surge line have been selected for monitoring using the Fatigue Monitoring Program.

The approach developed for managing the environmental effects of fatigue on the pressurizer surge line will include one or more of the following options:

1. Further refinement of the fatigue analysis to lower the CUF to below 1.0,

2. Repair of the affected locations,

3. Replacement of the affected locations, or

4. Manage the effects of fatigue by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC).

Should an inspection program be selected to manage environmentally-assisted fatigue for the surge line during the period of extended operation, inspection details such as scope, qualification, method, and frequency will be provided to the NRC for review and approval prior to entering the period of extended operation.

A3.4 Environmental Qualification of Electric Equipment

10 CFR 50.49, Environmental qualification of electric equipment important to safety for nuclear power plants, requires that selected electrical equipment that is relied upon to remain functional during and following a design basis event be environmentally qualified to perform its intended function. Equipment within the scope of the EQ rule has been identified in accordance with 10 CFR 50.49 paragraph (d) and are listed in the Ginna Station EQ Master List. Only the equipment qualification packages which indicate a qualified life of greater than 40 years were reviewed as a Time-Limited Aging Analysis (TLAA). Equipment qualification packages that indicate a qualified life of less than 40 years are not a TLAA as defined in 10 CFR 54.3 and therefore need not be discussed in the context of license renewal. To establish reasonable assurance that the safety related electrical equipment will perform its safety function when exposed to postulated harsh environmental conditions, licensees are required to develop an environmental qualification program. The program must demonstrate that the safety related electrical equipment required to perform the various safety related functions, identified in 10 CFR 50.49, are qualified to perform as intended. The program must maintain the environmental qualification of the equipment for its installed life. Periodic replacement and/or refurbishment of equipment are performed in order to maintain the qualified life of the device. The qualified life of an equipment type is that period of time the equipment is installed, under normal and abnormal plant operating conditions (thermal and radiation exposure), and still be expected to perform its intended function following a postulated design basis event.

The qualified life of an equipment type is determined using the ambient environmental conditions to which it is exposed for the predicted installation period as well as any internal heat rise and cyclic stresses.

EQ reanalyses have been performed to verify extension of EQ qualification to 60 years for most equipment, and shown to be acceptable per 10 CFR 54.21(c)(1)(i) or (c)(1)(ii). Calculations for the balance of the EQ components is continuing, and will be concluded prior to the period of extended operation.

A3.5 Concrete Containment Tendon Prestress

The Ginna Station containment structure is post-tensioned by 160 vertical tendons. The design for the containment provides for prestressing the concrete in the cylinder walls in the longitudinal direction with a sufficient compressive force to ensure that upon application of the design load combinations there will be no tensile stresses in the concrete due to membrane forces.

The prestressing forces of containment tendons decrease over time due to creep and shrinkage of concrete, and stress relaxation of the prestressing steel wires.

One hundred and thirty seven tendons were retensioned in 1979. The remaining twenty three tendons, which had been retensioned in 1969, were not included in the 1979 retensioning activity. Review of tendon surveillance lift-off data indicates that prestressing forces will remain at acceptable levels through the period of extended operation for all tendons except the group of twenty three which were retensioned in 1969. These tendons will be retensioned prior to the end of the current license period. Technical Specifications require that the results of the surveillance be compared with predicted values to verify that prestressing forces are maintained above the minimum design prestress levels.

Based on this review, the program will adequately manage loss of prestress in containment tendons during the extended period of operation in accordance with 10 CFR 54.21(c)(1)(iii).

3.5.1 **Containment Tendon Fatigue**

Fatigue tests were conducted on tendon wire materials in 1960 by an independent testing lab. The tests indicated that the tendons were capable of withstanding over 2 million cycles at stress levels between 135 and 158 Ksi. The test results were used to conclude that dynamic loads, considering especially pulsating loads resulting from an earthquake, do not jeopardize buttonhead anchorage.

The tendon fatigue test results indicate that the fatigue limit of the tendon wires exceeds, by many orders of magnitude, the total number of cycles that could accumulate through multiple seismic events over 60 years. The seismic fatigue evaluation remains valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

3.5.2 Containment Tendon Bellows Fatigue

The allowable radial and vertical displacements of the containment stainless steel tendon bellows are limited to two cycles per year for the 40-year life of the plant. This limits the total number of allowable bellows displacement cycles to 80. Since the completion of construction, displacements at the tendon bellows have occurred due to pressure testing and temperature changes in the cylindrical shell wall due to seasonal variations and reactor shutdown during refueling outages.

The fatigue usage factor of the tendon bellows has been calculated to be .004 over a 60-year period. Therefore, the structural integrity of the tendon bellows will be maintained through the period of extended operation. Thus, the original fatigue analysis remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A3.6 Containment Liner Plate and Penetration Fatigue

The containment liner, liner penetrations and liner steel components of the Ginna Station Containment Structure comply with the ASME Code Section III-1965 for pressure boundary and the AISC Code for structural steel. The containment liner and penetrations, including the equipment and personnel hatch penetrations, were designed as Class B Vessels. The Winter 1965 Addenda of ASME Section III, Subsection B, N-1314(a) requires that the containment vessel satisfy the provisions of Subsection A, N-415.1, “Vessels Not Requiring Analysis for Cyclic Operation,” in order that Subsection B rules be applicable.

ASME Section III, N-415.1 states that a fatigue analysis is not required, and it may be assumed that the peak stress intensity limit has been satisfied for a vessel or component by compliance with the applicable requirements for materials, design, fabrication, testing, and inspection, provided the service loading of the vessel or component meets all of six (6) conditions. An analysis was performed which verifies that each of the six conditions are satisfied for the period of extended operation. This analysis demonstrates that the liner and penetrations comply with the ASME Section III - 1965 Code Rules for fatigue through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

3.6.1 Containment Liner Anchorage Fatigue

A fatigue analysis of the fillet weld attaching the channel anchors to the liner was performed as part of the original containment design. The allowable fatigue stress of the attachment weld was set equal to the stress caused by static loading. This stress equals 13,600 psi and corresponds to 100,000 stress cycles.

A total of 100,000 stress cycles corresponds to more than 4 full stress cycles per day for 60 years. Fluctuations of temperature and pressure in containment on a daily basis are not significant enough in magnitude to cause four cycles of design basis stress at the liner anchorage weld each day. Therefore, the original fatigue analysis remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A3.7 Containment Liner Stress

The containment liner is carbon steel plate conforming to ASTM A442-60T Grade 60 with a minimum yield of 32,000 psi, and a buckling stress of 16,600 psi at operating conditions. The liner plate thickness is 1/4 in. for the base and 3/8 in. for the cylinder and dome. The liner stresses (meridional directions) were calculated to be 4500 psi compression based upon a prestress force of 0.70 fs. The concrete strain due to creep and shrinkage was established as being 320 x 10^{-6} in/in. This increases the liner stress to 14,100 psi at the end of 40 years.

The creep and shrinkage strain occurring over a 60-year plant life was calculated, and the resulting compressive liner stress due to both time-dependent and non-time dependent loads is determined to be 14,870 psi. This liner stress is less than the liner buckling stress of 16,600 psi and therefore the analysis has been projected to the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A3.8 Other Time-Limited Aging Analyses

A3.8.1 **Crane Load Cycle Limit**

The estimated number of load cycles for each crane was compared to the number of design load cycles. The comparison demonstrated that all estimated load cycle combinations were well below the upper design loading cycle limit. In addition, the average percent of the rated load lifted was well below 50% of the limit as set forth in the design criteria. Since the number of operating load cycles for the cranes will be fewer than the design cycles and the average percent of rated load lifted is less than 50% for the design load cycles, the crane load cycle limits will remain valid for the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

A3.9 Exemptions

The requirements of 10 CFR 54.21(c) stipulate that the application for a renewed license should include a list of plant-specific exemptions granted pursuant to 10 CFR 50.12 and that are based on time-limited aging analyses, as defined in 10 CFR 54.3. Each active 10 CFR 50.12 exemption has been reviewed to determine whether the exemption is based on a time-limited aging analysis. No plant-specific exemptions granted pursuant to 10 CFR 50.12 and based on a time-limited aging analysis as defined in 10 CFR 54.3 have been identified.

A4.0 TLAA SUPPORTING ACTIVITIES

A4.1 Concrete Containment Tendon Prestress

The prestressing forces generated by the containment wall tendons diminish over time due to stress relaxation of the steel tendon wires and shrinkage and creep of the surrounding concrete. The aging management program developed to monitor the prestressing tendon forces ensures that, by periodic surveillance lift-off tests, the trend lines of the measured prestressing forces meet the requirements of 10 CFR 50.55a(b)(2)(viii)(B). If the trend lines cross the predicted lower limits, corrective action such as retensioning will be taken.

A4.2 Environmental Qualification Program

Equipment environmental qualification has been reviewed and one of the following options was used for those components re-analyzed:

- The original environmental qualification qualified life has been shown to remain valid for the period of extended operation.
- The environmental qualification has been projected to the end of the period of extended operation. Reanalysis addresses attributes of analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions.

Calculations are continuing for the balance of Ginna Station EQ equipment using these methods. These analyses will be complete prior to the period of extended operation.

A4.3 Fatigue Monitoring Program

The program is a confirmatory program that monitors loading cycles due to thermal and pressure transients at selected locations on critical reactor coolant system components. The program provides means for evaluating transients using either a stress based or cycle based methodology. The program provides an analytical basis for confirming that the number of cycles established by the analysis of record will not be exceeded before the end of the period of extended operation.

The effects of the reactor coolant environment on component fatigue life are considered by evaluation of a sample of critical components that include, as a minimum, those components selected in NUREG/CR-6260 using the appropriate environmental fatigue correction factors. Formulas for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels.

APPENDIX B

AGING MANAGEMENT ACTIVITIES

APPENDIX B
Contents

APPENDIX B
Tables

B1.0 APPENDIX B - INTRODUCTION

B1.1 Overview

Aging management program descriptions are provided in this appendix for each program credited for managing aging effects based upon the aging management review results provided in Section 3.2 through Section 3.7.

Each of the aging management programs described in this section has been compared against the ten attributes which are described in Section A.1, “Aging Management Review - Generic,” Table A.1-1, “Elements of an Aging Management Program for License Renewal,” of the NUREG-1800, SRP-LR (Reference 1). A description of the program, relevant operating experience associated with the program, and a conclusion section is provided for each program. The conclusion section serves three purposes:

- it describes conformance to the NUREG-1801 program
- it describes enhancements to certain attributes as considered necessary to be consistent with the NUREG-1801 program, and
- it describes exceptions to certain attributes in the NUREG-1801 program not considered necessary to successfully manage aging concerns.

For aging management programs identified in NUREG-1801 which are not specifically used at Ginna Station, a description of alternative programs which manage the applicable aging effects are described.

For new programs not described in NUREG-1801 a description of the program, an evaluation of the ten attributes and a conclusion regarding the adequacy of the program for managing the effects of aging is provided.

The ten attribute definitions are:

Scope of Program

The specific program is identified. The scope of the program includes the specific structures and components for which the program manages the effects of aging.

Preventive Actions

The activities for prevention and mitigation of age-related degradation are described. For condition or performance monitoring programs that do not rely on preventive actions, this information is not provided.

Parameters Monitored or Inspected

The parameters to be monitored or inspected are identified and linked to the degradation of the structure or component intended function(s).

For a condition monitoring program, the designated parameter monitored or inspected is intended to detect the presence and extent of the identified aging effects.

For a performance monitoring program, a link is established between the degradation of the structure or component intended function(s) and the parameter(s) being monitored.

For prevention and mitigation programs, the parameters monitored are the specific parameters being controlled to prevent or mitigate aging effects.

Detection of Aging Effects

Aging effects are detected before there is a loss of the structure and component intended function(s). The parameters to be monitored or inspected are appropriate to ensure that the structure and component intended function(s) will be adequately maintained for the period of extended operation. Methods or techniques (e.g., visual, volumetric, surface) used for detection of aging effects are discussed. Plant-specific program documentation will include additional information such as frequency, sample size, data collection and timing of new/one-time inspections to ensure timely detection of aging effects.

Monitoring and Trending

Monitoring and trending activities are described, and are designed to provide predictability of the extent of degradation and thus effect timely corrective or mitigative actions.

Acceptance Criteria

The 10 CFR 50 Appendix B (Reference 2) acceptance criteria of the program and its basis is described. The acceptance criteria, against which the need for corrective action is generally evaluated, ensure that the structure and component intended function(s) are maintained under all CLB design conditions during the period of extended operation. Plant-specific program documentation will include the description of methodologies for analyzing results against applicable acceptance criteria. These acceptance criteria will provide for timely corrective action before loss of intended function described in the CLB.

Corrective Actions

This attribute describes 10 CFR 50 Appendix B required actions that will be taken when the acceptance criteria are not met. Timely corrective actions, including root cause determination and prevention of recurrence, are part of this attribute.

Confirmation Process

The confirmation process ensures that preventive actions are adequate and that appropriate corrective actions have been completed and are effective.

Administrative Controls

The administrative controls of the program are described. This includes a 10 CFR 50, Appendix B formal review and approval process.

Operating experience

Plant-specific operating experience, including past corrective actions resulting in program enhancements or additional programs, is reviewed and described as appropriate. This information provides objective evidence to support the conclusion that the effects of aging will be managed adequately so that the structure and component intended function(s) will be maintained during the period of extended operation.

The Ginna Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Appendix A.2 of NUREG-1800 (Reference 1).

The Quality Assurance Program includes the elements of corrective action, confirmation process, and administrative controls, and is applicable to the safety-related and non-safety-related structures, systems, and components that are subject to aging management review. In many cases, existing activities were found adequate for managing aging effects during the period of extended operation.

In some cases, aging management reviews revealed that existing programs should be enhanced to adequately manage the effects of aging. Also, in a few cases, new programs will be developed to provide reasonable assurance that aging effects are adequately managed. Each aging management program presented in this appendix is characterized as either an Aging Management Program or as a Time-Limited Aging Analyses Activity that has been credited by a time-limited aging analysis described in (Section 4.0)

B1.2 Operating Experience

Industry operating experience was incorporated into the License Renewal process through a review of industry documents to identify aging effects and mechanisms that could challenge the intended function of systems and structures within the scope of License Renewal. Review of plant specific operating experience was performed to identify aging effects. This review involved electronic database searches of historical information from Ginna Station as well as other information documented in plant records from as early as early as 1970 to 2002. In addition, discussions with system engineers and long time company employees were conducted for identification of any additional aging concerns.

For those materials and environments identified in NUREG-1801 (Reference 3), no additional aging effects requiring management were identified that were not already identified in the GALL. However, additional materials and environments were identified at Ginna Station that were not identified in the GALL but which had aging effects requiring management. These additional materials and environments and associated aging effects are identified in Section 3. The programs identified for aging management are discussed in this appendix.

B1.3 Aging Management Programs

The following aging management programs are described in the sections listed in this appendix. Site specific programs are indicated. All other programs are fully consistent with or are, with some exceptions, consistent with programs in NUREG-1801.

1. Aboveground Carbon Steel Tanks (Section B2.1.1)
2. ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection (Section B2.1.2)
3. ASME Section XI, Subsections IWE & IWL Inservice Inspection (Section B2.1.3)
4. ASME Section XI, Subsection IWF Inservice Inspection (Section B2.1.4)
5. Bolting Integrity (Section B2.1.5)
6. Boric Acid Corrosion (Section B2.1.6)
7. Buried Piping and Tanks Inspection (Section B2.1.7)
8. Buried Piping and Tanks Surveillance (Section B2.1.8)
9. Closed-Cycle (Component) Cooling Water System Surveillance (Section B2.1.9)
10. Compressed Air Monitoring (Section B2.1.10)
11. Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section B2.1.11)
12. Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits (Section B2.1.12)
13. Fire Protection (Section B2.1.13)
14. Fire Water System (Section B2.1.14)
15. Flow-Accelerated Corrosion (Section B2.1.15)
16. Fuel Oil Chemistry (Section B2.1.16)
17. Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section B2.1.17)
18. Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (Section B2.1.18)

19. Loose Part Monitoring (Section B2.1.19)
20. Neutron Noise Monitoring (Section B2.1.20)
21. One-Time Inspection (Section B2.1.21)
22. Open-Cycle Cooling (Service) Water System (Section B2.1.22)
23. Periodic Surveillance and Preventive Maintenance (Section B2.1.23)
24. Protective Coatings Monitoring and Maintenance Program (Section B2.1.24)
25. Reactor Head Closure Studs (Section B2.1.25)
26. Reactor Vessel Head Penetration Inspection (Section B2.1.26)
27. Reactor Vessel Internals (Section B2.1.27)
28. Reactor Vessel Surveillance (Section B2.1.28)
29. Selective Leaching of Materials (Section B2.1.29)
30. Spent Fuel Pool Neutron Absorber Monitoring (Section B2.1.30)
31. Steam Generator Tube Integrity (Section B2.1.31)
32. Structures Monitoring Program (Section B2.1.32)
33. Systems Monitoring (Section B2.1.33)
34. Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) (Section B2.1.34)
35. Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) (Section B2.1.35)
36. Thimble Tube Inspection (Section B2.1.36)
37. Water Chemistry Control (Section B2.1.37)

B1.4 Time-Limited Aging Analyses Support Activities:

1. Environmental Qualification Program (Section B3.1)
2. Fatigue Monitoring Program (Section B3.2)
3. Concrete Containment Tendon Pre-stress(Section B3.3)

B2.0 AGING MANAGEMENT PROGRAMS

Correlation between NUREG-1801 Generic Aging Lessons Learned (GALL) programs and Ginna programs are in Table B2.0-1. For the Ginna Programs, links to appropriate sections of this appendix are provided.

Table B2.0-1 Correlation Between GALL Programs and Ginna Programs

GALL ID NUMBER	GALL PROGRAM	GINNA PROGRAM
XI.M1	ASME Section XI Inservice Inspection, Subsections IWB, IWC, & IWD	ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection (Section B2.1.2)
XI.M2	Water Chemistry	Water Chemistry Control (Section B2.1.37)
XI.M3	Reactor Head Closure Studs	Reactor Head Closure Studs (Section B2.1.25)
XI.M4	BWR Vessel ID Attachment Welds	Not Applicable, Ginna is a PWR.
XI.M5	BWR Feedwater Nozzle	Not Applicable, Ginna is a PWR.
XI.M6	BWR Control Rod Drive Return Line Nozzle	Not Applicable, Ginna is a PWR.
XI.M7	BWR Stress Corrosion Cracking	Not Applicable, Ginna is a PWR.
XI.M8	BWR Penetrations	Not Applicable, Ginna is a PWR.
XI.M9	BWR Vessel Internals	Not Applicable, Ginna is a PWR.
XI.M10	Boric Acid Corrosion	Boric Acid Corrosion (Section B2.1.6)
XI.M11	Nickel-Alloy Nozzles and Penetrations	Reactor Vessel Head Penetration Inspection (Section B2.1.26)
XI.M12	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) (Section B2.1.34)
XI.M13	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) (Section B2.1.35)
XI.M14	Loose Part Monitoring	Loose Part Monitoring (Section B2.1.19)
XI.M15	Neutron Noise Monitoring	Neutron Noise Monitoring (Section B2.1.20)

GALL ID NUMBER	GALL PROGRAM	GINNA PROGRAM
XI.M16	PWR Vessel Internals	Reactor Vessel Internals (Section B2.1.27)
XI.M17	Flow-Accelerated Corrosion	Flow-Accelerated Corrosion (Section B2.1.15)
XI.M18	Bolting Integrity	Bolting Integrity (Section B2.1.5)
XI.M19	Steam Generator Tube Integrity	Steam Generator Tube Integrity (Section B2.1.31)
XI.M20	Open-Cycle Cooling Water System	Open-Cycle Cooling (Service) Water System (Section B2.1.22)
XI.M21	Closed-Cycle Cooling Water System	Closed-Cycle (Component) Cooling Water System (Section B2.1.9)
XI.M22	Boraflex Monitoring	Spent Fuel Pool Neutron Absorber Monitoring (Section B2.1.30)
XI.M23	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (Section B2.1.18)
XI.M24	Compressed Air Monitoring	Compressed Air Monitoring (Section B2.1.10)
XI.M25	BWR Reactor Water Cleanup System	Not Applicable, Ginna is a PWR.
XI.M26	Fire Protection	Fire Protection (Section B2.1.13)
XI.M27	Fire Water System	Fire Water System (Section B2.1.14)
XI.M28	Buried Piping and Tanks Surveillance	Buried Piping and Tanks Surveillance (Section B2.1.8)
XI.M29	Aboveground Carbon Steel Tanks	Aboveground Carbon Steel Tanks (Section B2.1.1)
XI.M30	Fuel Oil Chemistry	Fuel Oil Chemistry (Section B2.1.16)
XI.M31	Reactor Vessel Surveillance	Reactor Vessel Surveillance (Section B2.1.28)
XI.M32	One-Time Inspection	One-Time Inspection (Section B2.1.21)
XI.M33	Selective Leaching of Materials	Selective Leaching of Materials (Section B2.1.29)

GALL ID NUMBER	GALL PROGRAM	GINNA PROGRAM
XI.M34	Buried Piping and Tanks Inspection	Buried Piping and Tanks Inspection (Section B2.1.7)
XI.E1	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section B2.1.11)
XI.E2	Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits (Section B2.1.12) Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section B2.1.11)
XI.E3	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section B2.1.17)
XI.S1	ASME Section XI, Subsection IWE	ASME Section XI, Subsections IWE & IWL Inservice Inspection (Section B2.1.3)
XI.S2	ASME Section XI, Subsection IWL	ASME Section XI, Subsections IWE & IWL Inservice Inspection (Section B2.1.3)
XI.S3	ASME Section XI, Subsection IWF	ASME Section XI, Subsection IWF Inservice Inspection (Section B2.1.4)
XI.S4	10 CFR 50, Appendix J	ASME Section XI, Subsections IWE & IWL Inservice Inspection (Section B2.1.3)
XI.S5	Masonry Wall Program	Structures Monitoring Program (Section B2.1.32)
XI.S6	Structures Monitoring Program	Structures Monitoring Program (Section B2.1.32)
XI.S7	RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants	Structures Monitoring Program (Section B2.1.32)

GALL ID NUMBER	GALL PROGRAM	GINNA PROGRAM
XI.S8	Protective Coating Monitoring and Maintenance	Protective Coatings Monitoring and Maintenance Program (Section B2.1.24)
Chapter X		
X.M1	Metal Fatigue of Reactor Coolant Pressure Boundary	Fatigue Monitoring (Section B3.2)
X.E1	Environmental Qualification (EQ) of Electrical Components	Environmental Qualification Program (Section B3.1)
X.S1	Concrete Containment Tendon Prestress	Concrete Containment Tendon Prestress (Section B3.3)
NA	Plant Specific Program	Thimble Tube Inspection Program (Section B2.1.36)
NA	Plant Specific Program	Periodic Surveillance and Preventive Maintenance (Section B2.1.23)
N/A	Plant Specific Program	Systems Monitoring (Section B2.1.33)

B2.1 Aging Management Activities

B2.1.1 Aboveground Carbon Steel Tanks

Program Description

This program relies on periodic system walkdowns to monitor the condition of aboveground carbon steel storage tanks. These walkdowns include an assessment of the condition of tank surfaces protected by paints or coatings, although the coatings themselves are not credited to perform a preventive function. For storage tanks supported on earthen or concrete foundations, corrosion could occur in inaccessible locations, such as the tank bottom. For such inaccessible surfaces, one-time thickness measurements of the tank bottom performed from inside the tank are used to assess the tank bottom condition.

This program is not specifically used for aging management at Ginna Station. Inspection, testing and surveillance activities described under the scope of this program in NUREG-1801 are performed at Ginna Station by the following programs:

- Systems Monitoring (Section B2.1.33)
- One-Time Inspection (Section B2.1.21)

Operating Experience

In accordance with the guidance provided in Generic Letter 98-04, Ginna Station evaluated plant specifications for use of protective coatings inside containment (including those used on carbon steel tanks such as the accumulators). Historical walkdowns to evaluate areas of flaking and peeling paint have discovered isolated areas of degradation, but the quantity has been insufficient to pose a threat to the pump recirculation capability by plugging the sump screens.

Outside containment, periodic assessments of the outside surfaces of tanks are part of the System Engineer Maintenance Rule (10 CFR 50.65) walkdowns. No significant corrosion of tanks has been documented as a result of these walkdowns.

Ginna Station does not have an operating history of tank bottom thickness measurements, but will perform a one-time inspection of the Reactor Makeup Water Tank prior to the period of extended operation.

Conclusion

Results of a one-time inspection of the Reactor Makeup Water Tank bottom will be evaluated and dispositioned in accordance with the Ginna Station Corrective Action Program. In addition, continued inspection and surveillance activities performed during walkdowns in accordance with the Systems Monitoring Program will provide assurance that age-related degradation of external surfaces of aboveground carbon steel tanks will be adequately managed during the period of extended operation.

B2.1.2 ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection

Program Description

The Code of Federal Regulations, 10 CFR 50.55a, imposes the Inservice Inspection (ISI) requirements of ASME Boiler and Pressure Vessel Code (the Code), Section XI, for Class 1, 2, and 3 pressure-retaining components and their integral attachments in light-water cooled power plants. Inspection, repair and replacement of these components are covered in Subsections IWB, IWC, and IWD, respectively, in the 1995 Edition of the Code through 1996 Addenda. The program includes periodic visual, surface, and or volumetric examinations and leakage tests of all Class 1, 2, and 3 pressure-retaining components and their integral attachments including welds, pump casings, valve bodies, and pressure-retaining bolting.

The ASME Section XI Inservice Inspection Program in accordance with Subsections IWB, IWC, and IWD has been shown to be generally effective in managing the effects of aging in Class 1, 2, and 3 components and their integral attachments.

Ginna Station has maintained an Inservice Inspection Program in accordance with 10 CFR 50.55a and Technical Specification requirements. The Fourth Ten-Year Interval of the Ginna Station Inservice Inspection Program began on January 1, 2000 and was developed and prepared to meet the requirements of ASME Section XI, 1995 Edition, with 1996 Addenda.

B2.1.2.1 Operating Experience

A thorough review of industry operating experience relating to inservice inspections has revealed numerous incidents of primary pressure boundary degradation that have been reported through NRC generic communications. These incidents may be grouped in the following categories:

- Boric acid corrosion due to leakage at bolted closures and leakage caused by cracking of primary pressure boundary Alloy 600 components such as reactor vessel head CRDM nozzles;
- Cracking due to SCC in safety injection piping, instrument nozzles in safety injection accumulators, and safety-related stainless steel piping systems containing stagnant or essentially stagnant borated water;
- Crack initiation and growth due to thermal and mechanical loading in high-pressure injection and safety injection lines;
- Degradation of steam generator tubing due to PWSCC, ODSCC, IGA, wastage and pitting; denting and cracking of tubes due to carbon steel support plate corrosion; and pitting and cracking of steam generator shell welds.

Review of plant-specific operating experience revealed the following conditions that were discovered by ISI Program examinations:

- Bolting degradation detected by VT-1 examinations and boric acid leakage by VT-2 leakage exams;
- PWSCC, ODSCC, IGA and denting of Alloy 600 steam generator tubing by eddy current examinations;
- Shallow thermal fatigue cracks in S/G feedwater nozzle-to-pipe weld; and
- Original manufacturing flaw indications in the primary inlet nozzle-to-reactor vessel weld (N2B) and pressurizer lower head-to-shell girth weld. These indications were evaluated by fracture mechanics and determined to be acceptable.

The ISI Program at Ginna Station is continually upgraded to account for industry experience and research and is subject to periodic NRC inspections and self-assessments. The ISI Program has provided an effective means of assuring the pressure integrity of Ginna Station Class 1, 2 and 3 systems.

Conclusion

The ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program is consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section XI.M1, “ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection” (Reference 3). The continued implementation of the ISI Program provides reasonable assurance that aging effects will be managed such that the intended functions of Class 1, 2 and 3 pressure-retaining components and their integral attachments will be maintained during the license renewal period.

B2.1.3 ASME Section XI, Subsections IWE & IWL Inservice Inspection

Program Description

The Code of Federal Regulations, 10 CFR 50.55a, imposes the Inservice Inspection (ISI) requirements of ASME Boiler and Pressure Vessel Code (the Code), Section XI, Subsection IWE for steel containments (Class MC) and steel liners for concrete containments (Class CC). 10 CFR 50.55a also imposes the examination requirements of ASME Section XI, Subsection IWL for reinforced and prestressed concrete containments (Class CC). The full scope of Subsection IWE includes steel containment shells and their integral attachments; steel liners for concrete containments and their integral attachments; containment hatches and airlocks; seals, gaskets and moisture barriers; and pressure-retaining bolting. The scope of Subsection IWL includes reinforced concrete and unbonded post-tensioning systems.

Ginna Station has maintained an Inservice Inspection (ISI) Program in accordance with 10 CFR 50.55a and Technical Specification requirements. The Containment Program which outlines the First IWE and IWL Inservice Inspection Interval requirements for Ginna Station was implemented Sept. 9, 1998 and formally included in the ASME Section XI ISI Program.

The Ginna Station ASME Section XI, Subsection IWE & IWL Inservice Inspection Program (the IWE/IWL Program) manages aging of (a) steel liners of concrete containments and their integral attachments; containment hatches and airlocks; seals, gaskets and moisture barriers; and pressure retaining bolting, and (b) reinforced concrete containments and unbonded post tensioning systems. The primary inspection methods employed are visual examinations (VT-1, VT-3, VT-1C, VT-3C) with limited supplemental volumetric and surface examinations as necessary. Tendon anchorages and wires are visually examined. Tendon wires are tested for verification that minimum mechanical properties requirements are met. Tendon corrosion protection medium is analyzed for alkalinity content and soluble ion concentrations. Prestressing forces are measured in selected sample tendons.

B2.1.3.1 **Operating Experience**

A review of industry operating experience relating to degradation of steel containment components and concrete revealed occurrences of corrosion in steel containment shells and liner plates, and degradation of reinforced concrete and pre-stressing systems that have been reported through various NRC generic communications.

Review of plant-specific operating experience revealed the following conditions that were discovered during tendon surveillances and more recent IWE and IWL examinations:

- Loss of pre-stress in most containment tendons requiring re-tensioning of 137 tendons;
- Containment moisture barrier found to be out of conformance with drawing; loose insulation; non-conformance corrected by recaulking;
- Minor corrosion of steel containment liner; wall thickness verified by UT; restoration of protective paint coating;
- Low grease levels in certain tendon grease cans at top of containment; cans refilled;
- Corroded and leaking tendon fill-port piping; all fill ports repaired.

Continued inspections in accordance with the Ginna Station ASME Section XI, Subsections IWE & IWL Inservice Inspection Program provide an effective means for timely detection and correction of any degradation of the containment pressure boundary, concrete and post-tensioning system.

Conclusion

The ASME Section XI, Subsections IWE & IWL Inservice Inspection Program is consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Sections XI.S1, “ASME Section XI, Subsections IWE,” XI.S2, “ASME Section XI, Subsections IWL,“ and XI.S4, “10 CFR 50, Appendix J” (Reference 3). The continued implementation of the IWE/IWL Program provides reasonable assurance that aging effects will be managed such that the intended functions of the Containment will be maintained throughout the license renewal period.

B2.1.4 ASME Section XI, Subsection IWF Inservice Inspection

Program Description

The Code of Federal Regulations, 10 CFR 50.55a, imposes the Inservice Inspection (ISI) requirements of ASME Boiler and Pressure Vessel Code (the Code), Section XI, for Class 1, 2, 3 and MC piping and components and their associated supports. Inservice inspection of supports for ASME piping and components is addressed in Section XI, Subsection IWF in the 1995 Edition of the Code (with 1996 Addenda). The program includes periodic visual examinations of supports based on a sampling of the total support population. The sample size varies depending on the ASME Class. The largest sample size is specified for the most critical supports; the sample size decreases for the less critical supports (Class 2 and 3). Discovery of deficiencies during regularly scheduled inspections results in an expansion of inspection scope to assure that the full extent of the deficiencies is identified. Degradation that potentially compromises the support function or load capacity is identified for evaluation.

The ASME Section XI, Subsection IWF Inservice Inspection Program (the IWF Program) has been shown to be generally effective in managing the effects of aging in Class 1, 2, and 3 and MC piping and component supports.

Ginna Station has maintained an Inservice Inspection Program in accordance with 10 CFR 50.55a and Technical Specification requirements. The Fourth Ten-Year Interval of the Ginna Station Inservice Inspection Program began on January 1, 2000 and was developed and prepared to meet the requirements of ASME Section XI, 1995 Edition, with 1996 Addenda.

B2.1.4.1 Operating Experience

A review of industry and Ginna-specific operating experience relating to inservice inspection of piping and component supports has revealed incidents of misalignment, improper hot or cold positions on spring supports and constant load supports, arc strikes, weld spatter, and missing, detached, or loosened support items. These conditions have been corrected in accordance with the requirements of Subsection IWF.

The IWF Program at Ginna Station is continually upgraded to account for industry experience and research and is subject to periodic NRC inspections and self-assessments. The IWF sampling inspections have been effective in managing aging effects for Ginna Station ASME Class 1, 2, 3 and MC supports.

Conclusion

The ASME Section XI, Subsection IWF Inservice Inspection Program is consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Sections XI.S3, “ASME Section XI, Subsections IWF” (Reference 3). The continued implementation of the IWF Program at Ginna Station provides reasonable assurance that aging effects will be managed such that the intended functions of Class 1, 2, 3 and MC supports will be maintained during the license renewal period.

B2.1.5 Bolting Integrity

Program Description

The Ginna Station Bolting Integrity Program is a comprehensive program which addresses aging management requirements for all bolting on mechanical and structural components within the scope of license renewal. The Bolting Integrity Program is based on industry recommendations and EPRI guidelines for materials selection and mechanical properties, installation procedures, joint/gasket designs, lubricants and sealants, torque and closure requirements, and enhanced inspection techniques. The program includes periodic inspection of closure bolting for indications of cracking, loss of material and loss of preload. Consideration is given to guidance in NUREG-1339 and EPRI NP-5769 (with exceptions noted in NUREG-1339) for safety related bolting, and EPRI TR-104213 for pressure retaining and structural bolting.

The Bolting Integrity Program credits activities performed under the direction of other aging management programs for managing specific aging effects. These programs include the following:

- ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection (1995 Edition with 1996 Addenda) (Section B2.1.2)
- ASME Section XI, Subsection IWF Inservice Inspection (Section B2.1.4)
- Periodic Surveillance and Preventive Maintenance (Section B2.1.23)
- Boric Acid Corrosion (Section B2.1.6)
- Systems Monitoring (Section B2.1.33)
- Structures Monitoring Program (Section B2.1.32)

B2.1.5.1 **Operating Experience**

A thorough review of industry operating experience has revealed numerous incidents of primary pressure boundary degradation. Various NRC generic communications, including information notices, bulletins and generic letters have been issued on bolting degradation. The majority of these incidents have dealt with boric acid corrosion caused by leakage at bolted closures, stress corrosion cracking of high-strength bolts, and cracking due to fatigue. General corrosion of structural bolting located underwater or in humid environments has also been reported.

Review of plant-specific operating experience revealed the following incidents involving bolting degradation:

- Various incidents involving bolting degradation resulting from boric acid leakage at bolted joints detected by VT-1 visual examinations and VT-2 leakage exams;
- Failures of ASTM A 490 high-strength RCP leg-support anchor bolts due to stress corrosion cracking; other factors contributing to the failures were improper heat-treatment and excessive preload during original installation;
- Linear MT indications in machined reduced-section shank of five steam generator manway bolts attributed to incipient fatigue damage due to multiple loadings during tensioning, coincident with tool marks produced during machining.

EPRI reports NP-5769 and TR-104213 document programmatic guidance and recommendations for addressing industry bolting integrity issues. The Bolting Integrity Program has been developed and implemented in accordance with this guidance and therefore provides an effective means of ensuring bolting reliability during the license renewal period.

Conclusion

The Ginna Station Bolting Integrity Program is consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Sections XI.M18, “Bolting Integrity” (Reference 3). The continued implementation of the Bolting Integrity Program provides reasonable assurance that aging effects will be managed such that the intended function of closure bolting on Class 1, 2 and 3 and other pressure-retaining components, NSSS component supports, and structural bolting will be maintained during the license renewal period.

B2.1.6 Boric Acid Corrosion

Program Description

The current Ginna Station Boric Acid Corrosion Control Program is described in an Administrative Procedure. It is a comprehensive program that was developed and implemented to meet the recommendations of USNRC Generic Letter (GL) 88-05, to monitor the condition of the reactor coolant system (RCS) pressure boundary components for boric acid leakage. The program identifies carbon steel components within the RCS that are susceptible to corrosion from leakage of boric acid. It also provides for periodic visual inspection of adjacent components, structures, and supports for evidence of leakage and corrosion.

This program will be enhanced to account for boric acid corrosion of non-RCS components located in areas where there is the potential for boric acid leakage, including cable connections, cable trays and other susceptible SSCs.

B2.1.6.1 Operating Experience

A review of industry operating experience has revealed numerous incidents of RCS pressure boundary degradation due to corrosion of external surfaces caused by leakage of borated water. These incidents have ranged from leakage at bolted closures causing wastage of carbon and low-alloy steel bolting to leakage at reactor vessel head penetrations resulting in accumulation of boric acid deposits on the head and severe loss of head wall thickness near the penetrations. Recent NRC communications (BL 2002-01) have focused on reactor vessel head degradation from boric acid corrosion.

Review of Ginna-specific operating experience has revealed various incidents involving bolting degradation resulting from boric acid leaks at bolted closures that were detected in their early stages by VT-1 visual examinations and VT-2 leakage examinations. In these cases, the bolting was replaced as a precautionary measure. Recent inspections of the Ginna reactor vessel head and penetrations revealed no evidence of boric acid leakage. Implementation of the Boric Acid Corrosion Control Program at Ginna Station in response to the guidance in NRC GL 88-05 has been demonstrated to effectively manage the effects of boric acid corrosion on the intended function of RCS components.

Conclusion

The Ginna Station Boric Acid Corrosion Control Program will be consistent with NUREG-1801, GALL, Section XI.M10. The program will be enhanced to account for boric acid wastage of non-RCS components, including cable connectors and cable trays as well as other susceptible SSCs. The program provides reasonable assurance that the aging effects due to boric acid corrosion of SSCs within the scope of the program will be managed such that their intended function will be maintained during the license renewal period.

B2.1.7 Buried Piping and Tanks Inspection

Program Description

This aging management program includes preventive measures to mitigate corrosion and periodic inspections to manage the effects of corrosion on the pressure-retaining capacity of buried carbon steel piping and tanks. These preventive measures are in accordance with standard industry practice for maintaining protective coatings. Buried piping and tanks are inspected when they are excavated during maintenance activities and when a pipe is uncovered for any reason. These are considered inspections of opportunity.

This program is not specifically used for aging management at Ginna Station. The inspection activities described under the scope of this program in NUREG-1801 are performed at Ginna Station by the following program:

- One-Time Inspection (Section B2.1.21).

Operating Experience

Portions of several buried piping systems and tanks have been excavated at Ginna Station in connection with maintenance activities. Over the years, several sections of the fire-water yard loops have been inspected and replaced with upgraded materials, including piping, fittings, shut-off valves and hydrants. As far back as 1974, a section of the service water discharge header from the Auxiliary Building was excavated and inspected due to preparations for construction of the Standby Auxiliary Feedwater Building. Several other inspection opportunities arose as a result of corrective maintenance, and others, such as those in 1995, were performed during relocation of underground headers due to construction of the concrete pad for the crane used for replacement of steam generators. Portions of the underground service water header was also uncovered and inspected in 1995. Most recently, in November

2001, a yard hydrant and connecting piping/fittings were replaced due to flow blockage identified by periodic flow testing. In the fall of 2001, the Security Diesel Generator underground fuel oil storage tank was replaced with a new tank as a preventive measure. In these cases, the exterior surface condition of the components inspected or replaced was found to be in good condition.

The Emergency Diesel Generator underground fuel oil storage tanks are inspected and leak-tested periodically in accordance with the Periodic Surveillance and Preventive Maintenance Program. No indication of tank wall degradation has been detected by these inspections. An ultrasonic thickness examination (one time inspection) of these tanks will be performed prior to the end of the current license period to assess the condition of the tank wall.

These inspections have provided an effective means of assuring the structural integrity of buried piping and tanks at Ginna Station.

Conclusion

Continued inspections of opportunity and other one-time inspections will provide reasonable assurance that the effects of aging for buried piping and tanks will be adequately managed during the license renewal period.

B2.1.8 Buried Piping and Tanks Surveillance

Program Description

The purpose of this program is to define surveillance and preventive measures which could be used to mitigate corrosion by protecting the external surface of buried carbon steel piping and tanks. It is recommended by NUREG-1801 to use NACE standards RP-0285-95 and RP-0169-96.

RG&E does not employ these standards, or credit the surveillance and preventive measures referenced in these standards, as aging management programs. Thirty three years of operation, and inspection, testing, and surveillance activities, have demonstrated the effectiveness of current Ginna Station programs in maintaining the intended functions of buried carbon steel piping and tanks. Descriptions of these aging management programs employed currently at Ginna Station, and those to be employed, are provided in the following program descriptions:

- ASME Section XI, Subsections IWB, IWC, IWD, Inservice Inspection Program (Section B2.1.2)
- Water Chemistry Control (Section B2.1.37)

- Open-cycle Cooling (Service) Water System (Section B2.1.22)
- Fire Water System (Section B2.1.14)
- Fuel Oil Chemistry (Section B2.1.16)
- One-time Inspection (Section B2.1.21)
- Buried Piping and Tanks Inspection (Section B2.1.7)
- Structures Monitoring Program (Section B2.1.32)
- Periodic Surveillance/Preventive Maintenance (Section B2.1.23)
- Systems Monitoring (Section B2.1.33)

B2.1.9 Closed-Cycle (Component) Cooling Water System

Program Description

The Closed-Cycle (Component) Cooling Water System Program at Ginna Station applies to the Component Cooling Water (CCW) System.

The program includes (a) preventive measures to minimize corrosion and (b) surveillance testing and inspection to monitor the effects of corrosion on the intended functions of the system. The program relies on maintenance of system corrosion inhibitor concentrations within specified limits of Electric Power Research Institute [EPRI] TR-107396 to minimize corrosion. Surveillance testing and inspections for closed-cycle cooling water system components are performed to evaluate system and component performance. These measures ensure that the CCW system and components serviced by the CCW system are performing their functions acceptably.

B2.1.9.1 Operating Experience

An engineering activity was performed in 1997 to address external corrosion of uninsulated CCW piping, due to intermittent condensation. Prior to insulating the piping, UT readings were taken at selected locations, and no significant wall thinning was noted. Installation of the insulation has prevented this problem from re-occurring.

In 1998, the CCW heat exchangers were retubed after thirty years of service to correct significant wall loss of the tubes on the Service Water side. During the retubing process, an internal inspection was made of the CCW heat exchangers and they were found to be in excellent physical condition.

Conclusion

The Closed-Cycle (Component) Cooling Water System Program provides reasonable assurance that the aging effects will be managed so that the components within the scope of License Renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

There are differences between the parameters monitored at Ginna Station and those recommended by NUREG-1801. For example, EPRI TR-107396 is not referenced in Ginna procedures, and the only parameters monitored are pH, corrosion inhibitor concentrations, and radioactivity. Plant operating experience has not demonstrated the need to monitor the additional parameters in the EPRI report, such as corrosion products, calcium, potassium, or refrigerant chemicals.

Other parameters that are not monitored on each cooler or heat exchanger are the inlet and outlet temperatures and differential pressure; however, temperatures and pressures are monitored at selected locations throughout the system and differential pressure is monitored on the Service Water side of the CCW heat exchangers.

Relative to detection of aging effects, NUREG-1801 suggests the use of corrosion coupons. These are not used at Ginna Station; NDE is used at locations where loss of material may occur. Also, Ginna Station does not perform MIC testing on the chromated water in the CCW system; plant and industry experience indicates that the use of potassium dichromate has inhibited MIC growth.

These differences from the GALL have been evaluated and determined to be minor in terms of assuring proper functionality of system components.

B2.1.10 **Compressed AIr Monitoring**

Program Description

This program consists of inspection, testing, and monitoring of the instrument and service air systems to ensure that oil, water, rust, dirt, and other contaminants are kept within specified limits. The purpose of such actions is to ensure that license renewal intended functions could be accomplished for the period of extended operation.

This issue was discussed in detail in NRC Generic Letter 88-14. In RG&E's response to that letter, details were provided of how compressed air systems are operated, tested and maintained at Ginna Station. It was also stated that air-operated valves were verified to fail-safe on loss of air, and that therefore the compressed air systems at Ginna Station did not perform a safety function (letter from RG&E to NRC, June 17, 1991).

In accordance with the above discussion and as demonstrated by the results of the scoping process, it has been concluded that the Plant Air Systems are not within the scope of license renewal.

B2.1.11 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

Program Description

The purpose of the aging management program described herein is to provide reasonable assurance that the intended functions of electrical cables and connections that are not subject to the environmental qualification requirements of 10 CFR 50.49 (Reference 5) and are exposed to adverse localized environments caused by heat, radiation, or moisture will be maintained consistent with the current licensing basis through the period of extended operation. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service environment for the cable. An adverse variation in environment is significant if it could appreciably increase the rate of aging of a component or have an immediate adverse effect on operability. Conductor insulation materials used in cables and connections may degrade more rapidly than expected in these adverse localized environments. This program, as described, can be thought of as a sampling program. Selected cables and connections from accessible areas (the inspection sample) are inspected and represent, with reasonable assurance, all cables and connections in the adverse localized environments. If an unacceptable condition or situation is identified for a cable or connection in the inspection sample, a determination is made as to whether the same condition or situation is applicable to other accessible or inaccessible cables or connections. Cables and connections used in low-level signal applications that are sensitive to reduction in insulation resistance are included in the scope of this program. Technical information and guidance provided in NUREG/CR-5643 (Reference 14), IEEE Std. P1205-2000 (Reference 15), and SAND 96-0344 (Reference 16), and EPRI TR-109619 (Reference 17) are considered.

B2.1.11.1 Scope of Program

This inspection program applies to accessible electrical cables and connections as well as cables used in low-level signal applications that are sensitive to reduction in insulation resistance (e.g., radiation monitoring and nuclear instrumentation) within the scope of license renewal that are installed or stored in the following plant buildings/areas:

Auxiliary Building, Standby Auxiliary Feedwater Building, Control Building, All-Volatile-Treatment Building, Cable Tunnel, Diesel Generator Building, Intermediate Building, Reactor Containment, Service Building, Screen House, Turbine Building, Technical Support Center.

Plant buildings/areas not listed above that are used to store electrical cables and connections in the scope of license renewal for a specific, approved application (i.e. Appendix R equipment restoration) do not have adverse localized environments.

B2.1.11.2 Preventive Actions

This is an inspection program and no actions are taken as part of this program to prevent or mitigate aging degradation.

B2.1.11.3 Parameters Monitored/Inspected

Readily accessible non-EQ insulated cables and connections installed in the areas described in the scope of this program are visually inspected for moisture and cable and connection jacket surface anomalies such as embrittlement, discoloration, cracking or surface contamination. Cable and connection jacket surface anomalies are precursor indications of conductor insulation aging degradation from heat or radiation in the presence of oxygen and may indicate the existence of an adverse localized equipment environment. An adverse localized equipment environment is a condition in a limited plant area that is significantly more severe than the specified service condition for the insulated cable or connection.

B2.1.11.4 Detection of Aging Effects

Conductor insulation aging degradation from heat, radiation, or moisture in the presence of oxygen causes cable and connection jacket surface anomalies. Accessible electrical cables within the scope of license renewal and installed in plant areas described in the scope of this program are visually inspected at least once every 10 years. This is an adequate period to preclude failures of the conductor insulation since experience has shown that aging degradation is a slow process. A 10-year inspection frequency will provide two data points during a 20-year period, which can be used to characterize the degradation rate. The first inspection for license renewal is to be completed before the end of the current license period.

B2.1.11.5 **Monitoring and Trending**

The two 10-year inspections will provide data that can be used to assess a trend in the degradation rate of the cables.

B2.1.11.6 **Acceptance Criteria**

The accessible cables and connections are to be free from unacceptable, visual indications of surface anomalies, which suggest that conductor insulation or connection degradation exists. An unacceptable indication is defined as a noted condition or situation that, if left unmanaged, could lead to a loss of the intended function.

B2.1.11.7 **Corrective Actions**

Corrective actions are implemented at Ginna Station in accordance with the requirements of 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants” as described in the “Quality Assurance Program for Station Operation” (ND-QAP). Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

All unacceptable visual indications of cable and connection jacket surface anomalies are subject to an engineering evaluation in accordance with the plant corrective action program. Such an evaluation is to consider the age and operating environment of the component, as well as the severity of the anomaly and whether such an anomaly has previously been correlated to degradation of conductor insulation or connections. Corrective actions may include, but are not limited to, testing, shielding or otherwise changing the environment, or relocation or replacement of the affected cable or connection. When an unacceptable condition or situation is identified, a determination is made as to whether the same condition or situation is applicable to other accessible or inaccessible cables or connections.

B2.1.11.8 Confirmation Process

Confirmation of the effectiveness of the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B (Reference 2).

B2.1.11.9 Administrative Controls

Ginna Station QA procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B and will continue to be adequate for the period of extended operation.

B2.1.11.10 Operating Experience

Operating experience has shown that adverse localized environments caused by heat or radiation for electrical cables and connections may exist next to or above steam generators, pressurizers and hot process pipes, such as steam and feedwater lines. These adverse localized environments have been found to cause degradation of the insulating materials on electrical cables and connections that is visually observable, such as color changes or surface cracking. These visual indications can be used as indicators of degradation.

Conclusion

This Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will adequately manage the effects of aging so that there is reasonable assurance that these components will perform their intended functions in accordance with the current licensing basis during the period of extended operation. Accessible electrical cables within the scope of this program will be visually inspected at least once every ten years. This is considered an adequate frequency to preclude failure of the conductor insulation since experience has shown that insulation aging degradation is a slow process. The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program that will be consistent with the NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section XI.E1, “Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements” (Reference 3).

B2.1.12 Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits

Discussion

Rochester Gas and Electric (RG&E) believes that invoking the NUREG-1801 XI.E1, Electrical Cables not Subject to 10 CFR 50.49 Environmental Qualification Requirements program to manage the effects of aging in accessible non-EQ cable and connectors provides reasonable assurance that these SC's will perform their intended function during the period of extended operation.

While RG&E agrees in principle that instrument loop calibrations could detect cable degradation as described in NUREG-1801 and the referenced technical reports, RG&E does not agree that crediting instrument loop surveillance testing as part of a new program for managing those aging effects is beneficial.

The basis for this conclusion includes consideration of the following facts:

- Instrument loops are comprised of many discrete components. Each component is subject to unique factors and many components have the potential of introducing calibration variables (e.g. drift) that mimic the symptoms that would be expected from cable or connector insulation degradation. As acknowledged in NUREG-1801, when a loop is found out of calibration troubleshooting is performed and that trouble shooting would eventually include consideration of the instrument cable.
- Plant Operating Experience has shown that most cable or connector failures detected during instrument calibrations are typically due to maintenance activities themselves, not from component aging.
- As instrument loop components are calibrated, repaired or replaced, after adjustment any meaningful trending information that may yield indication of insulation degradation is effectively re-zeroed. The very act of calibrations themselves will, for a time, mask the symptoms of insulation degradation.
- There is no way to know with prescience why an instrument loop may be exhibiting a particular behavior. The only effective way to ensure that a cable or connector is not developing precursors symptomatic of aging effects that may impact the loop function is to periodically monitor the condition of the cable.

The surveillance of instrument loops is an important part of the plant-licensing basis. However, for the reasons described above, surveillance of instrument loops is not considered an effective tool for managing the effects of aging in the passive long-lived portions of an instrument loop. Experience has shown that the aging effects of cable and connector insulation occur very slowly. Therefore, routine maintenance, calibration and repair activities on the active components in an instrument loop initially work to mask indications of cable and cable and connector insulation degradation. Only after the active portions of a loop can no longer be adjusted to compensate for cable and connector degradation would the passive portions of the instrument loop become suspect. Surveillance provides meaningful information, but that information is primarily used to cause changes to the active portions of an instrument loop. The predominate cause of non-event driven degradation in cable and connector insulation is thermal aging. External inspection of cables and connectors and their host environments identifies the possibility of thermal aging long before instrument loop adjustments can no longer compensate for current leakage. Because of this, RG&E considers that the only legitimate way to ensure the continued functioning of the long-lived passive components are those inspection activities performed under the XI.E1 program, Electrical Cables not Subject to 10 CFR 50.49 Environmental Qualification Requirements.

B2.1.13 Fire Protection

Program Description

The Ginna Station Fire Protection Program includes provisions for aging management of fire barriers and fire pumps. The fire barrier inspection program requires visual periodic visual inspection and functional tests of fire barrier penetration seals, fire barrier walls, ceilings, and floors, and periodic visual inspection and functional tests of fire rated doors to ensure their operability is maintained. The program also requires the fire pumps to be periodically tested, with preventive maintenance and inspections performed to ensure their operability. The program also provides for periodic inspection and testing of the relay room halon fire suppression system.

B2.1.13.1 Operating Experience

A review of previous fire barrier inspection results, action reports, and maintenance work requests provides assurance that fire seals, barriers and walls remain intact to perform their intended function. These inspections have effectively identified event-driven degradation such as torn Hemyc wrap, damaged fire seals, and cracked mortar/caulk in walls consistent with use and operation of the facility. No evidence of age-related degradation has been detected.

Periodic inspection, preventive maintenance, and functional testing of fire pumps provided the data and trending necessary to replace the Diesel Fire Pump engine in 1994.

Inspection results and actions taken as a result of the Fire Protection Program continue to provide an effective means of ensuring the structural integrity of systems and components within the program scope during the license renewal period.

Conclusion

The Fire Protection Program is consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section XI.M26, “Fire Protection” (Reference 3). The program provides reasonable assurance that the aging effects for fire seals, fire barriers, fire pumps, and the halon system will be managed such that the intended function of the components within the scope of the program will be maintained during the license renewal period.

B2.1.14 **Fire Water System**

Program Description

The Fire Water System Program is implemented by the Ginna Station Fire Protection Program which includes provisions for aging management of the fire water system and associated components. These components include sprinklers, nozzles, fittings, hydrants, hose stations, standpipes, fire water storage tank, fire booster pump, etc. System and component testing is conducted in accordance with the applicable National Fire Protection Association (NFPA) codes and standards. The fire water system and associated components are normally maintained at required pressure and monitored such that a loss of system pressure is immediately detected and corrective actions initiated. In addition to the testing performed per NFPA codes, portions of the fire water system are subjected to full flow testing. Also, internal portions of the fire water system are visually inspected when disassembled for maintenance. Volumetric NDE inspections using appropriate techniques are performed on sections of the system piping to detect wall loss and fouling. The flow testing and visual and/or volumetric inspections assure that any wall thinning due to corrosion, microbiologically influenced corrosion (MIC), or biofouling are managed such that the system function is maintained.

B2.1.14.1 Operating Experience

Over the life of the plant, portions of the underground yard hydrant system have been replaced with upgraded components based on the results of flow tests and/or inspections. The fire water storage tank has been periodically inspected and preventive maintenance such as recoating the tank internals has been completed to provide additional protection.

Inspection results and actions taken as a result of the Fire Protection Program continue to provide an effective means of ensuring the structural integrity of systems and components within the program scope during the license renewal period.

Conclusion

The Ginna Fire Water System Program will be consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section XI.M27, "Fire Water System" (Reference 3). A review of previous fire water system inspection results, action reports and maintenance work requests, provides assurance that the system and associated components remain capable of performing their intended function in accordance with the fire protection program and applicable NFPA standards. This program will be enhanced to provide for replacement or representative sample testing of sprinklers with a service life of 50 years. This replacement/testing activity will be performed at 10 year intervals following the 50 year in-service testing.The program provides reasonable assurance that the aging effects for the fire water system and associated components will be managed such that the intended function of the components within the scope of the program will be maintained during the license renewal period.

B2.1.15 Flow-Accelerated Corrosion

Program Description

The Ginna Station Flow-Accelerated Corrosion Program is implemented by approved site specific procedures. Flow-Accelerated Corrosion (FAC) is one form of erosion/corrosion that can affect piping and pressure vessels with flowing water or wet-steam. The program is a comprehensive program that addresses erosion/corrosion control measures in accordance with Electric Power Research Institute (EPRI) guidelines in the Nuclear Safety Analysis Center (NSAC)-202L, Revision 2. The program includes the performance of: (a) an analysis to determine critical locations using the predictive CHECWORKS computer code, (b) limited baseline inspections to determine the extent of thinning at these locations, and (c) follow-up inspections to confirm the predictions, or repairing or replacing components as necessary.

B2.1.15.1 **Operating Experience**

Wall-thinning problems in single-phase systems have occurred in feedwater and condensate systems (NRC IE Bulletin No. 87-01 and NRC Information Notices (INs)81-28, 92-35, and 95-11), in two-phase piping in extraction steam lines (NRC INs 89-53 and 97-84), and in moisture separation reheater and feedwater heater drains (NRC INs 89-53, 91-18, 93-21, and 97-84).

Review of plant-specific operating experience is highlighted as follows:

- In July, 1986, the reactor was manually tripped due to a large steam leak in a moisture separator drain line elbow. The failure of the elbow was due to wall thinning related to FAC and was caused by impingement of steam from a level control valve located in close proximity to the elbow. The elbow was replaced with chrome-moly material and the piping/valve reconfigured to reduce the impingement.
- In March, 1989, erosion/corrosion program inspections found two instances of wall thinning related to FAC in SG carbon steel blowdown piping. The piping was subsequently replaced with chrome-moly piping materials during the 1990 and 1991 refueling outages.
- In June, 1992, a carbon steel pre-separator flash tank located in the extraction steam system experienced a fishmouth rupture. This caused an unisolable steam leak and resulted in a plant shutdown. The wall loss was attributed to erosion/corrosion (FAC) as a result of two-phased water/steam impingement. The impingement was caused by the configuration of the tank internals. The tank and an identical tank in a duplicate line and inlet piping were redesigned and replaced with chrome-moly components with stainless steel internals. The tanks and associated piping were also added to the erosion/corrosion inspection program.
- In July, 1995, erosion/corrosion inspections identified some additional portions of the SG blowdown system, that were previously replaced with chrome-moly piping, were found to have wall thinning from impingement due to FAC at higher rates than expected. The piping was replaced with stainless steel material and reconfigured to alleviate the impingement.

- In October, 1997, erosion/corrosion inspection found some localized thinning due to FAC in a control valve and reducer in the main feedwater system. The components were repaired.
- In March, 1999, visual inspection found wall thinning due to FAC in a moisture separator drain hole shell resulting from impingement of steam/moisture at the drains. The areas were repaired and the drain holes were reconfigured with stainless steel at new locations to eliminate the direct impingement.

The Ginna Station Flow Accelerated Corrosion Program has been revised and improved over the years as inspection data has been collected and trended. Field verification of materials of construction have been performed and used as inputs to CHECWORKS to update predicted rates of wall loss. New locations have been included in the program based on both industry and plant-specific operating experience.

Conclusion

The Ginna Station Flow-Accelerated Corrosion Program is consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section XI.M17, “Flow-Accelerated Corrosion” (Reference 3). The program provides reasonable assurance that the aging effects from FAC will be managed such that the intended function of the piping and components within the scope of the program will be maintained during the license renewal period.

B2.1.16 Fuel Oil Chemistry

Program Description

The Fuel Oil Chemistry Program includes (a) surveillance and maintenance procedures conducted in accordance with plant Technical Specifications to mitigate aging effects such as loss of material due to corrosion and fouling buildup on the internal surfaces of fuel oil storage tanks and associated components in systems that contain fuel oil, and (b) measures to verify the effectiveness of the surveillance/ maintenance activities and confirm the absence of aging effects. The program includes (a) surveillance and monitoring procedures for maintaining fuel oil quality by controlling contaminants in accordance with ASTM Standards D975, D1796, D4057 and D4176 and (b) periodic draining, cleaning and visual inspection of internal surfaces of storage tanks. Supplemental wall thickness measurements (i.e., by UT) may be required at locations where contaminants might accumulate, such as tank bottoms.

B2.1.16.1 Operating Experience

The Diesel Fuel Oil Chemistry Program has been implemented at Ginna Station since the commencement of plant operation. The underground storage tanks have been drained and inspected annually until 1993. Since 1993, annual pressure tests have been performed, and internal inspections are performed on a 10-year frequency. No evidence of degradation of the interior surfaces of either storage tank has ever been observed. No evidence of biological activity has ever been observed.

During the spring RFO in 1987, low day tank levels resulting in loss of normal makeup capacity to the EDGs were determined to be due to partially plugged fuel oil transfer pump suction strainers. The strainers were partially plugged with weld flux and a fibrous material. The weld flux was assumed to have come from original construction and the fibrous material from either cleaning rags or filter material. Corrective actions included draining and inspection of both fuel oil storage tanks, with no problems identified. The strainers were repeatedly cleaned until no further accumulation of debris occurred. In addition, new suction strainers were fabricated and installed, cleaning procedures were revised to prohibit exposure to fibrous contaminants, and additional in-line strainers and screen filters were installed.

A routine NRC inspection in 1990 concluded that the Ginna program for procurement, receipt, sampling and inspection of EDG fuel oil is considered to meet the guidelines of RG 1.137 and assures an adequate supply of proper quality fuel oil to the EDGs.

The Fuel Oil Chemistry Program and periodic storage tank inspections have proven effective in managing the effects of aging resulting from fuel oil contamination at Ginna Station.

Conclusion:

The Fuel Oil Chemistry Program at Ginna Station provides reasonable assurance that the effects of aging will continue to be managed such that no loss of intended function will occur during the period of extended operation. There are some differences between the Ginna Station program and the program described in GALL Section XI.M32. Biocides are not added to the fuel and particulate sampling is not conducted in accordance with modified ASTM D2276. However, plant-specific operating experience demonstrates that the existing program has effectively managed the effects of aging due to fuel oil contamination since the commencement of plant operation.

B2.1.17 Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

There are no inaccessible medium voltage (2kV - 15 kV) cables (e.g., installed in conduit or direct buried) within the scope of license renewal. Medium voltage cables and connections subject to an aging management review are not installed in environments that lead to the formation of water trees (i.e. not exposed to significant moisture) therefore no aging management program other than that described in GALL Section XI.E1 is required.

B2.1.18 Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems

Program Description

This program demonstrates that testing and monitoring programs have been implemented to ensure that cranes are capable of sustaining their rated loads.

It should be noted that many components of a crane system perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life. These components are screened out of the license renewal aging management process. This program is primarily concerned with structural components that make up the bridge, trolley, rails, stops, and lifting devices.

B2.1.18.1 Operating Experience

Draft NUREG-XXXX, “Technical Assessment, Generic Issue 186: Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants,” January 17, 2002, provides a comprehensive assessment of crane issues. There have been numerous crane incidents, some of which resulted in the publication of NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants.” Most crane failures are caused by human error (not following procedures, improper test) or design issues (poor engineering). Less than 10% of failures were due to improper maintenance, and most of these were due to electrical malfunctions. There is very little history of wear-related or corrosion-related degradation that has impaired the ability of cranes in the industry to perform their intended functions. A reevaluation of crane operations based on Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," 4/11/96, concluded that although there were some inconsistencies between crane operation and the licensing basis at some nuclear power plants, few changes were required by licensees in their operation of cranes (and none related to age-related degradation).

Only one major crane failure occurred at Ginna Station. During plant construction, a portion of the reactor vessel internals weighing 90 tons was dropped about 6 feet. The cause of failure was attributed to a crane brake failure (crane motor overheated and the electromagnetic brake failed). No experience with crane failures due to age-related degradation such as wear or corrosion has occurred.

Conclusion

The Ginna Station inspection of Overhead Heavy Load and Light Load (related to refueling) Handling Systems Program is consistent with NUREG-1801. The inspection and testing program applied to crane assemblies at Ginna Station provides assurance that the intended functions of the cranes will continue to be met during the period of extended operation.

B2.1.19 Loose Part Monitoring

Program Description

The purpose of this program is to rely on inservice monitoring to detect and monitor loose parts in LWR power plants. NUREG-1801 suggests that this program should include measures to monitor and detect metallic loose parts by using transient signals analysis on acoustic data generated due to loose parts impact.

Ginna Station does not employ a loose part monitoring system for the reactor vessel. There is a loose parts monitoring system employed for the steam generators, which is called the digital metal impact monitoring system (DMIMS). However, use of DMIMS is not considered to be an aging management program at Ginna Station - rather, it is a reactive measurement system to detect failed or FME components that have inadvertently entered the steam generators. Aging management programs related to the reactor coolant system and reactor vessel internals are described in:

- ASME Section XI, Subsections IWB, IWC, IWD, Inservice Inspection Program (Section B2.1.2)

- Water Chemistry Control (Section B2.1.37)
- Reactor Vessel Head Penetration Inspection (Section B2.1.26)
- Thermal Aging Embrittlement of CASS (Section B2.1.34)
- Reactor Vessel Internals (Section B2.1.27)
- Bolting Integrity (Section B2.1.5)
- Steam Generator Tube Integrity (Section B2.1.31)
- Reactor Vessel Surveillance (Section B2.1.28)
- One-time Inspection (Section B2.1.21)
- Periodic Surveillance/Preventive Maintenance (Section B2.1.23)
- Systems Monitoring (Section B2.1.33)
- Thimble Tube Inspection (Section B2.1.36)

B2.1.20 Neutron Noise Monitoring

Program Description

The purpose of this program is to rely on monitoring the excess neutron detector signals due to core motion to detect and monitor significant loss of axial preload at the core support barrel's upper support flange.

RG&E does not include this program as one of the Ginna Station aging management programs. Changes in the support structure are managed by the following aging management programs:

- ASME Section XI, Subsections IWB, IWC, IWD, Inservice Inspection Program (Section B2.1.2)
- Reactor Vessel Internals Program (Section B2.1.27)

Thirty three years of successful operation, including two 10-year detailed inservice inspections of the reactor vessel internals, have demonstrated that additional programs, such as neutron noise monitoring programs, are not required to ensure continued functionality of the core support barrel upper support flange.

B2.1.21 **One-Time Inspection**

Program Description

The Ginna Station One-Time Inspection Program will include measures to verify the effectiveness of an existing aging management program and confirm the absence of an aging effect. The One-Time Inspection Program will address potentially long incubation periods for certain aging effects and provide a means of verifying that an aging effect is either not occurring or is progressing so slowly as to have negligible effect on the intended function of the structure or component. The program elements will include (a) determination of appropriate inspection sample size based on materials of construction, environment, plausible aging effects, and operating experience, (b) identification of inspection locations, (c) selection of examination technique, with acceptance criteria, and (d) evaluation of results to determine the need for additional inspections or other corrective actions. The inspection sample will include locations where the most severe aging effect(s) would be expected to occur. Inspection methods will include visual (or remote visual), surface or volumetric examination, or other established NDE techniques.

For treated-water systems in stagnant or low flow areas, one-time inspections will provide verification of the effectiveness of water chemistry controls in managing effects of aging. For Class 1 piping and associated components less than 4" NPS in diameter that are not inspected by volumetric techniques during inservice inspections, one-time inspections will confirm that crack initiation and growth due to stress corrosion cracking (SCC) or cyclic loading is not occurring. In addition, one-time inspections will be used to determine whether (a) loss of material due to galvanic corrosion at galvanic couples and (b) loss of material due to selective leaching for gray cast iron or brass components represent significant aging effects requiring management in treated water systems.

B2.1.21.1 **Scope of Program**

The One-Time Inspection Program will be used to determine the acceptability of components that may be susceptible to various aging effects and to verify that unacceptable degradation is not occurring, thereby validating the effectiveness of an existing aging management program or confirming that there is no need to manage age-related degradation for the period of extended operation. The scope of this program includes the following:

- Verification of the effectiveness of the Water Chemistry Control Program for managing the effects of aging in stagnant or low flow portions of piping, or occluded areas of components, exposed to treated water environments;
- Managing cracking due to SCC or cyclic loading due to thermal fatigue in small bore Class 1 piping (< 4 inches NPS) that is directly connected to the reactor coolant system;
- Managing loss of material due to galvanic corrosion on the internal surfaces of piping and components in treated water systems at locations where galvanic couples are present;
- Managing loss of material and/or loss of structural integrity due to selective leaching on the internal surfaces of piping and components made of gray cast iron, bronze, or brass exposed to treated water or raw water environments.

B2.1.21.2 **Preventive Actions**

The One-Time Inspection Program will perform inspection activities only.

B2.1.21.3 **Parameters Monitored/Inspected**

The program will monitor parameters directly related to the degradation of a component, such as loss of material due to pitting, crevice, or galvanic corrosion, loss of material or structural integrity due to selective leaching, and cracking due to SCC or cyclic loading. Inspections will be performed in accordance with qualified NDE procedures and include visual, volumetric, and/or surface techniques.

B2.1.21.4 **Detection of Aging Effects**

The inspection sample size will be a representative sample of the system population, and, to the extent possible, will include bounding or lead components most susceptible to aging based on time in service, severity of operating environment, and operating experience. For small-bore Class 1 piping, other considerations are accessibility and exposure levels. Volumetric examinations will be used for examinations of small-bore piping and associated components since cracking would be expected to originate at the internal surface of the pipe. Other inspections for detection of loss of material due to pitting, crevice and galvanic corrosion will be performed using visual, surface or volumetric methods, as appropriate. In addition, hardness measurements may be performed to evaluate potential loss of structural and/or pressure boundary integrity due to selective leaching. The inspections will be performed in accordance with qualified procedures using qualified personnel, consistent with the ASME Code and 10 CFR 50, Appendix B.

The one-time inspections will be conducted prior to, but near the end of the current operating license period so as to allow sufficient time for mechanisms with long incubation periods to become active. Inspections will be scheduled to minimize the impact on plant operations.

B2.1.21.5 **Monitoring and Trending**

The One-Time Inspection Program does not provide guidance for monitoring and trending. However follow-up examinations will be required if unacceptable conditions are discovered, requiring expansion of sample size and locations of inspections.

B2.1.21.6 **Acceptance Criteria**

Any significant indications or relevant conditions of degradation will be evaluated in accordance with the Ginna Station Corrective Action Program. Required expansion of inspection scope and sample size will be determined by engineering evaluation. Criteria for minimum wall thickness will be based on ASME Code design requirements or approved engineering outputs.

B2.1.21.7 **Corrective Actions**

Corrective actions are implemented at Ginna Station in accordance with the requirements of 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants” as described in the “Quality Assurance Program for Station Operation” (ND-QAP). Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

B2.1.21.8 **Confirmation Process**

Confirmation of the effectiveness of the One Time Inspection Program will be accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

B2.1.21.9 Administrative Controls

Ginna Station QA procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B and will continue to be adequate for the period of extended operation.

B2.1.21.10 Operating Experience

The One-Time Inspection Program is a new program to be implemented before the end of the current operating period. The inspection scope and techniques that will be employed are consistent with industry practice and have proven effective for timely detection of aging effects.

Conclusion

The Ginna Station One-Time Inspection Program will be consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section XI.M32, "One-Time Inspection" (Reference 3). Implementation of the One-Time Inspection Program will provide reasonable assurance that effects of aging such as cracking, loss of material, and loss of structural/mechanical integrity will be managed such that no loss of intended function will occur during the period of extended operation.

B2.1.22 Open-Cycle Cooling (Service) Water System

Program Description

At Ginna Station, the open-cycle cooling water system is called the Service Water System. The Service Water System Reliability and Optimization Program (SWSROP) relies on implementation of the recommendations of the Nuclear Regulatory Commission (NRC) Generic Letter (GL) 89-13 to ensure that the effects of aging on the Service Water System will be managed for the extended period of operation. The program includes surveillance and control techniques to manage aging effects caused by biofouling, corrosion, erosion and silting in the Service Water System or structures and components serviced by the Service Water System.

B2.1.22.1 Operating Experience

Heat exchangers have experienced erosion/corrosion of end bells, biofouling build-up, and silt accumulation. Zebra mussels have been found and are controlled by the chlorination system, and periodic cleaning of the heat exchanger tubes. Piping systems have experienced corrosion, pitting, MIC, and sedimentation build-up especially in low flow areas and stagnant dead legs off the main flowstream. These are controlled by flushing, the chlorination system, and inspections. Cavitation/erosion of components is monitored by using established NDE methods and components have been repaired/replaced as necessary.

Ginna Station has also replaced and/or retubed many of these heat exchangers, most notably the Reactor Containment Fan Coolers (replaced), Emergency Diesel Lubricating Oil and Jacket Water heat exchangers (retubed), and the Component Cooling Water heat exchangers (retubed).

The guidance of NRC GL 89-13 has been implemented for approximately 10 years and has been effective in managing and monitoring the effects of aging due to biofouling, corrosion, erosion, and silting in components serviced by the Service Water System.

Conclusion

The Ginna “Service Water System Reliability and Optimization Program” implements the NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section XI.M20, “Open-Cycle Cooling Water System” (Reference 3). The SWSROP provides reasonable assurance that aging effects of the Service Water System will be managed so that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

The program is considered consistent with the GALL, with the following minor differences: 1) heat transfer tests are not performed on selected small heat exchangers which are periodically cleaned and inspected in accordance with the Periodic Surveillance and Preventive Maintenance Program, and 2) the SWSROP does not address protective coatings, which are not credited for aging management in the Ginna Service Water System.

B2.1.23 Periodic Surveillance and Preventive Maintenance

Program Description

The Periodic Surveillance and Preventive Maintenance Program has been a vital element of maintaining plant equipment and structure condition and ensuring their reliability at Ginna Station. The Periodic Surveillance and Preventive Maintenance Program is credited for managing aging effects such as loss of material, crack initiation, fouling buildup, and loss of seal for systems, structures, and components within the scope of license renewal. The program provides for visual inspection and examination of surfaces of selected equipment items and components, including fasteners, for evidence of defects and age-related degradation such as corrosion, wear, cracking, fouling, etc., on a specified frequency based on operational experience. Leak inspection of piping and components in selected portions of systems are also performed on a specified frequency. Additionally, the program provides for replacement or refurbishment of certain components on a specified frequency based on operational experience. The Periodic Surveillance and Preventive Maintenance Program is also used to verify the effectiveness of other aging management programs.

B2.1.23.1 **Scope of Program**

The Periodic Surveillance and Preventive Maintenance Program provides for visual inspections and surface examinations of certain piping, equipment and components in all plant systems within the scope of license renewal. Additionally, the Periodic Surveillance and Preventive Maintenance Program provides for replacement or refurbishment of certain components on a specified frequency.

B2.1.23.2 **Preventive Actions**

The inspection and testing activities required by the Periodic Surveillance and Preventive Maintenance Program are primarily monitoring activities. However, periodic replacement or refurbishment of components may be considered preventive in nature.

B2.1.23.3 **Parameters Monitored/Inspected**

The administrative procedures that govern the Periodic Surveillance and Preventive Maintenance Program provide instructions for monitoring systems, structures and components to permit early detection of degradation. The program provides for visual inspection and examination of surfaces of selected equipment items and components, including fasteners, for evidence of defects and age-related degradation such as loss of material due to corrosion and wear, cracking, fouling buildup, and leakage, on a specified frequency based on operational experience. Equipment or systems operating parameters, e.g., pressure, flow, and temperature, are monitored by performance tests. These tests are effective in detecting performance degradation that may be indicative of aging effects.

Current guidelines in operations, maintenance, and surveillance test procedures and plant work orders will be enhanced to provide explicit guidance on detection of applicable aging effects and assessment of degradation.

B2.1.23.4 **Detection of Aging Effects**

Aging effects such as loss of material due to corrosion and wear, cracking, loss of seal, etc., are detected by visual inspection of surfaces for evidence of leakage, material thinning, accumulation of corrosion products, and debris, Operations, maintenance, and surveillance test procedures and task descriptions will be enhanced to provide explicit guidance on detection of applicable aging effects and assessment of degradation.

Administrative procedures that govern the Periodic Surveillance and Preventive Maintenance Program provide for evaluation of frequency and appropriateness of Periodic Surveillance and Preventive Maintenance activities to assess effectiveness and compare with typical industry practices.

B2.1.23.5 **Monitoring and Trending**

The Periodic Surveillance and Preventive Maintenance Program provides for monitoring and trending of material condition and equipment performance. PSPM activity intervals are established to provide timely detection of degradation and are based on service environment as well as industry and plant-specific operating experience and manufacturers recommendations. Operations and maintenance procedures specify activities such as periodic plant walkdowns for monitoring systems, structures and components for early detection of degradation such as coatings failures, corrosion, cracking, leakage and physical condition, mechanical damage, and loose or missing hardware. Data from walkdowns are documented, trended and evaluated to identify and correct deficiencies. Periodic Surveillance and Preventive Maintenance intervals may be adjusted as necessary based on inspection results and industry experience.

B2.1.23.6 **Acceptance Criteria**

Operations, maintenance, and surveillance procedures and specific task instructions will be enhanced to include explicit instructions for detection of aging effects and definition of acceptance criteria. Degradations deemed to be unacceptable are addressed by the ACTION Reporting process under the Corrective Action Program.

B2.1.23.7 **Corrective Actions**

Any identified condition that is determined to be deficient or unacceptable is addressed and evaluated under the Corrective Action Program. Corrective actions are implemented at Ginna Station in accordance with the requirements of 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants” as described in the “Quality Assurance Program for Station Operation” (ND-QAP). Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

B2.1.23.8 **Confirmation Process**

Confirmation of the effectiveness of the Periodic Surveillance and Preventive Maintenance Program is accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B. The ACTION Report disposition process includes checks, follow-up inspections, and reviews to verify the adequacy of proposed corrective actions.

B2.1.23.9 **Administrative Controls**

Ginna Station QA procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B and will continue to be adequate for the period of extended operation.

B2.1.23.10 **Operating Experience**

Periodic Surveillance and Preventive Maintenance activities have been in place since Ginna Station began operation. Review of plant-specific operating experience reveals that significant numbers of ACTION Reports and Plant Work Orders have been generated to correct conditions identified as a result of

Periodic Surveillance and Preventive Maintenance Program activities. These activities have proven to be effective in maintaining the material condition of systems, structures and components and detecting unsatisfactory or degraded conditions.

Conclusion

The Ginna Station Periodic Surveillance and Preventive Maintenance Program, with enhancements identified above, has been evaluated using the generic program attributes identified in Appendix A of the SRP. The continued implementation of the Periodic Surveillance and Preventive Maintenance Program provides reasonable assurance that aging effects will be managed such that the intended function of systems and components within the scope of license renewal will be maintained during the extended period of operation.

B2.1.24 Protective Coatings Monitoring and Maintenance Program

Program Description

Proper maintenance of protective coatings inside containment (described as Service Level 1 in NRC Regulatory Guide 1.54, Rev. 1) is essential to ensure operability of post accident safety systems that rely on water recirculated through the containment emergency Sump “B.” Ginna Station maintains protective coatings inside containment in accordance with our program as described in our December 1, 1998 response to Generic Letter 98-04, to ensure that paint chips or flakes do not dislodge in a post-accident environment and cause unacceptable sump blockage.

This program is not considered a license renewal aging management program. However, to demonstrate compliance with the resolution of GSI-191, the 10 element attributes of NUREG-1801 are discussed below:

B2.1.24.1 Scope of Program

All coatings inside containment that are procured, applied, and maintained by Ginna Station or our contractors are within the scope of the program. In addition, on a periodic basis consistent with refueling outages, a comprehensive containment walkdown is performed to look for all loose or flaking protective coatings, to minimize the potential for clogging of the containment sump “B” screens.

B2.1.24.2 **Preventive Actions**

This program is preventive in nature. All loose protective coatings are removed from the containment prior to startup from each refueling outage. Flaking/peeling paint areas are evaluated, and repaired or replaced if it is considered to be necessary.

B2.1.24.3 **Parameters Monitored/Inspected**

For protective coatings, degradation is considered to be visible defects, such as blistering, cracking, flaking, peeling, rusting, and physical damage.

B2.1.24.4 **Detection of Aging Effects**

Ginna Station periodically conducts visual inspections inside containment. General conditions, including coating conditions, are observed during the VT-2 leakage examination of Class 1 components and piping prior to startup after each refueling outage and during the VT-2 leakage examination of Class 2 and 3 piping, supports, and attachments. General walkdowns by Operations, Performance Monitoring, Systems Engineering, Radiation Protection, and Maintenance personnel, as well as crane inspections prior to refueling outages, ensure a general awareness of conditions by a variety of observers. If a localized area of degraded coating is identified, that area is evaluated and scheduled for repair or replacement, as necessary. These observations, and the resulting repair/replacement activities, assure that the amount of Service Level 1 coatings which may be susceptible to detachment from the substrate during a LOCA event is minimized. Ginna Station has incorporated standard industry guidance on coatings in site-specific procedures.

Ginna Station has also completed the baseline inspection required by ASME Section XI, Subsections IWE and IWL. These inspections provided the parameters to be evaluated for (loose/missing parts, corrosion, erosion, etc.) on the containment steel and concrete surfaces.

B2.1.24.5 **Monitoring and Trending**

The routine inspections performed and evaluated as described in Paragraph B2.1.25.5 above are compared to previous inspections from earlier refueling stages. The need for repair/replacement of degraded protective coating is evaluated, and scheduled as considered necessary.

For IWE/IWL inspections, where no degradation or defects were identified, a five-year inspection interval is used.

B2.1.24.6 Acceptance Criteria

Visual observations of the condition of protective coatings are made by qualified Systems and Structural Engineers, performed in accordance with checklists provided in site-specific procedures for structural monitoring and system engineering walkdowns. Major degradation is documented on an ACTION Report (see B2.1.23.8 below), and repaired as required.

B2.1.24.7 Corrective Actions

Corrective actions are implemented at Ginna Station in accordance with the requirements of 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants” as described in the “Quality Assurance Program for Station Operation” (ND-QAP). Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

B2.1.24.8 Confirmation Process

Confirmation of the effectiveness of the Protective Coatings Monitoring and Maintenance Program is accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

B2.1.24.9 Administrative Controls

Ginna Station QA procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B and will continue to be adequate for the period of extended operation.

B2.1.24.10 Operating Experience

Although Ginna Station was designed and built prior to Regulatory Guide 1.54 and related ANSI Standard N101.4, great care was taken in the selection and application of protective coatings within containment.

In the construction of Ginna Station, contemporary standards were specified to ensure that protective coatings applied would perform their functions under environmental conditions experienced during operation and the design-basis accident and to do so without hazard of interfering with other nuclear components.

Construction painting standards included techniques for preparation of surfaces to be painted, sampling, thickness measurement and control, and a detailed paint schedule including components and paint materials for plant structures and equipment. In addition, separate specifications for the preparation, application, material, and paint sampling for the interior of the containment dome were followed.

The painting of the containment structure and components inside the containment was governed by a Westinghouse process specification. This specification covered the application of paint systems to equipment and structures in containments which use additive spray systems for fission product removal and/or containment cooling.

For Ginna Station, Service Level 1 coatings are subject to the requirements of site specific procedures. Adequate assurance that the applicable requirements for the procurement, application, inspection, and maintenance of protective coatings are implemented is provided by procedures and programmatic controls, approved under the R. E. Ginna Nuclear Power Plant Quality Assurance program.

Service Level 1 coatings used for new applications or repair/replacement activities are procured from a vendor with a quality assurance program meeting the applicable requirements of 10 CFR Part 50 Appendix B. The applicable technical and quality requirements that the vendor is required to meet are specified by Ginna Station. Acceptance activities are conducted in accordance with procedures that are consistent with ANSI N 45.2 requirements (e.g., receipt inspection, source surveillance, etc.). This specification of required technical and quality requirements combined with appropriate acceptance activities provides adequate assurance that the coatings received meet the requirements of the procurement documents.

The qualification testing of Service Level 1 coatings used for new applications or repair/replacement activities inside containment meets the applicable requirements contained in the standards and regulatory commitments referenced. Service Level 1 coatings for Ginna Station are procured from the Carboline Company. The Carboline Company was last assessed by a Quality Control Procurement Audit in Spring, 1999.

The surface preparation, application and surveillance during installation of Service Level 1 coatings used for new applications or repair/replacement activities inside containment also meet the applicable portions of the standards and regulatory commitments referenced. Documentation of completion of these activities is performed consistent with the applicable requirements.

The investigation of materials compatibility in the post-accident design-basis environment included an evaluation of protective coatings for use in the containment. The results of the protective coatings evaluation showed that several inorganic zincs, modified phenolics, and epoxy coatings are resistant to an environment of high temperature (320°F maximum test temperature) and alkaline sodium borate. Long-term tests included exposure to spray solution at 150°F to 175°F for 60 days, after initially being subjected to the design-basis accident cycle. The protective coatings, which were found resistant to the test conditions (that is, exhibited no significant loss of adhesion to the substrate nor formation of decomposition products), comprise virtually all of the protective coatings used in the Ginna containment. Hence, the protective coatings will not add deleterious products to the core cooling solution. Essentially all carbon steel surfaces are coated with Carbozinc-11 (inorganic zinc primer) and Phenoline 305 (modified phenolic top coat). Phenoline 305 protective coating is also used on concrete surfaces.

Several test panels of the types of protective coatings used at Ginna Station were exposed for two design-basis accident cycles and showed no deterioration or loss of adhesion with the substrate.

In the safety evaluation of the SEP Topic VI-1, Organic Materials and Post-Accident Chemistry dated February 19, 1982, plant design was reviewed with respect to the effect of paints and coatings under accident conditions. Phenolic based paints are among the most radiation resistant, remaining serviceable after radiation dosage in excess of 10^9 rad. For a severe Design Basis Accident (DBA), 10^8 rad would be a conservative dose estimate. Most painted areas are calculated to receive less than 10^7 rad.

On the basis of the above information, the NRC found, in SEP Topic VI-1, that there is reasonable assurance that the radiation, thermal, and chemical resistance of the organic coatings used in the plant is sufficiently high that deterioration under DBA conditions would not interfere with the operation of engineered safety features. Qualification tests demonstrated that the types of organic coating materials used in the containment will maintain their integrity and remain in serviceable conditions after exposure to the severe environmental conditions of a DBA.

Conclusion

The Protective Coatings Monitoring and Maintenance Program inside containment, although not developed in accordance with Regulatory Guide 1.54 and ASTM D5163-96, is consistent with the NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section XI.S8, “Protective Coatings Monitoring and Maintenance” (Reference 3). Our program was described in detail in our December 1, 1998 response to Generic Letter 98-04, and was accepted by the NRC in their letter of November 19, 1999. Although consistent with NUREG-1801 it is not considered an aging management program, but is described to demonstrate compliance with the resolution of GSI-191.

B2.1.25 Reactor Head Closure Studs

Program Description

This program includes (a) inservice inspection (ISI) in accordance with the requirements of the ASME Code, Section XI, Subsection IWB (1995 edition through the 1996 addenda), Table IWB2500-1, and (b) preventive measures to mitigate cracking.

The ISI portion of the program is described in its entirety in the program description for “ASME Section XI, Subsection IWB, IWC, IWD, Inservice Inspection”. (Section B2.1.2)

The reactor head closure studs are fabricated from ASME SA-320 Grade L43 (AISI 4340) low-alloy steel and thus are not susceptible to stress corrosion cracking (specified minimum yield strength of 105 ksi). A comprehensive discussion of this subject is provided in the program description for “Bolting Integrity” (Section B2.1.5).

B2.1.26 Reactor Vessel Head Penetration Inspection

Program Description

The program includes performing (a) primary water stress corrosion cracking (PWSCC) susceptibility assessment to identify susceptible components, (b) monitoring and control of reactor coolant water chemistry to mitigate PWSCC, and (c) inservice inspection (ISI) of reactor vessel head penetrations and bottom-mounted instrument tube penetrations, in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWB, Table IWB 2500-1 (1995 edition through the 1996 addenda) to detect PWSCC and its effect on the intended function of the component.

Primary water chemistry is monitored and maintained in accordance with the Water Chemistry Control Program, described in (Section B2.1.37) of this appendix.

B2.1.26.1 Scope of Program

The program is focused on managing the effects of crack initiation and growth due to PWSCC of the reactor vessel head and bottom-mounted instrumentation penetrations of the Ginna reactor vessel. In response to the industry-wide initiative relative to GL 97-01, “Degradation of CRDM/CEDM Nozzle and Other Vessel Closure Head Penetrations,” a comprehensive eddy current inspection of all the Alloy 600 vessel closure head penetrations was performed at Ginna Station in 1999. The bottom-mounted instrumentation penetrations are routinely examined in accordance with ASME Section XI, Subsection IWB-2500-1. Since these penetrations operate at a lower RCS temperature, there has been no significant degradation of these penetrations in the industry.

B2.1.26.2 Preventive Actions

Preventive measures to mitigate PWSCC are in accordance with the Water Chemistry Control Program. The eddy current inspections performed in 1999 were not preventive, but provided excellent indication that no cracking had occurred in these nozzles.

As a result of these examinations, and industry-wide concern as expressed in NRC Bulletins 2001-01 and 2002-01, Ginna Station has decided to replace the reactor vessel head and CRDM penetrations. This replacement is scheduled for the fall 2003 refueling outage.

B2.1.26.3 Parameters Monitored/Inspected

The purpose of the program is to detect PWSCC so as to control or repair degradation which could have a negative effect on the intended pressure boundary function of the reactor vessel head penetrations.

B2.1.26.4 Detection of Aging Effects

The purpose of the eddy current examinations performed in 1999 was to detect PWSCC-initiated cracks in the CRDM penetrations. No significant indications were discovered during these inspections.

The reactor vessel head is planned to be replaced in 2003, with Alloy 690TT material used for all penetrations. When installed, Ginna Station will continue to follow industry events and developments and reevaluate the type, need, and schedule for addition inspections.

B2.1.26.5 Monitoring and Trending

The 1999 volumetric examination indicated no significant degradation of the reactor vessel head penetrations. The next step in this process is to replace the reactor vessel head and CRDMs in 2003. Ginna Station will follow industry events and developments to determine further inspection activities.

B2.1.26.6 Acceptance Criteria

The results of the 1999 volumetric examinations were analyzed, and it was determined that there was no significant degradation of the reactor vessel head penetrations. Safe operation of the station could continue until the planned vessel head replacement in 2003.

B2.1.26.7 Corrective Actions

Corrective actions are implemented at Ginna Station in accordance with the requirements of 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants” as described in the “Quality Assurance Program for Station Operation” (ND-QAP). Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

B2.1.26.8 Confirmation Process

Confirmation of the effectiveness of the Reactor Vessel Head Penetration Program is accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

B2.1.26.9 Administrative Controls

Ginna Station QA procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B and will continue to be adequate for the period of extended operation.

B2.1.26.10 Operating Experience

Significant operating experience has been documented for PWSCC of Alloy 600 vessel head penetrations, particularly GL 97-01, and Bulletins 2001-01 “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,” 8/3/01 and 2002-01, “Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity, 3/18/02.” Comprehensive eddy current examinations were performed at Ginna Station in 1999 as an industry initiative in response to GL 97-01. As noted earlier, no significant degradation was evidenced. However, in response to industry concerns in this area. Ginna Station has proactively planned to replace the reactor vessel head, with penetrations using Alloy 690TT material. This replacement is scheduled for fall 2003.

Conclusion

The Ginna Station ASME Section XI ISI program has been effective in maintaining the intended function of the current reactor vessel upper and lower head penetrations. The type and extent of inspections to be performed under the Reactor Vessel Head Penetration Inspection Program for the new reactor vessel head will be determined as Ginna Station continues to follow industry events and developments.

B2.1.27 **Reactor Vessel Internals**

Program Description

The Ginna Station Reactor Vessel Internals Program includes (a) monitoring and control of reactor coolant water chemistry in accordance with the EPRI guidelines in TR-105714 to ensure the long-term integrity and safe operation of PWR reactor vessel internal (RVI) components, and will include (b) augmentation of the ASME Section XI, Subsection IWB, Table IWB-2500-1 Inservice Inspection (ISI) Program (1995 Edition with 1996 Addenda) for certain susceptible or limiting components or locations,. Detection of fine cracks in non-bolted components will be achieved by augmenting the ASME Section XI ISI Program with enhanced visual methods capable of resolving .0005 inch features of interest when cost effective techniques become available. Inspection and replacement of baffle-former bolts was performed in 1999, and the results are considered acceptable. There are no future plans for inspection/replacement of baffle-former bolts at Ginna Station. However, ongoing industry initiatives will be monitored and the Reactor Vessel Internals Program will be modified appropriately to incorporate industry lessons learned.

B2.1.27.1 **Scope of Program**

This program will be focused on managing the effects of crack initiation and growth due to stress corrosion cracking (SCC) and irradiation assisted stress corrosion cracking (IASCC), and loss of fracture toughness due to neutron irradiation embrittlement or void swelling. The program includes preventive measures for mitigation of SCC and IASCC. In addition, the program will include augmented ISI inspections to monitor the effects of cracking on internals components. Repair and/or replacement activities may be required to maintain RVI intended function. Since cracks would be expected to initiate at surfaces, augmented visual examinations would provide the required detection capability.

The program will include the following elements: (a) identification of the most susceptible or limiting RVI components and locations; (b) development of appropriate inspection techniques to permit detection and characterizing features of interest (fine cracks) and demonstration of effectiveness of the proposed techniques; and (c) implementation of the augmented inspections during the license renewal term.

The following RVI components are judged to be most susceptible to crack initiation and growth due to IASCC and loss of fracture toughness due to neutron irradiation embrittlement and/or void swelling:

- Lower core plate and fuel alignment pins;
- Lower support columns;
- Core barrel and core barrel flange in active core region;
- Baffle and former plates;
- Thermal shield and neutron panels;
- Bolting - lower support column, baffle-former, and barrel former

It is noted that loss of fracture toughness/cracking of cast austenitic stainless steel (CASS) RVI components due to synergistic effects of neutron embrittlement and thermal aging is not an aging effect requiring management for the Ginna Station internals since there are no RVI components within the scope of license renewal which are fabricated from CASS.

B2.1.27.2 Preventive Actions

The RVI program will be primarily an inspection program. However, monitoring and control of reactor coolant water chemistry parameters in accordance with EPRI TR-105714 is a preventive measure which reduces susceptibility of RVI components to crack initiation and growth due to SCC and IASCC.

B2.1.27.3 Parameters Monitored/Inspected

The program will monitor the effects of cracking on RVI components by detection and sizing of cracks using augmented ISI techniques such as enhanced visual and ultrasonic methods according to the requirements of ASME Section XI, Table IWB-2500-1.

B2.1.27.4 Detection of Aging Effects

Reactor vessel internal components are inspected in accordance with the inservice inspection requirements of ASME Section XI, Subsection IWB, Examination Category B-N-3 for all accessible surfaces of reactor core support structures that can be removed from the vessel. ASME Section XI specifies visual VT-3 examination to determine the general mechanical and structural condition of the component supports by (a) verifying parameters such as clearances, settings and physical displacements, and (b) detecting discontinuities and imperfections such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear or erosion.

Augmented examinations such as enhanced visual techniques will be required for detection of fine or tight cracks. The enhanced visual technique must be capable of .0005 inch resolution in order to detect flaws sufficiently small in size to preserve intended function. Detection of cracking in bolts and fasteners is generally not possible by visual techniques because cracking typically occurs at the bolt head/shank intersection, which is not accessible for visual examination. Ultrasonic examination techniques are required for detecting cracks in bolts and fasteners used to secure bolted connections such as baffle/former bolts and other bolts.

The VT-3 examination per ASME Section XI, Subsection IWB, Category B-N-3 is performed once per 10-year interval on each part of the RVI. Augmented examinations for detection of cracking in components susceptible to IASCC or loss of fracture toughness due to irradiation embrittlement will be scheduled as either periodic or one-time inspections. With respect to augmented examinations of component types susceptible to IASCC and irradiation embrittlement, components with the highest susceptibility (as determined by fluence, temperature and stress) will be selected for examination. If these leading components are found to be free of cracking, less susceptible components may not require examination. The scheduling of future augmented examinations will depend on the results of the initial examinations.

B2.1.27.5 **Monitoring and Trending**

The Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) Issue Task Group (ITG) on Reactor Vessel Internals is currently sponsoring research with the aim of acquiring more data on IASCC, SCC, neutron embrittlement, synergistic effects of neutron embrittlement plus thermal aging, void swelling, and stress relaxation of PWR RVI. Much of this data will be gained through analysis of material currently being irradiated in both research reactors and commercial reactors worldwide. The results of this data will assist Ginna Station in determining the scope and schedule of augmented examinations for cracking, change in dimensions, and loss of preload due to stress relaxation. Ginna Station will actively monitor the EPRI MRP ITG and Westinghouse Owners Group work so as to gain the full benefit of the data that will be generated.

B2.1.27.6 Acceptance Criteria

Any indication or relevant condition of degradation will be evaluated in accordance with IWB-3100, which refers to acceptance standards contained in IWB-3400 and IWB-3500.

B2.1.27.7 Corrective Actions

Repair and replacement activities are performed as required by ASME Section XI. Repairs are conducted in accordance with the requirements of IWB-4000 and replacements in accordance with IWB-7000.

Corrective actions are implemented at Ginna Station in accordance with the requirements of 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants” as described in the “Quality Assurance Program for Station Operation” (ND-QAP). Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

B2.1.27.8 Confirmation Process

Confirmation of the effectiveness of the Reactor Vessel Internals Program will be accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

B2.1.27.9 Administrative Controls

Ginna Station QA procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B and will continue to be adequate for the period of extended operation.

B2.1.27.10 Operating Experience

A thorough review of industry operating experience was performed, including NRC generic communications and other industry event reports. Two significant NRC communications reviewed were NRC Information Notice 98-11, “Cracking Of Reactor Vessel Internal Baffle Former Bolts In Foreign Plants,” and NRC Information Notice 84-18, “Stress Corrosion Cracking in PWR Systems.” Most of the industry operating experience reviewed has involved cracking of austenitic stainless steel baffle-former bolts, reportedly due to IASCC, and SCC of high-strength internals bolting. SCC of guide tube split pins fabricated from Alloy X-750 has also been reported. Split pin failures were attributed to improper heat treatment and excessive installation preloads, resulting in increased susceptibility to SCC.

A review of Ginna Station plant-specific experience with reactor vessel internals indicates that Ginna has responded proactively to industry experience with respect to reactor internals degradation. Two examples of this proactive response are as follows:

- Guide-tube split pins were preemptively replaced at Ginna during the 1986 refueling outage using a pin of improved design and heat treatment; however, there is no evidence that split pin failures actually occurred at Ginna.
- Augmented examination and preemptive replacement of selected baffle-former bolts was performed at Ginna in 1999. Out of a total population of 728 Type 347 stainless steel bolts, 639 were examined by UT. Approximately 9% of these bolts exhibited defect-like indications. A total of 56 Type 316 stainless steel replacement bolts were installed. These were bolts that contained defect-like indications and were part of a pre-qualified minimum bolt pattern for two-loop nuclear plants that was generated by the Westinghouse Owners Group (WCAP-15036). Maintaining the structural integrity of the bolts within this pattern assured compliance with requirements of ASME III, Subsection NG (1989).

Results of ongoing research being conducted by Electric Power Research Institute, Reactor Vessel Internals Issue Task Group (EPRI RI-ITG) on aging effects of reactor vessel internals will be closely followed to assure that guidance on corrective action for these aging effects is incorporated into aging management program activities at Ginna Station. Ginna Station also participates in Westinghouse Owner's Group activities related to reactor vessel internals.

Conclusion

The Ginna Station Reactor Vessel Internals Program is consistent with NUREG-1801 relative to monitoring and control of reactor coolant water chemistry in accordance with the EPRI Guidelines in TR-105714. RG&E is also committed to ASME Section XI, Subsection IWB (1995 Edition with 1996 Addenda). The current ASME Section XI ISI program is considered to provide reasonable assurance that aging effects will be managed such that the intended functions of reactor vessel internals components will be maintained during the license renewal period. That notwithstanding, RG&E will participate in industry activities concerning the development of augmented inspection techniques in order to visually inspect for fine cracks (0.0005 inch) and other changes in dimension in non-bolted components.

B2.1.28 Reactor Vessel Surveillance

Program Description

The Code of Federal Regulations, 10 CFR Part 50, Appendix H, requires that peak neutron fluence at the end of the design life of the vessel will not exceed 10^{17} n/cm^2 (E >1MeV), or that reactor vessel beltline materials be monitored by a surveillance program to meet the American Society for Testing and Materials (ASTM) E 185 Standard.

The existing reactor vessel material surveillance program provides sufficient material data and dosimetry to monitor irradiation embrittlement at the end of the period of extended operation, and to determine the need for operating restrictions on the inlet temperature, neutron spectrum, and neutron flux. The program was designed to meet the requirements of 10 CFR 50, Appendix H, and ASTM E-185-73. Capsules withdrawn after July 26, 1983, will be tested and the results reported in accordance with the 1982 revision of ASTM E-185 as required by 10 CFR 50, Appendix H. The program consists of six surveillance capsules (V, R, T, P, S, and N) positioned in the reactor vessel between the thermal shield and the reactor vessel wall. Capsule V was removed and tested in 1971, capsule R in 1974, capsule T in 1980, and capsule S in 1993. Capsule P is scheduled to be withdrawn at an estimated inside surface 52 effective-full-power-year fluence. Capsule N is a standby capsule and is scheduled to be withdrawn at one to two times the inside surface end-of-life fluence and stored (without testing).

Since it has been projected that the upper-shelf Charpy energy levels of the beltline weld materials may be less than 50 ft-lb at 52 EFPY of service, a low upper-shelf fracture mechanics evaluation has been performed to satisfy the requirements of Appendix G to 10 CFR Part 50. An additional capsule will be withdrawn at a neutron fluence equivalent to approximately 52 EFPY of exposure. This capsule will be stored to be made available should testing of the material specimens be required.

B2.1.28.1 Scope of Program

The program is focused on determining reactor pressure vessel loss of fracture toughness due to neutron irradiation embrittlement.

Time-Limited Aging Analysis topics related to this program include P-T Limit Curves, Upper Shelf Energy, Pressurized Thermal Shock, and LTOP setpoints. However, this program specifically addresses only Upper Shelf Energy. Other analyses are not included in the scope of this program and are addressed separately.

B2.1.28.2 Preventative Actions

This is a condition-monitoring program and therefore no preventative actions are taken.

B2.1.28.3 Parameters Monitored/Inspected

This program monitors accumulated neutron fluence from irradiated material specimens, and measures material fracture toughness and tensile strength.

B2.1.28.4 Detection of Aging Effects

Reduction in fracture toughness is the aging effect for the reactor vessel. Fracture toughness is measured against acceptance criteria to determine the extent of aging.

B2.1.28.5 Monitoring and Trending

Reactor vessel capsule neutron fluence, and Charpy V-notch upper shelf energy are monitored. This data, along with reactor vessel materials information are used as input to the analyses.

B2.1.28.6 Acceptance Criteria

Appendix G of 10 CFR Part 50 provides acceptance criteria for fracture toughness. In accordance with section IV.A.1.a of the appendix, Ginna Station has demonstrated that lower values of Charpy Upper-Shelf Energy will provide margins of safety against fracture equivalent to those required in Appendix G of Section XI of the ASME Code.

Accumulated neutron fluence for the capsule bounds the expected fluence for the beltline weld region through the period of extended operation

B2.1.28.7 Corrective Actions

Corrective actions are implemented at Ginna Station in accordance with the requirements of 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants” as described in the “Quality Assurance Program for Station Operation” (ND-QAP). Corrective actions are taken to prevent recurrence when the specified limits for fuel oil standards are exceeded or when water is drained during periodic surveillance. Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

B2.1.28.8 Confirmation Process

Confirmation of the effectiveness of the Reactor Pressure Vessel Surveillance Program is accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

B2.1.28.9 Administrative Controls

Ginna Station QA procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B and will continue to be adequate for the period of extended operation.

B2.1.28.10 Operating Experience

Analyses that are based on the program results have been revised over time. These analyses include P-T limits, Upper Shelf Energy, Pressurized Thermal Shock, and LTOP.

The reactor vessel surveillance data for the last two capsules demonstrates that reduction in upper shelf toughness is significantly less than that predicted by regulatory guide 1.99. In addition, the last surveillance capsule tested at an estimated 3.87×10^{19} n/cm2 showed no additional reduction in upper shelf energy than the capsule tests at an estimated 1.97×10^{19} n/cm2. The reactor vessel surveillance data tend to indicate that the actual reduction in upper shelf energy has not occurred at the predicted rate. This observation provides additional assurance that the reactor vessel will continue to perform its intended function throughout the period of extended operation.

Conclusion

The Ginna Station Reactor Vessel Surveillance Program is consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Sections XI.M31, “Reactor Vessel Surveillance”. The continued implementation of the Reactor Vessel Surveillance Program provides reasonable assurance that aging effects will be managed such that the intended function of the reactor pressure vessel will be maintained during the license renewal period.

B2.1.29 Selective Leaching of Materials

Program Description

This program ensures the integrity of components made of cast iron, bronze, and brass exposed to raw water (service water), treated water, or ground water at Ginna Station. The program utilizes visual inspections performed under the Periodic Surveillance/Preventive Maintenance Program (Section B2.1.23), or the One-Time Inspection Program (Section B2.1.21), to determine if selective leaching is occurring in susceptible components. The Periodic Surveillance/Preventive Maintenance Program is invoked for those potentially susceptible components which currently have a routine preventive maintenance activity. For potentially susceptible components which do not have a routine preventive maintenance activity, a one-time inspection will be performed.

Conclusion

The programs used to implement the selective leaching program at Ginna Station are similar to the program described in the GALL, with the following exception:

Hardness tests are not typically performed, although an assessment of the feasibility of performing hardness tests and the value of hardness data is made on a component specific basis.

B2.1.30 Spent Fuel Pool Neutron Absorber Monitoring

Program Description

The purpose of this program is to monitor the long-term performance of the borated stainless steel neutron absorber material used in the Ginna Station spent fuel pool.

Ginna Station also incorporates boraflex panels in the spent fuel pool. However, reliance on the neutron absorption capability of the boraflex panels was discontinued when the NRC approved License Amendment 79 on December 7, 2000. That amendment provided for reliance on soluble boron instead of the boraflex (credit for the borated stainless steel is still required).

B2.1.30.1 Scope of Program

The program monitors long term performance of the borated stainless steel (BSS) panels, using surveillance coupons comprised of the same material. The Ginna Station spent fuel pool currently has 828 locations employing (non-credited) boraflex panels, and 493 locations employing borated stainless steel.

B2.1.30.2 Preventive Actions

The Spent Fuel Pool Neutron Absorber Monitoring Program is a monitoring program only and specifies no preventive actions.

B2.1.30.3 Parameters Monitored/Inspected

The Spent Fuel Pool Neutron Absorber Monitoring Program ensures surveillance coupons are removed and evaluated as follows:

- Visual comparisons are made after the test coupons are cleaned and dried.
- Thickness measurements are taken at locations chosen to be representative of creviced/galvanically coupled areas and exposed surfaces.
- Weight measurements are taken of the test coupons using a balance capable of measuring 0.1 gram.

B2.1.30.4 Detection of Aging Effects

The BSS surveillance coupons in the spent fuel pool are periodically examined. The examinations consist of visual comparisons, thickness measurements, and weight measurements relative to reference samples that have not been exposed to the spent fuel pool environment.

B2.1.30.5 Monitoring and Trending

Parameters evaluated are recorded as directed in site specific procedures.

A schedule milestone for performing these evaluations has been determined:

- the completion of the first operating cycle following installation of the racks (milestone reached)

- the completion of every third additional operational cycle (Cycles 31, 34, 37, etc.).

B2.1.30.6 Acceptance Criteria

The comparisons are made by qualified personnel in the Ginna Station Reactor Engineering and Laboratory and Inspection Services/Chemistry sections. The values determined in the evaluation are compared to the reference values.

B2.1.30.7 Corrective Actions

Corrective actions are implemented at Ginna Station in accordance with the requirements of 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants” as described in the “Quality Assurance Program for Station Operation” (ND-QAP). Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

Any significant changes or results are to be documented in an ACTION Report per site-specific procedures and further investigated using appropriate analytical techniques. These results should also be reported to ATEA, the firm that designed, built, and installed the racks at Ginna Station

B2.1.30.8 Confirmation Process

Confirmation of the effectiveness of the Spent Fuel Pool Neutron Absorber Monitoring Program is accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

B2.1.30.9 Administrative Controls

Ginna Station QA procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B and will continue to be adequate for the period of extended operation.

B2.1.30.10 Operating Experience

A review of industry operating experience has revealed no evidence of significant age-related degradation of borated stainless steel material exposed to spent fuel pool environments.

The first examination of a coupon has been completed and the results documented in station correspondence. No evidence of degradation was found. The visual appearance of the coupons was excellent. The visual appearance and other measurements indicate that the borated stainless steel absorber panels exhibit good corrosion resistance in the spent fuel pool environment and will perform as expected over the remaining life of the racks.

Conclusion

The Spent Fuel Pool Neutron Absorber Monitoring Program is consistent with all the NUREG-1801 attributes, and is consistent with the processes required for the Boraflex Monitoring Program as described in NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section XI.M22, "Boraflex Monitoring" (Reference 3).

B2.1.31 Steam Generator Tube Integrity

Program Description

The Ginna Station Steam Generator Tube Integrity Program is a comprehensive program that incorporates the guidance of NEI 97-06 and EPRI TR-107569 and is credited for maintaining the integrity of the steam generator (SG) tubes. The program manages aging effects such as cracking due to primary water stress corrosion cracking (PWSCC), outside diameter stress corrosion cracking (ODSCC), intergranular attack (IGA), pitting, wastage, wear, fouling due to corrosion product buildup, mechanical degradation due to denting and impingement damage, and fatigue.

The program manages these aging effects/mechanisms through a balance of prevention, inspection, examination, assessment, evaluation, repair and leakage monitoring measures. The program is administered through a series of plant directives and interface procedures, as well as the plant technical specifications. Key program attributes include non-destructive examination (NDE), sludge lancing, primary and secondary water chemistry control, and primary-to-secondary leakage trending and monitoring.

B2.1.31.1 Operating Experience

Past industry and Ginna operating experience has led to sweeping changes in the programs required to maintain SG tube integrity. The NRC, EPRI and NEI have generated guidance to be used by the industry to provide effective controls. NEI has generated NEI 97-06 to incorporate lessons learned from plant operating experience and past SG tube inspection experience. Ginna Station has committed to provide a SG tube integrity program that meets these guidelines.

Ginna Station experienced many of the industry problems related to SG tube integrity in the original Westinghouse Series 44 SGs. These SGs had 28 years of operating experience. With initial startup chemistry being phosphate buffer control, these SGs experienced many of the same problems of other PWR SGs. In late 1974, the chemistry was changed over to AVT with the addition of full flow condensate demineralizers. In January, 1982, a tube rupture event occurred in the “B” SG at Ginna Station. The root cause was determined to be tube thinning due to wear caused by foreign material in the secondary side. The rate of SG tube degradation over the years, with subsequent loss of heat transfer surface due to plugging and sleeving, led to the decision to replace the SGs.

In June, 1996, the Ginna SGs were replaced with a new design, supplied by Babcock & Wilcox (B&W) International. The replacement SGs incorporate design and manufacturing improvements to reduce and/or prevent many of the problems that the industry has experienced. In addition, during the SG replacement, a modification was completed to reduce T_{AVG} in the RCS during normal operations with the intent to further prolong the life of the SG tube materials. To date, Ginna has completed SG Tube examinations and sludge lancing and secondary side foreign material/loose parts inspection in accordance with the program requirements during refuelings of 1997, 1999, and 2002 with no degradation observed. All tubes remain in service with the exception of one tube in each SG that were plugged during fabrication due to manufacturing defects. Industry experience associated with steam generators of similar design has identified tube wear due to fretting at AVBs. No evidence of wear damage has been detected in the Ginna replacement steam generators to date.

Conclusion

The Ginna Station Steam Generator Tube Integrity Program is consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section XI.M19, “Steam Generator Tube Integrity” (Reference 3). The program provides reasonable assurance that aging effects will be managed such that the intended function of the SG tubes will be maintained during the license renewal period.

B2.1.32 Structures Monitoring Program

Program Description

The Ginna Station Structures Monitoring Program is described in an Engineering Procedure. It is a comprehensive program that was developed and implemented to meet the regulatory requirements of the maintenance rule (10 CFR 50.65, USNRC Regulatory Guide 1.160, and NUMARC 93-01). The program includes masonry walls evaluated in accordance with NRC IEB 80-11, “Masonry Wall Design” and incorporates guidance in NRC IN 87-67, “Lessons learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11” and NRC Regulatory Guide 1.127, “Inspection of Water-Control Structures Associated with Nuclear Power Plants.” The program identifies the structures and structural components within the scope of the maintenance rule and license renewal, the performance criteria that are to be monitored, the frequency of inspections, and provides the controls to ensure that there is no loss of structure or structural component intended function.

B2.1.32.1 Operating Experience

Although the Ginna Station Structures Monitoring Program requirements have been developed and documented since 1995, plant inspection and maintenance of specific structures within the program has been on-going since initial operation. Structures such as buildings, supports, intakes, canals, etc., including roofs, block/masonry walls, liners, steel, etc. have been maintained periodically to ensure their intended function and have been upgraded consistent with regulatory requirements and industry experience.

The Ginna Station Structures Monitoring Program provides the controls necessary to ensure that any degradation can be detected and resolved prior to any loss of intended function for the period of extended operation.

Conclusion

The Ginna Station Structures Monitoring Program will be consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Sections XI.S5,” Masonry Wall Program,” XI.S6, “Structures Monitoring Program,” and XI.S7, “RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants” (Reference 3). Enhancements will be made to include additional structural components consistent with the scope described above. The program will provide reasonable assurance that the aging effects from external and internal environments on structures and structural components will be managed such that their intended function will be maintained during the license renewal period.

B2.1.33 Systems Monitoring

Program Description

The Ginna Station Systems Monitoring Program is a comprehensive program which addresses aging management requirements for piping, components and equipment in systems which are within the scope of license renewal. As part of the implementation of 10 CFR 50.65 (Maintenance Rule), specific guidelines for assessing the material condition of systems, structures, and components during System Engineer walkdowns were developed. The Systems Monitoring Program is credited for managing aging effects such as loss of material, cracking, and fouling buildup for normally accessible, external surfaces of piping, tanks, and other components and equipment within the scope of License Renewal. These aging effects are managed through visual inspection and monitoring of external surfaces for leakage and evidence of material degradation, such as corrosion, cracking, degradation of coatings, sealants and caulking, deformation, and debris and corrosion product buildup.

B2.1.33.1 Scope of Program

The Systems Monitoring Program provides inspection requirements and guidelines for monitoring the effectiveness of maintenance in accordance with 10 CFR 50.65 (the Maintenance Rule). The scope of the program includes the accessible surfaces including insulated portions of systems, components, and equipment (including welds and bolting) which are designated as maintenance rule systems and within the scope of license renewal. The Program is based on scheduled system walkdowns, health reports, and performance monitoring and trending analysis. The Program is conducted as part of the responsibilities of System Engineers.

B2.1.33.2 Preventive Actions

The Systems Monitoring Program is primarily a condition-monitoring program. However, timely identification of aging effects before significant structural or pressure-boundary degradation occurs may be considered preventive in nature.

B2.1.33.3 Parameters Monitored/Inspected

Surface conditions of system piping and components including visible portions of insulated components, equipment, supports and closure bolting are monitored through periodic visual examinations for evidence of leakage, corrosion, cracking, coating degradation, deformation, change in material properties of flexible connections and sealants, fouling and corrosion product build-up.

B2.1.33.4 Detection of Aging Effects

Visual inspections performed during System Engineer walkdowns provide the primary means for detection and quantification of aging effects and degradation. Additional guidance for identifying and evaluating the evidence of degradation will be included in appropriate plant procedures. Degradation that is deemed “Unacceptable” will be addressed using the Ginna Station Corrective Action process. The Systems Monitoring Program is designed for early detection of age-related degradation prior to system or component failure.

Accessible portions of maintenance rule and license renewal systems are required to be walked down once per quarter. Walkdowns are scheduled and performed so that the entire system is fully inspected within one operating cycle.

B2.1.33.5 Monitoring and Trending

Detailed system/equipment material condition inspections will be performed according to instructions in pertinent administrative procedures. Data from inspections performed during walkdowns is documented, trended and evaluated. The frequency of material condition inspections may be adjusted as necessary based on inspection results and industry experience.

B2.1.33.6 Acceptance Criteria

Administrative procedures will be enhanced to include visual inspection acceptance criteria. Guidance for the assessment of surface corrosion that includes consideration of design margins will be provided in the enhanced procedures. Additional guidance on evaluation of protective coatings will be included. Detection of pressure-boundary leakage requires assessment and appropriate correction.

B2.1.33.7 Corrective Actions

Corrective actions are implemented at Ginna Station in accordance with the requirements of 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants” as described in the “Quality Assurance Program for Station Operation” (ND-QAP). Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

B2.1.33.8 Confirmation Process

Confirmation of the effectiveness of the System Monitoring Program is accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

B2.1.33.9 Administrative Controls

Ginna Station QA procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B and will continue to be adequate for the period of extended operation.

B2.1.33.10 **Operating Experience**

System inspection requirements that have been in effect at Ginna Station since the mid 1990's in support of the Maintenance Rule have proved to be effective in maintaining the material condition of plant systems. A significant number of corrective actions have been processed as a result of System Engineer walkdowns. The Systems Monitoring Program will also be continually re-assessed and upgraded based on industry and plant-specific operating experience reviews.

Conclusion

The Ginna Station Systems Monitoring Program, with the enhancements identified above, has been evaluated using the generic program attributes identified in Appendix A of the SRP. The continued implementation of the Program provides reasonable assurance that aging effects will be managed such that the intended function of systems and components within the scope of license renewal will be maintained during the extended period of operation. In addition to specific enhancements identified in the attributes described above, additional systems/components, consistent with the scope of license renewal, will be included in a future revision of appropriate Ginna Station Procedures.

B2.1.34 **Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)**

Program Description

The potential exists for a loss of fracture toughness due to thermal aging of cast austenitic stainless steel (CASS) components. An evaluation of the susceptibility of CASS components at Ginna Station was made, based on the casting method, molybdenum content, and percent ferrite. It was determined that the CASS RCS elbows were susceptible to loss of fracture toughness due to thermal aging. A plant-specific flaw tolerance evaluation was conducted, and documented in WCAP-15837, “Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the R.E. Ginna Nuclear Power Plant for the License Renewal Program”, April 2002. The evaluation concluded that adequate fracture toughness exists for the RCS loop, including the cast elbows, for the period of extended operation (60 years).

A separate evaluation was made for the reactor coolant pump casings. In WCAP-15873, “A Demonstration of the Applicability of ASME Code Case N-481 to the Primary Loop Pump Casings of R.E. Ginna Nuclear Power Plant for the License Renewal Program”, May 2002, it was concluded that the primary loop pump casings are qualified to item (d) of ASME Code Case N-481 for the period of extended operation (60 years).

B2.1.35 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)

Program Description

The reactor vessel internals receive a visual inspection in accordance with ASME Code Section XI, Subsection IWB, Category B-N-3. This inspection is augmented to detect the effects of loss of fracture toughness due to thermal aging, neutron irradiation embrittlement, and void swelling of CASS reactor vessel internals.

Ginna Station has performed an evaluation to determine potentially susceptible components of the internals. No reactor vessel internal components made of CASS that serve a license renewal intended function have been identified.

Therefore, this program is not applicable to Ginna Station.

B2.1.36 Thimble Tubes Inspection

Program Description

This program manages the integrity of the incore neutron monitoring thimble tubes, which serve as a portion of the reactor coolant pressure boundary. As discussed in NRC Bulletin 88-09, “Thimble Tube Thinning in Westinghouse Reactors,” July 26, 1988, thimble tube wall-thinning can occur as a result of flow-induced vibration. This wear damage is detected at locations associated with geometric discontinuities or area changes along the reactor coolant flow path, such as areas near the lower core plate, the core support forging, the lower tie plate, and the vessel penetrations.

Periodic assessment of thimble tube wear, and corrective actions as needed, form the basis for this program.

B2.1.36.1 Scope of Program

All thirty-six thimble tubes are within the scope of this inspection program.

B2.1.36.2 Preventive Actions

As noted in operating experience below, the replacement of tube G-6 with chrome plating at the wear area constitutes a preventive action. In addition flushing of the tubes during refueling outages is also considered preventive in nature.

Eddy current examinations are performed on a periodicity consistent with the severity of wear damage for each thimble tube. When wall loss in a tube exceeds 55%, but less than 65%, the tube is repositioned such that wear is redistributed, or other corrective action is taken.

The eddy current examinations themselves are inspection/verification activities, and are thus not considered preventive.

B2.1.36.3 Parameters Monitored/Inspected

The eddy current examinations determine the wall thickness of the thimble tubes, allowing an assessment of the wear, and wear rate, of each tube in each location.

B2.1.36.4 Detection of Aging Effects

Thimble tube inspections are conducted using a methodology specified in a Ginna Station plant-specific procedure. This procedure requires the use of a Zetec MIZ-18 Multifrequency Eddy Current Testing System. These inspections provide indication of tube wear, and tube wear rate.

B2.1.36.5 Monitoring and Trending

Based on the results of a plant-specific analysis, examination results are compared to an upper allowable limit of 65% through-wall wear.

Eddy current examinations performed in 1988, 1989, 1990, 1991, and 1992 provided a basis for establishing the wear rates, and thus the inspection intervals, for thimble tubes. Based on those results, the inspection frequency and acceptance criteria are:

- previous indication 10% to less than 45% - every third refueling outage (approximately once every 4.2 years)
- previous indication 45% to less than 55% - every other refueling outage (approximately once every three years)
- previous indication 55% or greater - perform corrective action, if support plate wear is the suspected cause. For other indications, corrective action

will be taken at 65% or greater. Future inspection frequency will be every other or every third outage, as stated above.

- previous inspection never exceeded 10% through-wall - no specified frequency. Future inspections will be based on a Ginna Station periodic assessment.

B2.1.36.6 Acceptance Criteria

The acceptance criteria is provided in (Section B2.1.36.5).

B2.1.36.7 Corrective Actions

Corrective actions are implemented at Ginna Station in accordance with the requirements of 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants” as described in the “Quality Assurance Program for Station Operation” (ND-QAP). Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

B2.1.36.8 Confirmation Process

Confirmation of the effectiveness of the Thimble Tube Inspection Program is accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

B2.1.36.9 Administrative Controls

Ginna Station QA procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B and will continue to be adequate for the period of extended operation.

B2.1.36.10 Operating Experience

Thimble tube wear in Westinghouse reactors was documented in NRC IN 87-44, “Thimble Tube Thinning in Westinghouse Reactors,” and NRC Bulletin 88-09. In response to these notifications, eddy current examination of thimble tubes was performed annually from 1988 to 1992 at Ginna Station.

In 1990, thimble tube G-6 had indication of wear greater than 55%. Corrective action was taken by repositioning (moving worn areas away from the lower support plate by 1- 2") the tube.

Three other thimble tubes had indications noted in the 1997 examination that resulted in the need for corrective action (ACTION Report 97-1889). All four thimble tubes were replaced during the 1999 refueling outage. One thimble had an indication of intergranular attack. The conduit water was sampled, and analysis showed the presence of chlorides, fluorides, and sulfates in concentrations significantly above RCS water. These conduits were flushed during the thimble tube replacement. All other thimble tube conduits were flushed during the 2000 refueling outage.

During the 2000 refueling outage, inspection of tube G-6 again indicated degradation due to flow-induced vibration. This tube was replaced with a chrome-plated tube during the 2002 refueling outage.

Conclusion

Although the Ginna Station Thimble Tube Inspection Program is a plant-specific program consistent with our commitments to Bulletin 88-09, the attributes associated with the ASME Section XI, Subsection IWB, IWC, IWD Inservice Inspection Program (Section B2.1.2) are met.

B2.1.37 Water Chemistry Control

Program Description

The Water Chemistry Control Program mitigates damage caused by aging effects by controlling the internal environment of components in the primary, borated, and secondary water systems. Aging effects managed by this program include loss of material due to general, pitting and crevice corrosion, microbiologically influenced corrosion (MIC), stress corrosion cracking (SCC), and fouling due to corrosion product buildup. The program relies on monitoring and control of water chemistry based on the EPRI guidelines in TR-105714 (Reference 7) for primary systems chemistry and TR-102134 (Reference 8) for secondary systems chemistry.

For low-flow or stagnant portions of a system, a one-time inspection of selected components at susceptible locations provides verification of the effectiveness of the Water Chemistry Control Program. This inspection is covered within the scope of the One-time Inspection Program (Section B2.1.21). No verification inspections are required for intermediate and high flow regions.

The aging effects are managed by controlling concentrations of known detrimental chemical species such as chlorides, fluorides, sulfates and dissolved oxygen below the levels known to cause degradation. The program includes specifications for chemical species, sampling and analysis frequencies, and corrective actions for control of water chemistry.

B2.1.37.1 Operating Experience

The EPRI guideline documents have been developed based on industry experience and have been shown to be effective over time with their widespread use. Industry operating experience related to specific issues have been incorporated into subsequent revisions of these guidelines upon which the plant specific program is based.

A review of plant specific operating experience indicates that Ginna Station has experienced a single Level 3 excursion, which was in the Reactor Coolant System oxygen concentration after the Cycle 17 refueling startup. No other Level 3 excursions were found. An independent assessment of the primary and secondary chemistry programs at Ginna Station is routinely performed to confirm that the program is maintained within plant specifications and industry guidelines. Recommendations from these assessments have been used to improve plant chemistry and overall plant operations.

In 1996, Ginna station replaced the original Westinghouse model 44 steam generators with BWI replacement steam generators incorporating design enhancements and Inconel 690 TT tubing. The new design is less susceptible to many of the aging effects managed by this program. In addition, installation of a new reactor vessel closure head incorporating design enhancements and Inconel 690 TT penetrations is planned for the 2003 refueling outage.

Conclusion

The Ginna Station Water Chemistry Control program is consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Sections XI.M2, “Water Chemistry Control” (Reference 3), however, plant procedures and guidelines for water chemistry control are based on Revision 5 of EPRI TR-102134 (Reference 8) and Revision 4 of TR-105714 (Reference 7).

The existing water chemistry control program has demonstrated that the aging effects associated with applicable components have been adequately managed in the current operating term. Plant specific experience has been used to enhance the program over time, consistent with the requirements of the corrective action program. Further enhancements as described in this program will provide additional assurance that these components will perform their intended functions in accordance with the current licensing basis during the period of extended operation.

B3.0 TIME-LIMITED AGING ANALYSES SUPPORT ACTIVITIES

B3.1 Environmental Qualification Program

Program Description

The Nuclear Regulatory Commission (NRC) has established Environmental Qualification of Electrical Equipment requirements in 10 CFR Part 50, Appendix A, Criterion 4, and 10 CFR 50.49. 10 CFR 50.49 specifically requires that an EQ program be established to demonstrate that certain electrical components located in harsh plant environments (that is, those areas of the plant that could be subject to the harsh environmental effects of a loss of coolant accident [LOCA], high energy line breaks [HELBs] or post-LOCA radiation) are qualified to perform their safety function in those harsh environments after the effects of inservice aging. 10 CFR 50.49 requires that the effects of significant aging mechanisms be addressed as part of environmental qualification.

The Environmental Qualification (EQ) program manages component thermal, radiation, and cyclical aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods, for those components within the scope of the rule. As required by 10 CFR 50.49, EQ components not qualified for the current license term are to be refurbished, replaced, or have their qualification extended prior to reaching the age limits established in the evaluation.

Reanalysis of an aging evaluation to extend the qualification of components under 10 CFR 50.49(e) may be performed as part of an EQ program. Important attributes for the reanalysis of an aging evaluation include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met). These attributes are discussed in NUREG-1800 and section 4.4 of the License Renewal Application.

B3.1.1 Scope of Program

The Ginna Station EQ program applies to certain electrical components that are important to safety and could be exposed to harsh environment accident conditions, as defined in 10 CFR 50.49.

B3.1.2 Preventive Actions

10 CFR 50.49 does not require actions that prevent aging effects. Ginna Station EQ program actions that could be viewed as preventive actions include (a) establishing service conditions for components to extend equipment qualified life and (b) requiring specific installation, inspection, monitoring or periodic maintenance actions to maintain component aging effects within the bounds of the qualification basis.

B3.1.3 Parameters Monitored/Inspected

EQ component qualified life is based on testing and analysis not on condition or performance monitoring. However, pursuant to Regulatory Guide 1.89, Rev. 1, such monitoring programs are an acceptable basis to modify a qualified life through reanalysis. Monitoring or inspection of certain environmental conditions or component parameters may be used to ensure that the component is within the bounds of its qualification basis, or as a means to modify the qualified life.

B3.1.4 Detection of Aging Effects

10 CFR 50.49 does not require the detection of aging effects for in-service components. Monitoring of environmental conditions such as temperature in the immediate vicinity of EQ equipment is used to ensure that the component is within the bounds of its qualification basis, or as a means to modify the qualified life.

B3.1.5 Monitoring and Trending

10 CFR 50.49 does not require monitoring and trending of component condition or performance parameters to manage the effects of aging. However, Ginna Station EQ program actions do include monitoring how long qualified components have been installed, how often they are operated/energized, and what discrete environmental conditions such as temperature and radiation exist. Such monitoring is used to ensure

that select components are within the bounds of their qualification bases, or to modify the qualified life.

B3.1.6 Acceptance Criteria

10 CFR 50.49 acceptance criteria are that an inservice EQ component is maintained within the bounds of its qualification basis, including (a) its established qualified life and (b) continued qualification for the projected accident conditions. 10 CFR 50.49 requires refurbishment, replacement, or requalification prior to exceeding the qualified life of each installed device. Where monitoring has been used to modify a component qualified life, Ginna-specific acceptance criteria for operating in those conditions were established.

B3.1.7 Corrective Actions

If an EQ component is found to be outside the bounds of its qualification basis, corrective actions are implemented in accordance with the requirements of 10 CFR 50, App. B "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants", as described in the Ginna "Quality Assurance Program for Station Operation" (ND-QAP). When unexpected adverse conditions are identified during operational or maintenance activities that affect the environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions. When an emerging industry aging issue is identified that affects the qualification of an EQ component, the affected component is evaluated and appropriate corrective actions are taken, which may also include changes to the qualification bases and conclusions.

B3.1.8 Confirmation Process

Confirmation of the effectiveness of the Environmental Qualification Program is accomplished in accordance with the Ginna Station Corrective Action Program, the site QA procedures, review and approval processes and administrative controls, which are implemented in accordance with the requirements of 10 CFR 50, App. B.

B3.1.9 Administrative Controls

The EQ programs is implemented through the use of station policy, directives, and procedures. The EQ program will continue to comply with 10 CFR 50.49 throughout the renewal period, including development and maintenance of qualification documentation demonstrating reasonable assurance that a component can perform required functions during harsh accident conditions. EQ program documents identify the applicable environmental conditions for the component locations. EQ program qualification files are maintained at the plant site in an auditable form for the duration of the installed life of the component. EQ program documentation is controlled under the station's quality assurance program.

B3.1.10 Operating Experience

As a result of 10 CFR 50.49, Ginna Station installed extensive new environmentally qualified electrical equipment, in accordance with the “DOR Guidelines”. The type of equipment replaced or installed included transmitters, level switches, electrical cable, solenoid valves, connectors, splices, and LVDTs. Upgrades were also made to electrical motors, valve activators, and electrical penetrations.

Ginna maintains cognizance of emerging issues in EQ and aging management by actively participating in industry forums, including the Nuclear Utility Group on Equipment Qualification.

Conclusion

The Ginna Station Environmental Qualification Program provides reasonable assurance that compliance with 10 CFR 50.49 is maintained. EQ components are designed and maintained to perform their intended functions in a postulated post-accident, environment, following the effects of inservice aging for the period of extended operation.

B3.2 Fatigue Monitoring

Program Description

The Fatigue Monitoring Program is a newly incorporated program that is consistent with the NRC Generic Aging Lessons Learned (GALL) Report, Section X.M1, “Metal Fatigue of Reactor Coolant Pressure Boundary”. The Fatigue Monitoring Program is a confirmatory program that monitors loading cycles due to thermal and pressure transients for selected critical components. The program provides an analytical basis for confirming that the number of cycles established by the analysis of record will not be exceeded before the end of the period of extended operation. The effects of reactor coolant environment are considered through the evaluation of the seven component locations identified in NUREG/CR-6260 using the appropriate environmental fatigue factors. Formulae for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon or low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels, or other environmental fatigue factors appropriate to the material.

B3.2.1 Scope of Program

The scope of the Fatigue Monitoring Program includes those plant systems and components for which a cyclic or fatigue design basis exists. The specific systems and components included within the scope of the Fatigue Monitoring Program are identified below:

Reactor Pressure Vessel Closure Studs

Reactor Pressure Vessel Primary (Inlet and Outlet) Nozzles

Reactor Pressure Vessel at Core Support Pad

Steam Generator Tubesheet

Cold Leg (Accumulator) Safety Injection Nozzle

Pressurizer Upper Shell

Pressurizer Spray Nozzle

Pressurizer Surge Line Nozzle

Hot Leg Surge Line Nozzle

Pressurizer Surge Line

Pressurizer Heater Well Penetration

Reactor Coolant Piping Charging System Nozzles

Residual Heat Removal Hot Leg Suction Nozzles

Residual Heat Removal System Class 1 Piping

B3.2.2 **Preventive Actions**

The Fatigue Monitoring Program includes preventive measures to mitigate fatigue cracking of metal components of the reactor coolant pressure boundary, and will provide adequate margin against fatigue cracking of these components due to anticipated cyclic strains.

Tracking of operating transient cycles and maintaining the fatigue usage factor below the design code limit of 1.0, including the effects of reactor water environment, will provide adequate margin against fatigue cracking of reactor coolant system components due to anticipated cyclic strains.

B3.2.3 **Parameters Monitored, Inspected, and/or Tested**

The Fatigue Monitoring Program monitors plant transients that cause cyclic strains and are significant contributors to fatigue damage or crack growth. The Fatigue Monitoring Program consists of automated cycle counting to count the number of plant transients that cause significant fatigue damage. Fatigue usage factors are tracked for bounding component locations of the reactor coolant pressure boundary.

B3.2.4 Detection of Aging Effects

The Fatigue Monitoring Program provides for periodic updates of the plant cycle count and fatigue usage calculations. The metal fatigue aging effect will be monitored using FatiguePro^{TM}, which is an EPRI software product for plant transient monitoring and fatigue usage and fatigue crack growth calculations. Plant operating cycles will be tracked against design limits. Fatigue usage factors will be computed on an on-going basis for bounding components using plant instrument data.

B3.2.5 Monitoring and Trending

The Fatigue Monitoring Program includes monitoring the number and severity of plant design transients and an on-going fatigue analysis of a sampling of component locations whose level of metal fatigue is expected to be most adversely affected by the combined effects of plant cycles and reactor water environment. The monitored population includes each of the component locations identified in NUREG/CR-6260 for older vintage Westinghouse plants, as well as others listed in Section B3.2.1 above.

The Fatigue Monitoring Program will ensure that the extent of the fatigue aging effect is quantifiable on an on-going basis and that alternative or mitigative actions can be taken before there is a loss of any component’s intended function. The program monitors operating transients to-date, calculates fatigue usage factors to-date, and allows corrective measures to be implemented ahead of time to ensure that structural margins required by the Codes used in the original plant design are maintained throughout the operating life of the plant. The annual recording and assessment frequency ensures that normal operating transients that might occur during the plant operational period will not compromise these limits. The evaluation locations have been chosen to ensure that locations that might approach acceptance limits will be monitored. The program also includes provisions to identify deviations from expected usage factor accumulation so that appropriate corrective actions can be taken before structural margins are degraded to unacceptable levels.

B3.2.6 Acceptance Criteria

The acceptance criterion consists of maintaining the fatigue usage less than or equal to the design code allowable limit of 1.0, considering environmental fatigue effects. The acceptance criteria will ensure that original structural margins considered in the plant design are maintained throughout the operating period.

B3.2.7 Corrective Actions

Corrective actions are implemented in accordance with the requirements of 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants”, and ANSI N18.7-1976, “Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants”, as committed in the Ginna Station UFSAR. Provisions for timely evaluation of adverse conditions and implementation of any corrective actions required, including root cause analyses and prevention of recurrence where appropriate, are included in the corrective action program.

Corrective actions are implemented through the initiation of an ACTION Report in accordance with site-specific procedures.

The Fatigue Monitoring Program provides for corrective actions to prevent the fatigue usage factor from exceeding the design code limit of 1.0 during the period of extended operation. The Fatigue Monitoring Program utilizes FatiguePro™ both to perform an analysis of each monitored component location using actual plant data and to provide the basis for proactive action to maintain the fatigue usage factors below Code limits. Corrective actions include a review of additional affected component locations.

For component locations for which it can not be demonstrated that the fatigue usage factor remains below the design code limit of 1.0 during the period of extended operation, corrective actions can include a more rigorous analysis of the component to demonstrate that the design code limit will not be exceeded, repair or replacement of the component, or managing the effects of fatigue by an inspection program that has been reviewed and approved by the NRC (e.g. periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC).

B3.2.8 Confirmation Process

Confirmation of the effectiveness of the Fatigue Monitoring Program is accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

B3.2.9 Administrative Controls

Ginna Station QA procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B and will continue to be adequate for the period of extended operation.

B3.2.10 Operating Experience

The Fatigue Monitoring Program includes reviews of both industry and plant-specific operating experience regarding fatigue cracking for applicability to Ginna. These on-going reviews will be considered when selecting additional monitored components. As needed, additional inspections, or other analytical programs, will be used to ensure that unacceptable fatigue cracking does not occur, due to both anticipated and unanticipated transients.

Several industry issues have arisen which have provided knowledge about the propensity for fatigue cracking in Class 1 and 2 components. Starting in 1979, several PWRs experienced main feedwater piping and nozzle fatigue cracking resulting from thermal stratification cycling during conditions of low flow and hot standby. NRC Bulletin 79-13 dealt with fatigue cracking of steam generator feedwater nozzles (Class 2 component). As reported in 1980, through-wall cracking in the area of the auxiliary feedwater connection to the main feedwater piping occurred at Ginna Station. The J-tube design was implemented in the early 1980's. The currently installed replacement steam generators for Ginna include main feedwater nozzle features that minimize thermal stratification effects. Thus, the main feedwater nozzles are not included in the monitored locations.

NRC Bulletin 88-08 addressed the potential for fatigue cracking in normally stagnant piping systems attached to the reactor coolant system. A number of cold leg safety injection pipe cracking incidents (Farley-Unit 2 in 1987 and Sequoyah Unit 2 in 1996) and thermal sleeve cracking incidents (Trojan and McGuire Unit 1) have occurred at Westinghouse plants. This piping and nozzles were associated with the 10-inch accumulator line connection to the cold legs. The Fatigue Monitoring Program includes the nearby cold leg safety injection nozzles as monitored component locations. These locations are more fatigue-sensitive than the location evaluated in NUREG/CR-6260 and are considered to bound the NUREG/CR-6260 location. In addition, NDE of the most sensitive locations at Ginna Station has demonstrated that damage would not be expected due to inleakage-induced thermal transients.

Conclusion

The Ginna Station Fatigue Monitoring Program is consistent with GALL Section X.M1 “Metal Fatigue of Reactor Coolant Pressure Boundary” aging management program. However, corrective actions at Ginna Station may include management of the effects of fatigue by an inspection program (e.g. periodic non-destructive examination of the affected locations at defined inspection intervals) should an appropriate inspection technique and interval be developed and subsequently approved by the NRC.

B3.3 Concrete Containment Tendon Pre-stress

Program Description

In order to ensure the adequacy of prestressing forces in prestressed concrete containments during the extended period of operation, a Time-Limited Aging Analysis was performed. The results of this analysis indicated that continued monitoring and potential retensioning of the containment tendons may be necessary to ensure that the prestressing forces remain above the minimum required value for all tendons.

The aging management program (AMP) consists of an assessment of the results of inspections performed in accordance with the requirements of Subsection IWL of the ASME Section XI Code (Reference 19), as supplemented by the requirements of 10 CFR 50.55a(b)(2)(ix) or (viii) in the later amendment of the regulation. The assessment related to the adequacy of the prestressing force will consist of the establishment of (1) acceptance criteria and (2) trend lines. The acceptance criteria are developed consistent with the methodology of NRC Regulatory Guide 1.35.1 (Reference 21), and will normally consist of predicted lower limit (PLL) and the minimum required prestressing force, also called minimum required value (MRV). NRC Information Notice IN 99-10 (Reference 20) provides guidance for constructing the trend line. The goal is to keep the trend line above the PLL because, as a result of any inspection performed in accordance with ASME Section XI, Subsection IWL, if the trend line crosses the PLL, the existing prestress in the containment could go below the MRV soon after the inspection and would not meet the requirements of 10 CFR 50.55a(b)(2)(ix)(B) or 10 CFR 50.55a(b)(2)(viii)(B).

B3.3.1 **Scope of Program:**

The program addresses the assessment of containment prestressing force.

B3.3.2 **Preventive Actions*:***

Maintaining the prestress above the MRV, as described under program description above, will ensure that the structural and functional adequacy of the containment are maintained.

B3.3.3 **Parameters Monitored/Inspected:**

The parameters to be monitored are the containment prestressing forces in accordance with requirements specified in Subsection IWL of Section XI of the ASME Code, as incorporated by reference in 10 CFR 50.55a.

B3.3.4 **Detection of Aging Effects:**

This program detects the loss of containment prestressing forces.

B3.3.5 Monitoring and Trending:

The estimated and measured prestressing forces are plotted against time and the PLL, MRV, and trending lines developed for the period of extended operation.

B3.3.6 Acceptance Criteria:

The prestressing force trend lines indicate that existing prestressing forces in the containment would not be below the MRVs prior to the next scheduled inspection, as required by 10 CFR 50.55a(b)(2)(ix)(B) or 10 CFR 50.55a(b)(2)(viii)(B).

B3.3.7 Corrective Actions:

Corrective actions are implemented at Ginna Station in accordance with the requirements of 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants” as described in the “Quality Assurance Program for Station Operation” (ND-QAP). Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

B3.3.8 Conformation Process:

Confirmation of the effectiveness of the Concrete Containment Tendon Prestress Program is accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

B3.3.9 Administrative Controls:

Ginna Station QA procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B and will continue to be adequate for the period of extended operation.

B3.3.10 Operating Experience:

Ginna station retensioned 23 of the 160 vertical tendons 1000 hours after initial prestressing. Subsequent tests identified that tendon lift-off forces were generally lower than the predicted values. An investigation was started to determine the reason for the accelerated loss of lift-off forces. Prior to completing the investigation, Ginna retensioned the 137 tendons that were not originally retensioned. The investigation concluded that stress relaxation of the tendon wires was the only significant cause for the lower-than-predicted tendon forces. To quantify these findings, RG&E initiated a tendon stress relaxation test program that was conducted at the Fritz Engineering Laboratory of Lehigh University.

The Time-Limited Aging Analysis for the Evaluation of Loss of Prestress in Containment Tendons concluded that the initial retensioned set of 23 tendons should be retensioned prior to the end of the current licensing period to ensure that prestressing forces remain above the minimum required value in the period of extended operation.

Conclusion:

The Ginna Station Concrete Containment Tendon Prestress Program is consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section X.S1. The program provides reasonable assurance that the aging effect of loss of containment prestressing forces will be managed such that the tendon intended functions will be maintained during the period of extended operation.

Appendix B References

1. NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*, U.S. Nuclear Regulatory Commission, July 2001.

2. 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants”

3. NUREG-1801, *Generic Aging Lessons Learned (GALL) Report,* U.S. Nuclear Regulatory Commission, July 2001.

4. NUREG/CR-6260 (INEL-95/0045), “Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components,” U.S. Nuclear Regulatory Commission, March 1995.

5. 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants”

6. ANSI N18.7-1976, “Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants”.

7. EPRI TR-105714, “PWR Primary Water Chemistry Guidelines,” Revision 4, Electric Power Research Institute.

8. EPRI TR-102134, “PWR Secondary Water Chemistry Guidelines,” Revision 5, Electric Power Research Institute.

9. ASME Boiler and Pressure Vessel Code, Section XI, Appendix L “Operating Plant Fatigue Assessment,” 1995 Edition.

10. United States Nuclear Regulatory Commission, Bulletin 88-08, “Thermal Stresses in Piping Connected to Reactor Coolant Systems,” 6/22/1988, including Supplements 1, 2, and 3, dated 6/24/88, 8/4/88 and 4/11/89

11. NRC Information Notice 97-019, “Safety Injection System Weld Flaw at Sequoyah Nuclear Power Plant, Unit 2,” 4/18/97

12. NRC Information Notice 82-030, “Loss of Thermal Sleeves in Reactor Coolant System Piping at Certain Westinghouse PWR Power Plants,” 7/26/82

13. EPRI TR-107396, Closed Cooling Water Chemistry Guidelines, Electric Power Research Institute, Palo Alto, CA, November 1997.

14. NUREG/CR-5643, Insights Gained from Aging Research, U. S. Nuclear Regulatory Commission, March 1992.

15. IEEE Std. P1205-2000, IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations.

16. SAND 96-0344, Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable and Terminations, Sandia National Laboratories for the U. S. Department of Energy, September 1996.

17. EPRI TR-109619, Guideline for the Management of Adverse Localized Equipment Environments, Electric Power Research Institute, Palo Alto, CA, June 1999.

18. NEI 95-10, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 -The License Renewal Rule, Rev. 3, Nuclear Energy Institute, March 2001.

19. ASME Section XI, Rules for In-Service Inspection of Nuclear Power Plant Components, Subsection IWL, Requirements for Class CC Concrete Components of Light-Water Cooled Plants, 1992 Edition with 1992 Addenda; 1995 Edition with 1996 Addenda, The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

20. NRC Information Notice 99-10, Degradation of Prestressing Tendon Systems in Prestressed Concrete Containments, U. S. Nuclear Regulatory Commission, April 1999.

21. NRC Regulatory Guide 1.35.1, Determining Prestressing Forces for Inspection of Prestressed Concrete Containments, U. S. Nuclear Regulatory Commission, July 1990.

APPENDIX C

(Not Used for This Application)

APPENDIX C
Contents

C1.0 APPENDIX C - NOT USED

Appendix C is not used in this application.

APPENDIX D

TECHNICAL SPECIFICATION CHANGES

APPENDIX D
Contents

D2.0 APPENDIX D - TECHNICAL SPECIFICATIONS CHANGES

10 CFR 54.22, requires that an application for license renewal include any Technical Specification changes, or additions that are necessary to manage the effects of aging during the period of extended operation. A review of the information provided in this License Renewal Application and the Ginna Station Technical Specifications confirms that no changes to the Technical Specifications are necessary.

APPENDIX E

(Provided as Linked Document)

APPENDIX E
Contents

E1.0 APPENDIX E - ENVIRONMENTAL REPORT

This page provides a link to the Environmental Report.