License Renewal Application

Arkansas Nuclear One - Unit 1

PREFACE

The following discussion describes where information is located in the ANO-1 License Renewal Application.

Section 1.0 provides the administrative information required by 10CFR54.17 and 10CFR54.19.

Section 2.0 provides the scoping and screening methodology. Section 2.0 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. Section 2.0 identifies the set of plant-specific design basis events that were used to determine the systems, structures, and components within the scope of license renewal, consistent with the plant's current licensing basis. Tables 2.2-1 and 2.2-2 provide a list of the plant systems and structures, respectively, and identify those plant systems and structures that are within the scope of license renewal. Section 2.0 provides a description of systems, intended functions, and references to system boundary drawings. Tables 2.3-1, 2.3-6, 2.3-7, and 2.3-8 show the drawing numbers for the systems in the scope of license renewal. The drawings are provided in a separate submittal. Tables in Section 3.0 are referenced in Section 2.0.

Section 3.0 describes the results of the aging management reviews of the components and structures requiring aging management reviews. Furthermore, Section 3.0:

- identifies the components and structures subject to aging management review and their intended functions,
- describes or references the processes used to identify aging effects requiring management (Appendix C summarizes the process used to identify aging effects associated with non-Class 1 mechanical components, which encompasses engineered safeguards system, auxiliary system, and steam and power conversion system components),
- discusses the materials and environments which produce aging effects,
- identifies the aging effects requiring management,
- describes industry and operating experience with respect to the applicable aging effects, and
- identifies the aging management programs that will manage the aging effects requiring management.

Section 3.0 refers to the aging management programs but does not describe the aging management programs nor discuss how the programs will manage the aging effects requiring management. Appendix B describes these programs and provides the information necessary to demonstrate that the aging effects requiring management will be adequately managed. The tables in Section 3.0 provide a comprehensive summary of information concerning the aging effects requiring management for component and commodity groupings in the scope of license renewal. For the component and commodity groupings that make up the system or structure, the tables list intended function, material, environment, aging effect, and the aging management programs and activities.

Section 4.0 includes a list of time-limited aging analyses, as defined by 10CFR54.3. It includes the identification of the component or subject and an explanation of the time-dependent aspects of the calculation or analysis. Section 4.0 includes a demonstration 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 4.0 also states that no 10CFR50.12 exemptions involving a time-limited aging analysis as defined in 10CFR54.3 are required during the period of extended operation.

Appendix A, Safety Analysis Report Supplement, provides a summary description of the programs and activities 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 and Activities, 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. The ANO-1 programs and activities that are credited for managing aging are divided into new actions and existing actions.

Appendix C, Process for Identifying Aging Effects Requiring Aging Management for Non-Class 1 Mechanical Components, summarizes the process through which the applicable aging effects were identified and associated with the non-Class 1 mechanical components determined to be subject to an aging management review.

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.

The information in Section 2.0, Section 3.0, and Appendix B fulfills the requirements in 10CFR54.21(a). The information in Section 4.0 fulfills the requirements in 10CFR54.21(c). The information in Appendix A and Appendix D fulfills the requirements in 10CFR54.21(b) will be met. The supplement to the environmental report required by 10CFR54.21(d) and 10CFR54.22, respectively. Section 1.4 discusses how the requirements in 10CFR54.23 is provided with the ANO-1 LRA as a separate document.

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Acronyms and Abbreviations

AAC	Alternate AC
AC	Alternating Current
ACI	American Concrete Institute
ACW	Auxiliary Cooling Water
AEC	Atomic Energy Commission
AISC	American Institute for Steel Construction
AMSAC	ATWS Mitigation System Actuation Circuit
ANO	Arkansas Nuclear One
ANO-1	Arkansas Nuclear One, Unit 1
ANO-2	Arkansas Nuclear One, Unit 2
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transient Without SCRAM
B&PV	Boiler and Pressure Vessel
BAW (B&W)	Babcock and Wilcox
BWOG	Babcock and Wilcox Owners Group
BWST	Borated Water Storage Tank
CASS	Cast Austenitic Stainless Steel
CFR	Code of Federal Regulations
CIRSE	CRDM Nozzle PWSCC Inspection and Repair Strategic Evaluation
CLB	Current Licensing Basis
CPE	Chlorinated Polyethylene
CRDM	Control Rod Drive Mechanism
CRDSS	Control Rod Drive Service Structure
CSPE	Chlorosulfonated Polyethylene
CST	Condensate Storage Tank
CuNi	Copper-Nickel
C_vUSE	Charpy Upper-Shelf Energy
DBA	Design Basis Accident
DC	Direct Current
DH	Decay Heat
DHR	Decay Heat Removal
DOR	Division of Operating Reactors
DROPS	Diverse Reactor Overpressure Protection System
DSS	Diverse SCRAM System
EAI	Entergy Arkansas, Inc.
ECCS	Emergency Core Cooling System
ECP	Emergency Cooling Pond
EDG	Emergency Diesel Generator
EFPD	Effective Full Power Days
EFPY	Effective Full Power Years

Acronyms and Abbreviations

EFW	Emergency Feedwater
EHC	Electro-Hydraulic Control
EOI	Entergy Operations, Inc.
EP	Ethylene Propylene
EPDM	Ethylene Propylene Diene Monomer
EPR	Ethylene Propylene Rubber
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
EQ-TAP	EQ Task Action Plan
ER	Applicant's Environmental Report – Operating License Renewal Stage
ES	Engineered Safeguards
ESF	Engineered Safeguards
eV	Electron Volt
FIV	Flow Induced Vibration
FMR	Okonite Cable Trademark
FR	Flame Resistant
FR-EP	Flame Resistant Ethylene Propylene
FR-EPDM	Flame Retardant Ethylene Propylene Diene Monomer
FSAR	Final Safety Analysis Report
FTI	Framatome Technologies, Inc.
GL	Generic Letter
GLRP	Generic License Renewal Program
gpm	Gallons per Minute
GSI	Generic Safety Issue
H&V	Heating and Ventilation
HELB	High Energy Line Break
HEPA	High-Efficiency Particulate Air
HPI	High Pressure Injection
HVAC	Heating, Ventilation, and Air Conditioning
IASCC	Irradiation-Assisted Stress Corrosion Cracking
ICW	Intermediate Cooling Water
IE	Inspection and Enforcement
IEEE	Institute of Electrical and Electronics Engineers
IGA	Intergranular Attack
ILRT	Integrated Leak Rate Testing
IN	Information Notice
INPO	Institute of Nuclear Power Operations
IPA	Integrated Plant Assessment
IPCEA	Insulated Power Cable Engineers Association
IRT	Initial Reference Temperature
ISI	Inservice Inspection
ISFSI	Independent Spent Fuel Storage Installation

Acronyms and Abbreviations

IST	Inservice Test
ITG	Issues Task Group
ITT	International Telephone and Telegraph
IWB	Subsection IWB of ASME Section XI Code
IWC	Subsection IWC of ASME Section XI Code
IWD	Subsection IWD of ASME Section XI Code
IWE	Subsection IWE of ASME Section XI Code
IWF	Subsection IWF of ASME Section XI Code
IWL	Subsection IWL of ASME Section XI Code
kip	Kilopound
kJ	Kilojoule
ksi	Kilopounds per Square Inch
KV	Kilovolt
KVA	Kilovolt-Ampere
KW	Kilowatt
LBB	Leak Before Break
LLRWB	Low Level Radwaste Building
LLRT	Local Leak Rate Testing
LOCA	Loss of Coolant Accident
LPI	Low Pressure Injection
LR	License Renewal
LRA	License Renewal Application
MFW	Main Feedwater
mg	Milligram
MIC	Microbiologically Induced Corrosion
MIRVP	Master Integrated Reactor Vessel Surveillance Program
mm	Millimeter
MOV	Motor Operated Valve
MSSV	Main Steam Safety Valve
MU	Makeup
MUP	Makeup and Purification
MW	Megawatt
MW(e)	Megawatt Electric
MW(t)	Megawatt Thermal
NaOH	Sodium Hydroxide
NDE	Nondestructive Examination
NDTT	Nil-Ductility Transition Temperature
NEI	Nuclear Energy Institute (formerly NUMARC)
NFPA	National Fire Protection Association
Ni	Nickel
Np	Neptunium
NPRDS	Nuclear Plant Reliability Data System
NPS	Nominal Pipe Size

Acronyms and Abbreviations

NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUMARC	Nuclear Management and Resources Council (now NEI)
OTSG	Once-Through Steam Generator
ppb	Parts per Billion
P&ID	Piping and Instrumentation Diagram
P-T	Pressure-Temperature
PASS	Post-Accident Sampling System
PM	Preventive Maintenance
PORV	Power-Operated Relief Valve
PSAR	Preliminary Safety Analysis Report
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
Q	Safety-related
QA	Quality Assurance
Q-CST	Safety-Related Condensate Storage Tank
RAI	Request for Additional Information
RB	Reactor Building
RBS	Reactor Building Spray
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RG	Regulatory Guide
RI-ISI	Risk-Informed Inservice Inspection
RPV	Reactor Pressure Vessel
RT	Reference Temperature
RTE	Resistance Temperature Element
RV	Reactor Vessel
RVI	Reactor Vessel Internals
RVIAMP	Reactor Vessel Internals Aging Management Program
RVLMS	Reactor Vessel Level Monitoring System
SAR	Safety Analysis Report
SCC	Stress Corrosion Cracking
SER	Safety Evaluation Report
SFP	Spent Fuel Pool
SMAW	Shielded Metal Arc Welding
SOC	Statements of Consideration
SR	Silicone Rubber
SRP	Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants – LWR Edition
SS	Stainless Steel
SSC	System, Structure, Component
SSHT	Surveillance Specimen Holder Tubes

Acronyms and Abbreviations

SW	Service Water
TID	Total Integrated Dose
TLAA	Time-Limited Aging Analysis
TMI	Three-Mile Island
UBC	Uniform Building Code
USAR	Updated Safety Analysis Report
USAS	USA Standards Institute
USE	Upper Shelf Energy
V	Volt
XLPE	Cross-linked Polyethylene

1.0 ADMINISTRATIVE INFORMATION

1.1 PURPOSE AND GENERAL INFORMATION

As the current operating license holder for ANO-1, Entergy Operations has prepared this application to provide the technical information required by 10CFR54 for the submittal of a License Renewal Application. This LRA and its supporting Applicant's Environmental Report - Operating License Renewal Stage are intended to provide sufficient information for the NRC to complete its technical and environmental reviews. The LRA and ER are designed to allow the NRC to make the finding required by 10CFR54.29 in support of the issuance of a renewed operating license for ANO-1. Following is the general information required by 10CFR54.17 and 10CFR54.19.

1.1.1 Name of Applicant

Entergy Operations, Inc. (operator) and Entergy Arkansas, Inc. (owner)

1.1.2 Address of Applicant

Entergy Operations (ANO-1)
1448 State Road 333
Russellville, AR 72802

1.1.3 Description of Business or Occupation of Applicant

Entergy Operations is an operating subsidiary of the Entergy Corporation, which is an investor-owned utility. Entergy Operations is engaged in the production of electric power primarily for portions of the states of Arkansas, Mississippi, Louisiana, and Texas. As a major part of this electricity production, Entergy Operations operates five nuclear power plants with a combined capacity of approximately 4875 megawatts.

1.1.4 Organization and Management of Applicant

Entergy Operations and Entergy Arkansas are public utilities incorporated under the laws of the State of Delaware. The Entergy Operations and Entergy Arkansas principal offices are located in Jackson, Mississippi and Little Rock, Arkansas, respectively, at the following addresses:

Entergy Operations, Inc. 1340 Echelon Parkway Jackson, MS 39213	Entergy Arkansas, Inc. 425 West Capitol Avenue Little Rock, AR 72201

Entergy Operations and Entergy Arkansas are not owned, controlled, or dominated by any alien, a foreign corporation, or foreign government. Entergy Operations and Entergy Arkansas make this application on their own behalf and are not acting as an agent or representative of any other person.

The names and addresses of the Entergy Operations and Entergy Arkansas directors and principal officers, all of whom are citizens of the United States, are as follows:

Directors of Entergy Operations, Inc.

Name	Address
Jerry Yelverton (EOI Chairman)	Jackson, Mississippi
Donald C. Hintz	New Orleans, Louisiana
C. John Wilder	New Orleans, Louisiana

Principal Officers of Entergy Operations, Inc.

Name	Address
Jerry W. Yelverton President and Chief Executive Officer	Jackson, Mississippi
C. John Wilder Executive Vice President and Chief Financial Officer	New Orleans, Louisiana
John R. McGaha Executive Vice President and Chief Operating Officer	Jackson, Mississippi
C. Gary Clary Senior Vice President-Human Resources and Administration	New Orleans, Louisiana
William A. Eaton Vice President-Operations (Grand Gulf)	Port Gibson, Mississippi
Randall K. Edington Vice President-Operations (River Bend)	St. Francisville, Louisiana
Joseph T. Henderson Vice President and General Tax Counsel	New Orleans, Louisiana
C. Randy Hutchinson Vice President-Operations (Arkansas Nuclear One)	Russellville, Arkansas
Charles M. Dugger Vice President-Operations (Waterford-3)	Killona, Louisiana
Nathan E. Langston Vice President and Chief Accounting Officer	New Orleans, Louisiana
Steven C. McNeal Vice President and Treasurer	New Orleans, Louisiana
Fred W. Titus Vice President-Engineering	Jackson, Mississippi
Joseph L. Blount Secretary	Jackson, Mississippi

Directors of Entergy Arkansas, Inc.

Name	Address
Thomas J. Wright (EAI Chairman)	Little Rock Arkansas
Donald C. Hintz	New Orleans, Louisiana
C. John Wilder	New Orleans, Louisiana

Principal Officers of Entergy Arkansas, Inc.

Name	Address
Thomas J. Wright President and Chief Executive Officer	Little Rock, Arkansas
C. John Wilder Executive Vice President and Chief Financial Officer	New Orleans, Louisiana
C. Gary Clary Senior Vice President-Human Resources and Administration	New Orleans, Louisiana
Frank F. Gallaher Senior Vice President, Generation, Transmission and Energy Mgmt.	New Orleans, Louisiana
Michael G. Thompson Senior Vice President, General Counsel and Secretary	New Orleans, Louisiana
Cecil L. Alexander Vice President-State Governmental Affairs	Little Rock, Arkansas
Joseph T. Henderson Vice President and General Tax Counsel	New Orleans, Louisiana
Nathan E. Langston Vice President and Chief Accounting Officer	New Orleans, Louisiana
Steven C. McNeal Vice President and Treasurer	New Orleans, Louisiana

1.1.5 Class and Period of License Sought

Entergy Operations requests renewal of the Class 104b operating license (reference 10CFR50.21(b)(1)) for ANO-1 (License No. DPR-51) for a period of 20 years beyond the expiration of the current license. The current license expires at midnight on May 20, 2014. Entergy Operations also requests renewal of those source, special nuclear material, and byproduct licenses that are combined in the operating license.

The facility will continue to be known as ANO-1 and will continue to generate electric power. ANO-1 is designed to operate at core power levels up to 2568 MW(t), which corresponds to a maximum dependable output electrical generation capacity of 836 net MW(e).

1.1.6 Earliest and Latest Dates for Alterations, if Proposed

Entergy Operations does not propose to construct or alter a production or utilization facility in connection with this application.

1.1.7 Conforming Changes to the Standard Indemnity Agreement

10CFR54.19(b) requires that conforming changes to the Standard Indemnity Agreement, Appendix B of 10CFR140.92, to account for the expiration term of the proposed renewed license be included in the application. The current Standard Indemnity Agreement for ANO-1 states in Article VII that the agreement shall terminate at the time of expiration of that license specified in Item 3 of the attachment to the Standard Indemnity Agreement. Item 3 of the attachment to the Standard Indemnity Agreement, as revised by Amendment No. 6, lists DPR-51 as an applicable license number. Entergy Operations requests that conforming changes be made to Article VII of the Standard Indemnity Agreement, and/or Item 3 of the attachment to the Standard Indemnity Agreement, specifying the extension of the Standard Indemnity Agreement until the expiration date of the renewed ANO-1 Operating License. Should the license number be changed upon issuance of the renewed license, Entergy Operations requests that conforming changes be made to Item 3 of the attachment and to any other sections of the Standard Indemnity Agreement as appropriate.

1.1.8 Restricted Data Agreement

As required by 10CFR50.37 and 10CFR54.17(g), Entergy Operations, as a part of the application for a renewed license for ANO-1, 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 10CFR Part 25 and/or 10CFR Part 95. This application does not contain restricted data or other national security information.

1.2 Plant Description

The ANO site is located in southwestern Pope County, Arkansas, on a peninsula formed by Lake Dardanelle. ANO-1 was constructed from 1968 until 1974. The unit consists of a Babcock and Wilcox pressurized water reactor nuclear steam supply system designed to generate 2568 MW(t), or approximately 836 net MW(e). The current facility operating license for ANO-1 expires at midnight May 20, 2014. The ANO-1 station consists primarily of an individual reactor building, an auxiliary building, an intake structure, and a common turbine building (shared with ANO-2, a Combustion Engineering PWR NSSS).

Entergy Operations operates an independent spent fuel storage installation in accordance with 10CFR Part 72 at ANO. The independent spent fuel storage installation is an independent facility subject to separate licensing and renewal provisions under 10CFR Part 72. The independent spent fuel storage installation is not within scope of 10CFR Part 54 or this application.

1.3 Technical Information Required for an Application

10CFR54.21 requires four technical items to support an application for a renewed operating license. These are an integrated plant assessment (Sections 2.0 and 3.0), an evaluation of time-limited aging analyses (Section 4.0), a supplement to the ANO-1 Safety Analysis Report (sometimes referred to as the USAR or FSAR) which contains a summary description of the programs and activities for managing the effects of aging and the evaluation of the time-limited aging analyses (Appendix A), and current licensing basis changes during NRC review (Section 1.4)

In addition to the technical information, 10CFR54.22 requires applicants to submit any technical specification changes or additions necessary to manage the effects of aging during the period of extended operation (Appendix D). 10CFR54.23 requires the application to include a supplement to the environmental report (Applicant's Environmental Report – Operating License Renewal Stage).

The IPA, as defined by 10CFR54.3, is a licensee assessment that demonstrates that a nuclear power plant facility's structures and components requiring aging management review in accordance with 10CFR54.21(a) for license renewal have been identified. The IPA also demonstrates that the effects of aging on the functionality of such structures and components will be managed to maintain the current licensing basis during the period of extended operation. The ANO-1 IPA includes:

- identification of the structures and components within the scope of license renewal that are subject to an aging management review;
- identification of the aging effects applicable to these structures and components;
- identification of plant-specific programs and activities that will manage these identified aging effects; and
- a demonstration that these programs and activities will be effective in managing the effects of aging during the period of extended operation.

The ANO-1 IPA for license renewal, along with other information necessary to document compliance with 10CFR54, is maintained in an auditable and retrievable form in accordance with 10CFR54.37(a). The ANO-1 IPA is documented with site-specific reports and calculations which were generated in accordance with onsite administrative procedures. Also, note that references to the ANO-1 Technical Specifications and SAR are as of Amendment 202 and Amendment 15, respectively.

1.4 Current Licensing Basis Changes during NRC Review

Each year, following the submittal of the ANO-1 LRA, and at least three months before the scheduled completion of the NRC review, Entergy Operations will submit amendments to the application pursuant to 10CFR54.21(b). The revision will identify any changes to the current licensing basis that materially affect the contents of the LRA and any other aspects of the application.

2.0 STRUCTURES AND COMPONENTS SUBJECT TO AN AGING MANAGEMENT REVIEW

2.1 SCOPING AND SCREENING METHODOLOGY

2.1.1 Introduction

Section 2.0 describes the first major activity of the ANO-1 integrated plant assessment, the identification of structures and components subject to aging management review. The information provided in Section 2.0 is intended to meet the requirements of 10CFR54.21(a)(1) and 10CFR54.21(a)(2).

ANO-2 and the independent spent fuel storage installation structures, and components were not included in the IPA, since they have separate licenses from the ANO-1 operating license. However, certain common systems, structures, and components that are shared by ANO-1 and ANO-2 are included in the IPA, since they meet the criteria for being in scope for ANO-1.

For those systems, structures, and components within the scope of license renewal (defined by 10CFR54.4), 10CFR54.21(a)(1) requires a license renewal applicant to identify and list the structures and components subject to aging management review. 10CFR54.21(a)(2) further requires that the methods used to identify and list these structures and components be described and justified. The information in this section serves to satisfy these requirements. Overall, the information provided in Sections 2.0 and 3.0 provides the basis for the NRC to make the finding required by 10CFR54.29(a)(1).

The ANO-1 IPA is divided into the areas of mechanical, civil/structural, and electrical. The results of the assessment to identify the systems and structures within the scope of license renewal (scoping) are contained in Section 2.2. The results of the identification of the components and structures subject to aging management review (screening) are contained in Section 2.3 for mechanical components, Section 2.4 for structures, and Section 2.5 for electrical commodities.

2.1.2 Assessment Using Criteria in 10CFR54.4

The ANO-1 IPA followed the process recommended in NEI 95-10 (Reference 2.1-1) to determine the systems, structures, and components in the scope of license renewal. This section discusses the assessment performed to identify structures, systems, and components that satisfy the criteria in 10CFR54.4(a)(1), 10CFR54.4(a)(2), and 10CFR54.4(a)(3).

<u>Safety-Related Criteria Pursuant to 10CFR54.4(a)(1)</u>

10CFR54.4(a)(1) states that the systems, structures, and components within the scope of license renewal include:

Safety-related systems, structures, and components which are those relied upon to remain functional during and following design-basis events (as defined in 10CFR50.49(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 the guidelines in 10CFR50.34(a)(1) or 10CFR100.11 of this chapter, as applicable.

In the ANO-1 SAR Table 1-2, “safety-related” or “Q” is defined based on 10CFR Part 100, Appendix A, as the structures, systems, and components required to assure:

(1) the integrity of the reactor coolant pressure boundary,

(2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or

(3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite doses comparable to the guideline doses of 10CFR Part 100.

A summary level Q-list for ANO-1 structures and systems is provided in the ANO-1 SAR Table 1-2.

In the mid-1980’s, Entergy Operations implemented a component level Q-list project, which classified “Q” devices at the component level. This component level Q-list is maintained in a component database at ANO. The ANO-1 summary and component level Q-lists include those systems, structures, and components relied on to remain functional during or following design basis events described in SAR Chapter 14. The postulated events in SAR Chapter 14 include:

- Uncompensated operating reactivity changes
- Startup accident
- Rod withdrawal accident at rated power operation
- Moderator dilution accident
- Cold water accident

- Loss of coolant flow
- Stuck-out, stuck-in, or dropped control rod accident
- Loss of electric power
- Turbine overspeed
- Fuel loading errors
- Steam line failure
- Steam generator tube failure
- Fuel handling accident
- Rod ejection accident
- Loss of coolant accident
- Maximum hypothetical accident
- Waste gas tank rupture

In summary, the ANO-1 summary and component level Q-lists were used during the IPA to identify ANO-1 systems, structures, and components that are safety-related. Since the ANO-1 summary and component level Q-lists include safety-related systems, structures, and components for ANO-1, this process to identify systems, structures, and components meets the criteria of 10CFR54.4(a)(1).

Nonsafety-Related Criteria Pursuant to10CFR54.4(a)(2)

10CFR54.4(a)(2) states that the systems, structures, and components within the scope of license renewal include:

"All nonsafety-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) of this section."

At ANO-1, the majority of systems, structures, or components whose failure could impact a functionally related safety-related system are classified as safety-related. Therefore, except for a few cases, the systems and structures meeting these criteria have components on the ANO-1 Q-list and are included in the scope of license renewal under 10CFR54.4(a)(1).

Based on a review of the ANO-1 SAR and design documents, a few cases have been identified in which passive, long-lived, nonsafety-related components could impact safety-related functions.

- Spatially-related components in which the physical location could result in interaction between components. This includes seismic category 2 over 1 structures and components.
- Spent fuel pool liner is nonsafety-related as documented in the ANO-1 SAR, but has been included in the scope of license renewal. [Note: The acceptability of a nonsafety-related liner is acknowledged in footnote 1 of the NRC Standard Review

Plan (NUREG-0800), Section 9.1.2, “Spent Fuel Storage.” This footnote states that for operating licenses issued before November 17, 1977, analysis of a nonsafety-related liner was not required and is not considered necessary since stresses in the liner in the event of a safe shutdown earthquake will be low and, therefore, liner failure is unlikely.]

In addition, a few nonsafety-related components were included in the scope of license renewal, although they do not meet the criteria of 10CFR54.4(a)(2). These additional components are

- nonsafety-related valves and piping that are part of the pressure boundary for the main steam lines and steam generators inside the reactor building, and
- certain nonsafety-related valves and piping in the auxiliary building sump system that are credited for preventing offsite releases.

In summary, the few cases in which ANO-1 nonsafety-related components could impact safety-related functions have been identified and the associated components have been included in the scope of license renewal in accordance with the criteria of 10CFR54.4(a)(2).

<u>Other Scoping pursuant to 10CFR54.4(a)(3)</u>

10CFR54.4(a)(3) states that the systems, structures, and components within the scope of license renewal include:

All systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission’s regulations for fire protection (10CFR50.48), environmental qualification (10CFR50.49), pressurized thermal shock (10CFR50.61), anticipated transients without scram (10CFR50.62), and station blackout (10CFR50.63).

Scoping based on each of these regulations is addressed separately in the following sections:

<u>Commission's Regulations for Fire Protection (10CFR50.48)</u>

The original ANO-1 fire protection systems met General Design Criterion 3 of 10CFR Part 50, Appendix A. In 1980, the NRC issued 10CFR50.48 and 10CFR Part 50, Appendix R, to address new requirements. Three sections of 10CFR Part 50 Appendix R (Sections III.G, III.J, and III.O) impose new requirements on plants licensed prior to January 1, 1979, including ANO-1.

Section III.G required safe shutdown evaluations to ensure one train of safe shutdown equipment remained free of fire damage. A fire area reanalysis was performed at ANO-1 to evaluate the plant equipment required to place the plant in a safe shutdown condition for any single fire scenario. The fire analysis contains a listing of the ANO-1 components that can be used to place the plant in a safe condition following a fire. These components are in the license renewal scope.

Section III.J required emergency lighting for all necessary areas including access and egress. The emergency lighting system is included in the license renewal scope.

Section III.O required a seismic oil collection system be provided for the reactor coolant pumps. Entergy Operations requested and was granted an exemption from the requirements for a seismic reactor coolant pump oil collection system. The ANO-1 non-seismic reactor coolant pump oil collection system is in the license renewal scope.

In addition to the components discussed above, the ANO-1 component database includes nonsafety-related components (both mechanical and electrical) that are required to meet 10CFR50.48 requirements. These components are included in the license renewal scope.

For structural components and commodities, ANO-1 fire protection drawings and procedures were utilized to identify the fire doors, fire walls, and other fire barriers that are credited for compliance with 10CFR50.48 requirements. These structural components are in the license renewal scope.

Commission's Regulations for Environmental Qualification (10CFR50.49)

The Environmental Qualification Program at ANO is a centralized plant support program administered by design engineering in order to maintain compliance with 10CFR50.49. Electrical equipment located in a harsh environment which is important to safety, including safety-related (Q) equipment, nonsafety-related equipment whose failure could adversely affect safety-related equipment, and the necessary post-accident monitoring equipment is included in the scope of the program. The identification of equipment is specified by procedural controls and the component database is utilized to maintain an EQ equipment master list. The components in the EQ program were included in the ANO-1 license renewal scope.

Commission's Regulations for Pressurized Thermal Shock (10CFR50.61)

Pressurized thermal shock is an event, or transient, that causes a severe overcooling concurrent with, or followed by, significant pressure in the reactor vessel. The requirements in 10CFR50.61 identify specific operational limits for PTS pertaining to the belt-line region of the reactor vessel. If these limits are exceeded, the licensee must identify and submit a list of any plant modifications to systems, equipment, and operation to prevent a potential failure of the reactor vessel. ANO-1 will not exceed the PTS screening criteria during the license renewal period to 60 years, as addressed in Section 4.2.1. Since the PTS screening criteria are met, no structures, systems, or components are required for ANO-1 to comply with the PTS regulation. Therefore, no structures, systems, or components are included in the scope of license renewal to meet the PTS criterion of 10CFR54.4(a)(3).

Commission's Regulations for Anticipated Transients without Scram (10CFR50.62)

In 1990, Entergy Operations installed several instrumentation systems in ANO-1 for compliance with the Commission's regulations for anticipated transients without scram (10CFR50.62). A DROPS/DSS system was installed for a diverse reactor trip and a DROPS/AMSAC was installed for a backup actuation of EFW and a diverse main turbine trip. These systems place ANO-1 in compliance with 10CFR50.62 and provide plant protection in the event the reactor protection system fails to perform its function during an ATWS event. The components in the DROPS/DSS and the DROPS/AMSAC are included in the scope of license renewal.

Commissions Regulations for Station Blackout (10CFR50.63)

ANO-1 is committed to the regulatory requirements of 10CFR50.63 utilizing the design criteria/approach in NUMARC 87-00 (Reference 2.1-2) that is accepted by Regulatory Guide 1.155, "Station Blackout." An alternate AC diesel generator has been installed at ANO. The alternate AC diesel generator is in a separate structure and is totally independent of the other emergency power sources and their auxiliaries. The alternate AC diesel generator has its own air start system, fuel oil transfer pump/day tank, cooling, and lube oil subsystems. The mechanical components, electrical commodities, and structures that are needed for the alternate AC diesel generator to perform its function are included in the scope of license renewal.

In summary, the systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with NRC regulations for fire protection, EQ, PTS, ATWS, and station blackout, have been included in the scope of license renewal in accordance with the criterion of 10CFR54.4(a)(3).

2.1.3 Assessment Using Criteria in 10CFR54.21(a)(1)

10CFR54.21(a)(1) states that structures and components subject to an aging management review shall encompass those structures and components:

(i) that perform an intended function, as described in 10CFR54.4, without moving parts or without a change in configuration or properties and,

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

As part of the ANO-1 IPA, an assessment was performed using the ANO-1 SAR, component database, and design documents to identify the intended functions for the mechanical components, structures and structural components, and electrical commodities. The intended functions identified were based on the guidance of NEI 95-10.

Of the structures and components within the scope of 10CFR54.4, only those that are classified as long-lived and passive are subject to an aging management review. In other words, active components and those identified as having a periodic replacement schedule are not subject to an aging management review. The passive and long-lived structures and components at ANO-1 were identified using NEI 95-10 as a guide. The listing of identified structures and components and their intended functions is discussed in Section 2.3 for mechanical components, Section 2.4 for structures, and Section 2.5 for electrical commodities.

2.1.4 Generic Safety Issues

In accordance with the guidance in NEI 95-10, review of NRC generic safety issues as a part of the license renewal process is required to satisfy the finding required by 10CFR54.29. GSIs that involve an issue related to the license renewal aging management review or time-limited aging analysis evaluations are to be addressed in the LRA. Based on the NEI guidance, NUREG-0933, "A Prioritization of Generic Safety Issues" (Supplement 23), and the Oconee LRA, Entergy Operations has identified the following list of GSIs to be addressed in the ANO-1 LRA:

<u>GSI 23 – Reactor Coolant Pump Seal Failures</u>

GSI 23 has recently been closed by the NRC (Reference 2.1-3).

<u>GSI 168 – Environmental Qualification of Electrical Equipment</u>

This GSI is related to aging concerns with respect to environmental qualification of electrical equipment. Environmental qualification evaluations of electrical equipment are identified as time-limited aging analyses for ANO-1; therefore, this GSI is addressed in Section 4.4.69.

<u>GSI 173.A – Spent Fuel Storage Pool: Operating Facilities</u>

The only age-related issue associated with this GSI is the potential degradation of Boraflex, which was identified in NRC Generic Letter 96-04 and is identified as a time-limited aging analysis for ANO-1. Therefore, this GSI is addressed in Section 4.7.

<u>GSI 190 – Fatigue Evaluation of Metal Components for 60-Year Plant Life</u>

This GSI addresses fatigue life of metal components. Fatigue evaluation of metal components is identified as a time-limited aging analysis for ANO-1. Therefore, this GSI is addressed in Section 4.3.

2.1.5 Conclusion

The methods described in Sections 2.1.1 through 2.1.4 were used at ANO-1 to identify the systems, structures, and components that are within the scope of license renewal and that require an aging management review. This is consistent with the requirements of 10CFR54.4 and 10CFR54.21(a)(1), respectively.

The list of the systems and structures in the scope of license renewal is included in Section 2.2. The more detailed listing of components, structures, and commodity groups and the associated intended functions is discussed in Section 2.3 for mechanical components, Section 2.4 for structures, and Section 2.5 for electrical commodities.

2.1.6 References for Section 2.1

2.1-1 NEI 95-10, "*Industry Guidelines for Implementing the Requirements of 10CFR Part 54 – The License Renewal Rule,*" Revision 0, Nuclear Energy Institute, March 1996.

2.1-2 NUMARC 87-00, "*Guidelines and Technical Basis for NUMARC Initiatives Addressing Station Blackout at LWRs,*" Revision 1, Nuclear Management and Resources Council, August 1991.

2.1-3 NRC letter from A. Thadani to W. Travers, dated November 8, 1999.

2.2 Plant Level Scoping Results

Table 2.2-1 lists the ANO-1 plant systems and the results of the evaluation that was performed to determine what systems contain components that meet the 10CFR54.4 criteria for being in the scope of license renewal. The systems are listed as being in the scope of license renewal if any of the components in the system meet the criteria of 10CFR54.4 as discussed in Section 2.1.2. Therefore, many nonsafety-related systems, in addition to those nonsafety-related systems discussed in Section 2.1.2, are identified in the table as being in the scope of license renewal only due to a limited number of components that require review at the electrical or mechanical interfacing boundaries. Section 2.2.1 provides additional detail on the assessment results provided in Table 2.2-1.

Table 2.2-2 lists ANO-1 plant structures and identifies which structures meet the 10CFR54.4 criteria for being in the scope of license renewal. Section 2.2.2 provides additional detail on the assessment results provided in Table 2.2-2.

2.2.1 Mechanical and Electrical Systems

The list of ANO-1 systems (mechanical and electrical) within the scope of license renewal was created as discussed in Section 2.1 and is documented in Table 2.2-1. The following information provides additional detail on the assessment process.

The ANO-1 reactor coolant system is a typical B&W pressurized water nuclear steam supply system with a reactor vessel, two steam generators, four reactor coolant pumps, a pressurizer, and the connecting or interfacing piping as the primary components. The RCS is an ASME Class 1 system that is safety-related and is, therefore, in the scope of license renewal. The components that maintain the RCS pressure boundary are in the scope of license renewal.

The non-Class 1 mechanical systems determined to meet the 10CFR54.4 criteria were included in the scope of review. Many of these systems (such as high pressure injection, low pressure injection, core flood, reactor building spray, emergency feedwater, reactor building cooling, emergency diesel generators, hydrogen control, penetration room ventilation, control room ventilation, etc.) have important safety functions under accident conditions and are clearly in the scope of license renewal. Other systems are needed to support these systems (such as service water, fuel oil, etc.) and are in the scope of license renewal, since they are required to operate under accident conditions to support the safety-related systems.

Some non-Class 1 systems required for normal plant operation (such as main feedwater and main steam) have a limited portion of the system that performs a safety-related function, and therefore, a portion of the system is included in the scope of license renewal. Portions of the instrument air system, that are necessary for the operation of safety-related valves and dampers, are Q-listed and are included in the scope of license renewal. The portions of the condensate storage and transfer system required to support emergency feedwater system operation are Q-listed and in the scope of license renewal. Portions of the chilled water systems, that support operation of safety-related cooling units, are Q-listed and in the scope of license renewal.

A number of non-Class 1 mechanical systems are in the scope of license renewal *only* due to their reactor building penetrations (the remainder of the system was not in scope) and were grouped together for aging management review. Some vent, drain and sampling systems are in the scope of license renewal because of Q-listed components at the interface with safety-related systems.

The halon system, portions of the fire protection system required to support 10CFR50.48, and the alternate AC diesel generator and supporting equipment are in the scope of license renewal per 10CFR54.4(a)(3).

The ANO-1 electrical systems include an offsite power supply from the switchyard, two essential trains (red and green) of onsite electrical distribution that supply power to safety-related components, and a non-safety grade power supply for nonsafety-related equipment. The 500kV switchyard is a non-Q power supply that is not in the scope of license renewal (except for some interfacing components) since the emergency diesel generators are the credited safety-related power supplies. The main generator excitation is required for normal plant operation, but is non-Q and is not in the scope of license renewal (except for some interfacing components).

The ANO-1 reactor coolant pumps are supplied by non-Q 6.9kV busses. The 6.9kV switchgear is not in the scope of license renewal, except for interfacing components and portions of the breakers that are credited with tripping the breaker in the 10CFR50.48 analysis (to ensure the reactor coolant pumps do not continue to operate and cause a seal failure if seal cooling is lost due to a fire).

The ANO-1 4.16kV switchgear system can be powered from offsite or onsite sources. The safety grade portions of this system supply power to the larger safety grade pump motors (high pressure injection, low pressure injection, service water, emergency feedwater, etc.) and the safety grade 480V load centers that supply power to many safety-related valves, coolers, and pumps. Therefore, the safety grade portions of the 4.16kV switchgear and 480V load centers are Q-listed and in the scope of license renewal as identified in Table 2.2-1.

The essential 125V DC system, including the safety-related station batteries, and the essential 120V AC system are Q-listed and in the scope of license renewal since they supply power to safety-related equipment under accident conditions.

The safety-related instrumentation systems (reactor protection, emergency feedwater initiation and control, engineered safeguards actuation, etc.) are in the scope of license renewal since they are Q-listed and credited in accident analyses. The ATWS mitigation, diverse scram, and diverse reactor overpressure protection systems are in the scope of license renewal since they are credited in the anticipated transients without scram (10CFR50.62) analysis. Many of the other control systems that are required to support normal plant operation (electro-hydraulic control, integrated control system) are non-Q and are not in the scope of license renewal (except for some interfacing components).

The electrical components associated with the alternate AC diesel generator (including the DC power supply), the fire detection system, the electrical controls of the fire protection system and the portion of the emergency lighting required for 10CFR50.48 are in the scope of license renewal per 10CFR54.4(a)(3).

2.2.2 Structures

The list of ANO-1 structures within the scope of license renewal was created by reviewing the SAR, site plans and general arrangement drawings, and other plant-specific documents as discussed in Section 2.1. The identification of safety-related structures and non-safety-related structures whose failure could prevent safety-related systems, structures, or components from fulfilling their safety-related functions was based on the classification of each ANO-1 structure as documented in SAR Table 1-2.

ANO-1 structures are designated as either seismic category 1 or seismic category 2. ANO-1 seismic category 1 structures are those which prevent uncontrolled release of radioactivity and are designed to withstand design basis loadings without loss of function as defined in the ANO-1 SAR. This classification is consistent with the intent of 10CFR54.4(a)(1). Therefore, Entergy Operations has determined that ANO-1 seismic category 1 structures meet the criteria contained in 10CFR54.4(a)(1), and are within the scope of license renewal.

Seismic category 2 structures are those structures whose limited damage would not result in a release of radioactivity, would permit a controlled plant shutdown, but could interrupt power generation. Chapter 5 of the SAR states that seismic category 2 structures do not perform a nuclear safety-related function but that their failure could possibly affect the function of a safety-related system. This classification is consistent with the intent of 10CFR54.4(a)(2). Therefore, Entergy Operations has determined that some ANO-1 seismic category 2 structures/building portions meet the criteria contained in 10CFR 54.4(a)(2), and are within the scope of license renewal.

The list of ANO-1 structures also includes those within the scope of license renewal that meet the criteria of 10CFR54.4(a)(3).

Table 2.2-1 Listing of ANO-1 Mechanical and Electrical Systems

SYSTEM	IN SCOPE	LRA SECTION(S)
4.16 KV Switchgear	X	2.5
120 V Instrument AC	X	2.5
120 VAC	X	2.5
125 V DC	X	2.5
480V Load Centers	X	2.5
500 KV	X[1]	2.5
6.9 KV Switchgear	X[1]	2.5
AAC Building Ventilation	X	2.3.3.5, 2.5
Administration Building Heating and Ventilation		Not in scope of LRA
Alternate AC Diesel Generator	X	2.3.3.5, 2.5
Annunciator	X[1]	2.5
Area Radiation Monitoring	X[1]	2.5
Atmospheric Vents		Not in scope of LRA
ATWS Mitigation	X	2.5
Auxiliary Building Drains		Not in scope of LRA
Auxiliary Building Equipment Vents		Not in scope of LRA
Auxiliary Building Heating and Ventilation	X	2.3.3.12, 2.5
Auxiliary Building Sump	X[1]	2.3.3.4
Auxiliary Cooling Water	X[1]	2.5
Breathing Air	X[1]	2.3.2.7
Carbon Dioxide		Not in scope of LRA
Cardox	X[2]	2.3.3.6, 2.5
Cathodic Protection		Not in scope of LRA
Chemical Addition	X[1]	2.3.2.4, 2.3.2.6, 2.5
Chilled Water	X	2.3.3.9, 2.5
Chlorination		Not in scope of LRA
Circulating Water		Not in scope of LRA
Clean Liquid Radwaste	X[1]	2.3.2.7, 2.5
Condensate Demineralizer	X[1]	2.5
Condensate Storage & Transfer	X	2.3.4.4, 2.5
Condenser Vacuum		Not in scope of LRA
Control Rod Drive	X	2.3.1.5, 2.3.1.9, 2.5

Table 2.2-1 Listing of ANO-1 Mechanical and Electrical Systems

SYSTEM	IN SCOPE	LRA SECTION(S)
Control Room Ventilation	X	2.3.3.13, 2.5
Core Flood	X	2.3.2.1, 2.5
Decay Heat	X	2.3.2.2, 2.5
Dirty Liquid Radwaste	X[1]	2.5
Dirty Water Drain	X[1]	2.3.3.4
Diverse Reactor Over-Pressure Protection	X	2.5
Diverse SCRAM	X	2.5
Domestic Water		Not in scope of LRA
Electro-Hydraulic Control	X[1]	2.5
Emergency Diesel Generator	X	2.3.3.3, 2.5
Emergency Feedwater	X	2.3.4.3, 2.5
Emergency Feedwater Initiation and Control	X	2.5
Emergency Lighting	X	2.5
Emergency Operations Facility		Not in scope of LRA
Engineered Safeguards Actuation	X	2.5
Extraction Steam		Not in scope of LRA
Feedwater Pump Lube Oil		Not in scope of LRA
Fire Detection	X	2.5
Fire Protection	X	2.3.3.2, 2.5
Fuel Handling	X[4]	2.4.2.1
Fuel Oil	X	2.3.3.7, 2.5
Gas Collection Header	X[1]	2.3.2.8
Gaseous Effluent Radiation Monitoring		Not in scope of LRA
Gaseous Radwaste	X[1]	2.3.2.7, 2.5
Generator Seal Oil		Not in scope of LRA
Gland Steam		Not in scope of LRA
Halon	X	2.3.3.6, 2.5
Heat Tracing	X[1]	2.5
Heater Drains		Not in scope of LRA
Heater Vents	X[1]	2.3.2.7, 2.3.4.1
High Pressure Injection	X	2.3.2.3, 2.5

Table 2.2-1 Listing of ANO-1 Mechanical and Electrical Systems

SYSTEM	IN SCOPE	LRA SECTION(S)
Hydrogen	X[1]	2.5
Hydrogen Purge	X[1]	2.3.2.8, 2.5
Hydrogen Recombiners	X	2.3.2.8, 2.5
Inadequate Core Cooling	X	2.5
Incore Instrumentation	X	2.3.1.3, 2.5
Instrument Air	X[1]	2.3.3.8, 2.5
Integrated Control	X[1]	2.5
Intermediate Cooling	X[1]	2.3.2.7, 2.3.2.8, 2.5
Isophase Bus		Not in scope of LRA
Isophase Bus Cooling	X[1]	2.5
Laundry Radwaste		Not in scope of LRA
Low Pressure Injection	X	2.3.2.2, 2.5
Lube Oil	X[1]	2.5
Main Chiller Cooling Water		Not in scope of LRA
Main Feedwater	X	2.3.4.2, 2.5
Main Generator Excitation	X[1]	2.5
Main Steam	X	2.3.4.1, 2.5
Main, Auxiliary and Startup Transformers	X[1]	2.5
Makeup and Purification	X	2.3.2.3, 2.5
Meteorological		Not in scope of LRA
Miscellaneous Turbine Drains		Not in scope of LRA
Neutralizing Tank		Not in scope of LRA
Nitrogen Supply	X[1]	2.3.1.3, 2.3.2.7, 2.3.4.1, 2.5
Non-Nuclear Instrumentation	X	2.5
Nuclear Instrumentation	X	2.5
Oily Water Separator		Not in scope of LRA
Particulate Air Monitoring	X[1]	2.5
Penetration Room Ventilation	X	2.3.3.11, 2.5
Plant Computer		Not in scope of LRA
Plant Heating	X[1]	2.3.2.7, 2.5
Plant Makeup	X[1]	2.5
Plant Performance Analysis		Not in scope of LRA
Post Accident Sampling	X[1]	2.3.1.3, 2.3.2.2, 2.5

Table 2.2-1 Listing of ANO-1 Mechanical and Electrical Systems

SYSTEM	IN SCOPE	LRA SECTION(S)
Process Radiation Monitoring.	X[1]	2.5
Quindar		Not in scope of LRA
Radiation Monitoring		Not in scope of LRA
Radiation Dose Assessment		Not in scope of LRA
Reactor Building Drains	X	2.3.1.3, 2.3.3.4
Reactor Building Heating and Ventilation	X	2.3.2.5, 2.5
Reactor Building Purge	X[1]	2.3.2.5
Reactor Building Spray	X	2.3.2.4, 2.3.3.4, 2.5
Reactor Building Sump	X	2.3.3.4, 2.5
Reactor Building Vents	X	2.3.1.9
Reactor Coolant	X	2.3.1, 2.3.3.4, 2.5
Reactor Coolant Pump (Support Equipment)	X	2.5
Reactor Core	X[3]	
Reactor Protection	X	2.5
Regenerative Waste	X[1]	2.5
Reheat Steam		Not in scope of LRA
Resin Transfer		Not in scope of LRA
Safety Parameter Display	X[1]	2.5
Sampling	X[5]	2.5
Screen Wash	X[1]	2.5
Security	X[1]	2.5
Seismic Monitoring		Not in scope of LRA
Service Air	X[1]	2.3.2.4
Service Water	X	2.3.3.10, 2.5
Sewage Treatment		Not in scope of LRA
Smart Auto Signal Selection		Not in scope of LRA
Spent Fuel	X	2.3.3.1, 2.5
Spent Resin		Not in scope of LRA
Startup Boiler		Not in scope of LRA
Turbine Building Ventilation	X[4]	2.4.6.2
Turbine Building Sump		Not in scope of LRA
Turbine Generator	X[1]	2.5

Table 2.2-1 Listing of ANO-1 Mechanical and Electrical Systems

SYSTEM	IN SCOPE	LRA SECTION(S)
Vibration and Loose Parts Monitoring		Not in scope of LRA
Waterbox Vacuum		Not in scope of LRA

NOTES:

'X' – Denotes system is within the scope of license renewal

1. A small portion of the system is in scope
2. Halon portion only
3. Not subject to aging management review due to periodic component replacement
4. Structural portions only
5. Mechanical components in scope are contained in several systems (Refer to scoping drawings).

Table 2.2-2 Listing of ANO-1 Structures		
STRUCTURE	**IN SCOPE**	**CORRESPONDING LRA SECTION(S)**
Administration Building		Not in scope of LRA
Alternate AC Diesel Generator Building Foundation	X	2.4.6.1
Auxiliary Building	X[2]	2.4.3
Boathouse		Not in scope of LRA
Borated Water Storage Tank Foundation	X[1]	2.4.6.1
Bottle Storage Building		Not in scope of LRA
Bulk Fuel Oil Storage Tank Foundation	X	2.4.6.1
Caustic Acid Building		Not in scope of LRA
Central Support Building		Not in scope of LRA
Chemical Flush Discharge Pond		Not in scope of LRA
Chemical Treatment Building		Not in scope of LRA
Condensate Storage Tank Foundation and Pipe Trenches	X	2.4.6.1
Controlled Access #3		Not in scope of LRA
Crafts Fabrication Shop		Not in scope of LRA
Deluge Valve Building		Not in scope of LRA
Electrical Manholes	X	2.4.6.1
Emergency Cooling Pond/ Intake/Discharge Canals	X	2.4.5
Emergency Diesel Fuel Oil Storage Tank Vault	X	2.4.6.1
Engineering Building		Not in scope of LRA
Fire Fighting Equipment Hose Houses next to Hydrant H1 through H11.		Not in scope of LRA
Fire Training Smoke House		Not in scope of LRA
Generation Support Building		Not in scope of LRA
Hydrogen and CO_2 Building		Not in scope of LRA
Incinerator Building		Not in scope of LRA
Intake Structure	X[2,3]	2.4.4
LLRWB SPING Unit Shelter		Not in scope of LRA

Table 2.2-2 Listing of ANO-1 Structures		
STRUCTURE	**IN SCOPE**	**CORRESPONDING LRA SECTION(S)**
Maintenance Building		Not in scope of LRA
North Gate Guard House		Not in scope of LRA
Off-Site Fabrication Building		Not in scope of LRA
Oily Water Separator Building		Not in scope of LRA
Old Administration Building		Not in scope of LRA
Paint and Lube Oil Storage Building		Not in scope of LRA
Post Accident Sampling System Building	X	2.4.3
Radioactive Waste Building		Not in scope of LRA
Radwaste Storage Building		Not in scope of LRA
Reactor Building	X[2]	2.4.1 and 2.4.2
Secondary Guard House		Not in scope of LRA
Security Compliance Building		Not in scope of LRA
Start-up Boiler Building		Not in scope of LRA
Sullair Air Compressor Building		Not in scope of LRA
Technical Support Building		Not in scope of LRA
Turbine Building	X[4]	2.4.3 and 2.4.6.2
Vacuum Degasifier Building		Not in scope of LRA
Warehouse No. 1 through Warehouse No. 7		Not in scope of LRA

Notes:

‘X’ – Denotes structure is within the scope of license renewal.

1. The Borated Water Storage Tank (T3) sits on a slab that is part of the Auxiliary Building.
2. Includes associated structural components and commodities (i.e., supports, spent fuel pool) within the scope of license renewal.
3. Category 1 portions.
4. Limited areas containing 10CFR50.48-required fire barriers are in-scope.

2.3 MECHANICAL SYSTEMS SCOPING AND SCREENING RESULTS

2.3.1 Reactor Coolant System Mechanical Components

2.3.1.1 Description of the Process to Identify Reactor Coolant System Components Subject to Aging Management Review

The determination of mechanical systems within the scope of license renewal is made by initially identifying ANO-1 mechanical systems and then reviewing them to determine which ones satisfy one or more of the criteria contained in 10CFR54.4. This process is described in Section 2.1 and the results of the mechanical systems review are contained in Section 2.2. Section 2.3.1 contains the information required by 10CFR54.21(a)(1) and 10CFR54.21(a)(2) for the ANO-1 RCS components that are subject to aging management review for license renewal.

The RCS is within the scope of license renewal because it is relied upon to remain functional during and following design basis events, and it is relied upon in safety analyses and plant evaluations to perform a function that demonstrates compliance with the NRC regulations for fire protection, environmental qualification, anticipated transients without scram, and station blackout. RCS components are designed to maintain functional integrity during seismic events.

The method used to determine the RCS structures and components subject to aging management review is consistent with the guidance in NEI 95-10 (Reference 2.3-1). The list of those RCS components that are subject to aging management review was made by reviewing ANO-1 flow diagrams for the RCS and marking the ANO-1 ISI Class 1 boundary. For ANO-1, the Class 1 ISI boundary and Class 1 design boundary are equivalent. Evaluation boundaries for the portions of the RCS within the scope of license renewal are shown on the flow diagrams listed in Table 2.3-1.

The RCS mechanical components subject to aging management review were identified by reviewing the following documentation: ANO-1 piping and instrumentation diagrams, SAR Chapters 3.0, 4.0, and 5.0, and ANO-1 Upper Level Documents. RCS mechanical components subject to aging management review include B&W designed vessels (i.e., reactor vessel and control rod drive mechanism pressure boundary, pressurizer, and once-through steam generators) and reactor vessel internals, reactor coolant pumps, and Class 1 piping and valves.

ANO-1 Class 1 piping includes B&W-supplied piping (i.e., main coolant, pressurizer surge, pressurizer spray, and incore monitoring system) and Bechtel-supplied piping (i.e., vents, drains, instrumentation lines, and Class 1 portions of ancillary systems attached to the B&W scope of supply and the reactor coolant pumps). Ancillary systems include decay heat/low pressure injection, core flood, high pressure injection/makeup and purification, and reactor building isolation. Within the Bechtel-supplied piping attached to the B&W scope of supply, the Class 1 RCS boundary extends to the second isolation valve with the exception of the pressurizer code safety valves.

Other RCS components within the scope of license renewal include non-Class 1 instrumentation tubing, piping, valves, and the letdown coolers.

The B&W-supplied vessels were designed in accordance with ASME Section III Class A (Reference 2.3-2), 1965 Edition, with Addenda through the Summer of 1967. Reactor coolant pumps were designed in accordance with ASME Section III, 1968 Edition, with no Addenda; however, the pumps were not code stamped. The RCS piping supplied by B&W and Bechtel was designed to Nuclear Piping Code USAS B31.7 (Reference 2.3-3), dated February 1968 and as corrected for Errata under date of June 1968. Subsequent to the original design, modifications to the RCS can be made utilizing later appropriate ASME Section III Code sections if they have been reconciled. The design of the reactor vessel internals meets the intent of ASME Section III with qualification of the design accomplished through a combination of analysis and testing. The letdown coolers were designed in accordance with ASME Section III Class C for the tube side and ASME Section VIII (Reference 2.3-4) for the shell side.

Component intended functions have been determined based on a review of the ANO-1 SAR and design documents. Components within the boundary of the RCS that perform their intended functions without moving parts or without a change in configuration or properties are listed in Table 3.2-1, along with the intended functions they must maintain. Entergy Operations actively participated in a BWOG effort that developed a series of topical reports whose purpose was to demonstrate that the aging effects for RCS components are adequately managed for the period of extended operation. The following is a list of the BWOG topical reports applicable to the RCS at ANO-1 that have been approved by the NRC:

- BAW-2243A, Reactor Coolant System Piping (Reference 2.3-5)
- BAW-2244A, Pressurizer (Reference 2.3-6)
- BAW-2251A, Reactor Vessel (Reference 2.3-7)
- BAW-2248A, Reactor Vessel Internals (Reference 2.3-8)

NRC approved reports may be incorporated by reference pursuant to 10CFR54.17(e) provided the conditions of approval contained in the safety evaluation of the specific report are met. These reports have been incorporated by reference into the ANO-1 LRA as discussed in Section 2.3.1.2. Each of the components of the RCS that are subject to aging management review are described in Sections 2.3.1.3 through 2.3.1.9.

2.3.1.2 Process to Incorporate Approved BWOG Topical Reports by Reference

Entergy Operations used the following process to incorporate approved BWOG topical reports by reference into the ANO-1 LRA.

(1) *Comparison of the component intended functions for the RCS components under review***.** The ANO-1-specific component screening review first identifies the component intended functions and then compares these functions to those identified in the generic BWOG topical reports. Differences are noted and justification for the variances provided.

(2) *Identification of the items that are subject to aging management review.* ANO-1 drawings and pertinent design and field change data are reviewed. The process establishes the full extent to which the scope of the generic BWOG topical reports bound the ANO-1 RCS components.

(3) Identification of the applicable aging effects. An independent assessment of the applicable aging effects is performed by reviewing plant operating environment, operating stresses (qualitative), and plant-specific operating experience. This reveals potential aging effects not identified in the generic BWOG topical reports. Aging effects for items that are determined to be subject to aging management review that were not identified in the generic BWOG topical reports are evaluated.

The results of steps (1) and (2) are provided in Sections 2.3.1.3 through 2.3.1.9, while the results of step (3) are provided in Section 3.2.

2.3.1.3 Reactor Coolant System Piping

For ANO-1, the following components are within the reactor coolant pressure boundary: reactor vessel, once-through steam generators (primary side), pressurizer, reactor coolant pump, main coolant piping and portions of systems attached to these components. The attached systems that contain Class 1 components include the core flood system, makeup/high pressure injection system, and decay heat/low pressure injection system. In addition, vents, drains, and instrumentation lines also contain Class 1 components. RCS piping includes piping (including fittings, branch connections, safe ends, and thermal sleeves), pressure retaining parts of RCS valves, and bolted closures and connections. Additional descriptions of the RCS piping are contained in SAR Section 4.2.2.4.

Non-Class 1 portions of the systems attached to the RCS are discussed in the following sections.

Section 2.3.2.1-Core Flood

Section 2.3.2.2-Low Pressure Injection/Decay Heat

Section 2.3.2.3-High Pressure Injection/Makeup and Purification

Section 2.3.2.7-Reactor Building Isolation

As noted in Section 2.3.1.1, BWOG topical report BAW-2243A has been approved by the NRC for use by applicants for a renewed operating license. As a result of NRC review of this report, several renewal applicant action items were identified. These action items are described in Section 4.1 of the safety evaluation issued by the NRC regarding BAW-2243A. The ANO-1 specific responses to these action items relevant to the identification of RCS piping components subject to aging management review are provided in Table 2.3-2.

As summarized in Section 2.3.1.1, Entergy Operations participated in the development of BAW-2243A by providing ANO-1 specific design and operational information.

Entergy Operations has reviewed the current design and operation of the ANO-1 RCS piping using the process described in Section 2.3.1.2 and confirms that the ANO-1 Class 1 piping is bounded by the description of Class 1 piping contained in BAW-2243A with regard to materials and operating environment. ANO-1 specific findings regarding RCS piping are contained in the following paragraphs.

The fast response RTE connections include a thermowell mounted within the mounting boss. The thermowell, which is constructed from Type 304 austenitic stainless steel, was omitted from the scope of BAW-2243A. In addition, the evaluation boundary in BAW-2243A did not include non-Class 1 instrumentation tubing that connects the second isolation valve to the instrumentation. These items are part of the RCS pressure boundary at ANO-1, are constructed from austenitic stainless steel, and are evaluated in Section 3.2. Other items that were not within the scope of BAW-2243A include the letdown coolers and reactor vessel leakage monitoring pipes that are connected to the reactor vessel.

Letdown Coolers

The letdown coolers are not within the scope of BAW-2243A but are subject to aging management review at ANO-1. The letdown coolers are heliflow shell and tube heat exchangers with spiral Type 304 stainless steel tubes and manifolds, carbon steel casing shells, and carbon steel casing end plates. The tube side was designed in accordance with ASME Section III, Class C, and the shell side was designed in accordance with ASME Section VIII. The primary water enters the tubes at approximately 555°F during normal plant operation and is cooled to approximately 120°F by intermediate cooling water (treated water) flowing through the shell. The intermediate cooling water enters at approximately 95°F and exits at less than 175°F. Both coolers are in service during normal plant operation with a relatively constant intermediate cooling water flow rate. The total letdown flow rate that is split between the coolers is manually varied anywhere between 45 and 140 gpm as required for RCS inventory control. The letdown flow through the coolers may be manually or automatically terminated.

Reactor Vessel Leakage Monitoring Piping

The 1-inch, schedule 160, reactor vessel leakage monitoring system piping attached to the reactor vessel is Class 3 at ANO-1. The lines do not support the RCS pressure boundary and were not in the scope of BAW-2243A (RCS Piping Report) or BAW-2251A (Reactor Vessel Report). If the reactor vessel closure flange O-rings fail and RCS fluid is introduced into the monitoring piping, leak flow would be limited since the 1/2-inch diameter hole in the vessel flange, which connects the region between the O-rings to the monitoring pipe, is less than the ID of the monitoring pipe. Therefore, the reactor vessel leakage monitoring piping is not subject to aging management review since the piping does not directly support the RCS pressure boundary.

2.3.1.4 Pressurizer

The pressurizer is a vertical cylindrical vessel with a surge line penetration connecting the surge line to the hot leg piping. The pressurizer contains electric heaters in its lower section and a water spray nozzle in its upper section. Since sources of heat in the RCS are interconnected by piping with no intervening isolation valves, relief protection is provided on the pressurizer. Overpressure protection consists of two code safety valves and one power operated relief valve.

Piping attached to the pressurizer is Class 1 up to and including the second isolation valve (with the exception of the pressurizer code safety valve) and is discussed in Section 2.3.1.3. Additional descriptions of the ANO-1 pressurizer are contained in SAR Section 4.2.2.3 and BAW-2244A. The pressurizer is shown on SAR Figure 4-14.

As noted previously in Section 2.3.1.1, BWOG topical report BAW-2244A has been approved by the NRC for use by applicants for a renewed operating license. Entergy Operations has reviewed the current design and operation of the ANO-1 pressurizer using the process described in Section 2.3.1.2 and has confirmed that the ANO-1 pressurizer is bounded by the description contained in BAW-2244A. As a result of NRC review of BAW-2244A, several renewal applicant action items were identified. These Action Items are described in Section 4.1 of the safety evaluation issued by the NRC concerning BAW-2244A (Reference 2.3-6). The ANO-1 specific responses to the renewal applicant action items relevant to the pressurizer are provided in Table 2.3-3.

2.3.1.5 Reactor Vessel

The reactor vessel consists of the cylindrical vessel shell, lower vessel head, closure head, nozzles, interior attachments, and associated pressure retaining bolting. Coolant enters the reactor through the inlet nozzles, passes down through the annulus between the thermal shield and vessel inside wall, reverses at the lower head, passes up through the core, turns around through the plenum assembly, and leaves the reactor vessel through the outlet nozzles. The ANO-1 reactor vessel is shown on SAR Figure 4-4.

The reactor vessel has two outlet nozzles through which the coolant is transported to the steam generators and four inlet nozzles, through which coolant enters the reactor vessel from the discharge of the reactor coolant pumps. Two smaller nozzles located between the inlet nozzles serve as inlets for decay heat removal and emergency core cooling water injection. Instrumentation nozzles penetrate the lower vessel head. Piping attached to the reactor vessel is discussed in BAW-2251A (Reference 2.3-7) and covered in Section 2.3.1.3. The reactor vessel support skirt and control rod drive service structure are addressed in Section 2.4.2.1.

Control rod drive mechanisms are attached to flanged nozzles, which penetrate the closure head. The active portions of the control rod drive mechanisms are not within the scope of license renewal; however, the control rod drive motor tube assemblies and closure insert and vent assemblies are subject to aging management review and are discussed in Section 2.3.1.9. One of the ANO-1 CRDMs was removed to install a reactor vessel level monitoring probe. The reactor vessel level monitoring probe is discussed in Section 2.3.1.6. Additional descriptions of the reactor vessel are contained in SAR Section 4.2.2.1 and BAW-2251A.

As noted previously in Section 2.3.1.1, BWOG topical report BAW-2251A has been approved by the NRC for use by applicants for a renewed operating license. Entergy Operations has reviewed the current design and operation of the reactor vessel using the process described in Section 2.3.1.2 and has confirmed that the ANO-1 reactor vessel is bounded by the description contained in BAW-2251A.

As a result of NRC review of BAW-2251A, several renewal applicant action items were identified. These action items are described in Section 4.1 of the safety evaluation issued by the NRC concerning BAW-2251A (Reference 2.3-7). The ANO-1 specific responses to the renewal applicant action items relevant to the reactor vessel are provided in Table 2.3-4.

2.3.1.6 Reactor Vessel Internals

The reactor vessel internals consist of two structural subassemblies located within the reactor vessel. These two subassemblies of the internals are the plenum assembly and the core support assembly. The reactor vessel internals can be removed during refueling outages when necessary. Descriptions of the reactor vessel internals for ANO-1 are contained in BAW-2248A and in SAR Section 3.2.4.1. The reactor vessel internals are shown on SAR Figure 3-59.

Entergy Operations has reviewed the current design and operation of the ANO-1 reactor vessel internals using the process described in Sections 2.3.1.1 and 2.3.1.2 and has determined that the ANO-1 internals have three additional intended functions that were not listed in BAW-2248A:

- supporting the reactor vessel level monitoring probe,
- providing gamma and neutron shielding, and
- providing support for the surveillance specimen assemblies in the annulus between the thermal shield and the reactor vessel wall.

One of the ANO-1 CRDMs was removed and the control rod guide assembly in the plenum was modified to accept the reactor vessel level monitoring probe. Support of the monitoring probe is an additional intended function of the reactor vessel internals. The items that support the reactor vessel level monitoring probe are fabricated from Type 304L austenitic stainless steel and are evaluated in Section 3.2.5.

The thermal shield, thermal shield upper restraint and associated bolting, which are all fabricated from austenitic stainless steel, support the intended function “provide gamma and neutron shielding.” These items are subject to aging management review and are evaluated in Section 3.2.5.

In addition, portions of the ANO-1 surveillance specimen holder tubes are attached to the internals. Although all the specimens have been removed, portions of the shroud tube and the supports that are bolted to the core support shield remain. These items only have the function of remaining secured to prevent loose parts in the RCS. This function will be considered applicable to the remaining portions of the surveillance specimen holder tubes. These items are evaluated in Section 3.2.5.

As a result of NRC review of BAW-2248A, several renewal applicant action items were identified. The BAW-2248A applicant action items are described on Section 4.1 of the safety evaluation issued by the NRC concerning BAW-2248A (Reference 2.3-8). The

ANO-1 specific responses to the renewal applicant action items relevant to the reactor vessel internals are provided in Table 2.3-5.

2.3.1.7 Once-Through Steam Generators

ANO-1 has two once-through steam generators. Each is a vertical, straight-tube, once-through, counterflow, shell-and-tube heat exchanger with shell-side boiling. The steam generator consists of upper and lower hemispherical heads welded to tubesheets that are separated by a shell assembly. Over 15,000 straight Alloy 600 tubes are held in alignment by fifteen tube support plates. The once-through steam generator is shown on SAR Figure 4-5.

Primary coolant from the reactor enters the steam generator through a single inlet nozzle in the top of the upper head. Coolant flows downward through the straight parallel tubes, is cooled by the secondary coolant on the shell side, and then exits through two outlet nozzles in the lower head. Secondary coolant enters through a ring of ports that penetrate the shell approximately midway up the shell assembly. The feedwater travels downward through an annulus between the lower baffle and the shell. Near the lower tubesheet, the feedwater turns inward and then flows upward around the tubes and through the tube support plates. As the feedwater absorbs heat from the primary coolant, it boils and then becomes superheated. The dry steam exits the steam generator through two steam outlet nozzles just above the feedwater inlet ports.

The intended functions of the once-through steam generators include maintaining primary pressure boundary, maintaining secondary pressure boundary, providing heat transfer from the primary fluid to the secondary fluid, and reactor building isolation. Once-through steam generator items that are subject to aging management review include the hemispherical heads, secondary shell, tubes, plugs, mechanical sleeves, tubesheets, primary nozzles, primary manway and inspection port assemblies, main and auxiliary feedwater nozzles, main and auxiliary feedwater header and riser piping, steam outlet nozzles, instrumentation nozzles, temperature sensing connections, drain nozzles, secondary manway and inspection port covers, associated pressure retaining bolting, and integral attachments inspected in accordance with ASME Section XI (Reference 2.3-9), Subsections IWB and IWC. Class 1 RCS piping attached to the primary once-through steam generator nozzles, including the welded joints, is addressed in Section 2.3.1.3. Secondary piping attached to the once-through steam generator nozzles, including the main and auxiliary feedwater headers and riser piping, is addressed in Section 2.3.4.2. The steam generator supports are addressed in Section 2.4.2.

Once-through steam generator items that do not support an intended function and that are not subject to aging management review include weld deposit pads on the external shell of the generator that are used for insulation supports, shell thermocouples, and grounding lugs; an internal support ring that is attached to the inside shell of the secondary side, secondary internal baffles, support plates, variable orifice plate, and tube stabilizers; and gaskets used in bolted connections at manways inspection ports, and main and auxiliary feedwater inlet piping.

Once-through steam generator items fabricated from low-alloy steel include the hemispherical heads, transition ring, tubesheets, and pressure retaining bolting.

Items fabricated from carbon steel include primary inlet and exit nozzles, secondary shell, secondary outlet nozzles, main and auxiliary feedwater header and riser piping, primary and secondary manway covers, primary and secondary inspection port covers, secondary vent nozzles, drain nozzles, level sensing nozzles, and main and auxiliary feedwater nozzles. Items fabricated from Alloy 600 include the primary drain nozzle, nozzle dam support rings, tubes, plugs, sleeves, and secondary temperature sensing connections. The once-through steam generators were designed as Class A vessels in accordance with ASME Section III, 1965 Edition, with Addenda through Summer of 1967, (Reference 2.3-2)

2.3.1.8 Reactor Coolant Pumps

The reactor coolant pumps propel the reactor coolant through the reactor core, piping, and steam generators. The four reactor coolant pumps are required during normal full power operation. The four reactor coolant pumps installed at ANO-1 are Byron-Jackson pumps. The reactor coolant pumps were designed, fabricated, tested, and inspected as Class A vessels in accordance with ASME Section III 1968 Edition. The RCPs were not code stamped.

The intended function of the reactor coolant pumps is to maintain the RCS pressure boundary. The reactor coolant pump items that support the intended function and are subject to aging management review include the casing, cover, integral seal injection heat exchangers, and pressure-retaining bolting. Non-Class 1 piping, instrumentation, and other components attached to the reactor coolant pump are addressed in Section 2.3.2. Class 1 piping connected to the pump, including the welded joints is discussed in Section 2.3.1.3. The portion of the reactor coolant pump rotating element above the pump coupling, the electric motor, and the flywheel are not subject to aging management review in accordance with 10CFR54.21(a)(1).

The reactor coolant pump casings include not only the casings themselves, but also the bolted closures and connections. These are constructed of stainless steel, except for the pressure retaining bolting which is fabricated from low-alloy steel. The upper and lower halves of the Byron-Jackson pump casings are cast austenitic stainless steel.

The pump cover is a generic term used to describe the pressure-retaining closure to the pump casing. The cast austenitic stainless steel cover serves as a housing for the mechanical seal, radial bearing, thermal barrier, and recirculating impeller. The cover is clamped between the carbon steel driver mount and the stainless steel pump casing.

Bolting used to secure the cover to the case includes cover-to-case studs and nuts, which are fabricated from low-alloy steel. Bolting used to secure the seal housing and/or seal glands to the cover includes studs and nuts. These bolting materials are less than two inches in diameter and are fabricated from low-alloy steel.

Each reactor coolant pump is supported by the cold leg piping during all modes of operation. The weight of each reactor coolant pump motor is supported by two vertical constant load supports, which are addressed in Section 2.4.2.1. Additional descriptions of the ANO-1 reactor coolant pumps are contained in SAR Section 4.2.2.5. SAR Figure 4-7 is a drawing of the ANO-1 RCP design.

2.3.1.9 Control Rod Drive Mechanism Pressure Boundary

Control rod drive mechanism motor tube assemblies, closure insert assemblies, and vent assemblies provide the reactor coolant pressure boundary around the control rod drive mechanisms. During normal operation, the control rod drive mechanism motor tube assemblies are filled with borated reactor coolant at the system operating pressure. Thermal barriers in the lower motor tube mechanism and the control rod drive mechanism cooling system maintain the temperatures in the housings below system temperature.

Control rod drive mechanism motor tube assemblies were designed, fabricated, tested, and inspected in accordance with ASME Section III 1965 Edition and Summer 1967 Addendum. The material of construction is stainless steel or Alloy 82/182 clad low-alloy steel.

Two different designs of control rod drive mechanisms are currently in use at ANO-1, Type B and Type C. The control rod drive mechanisms themselves are active and not subject to aging management review for license renewal. The CRDM items subject to aging management review include the motor tube assemblies, closure insert and vent assemblies, associated bolting, and the Reactor Vessel Level Monitoring System adapter flange assembly.

2.3.2 Engineered Safeguards

ANO-1 refers to the engineered safety features as the engineered safeguards. Engineered safeguards consist of systems and components designed to function under accident conditions to minimize the severity of an accident or to mitigate the consequences of an accident. In the event of a loss of coolant accident, the ES provide emergency coolant to assure structural integrity of the core, to maintain the integrity of the reactor building, and to reduce the concentration of fission products expelled to the reactor building atmosphere. The engineered safeguards are described in SAR Chapter 6.

The following systems are included in this section:

- Core Flood
- Low Pressure Injection/Decay Heat
- High Pressure Injection/Makeup and Purification
- Reactor Building Spray
- Reactor Building Cooling and Purge (including reactor building heating and ventilation and portions of reactor building purge)
- Sodium Hydroxide (including chemical addition)
- Reactor Building Isolation
- Hydrogen Control (including hydrogen purge and hydrogen recombiners)

Scoping drawings, consisting of the piping and instrumentation diagrams for the ES systems, are listed in Table 2.3-6, and are provided as attachments to 1CAN010002 (Reference 2.3-10). These drawings have been highlighted to show the portions of the ES systems that are within the scope of license renewal. The boundaries indicated on the drawings are primarily based on the system designations in the component database. The use of the component database in conjunction with the system P&IDs assures that duplicate reviews and omission of components do not occur at the interface boundaries. The mechanical components and their intended functions for the ES systems are identified in Table 3.3-1 through Table 3.3-8.

2.3.2.1 Core Flood

The core flood system is described in SAR Sections 6.1.2.1.3 and 6.1.3.3. The safety function of the core flood system is to provide core cooling after intermediate and large break LOCAs. The core flood system components within the scope of license renewal and subject to aging management review include two core flood tanks and piping and components up to the reactor coolant system boundary. The intended function within the scope of license renewal is to maintain system pressure boundary integrity.

2.3.2.2 Low Pressure Injection/Decay Heat

The low pressure injection system is described in SAR Sections 6.1.3.2 and 6.1.2.1.2 while the decay heat system is described in SAR Section 9.5. The LPI/DH system is a dual-purpose system. This system operates as the DH system to remove decay heat from the core and sensible heat from the RCS during the latter stages of cooldown. The LPI system injects borated water into the reactor vessel to cool the core in the event of a LOCA.

The LPI system has the following safety functions.

- Inject borated water from the BWST during postulated large break LOCA
- Provide long term cooling following a LOCA by recirculating injection water from the reactor building sump
- Supply recirculated water from the reactor building sump to the suction of the high pressure injection pumps if RCS pressure is too high to allow the LPI pumps to function
- Supply injection water from the BWST to the DH/LPI pumps as well as the high pressure injection and the reactor building spray pumps (the BWST floods the reactor building basement to a level that will allow for recirculation from the reactor building sump under accident conditions)
- Provide water that is free of entrained air from the screened reactor building sump, when the BWST is depleted.

The decay heat system is credited in the 10CFR50.48 analysis with the capability of attaining cold shutdown. Therefore, the DH system has a function to remove decay heat from the reactor core and sensible heat from the RCS during the latter stages of cooldown such that fuel design limits and design conditions of the RCS pressure boundary are not exceeded. The DH system also supports the following functions.

- Circulating reactor coolant to prevent boron stratification and to minimize the effects of a boron dilution event
- Providing an alternate supply of borated water from the BWST for volume contraction during cooldown to cold shutdown
- Providing cooling, inventory addition, and instrumentation for loss of decay heat removal events

The following LPI/DH components are within the scope of license renewal and subject to aging management.

- The decay heat system piping that passes through the reactor building penetrations including the injection lines, the drop line, the pressurizer auxiliary spray line and the emergency sump lines (these portions of the system perform a reactor building isolation function and are within the scope of license renewal)
- The DH drop line valves, the DH coolers, the DH cooler isolation valves, and the decay heat pumps

- The BWST, the BWST supply header, and the injection lines up to the outboard RCS pressure boundary valve of the low-pressure injection lines, as well as the suction supply piping to the high pressure injection system
- The piping and components from the reactor building sump, including some piping and components that are part of the PASS system, which is used for post LOCA sump sampling (The sump screens and the vortex breakers are reviewed in Section 2.3.3.4.)
- The oil side of the LPI pump lube oil coolers (the service water side of the coolers is evaluated in Section 2.3.3.10).

The intended function within the scope of license renewal is to maintain the pressure boundary integrity. For the heat exchangers, heat transfer is a passive function within the scope of license renewal.

2.3.2.3 High Pressure Injection/ Makeup and Purification

The safety function of the high pressure injection system is to provide high pressure injection into the RCS during emergency conditions. This system is normally operated as part of the MUP system. During normal operations, the MUP system performs various functions in support of the RCS. The HPI system is described in SAR Sections 6.1.2.1.1 and 6.1.3, while the MUP system is described in SAR Section 9.1.

The HPI/MUP system has the following safety functions.

- Inject borated water from the BWST during postulated accidents such as the small break LOCA
- Provide long term cooling following small break LOCAs by recirculating injection water from the reactor building sump

The HPI/MUP system is credited in the 10CFR50.48 analysis with the capability for RCS makeup and pressure control. Some of the system valves must remain closed to prevent a direct RCS leak path in the event of a fire. The HPI/MUP system also supports the following functions.

- Provide inventory to the RCS during operational transients such as reactor trips and overcooling events
- Provide a backup inventory supply to the RCS during a loss of decay heat removal event
- Provide core cooling following a total loss of feedwater event via feed and bleed cooling of the RCS
- Provide an auxiliary means to spray the pressurizer steam space when normal spray is not available

The following HPI/MUP components are within the scope of license renewal and subject to aging management.

- The seven mechanical reactor building penetrations necessary for meeting reactor building isolation requirements

- The Class 1 RCS pressure boundary that extends to the second isolation valve off of the RCS (For the letdown line, this is downstream of the letdown coolers. The letdown coolers and the Class 1 valves are reviewed in Section 2.3.1.3.)
- The HPI piping from the BWST supply header up to the outboard RCS pressure boundary valve of the injection lines, and all portions of the system needed to support high pressure injection, including the suction supply from the low pressure injection system
- The oil-side of the HPI pump oil coolers (the service water side of the coolers is evaluated in Section 2.3.3.10.)

The intended function within the scope of license renewal is maintaining system pressure boundary integrity. For the heat exchangers, heat transfer is a passive function within the scope of license renewal.

2.3.2.4 Reactor Building Spray

The reactor building spray system is described in SAR Section 6.2. The safety function of the RBS system is to reduce reactor building pressure following accidents that pressurize the reactor building. This reduction in pressure reduces the driving force for leakage of radioactive materials from the reactor building following a LOCA. The RBS system also reduces the concentration of fission products in the reactor building atmosphere following a LOCA.

The components within the scope of license renewal and subject to aging management review consist of both trains of pumps and supporting equipment (lube oil coolers and seal water cyclone separators), piping, valves, and spray headers. The interfacing systems that form part of the RBS system pressure boundary are also included. This includes the interfacing valves from the "spared in place" sodium thiosulfate tank, the interfaces with the service air system, and the vents and drains off the RBS pump casings. The sodium hydroxide system is covered in Section 2.3.2.6.

The intended function within the scope of license renewal is to maintain RBS system pressure boundary integrity. For the bearing heat exchangers, heat transfer is a passive function within the scope of license renewal.

2.3.2.5 Reactor Building Cooling and Purge

The reactor building cooling and purge system is described in SAR Section 5.2.6. The safety function of the reactor building cooling system is to reduce the post-accident pressure and temperature in the reactor building and provide mixing of the reactor building atmosphere following a loss of coolant accident. During normal plant operation, the system must maintain the reactor building temperature below the maximum allowed for the equipment and below the accident analyses initial temperature assumptions. The reactor building purge has no safety function; however, the penetrations in this system must maintain reactor building integrity under accident conditions.

The following components are within the scope of license renewal and subject to aging management review.

- The four safety-related reactor building coolers
- The service water cooling coils, the fan/cooler housings and the discharge duct work, including the duct relief valves that prevent damage to the ductwork during a rapid building pressurization
- The reactor building isolation valves and piping at the two penetrations in the reactor building purge system

The intended function within the scope of license renewal is to maintain integrity. For the heat exchangers, heat transfer is a passive function within the scope of license renewal.

2.3.2.6 Sodium Hydroxide

The safety function of the sodium hydroxide system is to provide a solution of sodium hydroxide to the ECCS suction headers. The increased pH improves iodine absorption and retention in the water, thereby minimizing the gaseous iodine and the offsite dose following a LOCA. The sodium hydroxide system is described in SAR Section 6.2. The components within the scope of license renewal and subject to aging management review are the sodium hydroxide tank and associated piping and components from the tank to the ECCS suction headers that are required to maintain the pressure boundary. The intended function within the scope of license renewal is to maintain the system pressure boundary integrity.

2.3.2.7 Reactor Building Isolation

The reactor building isolation system is described in SAR Section 5.2.5. A safety function of the reactor building isolation system is to allow passage of required fluids, materials, personnel, electrical signals, and electrical power across the reactor building boundary. In addition, the reactor building isolation system seals penetrations that are not required for operation to provide a fission product barrier between the inside of the reactor building and the outside environment. This capability is also required for the nonsafety-related systems that penetrate the reactor building. Therefore, the penetrations have a reactor building isolation function in addition to their system function.

The components within the scope of license renewal and subject to aging management review are the 20 mechanical penetration components and piping that are not included in other sections of the ANO-1 LRA. These penetrations involve the following.

- Intermediate cooling water, nitrogen, breathing air, plant heating, and gaseous radwaste
- Core flood: tank sampling and makeup and nitrogen pressurization
- Sampling system: steam generator secondary sampling and quench tank sampling
- Condensate storage and transfer: condensate transfer supply to quench tank
- Liquid radwaste: quench tank drain

- Heater vents system: steam generator secondary drains
- Integrated leak rate test connection

The intended function within the scope of license renewal is to maintain system pressure boundary integrity.

2.3.2.8 Hydrogen Control

The hydrogen control system is described in SAR Section 6.6. The safety function of the hydrogen control system is to provide a direct reading of the hydrogen concentration in the reactor building using the hydrogen analyzer system and to reduce the hydrogen concentration following a LOCA using the hydrogen recombiner system.

The hydrogen control system components within the scope of license renewal and subject to aging management review are the penetrations, the passive mechanical components of the hydrogen samplers and the piping to and from the hydrogen samplers. The piping to the analyzers uses a portion of the hydrogen purge system, and one of the boundary valves is in the gas collection header system. The passive mechanical components of the hydrogen recombiners are also in the scope of license renewal and subject to aging management review. The control power cabinets in the penetration room and the passive electrical components of the recombiners are addressed in Section 2.5.

The intended function within the scope of license renewal is to maintain the system pressure boundary integrity. For the heat exchangers, heat transfer is a passive function within the scope of license renewal.

2.3.3 Auxiliary Systems

The following systems are included in this section.

- Spent fuel
- Fire protection
- Emergency diesel generator
- Auxiliary building sump and reactor building drains
- Alternate AC diesel generator
- Halon
- Fuel oil
- Instrument air
- Chilled water
- Service water
- Penetration room ventilation
- Auxiliary building heating and ventilation
- Control room ventilation

Scoping drawings, consisting of the piping and instrumentation diagrams for the auxiliary systems, are listed in Table 2.3-7, and are provided as attachments to 1CAN010002 (Reference 2.3-10). These drawings have been highlighted to show the portions of the auxiliary systems within the scope of license renewal. The boundaries indicated on the drawings are primarily based on the system designations in the component database. The use of the component database in conjunction with the system P&IDs assures that duplicate reviews and omission of components do not occur at the interface boundaries. The mechanical components and their intended functions for the systems in this section are identified in Table 3.4-1 through Table 3.4-13.

2.3.3.1 Spent Fuel

The spent fuel system is described in SAR Section 9.4.1. The safety functions of the spent fuel system are to maintain an adequate water level in the pool for cooling and shielding and to maintain the subcritical margin. The emergency makeup supply to the spent fuel pool from the service water system does not utilize the spent fuel pool cooling system piping and is therefore, reviewed with the service water system in Section 2.3.3.10.

The spent fuel pool stainless steel liner and the spent fuel pool gates are in the scope of license renewal and subject to aging management review. The liner protects the concrete walls from direct contact with the borated water and maintains the leak tightness of the pool.

The intended function for the spent fuel pool liner within the scope of license renewal is to maintain the system pressure boundary integrity.

Other spent fuel system components within the scope of license renewal and subject to aging management review are the following.

- The spent fuel racks, which support the spent fuel assemblies
- A mechanical reactor building penetration used for filling and draining of the fuel transfer canal (The pipe and valves are addressed in this section, while the penetration assembly is addressed in Section 2.4.1.1)
- The fuel transfer tube, which is a reactor building penetration (A blind flange is installed on the tube in the reactor building to ensure reactor building integrity is maintained during power operation. The fuel transfer tube and the blind flange are addressed in this section, while the penetration assembly is addressed in Section 2.4.1.1)
- The Boraflex neutron absorber material, which helps to maintain the subcritical margin (Boraflex is addressed in Section 4.7)

2.3.3.2 Fire Protection

The fire protection system is described in SAR Section 9.8.1. The safety function of the fire protection system is to minimize the effects of fires on structures, systems, and components important to safety as required by 10CFR Part 50 Appendix A, General Design Criteria 3. In accordance with 10CFR Part 54, the components required for compliance with 10CFR50.48 are in the scope of license renewal.

The following fire protection system components are within the scope of license renewal and subject to aging management review.

- The electric motor driven fire pump
- The diesel driven fire pump, including the engine gearbox oil cooler, the jacket water heat exchanger and the lube oil cooler (The fuel oil portions of the system are covered in Section 2.3.3.7.)
- The fire water distribution system including the portion of the outside loop, hose stations, standpipes, sectional control valves, and isolation valves that are required for protection of safety-related areas (Although the hose stations are included, the hoses are not normally part of the pressure boundary and are therefore outside of the scope of license renewal.)
- Sprinkler systems protecting safety-related areas, including piping, control valves, and sprinkler heads
- Sprinkler systems required to meet 10CFR50.48 requirements, including piping, control valves, and sprinkler heads

The intended function within the scope of license renewal is to maintain the system pressure boundary integrity. The diesel fuel pump lube oil coolers also have the passive component function of heat transfer to cool the oil.

2.3.3.3 Emergency Diesel Generator

The emergency diesel generators are described in SAR Section 8.3.1.1.7. The safety function of the EDGs is to supply the engineered safeguards bus loads following a design basis accident. The EDGs are also required to be available following a fire and are considered components required to comply with 10CFR50.48.

The following EDG system components are within the scope of license renewal and subject to aging management review .

- Safety-related portions of the EDG starting air subsystem from the receivers to the EDG assembly
- EDG lubrication subsystem components
- EDG combustion air intake and exhaust subsystem components
- EDG cooling water subsystem components
- The fuel oil system including the EDG fuel oil components will be evaluated in Section 2.3.3.7
- The service water side of the EDG heat exchangers is addressed in Section 2.3.3.10

The intended function within the scope of license renewal is to maintain pressure boundary integrity. Heat transfer is also a function in the scope of license renewal for the heat exchangers.

2.3.3.4 Auxiliary Building Sump and Reactor Building Drains

The ANO-1 auxiliary building sump and reactor building drain systems consist of the floor and equipment drains, piping, valves, sumps and tanks that collect liquids from the reactor building and auxiliary building for processing or disposal.

The following are the safety-related functions of the auxiliary building sump and reactor building drains:

- The reactor building penetrations contain the radioactivity in the reactor building following a LOCA. The penetrations are in the reactor building sump and reactor building drain systems.
- The screens on the reactor building sump and floor drains prevent debris from entering the reactor building sump and interfering with recirculation post LOCA. The screens are in the dirty water drain system.
- The anti-vortex device on the reactor building sump prevents vortexing that could occur under accident conditions when the reactor building is flooded and recirculation of the reactor building water is underway. The anti-vortex device is part of the reactor building sump system.
- The decay heat pump room drains and isolation valves ensure the radioactive liquids that could be present in the decay heat room following a LOCA do not

escape to other portions of the auxiliary building. These drains are in the auxiliary building sump system.

- The reactor coolant pump motor oil leakage collection tanks, piping and valves collect oil leakage from the reactor coolant pump motors in order to reduce the chance of a fire. This function is not safety-related, but is required by 10CFR50.48, and does require pressure boundary integrity to prevent a fire involving leaking oil. These components are part of the reactor coolant system.

The following auxiliary building sump and reactor building drain system components are within the scope of license renewal and subject to aging management review.

- Mechanical components associated with penetrations that are required for reactor building isolation
- Reactor building sump inlet and anti-vortexing screen and the individual screens on the floor drains that drain into the reactor building sump that prevent debris from entering the sump and interfering with recirculation post LOCA. (The concrete sump and the structural steel for the sump including the screen structural steel supports and sump divider plate are evaluated as structural components)
- The valves and piping that isolate the decay heat pump rooms, which are credited as part of the room pressure boundary for offsite dose calculations
- The reactor coolant pump motor oil leakage collection tanks and piping that are specifically required by 10CFR50.48

The intended function within the scope of license renewal is to maintain the system pressure boundary integrity.

2.3.3.5 Alternate AC Diesel Generator

The alternate AC generator is a 4400 kW diesel generator installed in response to the regulatory requirements of 10CFR50.63, "Loss of All Alternating Current Power," to provide backup power in the event of a station blackout at ANO-1 or ANO-2.

The AAC generator system is required by 10CFR50.63, but it does not have a safety-related function. The AAC generator system is credited for providing power during a loss of off-site power concurrent with a loss of the EDGs (i.e., station blackout).

The following AAC generator system components are within the scope of license renewal and subject to aging management.

- Portions of the pressure boundary of the AAC generator starting air subsystem including the receivers to the AAC generator assembly
- The AAC generator combustion air intake and exhaust subsystems
- Components required to maintain the pressure boundary of the AAC generator cooling water subsystem
- Components required to maintain the pressure boundary of the AAC generator lubrication subsystem

- The two engine room exhaust fans that provide the necessary cooling when the engine is in service and the corresponding inlet air dampers
- An exhaust fan that provides the necessary cooling for the switchgear room and the corresponding inlet air damper

The AAC generator building is covered in Section 2.4.6.1

The fuel oil subsystem components are evaluated in Section 2.3.3.7

The intended function within the scope of license renewal is to maintain the system pressure boundary integrity. For the heat exchangers, heat transfer is also a passive function within the scope of license renewal.

2.3.3.6 Halon

The halon system is described in SAR Section 9.8.2. The ANO-1 halon fire system equipment provides fire protection for the ceiling and false floor of the ANO-1 control room as required by 10CFR50.48. Most of the halon system is within the scope of license renewal and subject to aging management review. Specifically, the halon cylinders, actuation valves, pilot piping, manual actuator cylinders and valves, discharge piping, and outlet nozzles are within scope. The passive electrical portions of the system are evaluated in Section 2.5. The bottle racks, supports for the system, ceiling tiles, marinite boards, concrete walls, concrete and false floor components which are required to enclose these areas and allow the effective use of halon, are addressed in Section 2.4.3. The intended function within the scope of license renewal is to maintain the system pressure boundary integrity.

2.3.3.7 Fuel Oil

The fuel oil system is described primarily in SAR Section 8.3.1.1.7.2. The safety function of the fuel oil system is to store and supply fuel oil to the diesel-driven components. The emergency diesel fuel tanks and the EDG day tank have the safety-related function of storing and supplying the emergency diesel generators with fuel oil. The bulk fuel oil storage tank has the nonsafety-related function of storing and supplying fuel oil to nonsafety-related equipment, including the AAC generator and the diesel fire pump day tanks that are in the scope of license renewal. The AAC day tank has the function of storing and supplying fuel oil to the AAC diesel generator in accordance with station blackout commitments. The diesel fire pump day tank function is to store and supply fuel oil to the fire protection diesel as required to satisfy 10CFR50.48 requirements.

The EDG fuel tanks, EDG day tanks, and the safety-related equipment and piping that supports the transfer of fuel to the EDG are within the scope of license renewal and subject to aging management review. Since the bulk fuel oil storage tank has been credited with an extended (4.5 day) supply for the alternate AC generator, the bulk fuel oil storage tank, the AAC diesel generator day tank, and the equipment and piping that supports the transfer of fuel to the AAC diesel generator injectors are included. The diesel driven fire pump day tank and the equipment and piping that supports the transfer of fuel to the diesel fire pump are also included since they are required to satisfy 10CFR50.48.

The intended function within the scope of license renewal is to maintain the system pressure boundary integrity. For the heat exchangers, heat transfer is also a passive function within the scope of license renewal.

2.3.3.8 Instrument Air

The instrument air system is described in SAR Section 9.9. The instrument air system is designed to provide a reliable supply of dry, oil-free, compressed air for pneumatic equipment operation. Although the majority of the instrument air system is not safety-related, some safety-related components utilize instrument air for operation of their pneumatic actuators.

The safety-related components that utilize instrument air were reviewed under the scope of license renewal. These components are in the instrument air system as well as other systems. Since many components are designed to fail to the desired post accident condition upon loss of air supply, the integrity of the pressure boundary is not required. However, pressure boundary integrity is required for the following components and thus these components are subject to aging management review.

- The portions of the instrument air system that are part of the reactor building penetration for the instrument air supply to the reactor building, since maintaining the pressure boundary integrity of these components is required for the penetration to perform its intended function
- The instrument air supply to the intermediate cooling water supply valve for the RCP motor air and lube oil coolers (The cooling water supply valve provides reactor building isolation and is provided with a double acting pneumatic cylinder that requires air pressure to reposition the valve. An accumulator is provided in the event of a loss of instrument air. Maintaining the pressure boundary integrity of the accumulator and components between the accumulator and the valve actuator is required.)
- The instrument air supply to the intermediate cooling water supply and return valves for the letdown coolers and RCP seal coolers (These valves provide reactor building isolation and are provided with double acting pneumatic cylinders that require air pressure to reposition the valves. Accumulators are provided in the event of a loss of instrument air, and maintaining the pressure boundary integrity of the accumulators and components between the accumulators and the valve actuators is required.)
- The outside air dampers for the two emergency fan filter units in the control room ventilation system have a function to reposition. High pressure carbon dioxide bottles are provided that can be manually selected as the pneumatic supply for these dampers. Maintaining the pressure boundary integrity is a required function for the carbon dioxide bottles and the portion of the system from the carbon dioxide bottles to the valve actuators

The intended function within the scope of license renewal is to maintain the system pressure boundary integrity for those portions of the system where pressure boundary integrity is required for the component to perform its intended safety function.

2.3.3.9 Chilled Water

The chilled water systems provide chilled water to the cooling coils of a variety of room and area ventilation units. The auxiliary building electrical room emergency chillers are the only safety-related chillers within the scope of license renewal. These chillers have the safety-related function to supply chilled water for emergency cooling to coolers that service the safety-related electrical equipment located in the auxiliary building electrical equipment rooms.

Two emergency chillers, the internal surfaces of the six cooling coils supplied by the chillers, as well as the associated valves and piping are subject to aging management review.

The fan/coil housing assemblies, the external surfaces of the cooling coils, the ductwork and fire dampers in the ductwork are covered in Section 2.3.3.12.

The main chilled water system reactor building penetrations piping and valves, are also included since they provide a safety function of reactor building isolation.

The intended function within the scope of license renewal is to maintain the system pressure boundary integrity. For the coolers, heat transfer is also a passive function within the scope of license renewal.

2.3.3.10 Service Water

The service water system is described in SAR Section 9.3. The safety function of the service water system is to transfer heat from safety-related components to an ultimate heat sink, (Lake Dardanelle or the emergency cooling pond). The safety-related service water system provides the emergency supply of water to the emergency feedwater pumps and the spent fuel pool. The service water system is credited in the fire analysis and is required to meet the requirements of 10CFR50.48. If the source of water from Lake Dardanelle is lost, the emergency cooling pond can supply the water to the intake structure for use by the fire pumps.

All passive, long-lived safety-related components and piping in the service water system are within the scope of license renewal and subject to aging management review. In addition, the piping to and from the emergency cooling pond and the sluice gates are included. The intake structure and the emergency cooling pond are evaluated in Sections 2.4.4 and 2.4.5. The individual coolers and heat exchangers supplied by the service water system are reviewed in conjunction with the system being cooled. The service water side of each cooler is evaluated in this section. The interfacing side of each cooler is evaluated in its applicable section. The service water system has four mechanical reactor building penetrations that must meet reactor building isolation requirements. The pipe and valves are reviewed in this section, while the penetration assembly is addressed in Section 2.4.1.1.

The intended function of the service water components and piping within the scope of license renewal is to maintain the service water system pressure boundary integrity. The heat exchangers also have the required function of heat transfer.

2.3.3.11 Penetration Room Ventilation

The penetration room ventilation system is described in SAR Section 6.5.2.1. The safety function of the penetration room ventilation system is to collect and process the radioactivity released to the penetration areas due to post LOCA reactor building leakage to assure the 10CFR Part 100 dose values are not exceeded. The components in the penetration room ventilation system within the scope of license renewal and subject to aging management review include safety-related portions of the system necessary to support emergency operation of the system. This includes the exhaust fans, the pre-filters, the HEPA filters, the adsorber filters, the ductwork and the dampers in the flow path, as well as the dampers that isolate the normal ventilation system. The penetration room floor drain check valves limit backflow into the rooms to aid the penetration room ventilation system in drawing a slight vacuum in these rooms under accident conditions. These drains are in the penetration room ventilation system and are assessed in Section 2.3.3.4.

The intended function within the scope of license renewal is to maintain the system pressure boundary integrity

2.3.3.12 Auxiliary Building Heating and Ventilation

The auxiliary building heating and ventilation system is described in SAR Section 9.7.2.1. The safety function of the auxiliary building heating and ventilation system is to provide a suitable environment for those areas of the auxiliary building which contain equipment requiring cooling post accident. Some of the fire dampers in the auxiliary building heating and ventilation system also have the function of closing in the unlikely event of a fire to meet 10CFR50.48 requirements.

The components within the scope of license renewal and subject to aging management review include safety-related portions of the system necessary to support emergency operation of the subsystems. This includes the ductwork, damper bodies, cooler housings, blower housings, and the components that maintain the system flow path for the subsystems that are safety-related. This includes the following:

- emergency diesel generator exhaust fans and inlet dampers
- auxiliary building electrical room unit coolers
- switchgear room unit coolers
- decay heat removal room unit coolers
- makeup pump room coolers

The auxiliary building doors, walls and piping penetrations are evaluated in Section 2.4.3. Although fire dampers are evaluated in Section 2.4.6.2, the fire dampers that form part of the pressure boundary for these systems are addressed in this section.

The intended function within the scope of license renewal is to maintain the system pressure boundary integrity. For the coolers (other than makeup pump room coolers), heat transfer is also a passive function within the scope of license renewal.

2.3.3.13 Control Room Ventilation

The control room ventilation system is described in SAR Section 9.7.2.1. The safety-related function of the control room ventilation system is to isolate the control room under accident conditions. The emergency unit coolers, emergency compressor/condensing units and the emergency fan/filter units have a safety-related function of providing a suitable environment for the control room operators and for equipment that requires cooling post accident. There are fire dampers and several temperature elements on the charcoal filters that are credited in the fire analyses, but the majority of the control room ventilation system components are not required by 10CFR50.48.

The components within the scope of license renewal and subject to aging management review include safety-related portions of the system necessary to support the emergency operation of the subsystems. This includes the dampers that isolate the normal ventilation and ductwork, damper bodies, cooler housings, blower housings and other components that maintain the system flow path for the emergency ventilation equipment. The major equipment in the scope of license renewal and subject to aging management review includes:

- normal control room ventilation isolation dampers
- control room emergency unit coolers
- emergency compressor/condensing units
- electrical equipment room 2150 emergency cooling units – included because of their function to maintain the freon pressure boundary
- emergency fan filter units

Fire dampers are covered in Section 2.4.6.2, however the fire dampers that form part of the pressure boundary for the safety-related portions of the control room ventilation system are addressed in this section. The heat exchangers of the emergency compressor/condensing units are exposed to raw water from the service water system and are evaluated in Section 2.3.3.10.

The intended function within the scope of license renewal is to maintain the system pressure boundary integrity. For the coolers, heat transfer is also a passive function within the scope of license renewal.

2.3.4 Steam and Power Conversion Systems

The following systems are included in this section:

- Main Steam
- Main Feedwater
- Emergency Feedwater
- Condensate Storage and Transfer

Scoping drawings, consisting of the P&IDs for the steam and power conversion systems, are listed in Table 2.3-8, and are provided as attachments to 1CAN010002 (Reference 2.3-10). These drawings have been highlighted to show the portions of the steam and power conversion systems within the scope of license renewal. The boundaries indicated on the drawings are primarily based on the system designations in the component database. The use of the component database in conjunction with the system P&IDs assures that duplicate reviews and omission of components do not occur at the interface boundaries. The mechanical component groups and their intended functions for the systems in this section are identified in Table 3.5-1 through Table 3.5-4.

2.3.4.1 Main Steam

The main steam system is described in SAR Section 10.3. The ANO-1 main steam system is primarily a nonsafety-related system, with the majority of the system components outside the scope of license renewal. However, the nonsafety-related small bore piping and components attached to the steam generator shell, which perform a system pressure boundary function, are in scope. This includes valves that are part of the heater vent system. The portion of the main steam system piping that is safety-related is the portion of the piping between the steam generators and the main steam isolation valves, including the steam supply to the EFW turbine, as well as the nitrogen supply to the steam generators.

The main steam system has the safety-related functions of removing heat from the RCS, protecting the RCS and the steam generators from over-pressurization, providing for the isolation of the steam generators during a postulated steam line break, and providing a steam supply to the emergency feedwater turbine.

The components in the main steam system subject to aging management review include the piping, vent, and drain valves from the steam generators up to the main steam isolation valves and EFW turbine steam supply piping. This includes the main steam safety valves, the atmospheric dump and block valves, and the main steam isolation valves. The primary intended function within the scope of license renewal is pressure boundary integrity.

2.3.4.2 Main Feedwater

The main feedwater system is described in SAR Section 10.4.7. The main feedwater system consists of two trains of pumps, feedwater heaters, and the associated piping and valves that supply feedwater to the steam generators to support normal plant operation. The ANO-1 main feedwater system is largely a nonsafety-related system, and therefore, the majority of the system components are outside of the scope of license renewal. The portion of the main feedwater system that is safety-related is the portion of the piping between the main feedwater isolation valves and the steam generators. Other portions of the main feedwater system are nonsafety-related and are outside of the scope of license renewal.

The main feedwater isolation valves isolate the feedwater line during a main steam or a main feedwater line break. The closure of the valves is an active function, but the valve bodies and piping in the safety-related portions of the system must maintain the main feedwater system pressure boundary integrity.

The components in the main feedwater system subject to aging management review include the main feedwater isolation valves and the piping, vent, and drain valves in the piping up to the steam generator ring headers. The primary intended function within the scope of license renewal is pressure boundary integrity.

2.3.4.3 Emergency Feedwater

The ANO-1 emergency feedwater system is described in SAR Section 10.4.8. The EFW system consists of two trains of pumps and the associated piping and valves that supply feedwater to the steam generators if the main feedwater supply is lost. One EFW pump is motor driven and the other is turbine driven. The EFW pumps can take suction from the safety-related condensate storage tank, the nonsafety-related condensate storage tank, the service water system, or the ANO-2 condensate storage tanks. Each EFW pump has discharge piping to both steam generators. The system provides a backup source of feedwater to the steam generators as required to assure that core decay heat and primary system residual heat can be removed. The EFW system removes decay heat until the plant has been cooled and depressurized sufficiently to permit use of the decay heat system. The components in the EFW system subject to aging management review include the EFW discharge piping and valves, the EFW pumps, the safety-related portion of the minimum recirculation lines, and the piping and valves in the discharge up to the EFW headers on the steam generators. The main steam supply valves to the EFW turbine and the steam supply piping downstream of the valves are included in the scope of this review. The EFW headers and nozzles at the steam generators have been included in the scope of the steam generator aging management review.

The primary intended function within the scope of license renewal is pressure boundary integrity. For the heat exchangers in the scope of review, heat transfer is also considered a passive function within the scope of license renewal.

2.3.4.4 Condensate Storage and Transfer System

The ANO-1 condensate storage and transfer system consists of the condensate storage tank, the safety-related condensate storage tank and the system piping and valves to supply water from the condensate storage tanks to the secondary plant systems at ANO-1. The condensate storage and transfer system provides a source of demineralized water to the secondary plant of ANO-1. The safety-related condensate storage tank and the associated piping have the safety significant function of providing the initial (preferred) source of water to the emergency feedwater pumps. These functions are discussed in SAR Section 10.4.8.

The passive mechanical components in the condensate storage and transfer system scope that are required to be included in the scope of license renewal and subject to aging management review include the safety-related condensate storage tank and the piping that maintains the pressure boundary of the condensate storage system up to the emergency feedwater pumps. The primary intended function within the scope of license renewal is pressure boundary integrity.

2.3.5 References for Section 2.3

2.3-1 NEI 95-10, "*Industry Guidelines for Implementing the Requirements of 10CFR Part 54 –The License Renewal Rule*," Revision 0, Nuclear Energy Institute, March 1996.

2.3-2 ASME Boiler and Pressure Vessel Code, Section III, "*Rules for Construction of Nuclear Power Plant Components,*" American Society of Mechanical Engineers.

2.3-3 USAS B31.7, "*Nuclear Power Piping,*" USA Standards Institute.

2.3-4 ASME Boiler and Pressure Vessel Code, Section VIII, "*Pressure Vessels*," American Society of Mechanical Engineers.

2.3-5 BAW-2243A, "*Demonstration of the Management of Aging Effects for the Reactor Coolant System Piping*," The BWOG Generic License Renewal Program, June 1996.

2.3-6 BAW-2244A, "*Demonstration of the Management of Aging Effects for the Pressurizer*," The BWOG Generic License Renewal Program, December 1997.

2.3-7 BAW-2251A, "*Demonstration of the Management of Aging Effects for the Reactor Vessel*," The BWOG Generic License Renewal Program, June 1996.

2.3-8 BAW-2248A, "*Demonstration of the Management of Aging Effects for the Reactor Vessel Internals*," The BWOG Generic License Renewal Program, December 1999.

2.3-9 ASME Boiler and Pressure Vessel Code, Section XI, "*Rules for In-Service Inspection of Nuclear Power Plant Components*," American Society of Mechanical Engineers.

2.3-10 1CAN010002, Letter from J. Vandergrift (ANO) to NRC, "*LRA Boundary Drawings*," submitted with the ANO-1 LRA.

Table 2.3-1 Reactor Coolant System P&IDs		
LRP&ID	**Sheet**	**Revision**
LRA-M-230	1	Revision 0
LRA-M-230	2	Revision 0
LRA-M-231	2	Revision 0
LRA-M-232	1	Revision 0
LRA-M-237	1	Revision 0

Table 2.3-2 RCS Piping Applicant Actions Items from Section 4.1 of BAW-2243A SER	
Renewal Applicant Action Item	**ANO-1 Specific Response**
When incorporating the BWOG topical report in its renewal application, the license renewal applicant is to verify that its plant is bounded by the topical report.	Entergy Operations participated in the development of BAW-2243A by providing ANO-1 specific design and operational information. Entergy Operations has reviewed the current design and operation of the ANO-1 RCS piping using the process described in Section 2.3.1.2. The results of the RCS piping review are reported in Section 2.3.1.3.
Further, the renewal applicant is to commit to programs described as necessary in the report to manage the effects of aging during the period of extended operation on the functionality of the RCS piping components.	Descriptions of these programs are provided in Section 3.2 and Appendix B.
A summary description of these programs is to be provided in the license renewal FSAR supplement in accordance with 10CFR54.21(d).	Summary descriptions of these programs are provided in Appendix A.
Any deviations from the aging management programs described within this report as necessary to manage the effects of aging during the period of extended operation to maintain the functionality of RCS piping components or other information presented in the report, such as materials of construction and edition of the ASME Section XI code (including mandatory appendices), will have to be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10CFR54.21(a)(3).	No deviations from the aging management programs described in BAW-2243A or other information presented in the report have been identified by Entergy Operations.
Further, the BWOG defers the development of details of the inspection of (1) the Alloy 82/182 clad hot leg segment and plant selection for that inspection, and (2) the sample inspection of small bore RCS piping, to the renewal applicant referencing this topical report. The renewal applicant will have to provide details of these two augmented inspection programs in its renewal application for	Descriptions of these programs are provided in Section 3.2 and Appendix B.

Table 2.3-2 RCS Piping Applicant Actions Items from Section 4.1 of BAW-2243A SER	
Renewal Applicant Action Item	**ANO-1 Specific Response**
staff review and approval.	
The BWOG elected to exclude TLAAs applicable to the RCS piping components from the scope of the topical report and indicated that they will be resolved on a plant-specific basis. Thus, any renewal applicant referencing this report will have to evaluate TLAAs applicable to the RCS piping components in its renewal application in accordance with the requirements in 10CFR54.21(c).	Evaluations of ANO-1 specific TLAAs are provided in Section 4.0.
Additionally, since the staff does not make any finding relative to whether the BWOG report constitutes the complete list of RCS piping components subject to an aging management review or the adequacy of a scoping methodology, individual plant applicants will need to identify and list structures and components subject to an aging management review and a methodology for developing this list as part of their license renewal applications.	A list of RCS components that are subject to aging management review is provided in Section 2.3. The individual components are also listed in ANO-1 specific documents maintained onsite. The methodology for developing and maintaining this list of components is consistent with the guidance contained in NEI 95-10, Revision 0.

Table 2.3-3 Pressurizer Applicant Action Items from Section 4.1 of BAW-2244A SER	
Renewal Applicant Action Item	**ANO-1 Specific Response**
(1) The renewal applicant is to verify that its plant is bounded by the topical report. This includes confirming that the design of the pressurizer is consistent with that described in the report such that no important pressurizer components exist that have not been addressed in the report.	Entergy Operations participated in the development of BAW-2244A by providing ANO-1 specific design and operational information. Entergy Operations has reviewed the current design and operation of the ANO-1 pressurizer using the process described in Section 2.3.1.2 and confirms that the ANO-1 pressurizer is bounded by the description contained in BAW-2244A.
(2) The renewal applicant is to commit to programs identified as necessary in the report to manage the effects of aging on the functionality of the pressurizer.	Descriptions of these programs are provided in Section 3.2 and Appendix B.
(3) A summary description of these programs is to be provided in the license renewal FSAR supplement in accordance with 10CFR54.21(d).	Summary descriptions of these programs are provided in Appendix A.
(4) Any deviations from the aging management programs described within this report as necessary to manage the effects of aging during the period of extended operation to maintain the functionality of the pressurizer or other information presented in the report, such as materials of construction and edition of the ASME Section XI code (including mandatory appendices), will have to be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10CFR54.21(a)(3).	No deviations from the aging management programs described in BAW-2244A or other information presented in the report have been identified by Entergy Operations.
(5) Since the BWOG defers the development of details of the additional sample volumetric inspection program of small-bore nozzles and safe ends to the renewal applicant referencing this topical report, the renewal applicant will have to provide details of the additional sample inspection program in its renewal application for staff review and approval.	A description of this program is provided in Section 3.2 and Appendix B.

Table 2.3-3 Pressurizer Applicant Action Items from Section 4.1 of BAW-2244A SER	
Renewal Applicant Action Item	**ANO-1 Specific Response**
(6) Since the BWOG elected to exclude TLAAs applicable to the pressurizer from the scope of the topical report and indicated that they will be resolved on a plant-specific basis, any renewal applicant referencing this report will have to evaluate TLAAs applicable to the pressurizer in its renewal application in accordance with the requirements in 10CFR54.21(c).	Evaluations of ANO-1 specific TLAAs are provided in Section 4.0.
BAW-2244 Open Items (Section 4.2)	
(1) Cracking of Stainless Steel Cladding Inside the Pressurizer Vessel (Discussed in section 3.2.1 of the SER) The staff notes that cracking in cladding could potentially propagate into the base metal material and should be addressed by an aging management program. Industry experience at one site has shown that this is a potential aging effect. The staff maintains that cracking of the stainless steel is a potential aging effect that must be addressed by an aging management program for the period of extended operation. A program to provide a reasonable demonstration of the integrity of the pressurizer cladding could be a one-time inspection for license renewal. The inspection should include the cladding and any attachment welds to the cladding. The additional inspection would provide information on the condition of the cladding or, if cracking is discovered, the condition of the underlying base metal as a result of the cracked cladding. The staff notes that the inspection technique chosen (e.g., visual, surface, or volumetric) must be capable of determining the condition of the cladding and must be submitted for staff review and approval. Without such additional aging management program activities, the staff cannot conclude that all aging effects applicable to the pressurizer vessel cladding have been adequately	The ANO-1 program to address pressurizer cladding cracking is provided in the new program entitled "Pressurizer Examinations," which is described in Section 3.2 and Appendix B.

Table 2.3-3 Pressurizer Applicant Action Items from Section 4.1 of BAW-2244A SER	
Renewal Applicant Action Item	**ANO-1 Specific Response**
addressed by the aging management programs delineated in BAW-2244.	
(2) Aging management of pressurizer heater penetration welds (discussed in Section 3.3.2.2.3 of this SER) The staff regards the provision for examination of pressurizer heater penetration welds in ASME Code, Section XI, ISI Examination Category B-E as applicable to pressurizer heater partial-penetration welds. The BWOG considers the Examination Category B-E requirement not applicable to the B&W design because Examination Category B-E concerns pressure-retaining partial-penetration welds in vessels. The BWOG stated that, "Although the 'Parts Examined' listing under Item B4.20 of Examination Category B-E uses the term 'Heater Penetration Welds,' the 'Extent and Frequency of Examination' specifically requires only 'All Nozzles' to have examination." "There are no heater penetration nozzles or pressure-retaining heater nozzle partial-penetration welds in the vessels of the B&W pressurizer design." The staff disagrees with the BWOG assessment. The B&W pressurizer heaters are inserted through holes in the pressurizer heater bundle diaphragm plates and the heater sheaths (or heater sleeves at ONS-1 and TMI-1) are attached to the diaphragm plates on the inside by partial-penetration welds. The staff does not believe that the B&W heater penetrations are sufficiently different from other vendor designs, except that the B&W heaters are mounted horizontally on the diaphragm plates inserted through the side of the pressurizer shell, while other vendor designs mount the heaters vertically, inserted through the bottom of the pressurizer. In addition, Examination Category B-E explicitly states that the	The ANO-1 program to address inspection of heater penetration welds is provided in the new program entitled "Pressurizer Examinations," which is described Section 3.2 and Appendix B.

Table 2.3-3 Pressurizer Applicant Action Items from Section 4.1 of BAW-2244A SER	
Renewal Applicant Action Item	**ANO-1 Specific Response**
pressurizer heater penetration welds are to be examined. Therefore, the staff considers the pressurizer heater partial-penetration welds pressure-retaining, and subject to the requirements set forth in ASME Code, Section XI, ISI Examination Category B-E. Operating experience has also shown that pressurizer heater partial-penetration welds are susceptible to cracking. To provide reasonable assurance that cracking of the heater penetration welds and the heater sheath-to-sleeve welds (ONS-1 and TMI-1) will be managed during the period of extended operation, the staff is requesting an additional, more intrusive inspection technique. Specifically, the staff will consider ASME Code, Section XI, ISI Examination Category B-E together with an inspection program consisting of surface examinations (the criteria and technique of which would be developed at a later date and subject to staff approval) for the pressurizer partial-penetration heater sheath-to-heater bundle diaphragm plate welds, heater sleeve-to-heater bundle diaphragm plates welds and heater sheath-to-heater sleeve welds acceptable for managing the effects of cracking for the period of extended operations.	

Table 2.3-4 Reactor Vessel Applicant Action Items from Section 4.1 of BAW-2251A SER

Renewal Applicant Action Item	ANO-1 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 vessel components. 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 vessel 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 10CFR54.21(a)(3) and (c)(1).	Entergy Operations participated in the development of BAW-2251A by providing ANO-1 specific design and operational information. Entergy Operations has reviewed the current design and operation of the ANO-1 reactor vessel using the process described in Section 2.3.1.2 and confirms that the ANO-1 reactor vessel is bounded by the description contained in BAW-2251A.
(2) A summary description of the programs and evaluation of TLAAs is to be provided in the license renewal FSAR supplement in accordance with 10CFR54.21(d).	Summary descriptions of these programs are provided in Appendix A.

Table 2.3-4 Reactor Vessel Applicant Action Items from Section 4.1 of BAW-2251A SER	
Renewal Applicant Action Item	**ANO-1 Specific Response**
(3) Since the staff has not made any findings on whether the BWOG topical report provides the complete list of reactor vessel components subject to an aging management review or whether the scoping methodology is adequate, individual plant applicants will need to provide a comprehensive list of structures and components subject to an aging management review and the methodology for developing 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 topical report are required to be addressed in the license renewal application.	Entergy Operations has reviewed the current design and operation of the ANO-1 reactor vessel using the process described in Section 2.3.1.2 and has confirmed that the ANO-1 reactor vessel is bounded by the description contained in BAW-2251A. No additional RV items were identified as subject to aging management review.
(4) The BWOG has determined that the lower CRDM service support structure, including the weld that connects the lower CRDM service support skirt to the reactor vessel closure head, and the reactor vessel support skirt, including the weld that connects the reactor vessel support skirt to the transition forging, are subject to an aging management review for license renewal. However, the BWOG has decided to exclude them from the scope of the topical report. Thus, a renewal applicant needs to address them in its license renewal application.	The CRDM service support structure and the RV skirt are evaluated in Section 2.4.2.1
(5) The license renewal application for Oconee needs to address the fatigue evaluation of the reactor vessel studs on a plant-specific basis.	This renewal applicant action item is not applicable to ANO-1.

Table 2.3-4 Reactor Vessel Applicant Action Items from Section 4.1 of BAW-2251A SER	
Renewal Applicant Action Item	**ANO-1 Specific Response**
(6) A license renewal applicant needs to discuss the plant-specific methodology and instrumentation used to assess the number of operational transients in its renewal application for staff review. The staff review will also include the number of operating cycles applicable to the reactor vessel studs.	The ANO-1 program that monitors operational transients is described in Section 4.3.5.
(7) The BWOG identifies flaw growth acceptance in accordance with the ASME Section XI ISI program as a TLAA, but indicates that flaw growth acceptance evaluation is plant-specific, is not within the scope of the report, and will be resolved on a plant-specific basis. Thus, a license renewal applicant needs to address it in the renewal application.	The ANO-1 program to manage analytical evaluation of flaws is described in Section 4.3.6.
(8) Alloy 600 components in the reactor vessel such as CRDM housings and other penetrations may be subject to crack initiation and growth. The BWOG originally proposed to use the ASME Section XI program, supplemented by leak detection and surveillance of boric acid, to manage cracking of Alloy 600 components. In an April 1, 1997, response to the staff's request for additional information concerning Generic Letter 97-01, "Stress Corrosion Cracking of Control Rod Drive Mechanisms and Other Vessel Head Penetrations," the BWOG stated: "Each participating plant will address additional requirements for RV head penetrations, including closure head penetrations less than 2 inch N.S. (i.e., thermocouple nozzles at TMI-1 and ONS-2)." Thus, a license renewal applicant referencing the topical report will need to submit its plant-specific program to manage cracking of Alloy 600 components in the reactor vessel in its renewal application for staff review.	The ANO-1 programs that address Alloy 600 penetrations for the period of extended operation include the "Alloy 600 Aging Management Program" and the "CRDM Penetration and Other Vessel Head Penetration Program," which are described in Appendix B.

Table 2.3-4 Reactor Vessel Applicant Action Items from Section 4.1 of BAW-2251A SER	
Renewal Applicant Action Item	**ANO-1 Specific Response**
(9) During the review of the topical report, the staff had a question regarding the need to update the reactor vessel fracture toughness estimates with new data as it become available. In its August 11, 1997, RAI response, the BWOG states: "Each license renewal applicant will define a process to ensure that the time-dependent parameters used in the TLAA evaluations reported in BAW-2251 are tracked such that the TLAA remains valid through the period of extended operation. The process will be defined on a plant-specific basis at the time of the licensee renewal application." Thus, a license renewal applicant needs to describe such a process in its application for staff review. If new information affects the conclusions of the topical report for the applicant's plant, the applicant needs to update its TLAA evaluations as appropriate and provide the updated evaluations in its renewal application for staff review.	See the ANO-1 Reactor Vessel Integrity Program described in Appendix B.
(10) In its August 11, 1997, RAI response, the BWOG indicated that Oconee Unit 2 and TMI Unit 1 will provide updated predictions of RT_{PTS} *for welds WF-25 and SA-1526, respectively, when the plant-specific application for license renewal is submitted. For plants with an* RT_{PTS} *value for 48 EFPY exceeding the corresponding PTS screening criterion, a license renewal applicant must address the requirements in 10CFR50.61(b)(3) by developing, and requesting staff approval for reasonably practicable flux reduction programs to avoid exceeding the PTS criterion.*	This renewal applicant action item is not applicable to ANO-1.

Table 2.3-4 Reactor Vessel Applicant Action Items from Section 4.1 of BAW-2251A SER	
Renewal Applicant Action Item	**ANO-1 Specific Response**
(11) If an applicant has installed flow stabilizers using Alloy 600 and/or Alloy 82/182 weld material, the applicant must include the flow stabilizers in its Alloy 600 aging management program. Alloy 600 and Alloy 82/182 weld materials are susceptible to cracking in primary water environments.	The flow stabilizers at ANO-1 are made from austenitic stainless steel and attached to the cladding using stainless steel weldments.
(12) Embrittlement of the reactor vessel will be managed to ensure intended functions of the reactor vessel for 60 years. For the staff to determine if the plant could be operated for 60 years, an applicant must show that an operating window will be available between the pressure-temperature limits and the net positive suction curves for the RCPs for 60 years. Otherwise, the applicant will propose aging management activities to minimize the extent of embrittlement, or other alternatives, to permit safe plant operation for 60 years. Should the applicant show that the reactor could only be operated for a time period less than 60 years, the duration of the renewed license, if granted, would be limited to that time period.	ANO-1 developed 48 EFPY pressure-temperature limits in accordance with the requirements of ASME Section XI, Appendix G, as modified by Code Case N-588 for circumferential flaws in welds and by Code Case N-640 for the use of K_{IC} fracture toughness curve. The operating window at 48 EFPY exceeds the current P-T operating window, which has been approved by the NRC for 31 EFPY. The increased operating window is attributed to the use of Code Cases N-588 and N-640.
(13) The neutron fluence must be experimentally monitored by ex-vessel or in-vessel dosimetry, and if modifications to the design and operation of the plant changes either the neutron energy spectrum, gamma heating or the reactor inlet temperature, as discussed in section 3.3.4.1 of this safety evaluation, the licensee must notify the NRC and propose a program to determine the impact of the modifications.	Reactor vessel fluence monitoring is addressed in the ANO-1 Reactor Vessel Integrity Program that is described in Appendix B.

Table 2.3-5 RV Internals Applicant Action Items from Section 4.1 of BAW-2248A SER	
Renewal Applicant Action Item	**ANO-1 Specific Response**
(1) The license renewal applicant is to verify that the critical parameters for the plant are 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 vessel internals components. The applicant for license renewal will be responsible for describing any such commitments and proposing the appropriate regulatory controls. 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 vessel internal 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 10CFR54.21(a)(3) and (c)(1).	ANO-1 participated in the development of BAW-2248A by providing ANO-1 specific design and operational information. Entergy Operations has reviewed the current design and operation of the reactor vessel internals using the process described in Section 2.3.1.2 and confirms that ANO-1 is bounded by the description contained in BAW-2248A with regard to critical plant parameters. Reactor vessel internals items that ANO-1 identified as subject to aging management review that were not within the scope of BAW-2248A are described in Section 2.3.1.6. The ANO-1 aging management programs for the reactor vessel internals are described in Appendix B.
(2) A summary description of the programs and evaluation of TLAAs is to be provided in the license renewal FSAR supplement in accordance with 10CFR54.21(d).	Summary descriptions of these programs are provided in Appendix A.

Table 2.3-5 RV Internals Applicant Action Items from Section 4.1 of BAW-2248A SER	
Renewal Applicant Action Item	**ANO-1 Specific Response**
(3) The license renewal applicant must identify whether an intended function of the RVI is to provide shielding for the RPV. If not an intended function, the license renewal applicant should provide justification for that conclusion. Should a license renewal applicant determine that the RVI's intended function is to provide shielding for the RPV, then the items that support this intended function, such as, the thermal shield and the thermal shield upper restraint assemblies, must be identified and reviewed in accordance with 10CFR54.21(a)(3).	Reactor vessel internals intended functions are discussed in Section 2.3.1.6. Three additional intended functions were identified by ANO-1. These additional intended functions and items that support these functions are addressed in Section 2.3.1.6.
(4) Applicants must commit to participation in the BWOG RVIAMP, and any other industry programs as appropriate, to continue the investigation of potential aging effects for RVI components and to establish monitoring and inspection programs for RVI components. The applicant shall provide the NRC with either annual or periodic updates (after completion of significant milestones) on the status of the RVIAMP, commencing within one year of the issuance of the renewed license.	ANO-1 will participate in appropriate BWOG and/or industry level programs to ensure that the RV internals intended functions are maintained consistent with the CLB during the period of extended operation. ANO-1 will provide written reports to the staff upon completion of significant RVIAMP milestones commencing within one year of the issuance of the renewed license.
(5) The applicant must describe plans for augmented inspection of RVI components for management of SCC/IASCC and loss of fracture toughness (neutron embrittlement) of the RVI components. This description should specify the sample size, the examination method, acceptance criteria and timing of the inspection, or the process to be used to specify these items.	See the ANO-1 Reactor Vessel Internals Aging Management Program described in Appendix B.

Table 2.3-5 RV Internals Applicant Action Items from Section 4.1 of BAW-2248A SER	
Renewal Applicant Action Item	**ANO-1 Specific Response**
(6) According to the BWOG, one of its objectives in BAW-2248 states, "It is intended that NRC review and approval of this report will allow that no further review of the matters described herein will be needed when the report is incorporated by reference in a plant specific renewal license application." The license renewal applicant must address the baffle-former bolt cracking issues addressed in Section 3.3.2 of this SE pertaining to Refs. 4 and 5, with regard to the ITG project, initiated after April 23, 1998, to address generic RVI materials issues. The BWOG indicates this industry effort resulted in subsequent changes in the BWOG RVI aging management program. The ITG is currently addressing the issues of cracking of baffle bolts. The BWOG indicates that the changes in the aging management program now requires the applicants to be responsible for using the industry ITG project developed information to determine the necessary steps (e.g., inspection, operability determinations, and replacements) for the management of the applicable baffle bolt aging effects.	See the ANO-1 Reactor Vessel Internals Aging Management Program described in Appendix B.
(7) The applicant must describe plans for augmented inspection of RVI components for management of loss of fracture toughness due to thermal aging embrittlement of the RVI components. This description should specify the sample size, the examination method, acceptance criteria and timing of the inspection, or the process to be used to specify these items.	See the ANO-1 Reactor Vessel Internals Aging Management Program described in Appendix B.

Table 2.3-5 RV Internals Applicant Action Items from Section 4.1 of BAW-2248A SER	
Renewal Applicant Action Item	**ANO-1 Specific Response**
(8) The applicant must describe plans for management of stress relaxation for bolted closures of the RVI. This description should specify the critical locations, and monitoring and inspection techniques, and timing of the inspection, or the process to be used to specify these items.	See the ANO-1 Reactor Vessel Internals Aging Management Program described in Appendix B.
(9) The applicant must address aging management of void swelling. An adequate aging management program (AMP) would include participation in industry program(s) to address the significance of void swelling (either individually or through an owners or industry group), a commitment to develop a sufficient inspection program (including the basis, methods, locations to be examined, timing, frequency and acceptance criteria) for management of the issue based upon the results of the industry programs, and a commitment to implement the inspection program prior to the end of the current license period.	Change of dimensions by void swelling is included in the ANO-1 Reactor Vessel Internals Aging Management Program, which is described in Appendix B.
(10) If flaws have been detected in the reactor vessel internals, a TLAA plant-specific evaluation must be performed to determine the flaw growth acceptance in accordance with the ASME B&PV Code, Section XI, inservice inspection requirements.	No flaws requiring analytical evaluation have been discovered in the inspections of the reactor vessel internals at ANO-1.
(11) The applicant must address the plant-specific plans to continue monitoring and tracking design transient occurrences.	See the description of the ANO-1 Transient Cycle Logging Program in Section 4.3.5.

Table 2.3-5 RV Internals Applicant Action Items from Section 4.1 of BAW-2248A SER	
Renewal Applicant Action Item	**ANO-1 Specific Response**
(12) Plant-specific analysis is required to demonstrate that, under loss-of-coolant-accident (LOCA) and seismic loading, the internals have adequate ductility to absorb local strain at the regions of maximum stress intensity and that irradiation accumulated at the expiration of the renewal license will not adversely affect deformation limits. The RVIAMP must develop data to demonstrate that the internals will meet the deformation limits at the expiration of the renewal license.	A plant-specific analysis will be performed to demonstrate that under LOCA and seismic loading, the internals have adequate ductility to absorb local strain at the regions of maximum stress intensity and that irradiation accumulated at the expiration of the renewal license will not affect deformation limits. Data will be developed to demonstrate that the internals will meet the deformation limits at the expiration of the renewed license.

Table 2.3-6 Piping and Instrument Diagrams (P&IDs) – Evaluation Boundaries of Engineered Safeguards		
P&ID	**Sheet Number**	**Revision**
Core Flood System		
LRA-M-230	1	Rev. 0
LRA-M-236	1	Rev. 0
Low Pressure Injection/Decay Heat System		
LRA-M-230	1	Rev. 0
LRA-M-231	1	Rev. 0
LRA-M-232	1	Rev. 0
High Pressure Injection/Makeup and Purification System		
LRA-M-230	1	Rev. 0
LRA-M-231	1	Rev. 0
LRA-M-231	2	Rev. 0
LRA-M-231	3	Rev. 0
LRA-M-238	1	Rev. 0
LRA-M-238	2	Rev. 0
Reactor Building Spray System		
LRA-M-236	1	Rev. 0
Reactor Building Cooling and Purge System		
LRA-M-261	1	Rev. 0
Sodium Hydroxide System		
LRA-M-232	1	Rev. 0
LRA-M-233	1	Rev. 0
Reactor Building Isolation System		
LRA-M-206	1	Rev. 0
LRA-M-215	1	Rev. 0
LRA-M-218	5	Rev. 0
LRA-M-220	3	Rev. 0
LRA-M-230	1	Rev. 0
LRA-M-230	2	Rev. 0
LRA-M-233	1	Rev. 0
LRA-M-234	1	Rev. 0
LRA-M-234	2	Rev. 0
LRA-M-236	1	Rev. 0
LRA-M-237	1	Rev. 0
Hydrogen Control System		
LRA-M-237	4	Rev. 0
LRA-M-261	1	Rev. 0
LRA-M-261	3	Rev. 0

Table 2.3-7 Piping and Instrument Diagrams (P&IDs) – Evaluation Boundaries of Auxiliary Systems		
P&ID	**Sheet Number**	**Revision**
Spent Fuel Pool		
LRA-M-232	1	Rev. 0
LRA-M-235	1	Rev. 0
Fire Protection		
LRA-M-219	1	Rev. 0
LRA-M-2219	5	Rev. 0
Emergency Diesel Generator		
LRA-M-217	2	Rev. 0
LRA-M-217	3	Rev. 0
LRA-M-217	4	Rev. 0
Auxiliary Building Sump and Reactor Building Drains		
LRA-M-213	1	Rev. 0
LRA-M-213	2	Rev. 0
LRA-M-214	3	Rev. 0
LRA-M-232	1	Rev. 0
LRA-M-238	1	Rev. 0
LRA-M-238	2	Rev. 0
Alternate AC Diesel Generator		
LRA-M-2241	1	Rev. 0
LRA-M-2241	2	Rev. 0
LRA-M-2241	4	Rev. 0
LRA-M-2241	5	Rev. 0
LRA-M-2260	4	Rev. 0
Halon		
LRA-M-219	2	Rev. 0
Fuel Oil		
LRA-M-217	1	Rev. 0
LRA-M-217	2	Rev. 0
LRA-M-217	3	Rev. 0
LRA-M-219	1	Rev. 0
LRA-M-2241	3	Rev. 0
Instrument Air		
LRA-M-206	2	Rev. 0
LRA-M-210	1	Rev. 0
LRA-M-213	2	Rev. 0
LRA-M-215	1	Rev. 0
LRA-M-218	4	Rev. 0
LRA-M-222	1	Rev. 0
LRA-M-230	2	Rev. 0
LRA-M-232	1	Rev. 0

Table 2.3-7 Piping and Instrument Diagrams (P&IDs) – Evaluation Boundaries of Auxiliary Systems		
P&ID	**Sheet Number**	**Revision**
LRA-M-233	1	Rev. 0
LRA-M-234	1	Rev. 0
LRA-M-234	2	Rev. 0
LRA-M-237	1	Rev. 0
LRA-M-261	1	Rev. 0
LRA-M-262	1	Rev. 0
LRA-M-262	2	Rev. 0
LRA-M-262	3	Rev. 0
LRA-M-262	4	Rev. 0
LRA-M-263	1	Rev. 0
Chilled Water		
LRA-M-221	2	Rev. 0
LRA-M-222	1	Rev. 0
Service Water		
LRA-M-204	3	Rev. 0
LRA-M-209	1	Rev. 0
LRA-M-210	1	Rev. 0
LRA-M-221	2	Rev. 0
Penetration Room Ventilation		
LRA-M-264	1	Rev. 0
Auxiliary Building Heating and Ventilation		
LRA-M-262	3	Rev. 0
LRA-M-262	4	Rev. 0
LRA-M-263	2	Rev. 0
LRA-M-263	3	Rev. 0
Control Room Ventilation		
LRA-M-2221	2	Rev. 0
LRA-M-263	1	Rev. 0

Table 2.3-8 Piping and Instrument Diagrams (P&IDs) – Evaluation Boundaries of Steam and Power Conversion Systems		
P&ID	**Sheet Number**	**Revision**
Main Steam System		
LRA-M-204	6	Rev. 0
LRA-M-206	1	Rev. 0
LRA-M-206	2	Rev. 0
Main Feedwater System		
LRA-M-206	1	Rev. 0
Emergency Feedwater System		
LRA-M-204	3	Rev. 0
LRA-M-204	6	Rev. 0
LRA-M-206	1	Rev. 0
Condensate Storage and Transfer System		
LRA-M-204	3	Rev. 0
LRA-M-204	5	Rev. 0

2.4 STRUCTURES AND STRUCTURAL COMPONENTS SCOPING AND SCREENING RESULTS

2.4.1 Reactor Building

The determination of ANO-1 structures within the scope of license renewal is made by initially identifying structures and then reviewing each structure to determine which ones satisfy one or more of the criteria contained in 10CFR54.4. This section contains the information required by 10CFR54.21(a)(1) and (a)(2) for the ANO-1 reactor building (which provides the containment function) structural components that are subject to aging management review for license renewal.

The ANO-1 reactor building is identified as a seismic category 1 structure in the ANO-1 SAR Section 5.2. Seismic category 1 structures are those which prevent uncontrolled release of radioactivity and are designed to withstand design basis loading conditions without loss of function. Accordingly, seismic category 1 structures meet the criteria of 10CFR54.4(a)(1) and are within the scope of license renewal. Therefore, the ANO-1 reactor building is within the scope of license renewal. A portion of the reactor building serves as an important element in the radioactive release line-of-defense and therefore, receives special focus in the integrated plant assessment. The reactor building includes the concrete reactor building structure, liner, and penetrations.

The ANO-1 reactor building is a composite structure consisting of a post-tensioned, reinforced concrete structure with cylindrical wall, a flat foundation slab, and a shallow dome roof. SAR Figure 5-1 is an illustration of the ANO-1 prestressed concrete reactor building. The reactor building completely encloses the reactor and the associated RCS along with other electrical, mechanical and structural components. The cylinder wall integrity is provided by a post-tensioning system consisting of horizontal and vertical tendons in the cylinder wall. Dome integrity is provided by three sets of tendons with each set oriented 120 degrees from the other. The concrete foundation slab is conventionally reinforced. The entire structure is internally lined with a carbon steel liner plate to assure a high degree of leak tightness.

The reactor building structure is subdivided into component groupings for the aging management review. Many structural components are not typically associated with unique equipment identifiers and thus, are not individually identified during the identification of components subject to aging management review. Specific structural component identifiers are not needed because the aging management review process and resulting programmatic oversight was performed across an entire component grouping.

The intended functions of the reactor building were determined by reviewing information contained in the ANO-1 SAR and ANO-1 engineering documents, as well as NEI 95-10 (Reference 2.4-1). The reactor building and its structural components fulfill the following intended functions.

- Provide essentially leak tight barriers to prevent uncontrolled release of radioactivity

- Provide structural support or functional support to safety-related systems, structures, and components. Specifically, for the post-tensioning systems, this function means to impose compressive forces on the concrete reactor building structure to resist, with no loss of structural integrity, the internal pressure resulting from a DBA.
- Provide shelter or protection to safety-related equipment (including radiation shielding)
- Provide rated fire barriers to confine or retard a fire from spreading to or from adjacent areas
- Serve as external missile barriers
- Provide structural or functional support to nonsafety-related equipment, failure of which could directly prevent satisfactory accomplishment of required safety-related functions
- Provide a heat sink during DBA or station blackout conditions

The ANO-1 reactor building structural components within the scope of 10CFR Part 54 were reviewed to determine those components subject to an aging management review in accordance with 10CFR54.21(a)(1). An aging management review of a structural component is required if the component performs an intended function without moving parts or without a change in configuration or properties (i.e., passive) and if it is not subject to replacement based on a qualified life or specified time period (i.e., long-lived). Consistent with the guidance provided in NEI 95-10, the reactor building structural components within the scope of license renewal are long-lived and passive and require an aging management review.

A listing of the reactor building passive, long-lived components and unique commodities subject to an aging management review, and their intended function(s), is provided in Table 3.6-2. This list has been derived from individual reactor building components identified in ANO-1 specific documents maintained onsite. Components which do not perform an intended function are not within the scope of license renewal.

The reactor building structural components have been divided into three groups based on material of construction and component-level function. They are steel, concrete, and the post tensioning system. These component groups are described in the following sections.

2.4.1.1 Steel Components

Liner Plate

The interior of the reactor building is lined with a steel liner plate of welded construction. The liner plate covers the dome, the cylinder wall and also runs between the floor and the foundation slab to form an essentially leak tight barrier. The ANO-1 reactor building liner plate is an ASTM A36 (Reference 2.4-2) or A516 (Reference 2.4-3) plate attached to the concrete by means of an angle grid system of ASTM A36 material stitch welded to the liner plate and embedded in the concrete. The liner plate is anchored in both the longitudinal and hoop directions.

The anchor spacing and welds are designed to preclude failure of an individual anchor. The frequent anchoring is designed to prevent significant distortion of the liner plate during accident conditions and to ensure that the liner maintains its essentially leak tight integrity.

Before the reactor building penetrations were embedded in concrete, they were continuously welded to the liner plate. The entire length of every weld was leak tested following fabrication. Radiographs were taken of at least one foot out of every 50 feet of welding completed by each welder during fabrication.

The liner plate is coated on the inside with inorganic zinc primer for corrosion protection. There is no coating on the side in contact with the concrete. At the penetrations, the liner plate is thickened to reduce stresses in accordance with the ASME Code (Reference 2.4-4). The liner was designed as a free standing vessel for erection loads and was used as the internal form for the concrete. The liner plate is thickened at large attachments, such as the polar crane brackets, to accommodate strength and welding requirements for the attachments and anchors. The general liner configuration is shown in SAR Figure 5-1.

ASME Section III (Reference 2.4-5) is used as the basis for establishing allowable liner plate strains and stresses. ASME Section III requires that the liner material be prevented from experiencing significant distortion due to thermal loads and that stresses be considered from a fatigue standpoint.

Anchors/Embedments/Attachments

Anchors and embedments are steel commodities, such as angles and anchor studs that are welded to the liner and anchor the liner to the reactor building concrete shell. A typical liner anchor is shown in SAR Figure 5-28. In addition, other anchors and embedments are provided to transfer loads into the concrete cylinder wall or foundation mat from attachments to the liner. In these cases, a thickened insert plate is welded to the liner and is used as the point of attachment for the anchor. The polar crane bracket is anchored to the concrete shell by a welded plate assembly embedded in the concrete.

The anchors and embedments maintain the essentially leak tight barrier by preserving the integrity of the liner. The load carrying capacity of these anchor is also required to assure that the supported equipment, such as the polar crane or the steam generators, can continue to perform safely.

Attachments to the liner that are integral with the liner and concrete structure (i.e., attachment has corresponding anchor in concrete) are connected to the inside face of the liner and thus exposed to the interior of the reactor building. The polar crane brackets are examples of attachments to the liner.

The polar crane brackets consist of welded carbon steel plates, constructed of the same material as the liner. The Polar crane brackets were inspected using requirements similar to those for the liner. Other attachments to the liner include structural steel attachments that are welded directly to the liner to support various structures and components.

Attachment welds are not considered to be within the evaluation boundary of the reactor building. However, these attachment welds are considered to be within the evaluation boundary of reactor building internal structural components that are addressed in Section 2.4.2.

<u>Personnel Hatches</u>

Two hatches are provided into the reactor building for personnel access and egress (see SAR Figure 5-3). The larger personnel hatch is used as the primary access to the reactor building. The smaller personnel hatch is used for emergency egress. The personnel hatches are double-door, welded steel assemblies. The hatches are designed to withstand reactor building design conditions with either, or both, doors closed and locked. The doors open toward the center of the reactor building, preventing unseating of the doors during reactor building pressurization. The personnel hatches may be individually pressurized to demonstrate leak tightness. Quick-acting, equalizing valves connect the air within each personnel hatch with the air inside and outside of the reactor building. These valves equalize the pressures on either side of a hatch door when it is operated. The equalizing valves are active components of the hatches and do not require an aging management review. Functionality of the equalizing valves is verified periodically when the hatches are pressurized and tested for leakage.

The personnel hatches contain operating mechanisms, which include gears, latches, hinges, linkages, etc., to open and close the doors of the hatch. These operating mechanisms perform their function with moving parts and with a change of configuration. Both the larger personnel hatch and the smaller emergency hatch are required to be operable, as defined by ANO-1 Technical Specifications. Surveillance requirements are also included in ANO-1 Technical Specifications. Actions are required to be taken, up to and including plant shutdown, if one or more of the hatch doors become inoperable.

The Statement of Considerations of the final 10CFR Part 54 rule states that:

"… many licensee programs that ensure compliance with technical specifications are based on surveillance activities that monitor performance of systems, structures, and components that perform active functions. As a result of the continued applicability of existing programs and regulatory requirements, the Commission believes that active functions of systems, structures, and components will be reasonably assured in any period of extended operation."

Also, 10CFR54.21(a)(1)(i) states that:

"structures and components subject to an aging management review shall encompass those structures and components that perform an intended function, as described in 10CFR54.4, without moving parts or without a change in configuration or properties…".

Accordingly, since the hatch operating mechanisms perform their intended functions with moving parts and a change of configuration, Entergy Operations has determined that the reactor building hatch operating mechanisms are not subject to an aging management review.

The two doors of each personnel hatch are interlocked to ensure that reactor building integrity is always maintained by one door being completely closed before the other door can be opened. The interlocking system has a bypass, allowing the doors to open simultaneously during plant cold shutdown. The interlock system is an active component of the personnel hatch and is not subject to an aging management review. Serviceability of the interlock system is verified during periodic personnel hatch leakage testing as well as during periodic maintenance.

Each personnel hatch door is provided with flexible seals. The seals are replaced when warranted by their condition. The seals are not long-lived components and therefore do not require an aging management review.

The hatches were designed and fabricated in accordance with the ASME Section III requirements for Class B vessels. The plate materials that comprise the personnel hatches' pressure vessel components are painted carbon steel complying with material specification ASTM A516, Grade 70, made to ASTM A300 (Reference 2.4-6) specification, for fine grained materials with ductile material properties suitable for low temperature use.

Equipment Hatch

A single equipment hatch as shown in SAR Figure 5-3 is provided for the reactor building. The equipment hatch design and fabrication conform to the ASME Code for Class B vessels. As with the personnel hatches, the equipment hatch was fabricated using A516 Grade 70, painted carbon steel made to ASTM A300 specification.

The equipment hatch is furnished with a double sealed flange and bolted, dished head. The barrel portion of the equipment hatch is thicker than required based on permissible stresses. The space between the double seals on the equipment hatch flange can be pressurized for local leakage testing. As with the personnel hatches, the flexible seals are tested and replaced when warranted by condition. The seals are not long-lived, passive components and do not require an aging management review.

Mechanical Penetrations

The penetrations through the reactor building pressure boundary are designed to maintain the essentially leak tight barrier to prevent uncontrolled release of radioactivity. In addition to supporting the essentially leak tight barrier function, each penetration performs service related functions. Penetrations may also serve as support points for piping passing through the reactor building pressure boundary.

Penetration plate and sleeve material is ASTM A516 Grade 70 material. The plate material is also fabricated to specification ASTM A300.

Mechanical penetrations provide the means for passage of process piping transmitting liquids or gases across the reactor building boundary. A typical mechanical piping penetration, a single barrier piping penetration with a single closure between the process pipe and the reactor building liner, is shown in SAR Figure 5-3.

The penetrations are solidly anchored to the reactor building wall or foundation slab, precluding any requirements for expansion bellows. In accordance with the design requirement of ASME Section III, the piping penetration reinforcing plates and the pipe closure weldments were stress relieved.

A mechanical penetration's boundary for the LRA includes the entire penetration assembly, including the weld to the process piping, but not the process piping within the penetration. Penetrations are designed to maintain the adjacent concrete within an acceptable temperature range. Bellows are not installed in the reactor building penetrations at ANO-1. The reactor building evaluation boundary is shown on ANO-1 SAR Figure 5-3, Detail 5. Spare penetrations consist of a sleeve with welded end cap closure(s) or bolted blind flange plate(s) with gaskets at both ends of the penetration sleeve. During an outage, spare penetrations can readily be converted into additional permanent mechanical or electrical penetrations. The entire spare penetration assembly is considered in the ANO- 1 LRA.

Electrical Penetrations

Electrical penetrations provide the means for electrical and instrumentation conductors to cross the reactor building pressure boundary, while maintaining the essentially leak tight barrier. An electrical penetration through the reactor building is shown in SAR Figure 5-2, Detail 3. The LRA scope includes the metallic components of the electrical penetration that are part of the reactor building's essentially leak tight barrier. The inside steel header plate for the electrical terminals is included in the scope. The wiring, sealing compound, fixtures to hold the sealing compound, and seal welds of the fixtures to the header plate are addressed in environmental qualification reports. Environmental qualification of electrical penetrations is addressed in Section 4.4.

Fuel Transfer Tube

A fuel transfer tube penetrates the reactor building, linking the refueling canal inside the reactor building with the fuel transfer canal in the fuel handling building. This is the underwater pathway for moving fuel assemblies into and out of the reactor building during refueling operations. As part of the reactor building, the tube must assure the essentially leak tight barrier function for the design basis conditions.

The fuel transfer tube arrangement is shown in SAR Figure 5-2, Detail 10. As shown in the figure, the closure between the transfer tube and the sleeve that is integrally welded to the reactor building liner, consists of a circular plate shop welded to the tube and a short segment of pipe to mate with the sleeve.

The transfer tube, blind flange, and gate valve are part of the spent fuel pool system and are addressed in Section 2.3.3.1 and Table 3.4-1.

2.4.1.2 Concrete

Dome and Cylinder Walls

The reinforced concrete dome and cylinder walls are prestressed by a post-tensioning system, as shown in SAR Figure 5-1. The combined strength provided by the concrete, conventional reinforcing steel, and the post-tensioning system is used to satisfy the design loads. Although these three material components act together as one composite system, the post-tensioning system is addressed as a separate component because it is installed and stressed after the reinforced concrete components are complete and because of the unique tendon surveillance program.

Conventional reinforcing is provided near the surface of the cylinder walls and dome primarily to resist local moment and shear loads at discontinuities and for temperature and shrinkage crack control. The conventional reinforcing is accounted for in the strength design of the concrete sections for the internal shear forces and moments resulting from the design loadings.

The concrete sections are thickened and the conventional reinforcing steel is increased at the structural discontinuities to account for the increased stresses in those local areas. Primary structural discontinuities occur at the base of the cylinder and at the transition of the cylinder walls and dome to the ring girder. The ring girder serves as the anchorage area for the upper end of the vertical tendons and for both ends of the dome tendons. Six vertical buttresses are provided along the exterior face of the cylinder to serve as the anchorage points for the hoop tendons. The hoop tendons extend for 120 degrees of arc. Supplementary reinforcing steel is provided at tendon anchorage zones to account for the local forces at the anchorage. The concrete cylinder walls are also thickened and additional reinforcing is provided locally at the equipment hatch to account for the flow of forces in the walls around the relatively large diameter opening required for the hatch. Additionally, the concrete dome is coated with an elastomeric silicone rubber on the exterior to protect the dome from weathering conditions.

Floor

A reinforced concrete floor is provided in the reactor building, above the embedded portion of the liner plate, to protect the liner plate from punctures that could breach the essentially leak tight barrier.

Foundation Slab

The conventionally reinforced concrete foundation slab serves as the structural foundation support for the reactor building. The vertical tendons extend through the foundation slab thickness and are anchored on the underside of the slab. A reinforced concrete enclosure, the lower tendon access gallery, shown in SAR Figure 5-1, is provided at the underside of the foundation slab for access to the lower vertical tendon anchorages for tendon installation and surveillance purposes. The lower tendon access gallery and the foundation slab are constructed of separate concrete pours with horizontal and vertical isolation joints provided. The lower tendon access gallery does not support the intended functions of the reactor building and is therefore, not within the scope of license renewal.

2.4.1.3 Post-Tensioning System

An elevation section of the ANO-1 reactor building (SAR Figure 5-1) shows the orientation of the tendons. A section of a typical post-tensioned tendon assembly is also shown in SAR Figure 5-1.

The reactor building cylinder wall is prestressed by 102 vertical tendons anchored at the top surface of the upper ring girder at the top of the concrete cylinder and at the bottom of the foundation slab. The reactor building cylinder wall is also prestressed with three groups of 57 hoop tendons, minus two tendons that were not installed due to obstructed sheathing. These tendons encompass an arc of 240° for a total of 169 tendons anchored at the three vertical buttresses.

The reactor building dome is prestressed by three groups of 30 tendons oriented at 120° to each other, for a total of 90 tendons anchored at the vertical face of the upper ring girder. Each tendon consists of 186 wires bundled together. Conduits and bearing plates are cast into the concrete shell to receive the tendons, which were installed after construction of the reinforced concrete was complete. The tendons are continuous from anchorage to anchorage, being deflected around penetrations. The post-tensioning system is shown on SAR Figure 5-4. The design of the tendon system provides for the loss of any three adjacent tendons in any of the groups without significantly affecting the load carrying capacity of the reactor building.

The Birkenmeier Brandestinin Ros Vogt system uses parallel wires with cold-formed buttonheads at the ends, which bear upon a perforated steel anchor head, thus providing a positive mechanical means for transferring the prestress force into the concrete shell. Extensive prototypical static, dynamic, and low-temperature testing have been performed on the Birkenmeier Brandestinin Ros Vogt anchorage system to assure that the ultimate capacity of the tendons can be developed.

The post-tensioning system is the primary means of satisfying the controlling design loads of the structure, although the conventional mild steel reinforcing is taken into account when checking representative sections of the structure’s internal forces and moments resulting from the load combinations. The tendon stress remains in the elastic range for the controlling design load combinations.

2.4.2 Reactor Building Internals

The reactor building internals are reinforced concrete and steel structures, supported by the reactor building. The reactor building internals consist of the reactor cavity, two steam generator compartments, and a fuel transfer canal that is located between the two steam generator compartments and above the reactor cavity. The reactor cavity houses the reactor pressure vessel and serves as a radiation shield. Also in the cavities are structural steel, platforms, ladders, and grating for access to the various components for inspection and maintenance.

The steam generator compartments house the steam generators and other components of the RCS, including the reactor coolant pumps. The pressurizer is housed in a separate compartment adjacent to and integral to a steam generator compartment. The primary function of the steam generator compartment walls, commonly referred to as “D-rings” is to serve as secondary shield walls in order to resist the pressure jet loads due to pipe rupture. The dynamic effects caused by the impingement of the jet force on the “D-rings” walls are also considered in the design. The reactor building internals design also considers the effects of radiation and radiation generated heat.

The reactor building internals comprise various structural components, which support or protect in-scope system components or equipment. The reactor building internals fulfill the following intended functions.

- Provide structural support or functional support to safety-related equipment
- Provide shelter or protection to safety-related equipment (including radiation shielding)
- Provide rated fire barriers to confine or retard a fire from spreading to or from adjacent areas
- Serve as internal missile barriers
- Provide structural or functional support to nonsafety-related equipment, failure of which could directly prevent satisfactory accomplishment of required safety-related functions
- Provide a heat sink during DBA or station blackout conditions

A listing of the reactor building internals passive, long-lived components and unique commodities subject to an aging management review, and their intended functions, is provided in Table 3.6-3.

2.4.2.1 Steel

Illustrations of the reactor building internals steel components are provided in SAR Figure 5-8. Structural carbon steel is provided in the reactor building to allow access to the various elevations and areas inside reactor building for inspection and maintenance. The steel also provides support for several nuclear safety-related components, including core flood tanks, reactor building cooling units, emergency core cooling system piping, and electrical instrumentation, control and power.

Part of the floor surface is reinforced concrete, and the remainder is galvanized steel grating. The floor beams are supported by columns or by attachments to the exterior surface of the secondary shield wall. Structural steel, welded to the liner plate, also provides grating support. Attachment welds to the liner plate are within the evaluation boundary of the LRA.

Cranes

The ANO-1 reactor building internals also support various cranes for different maintenance applications. Illustrations of the reactor building crane components are provided in SAR Figure 9-48. The following reactor building crane components are within the scope of license renewal and subject to aging management review.

- Main fuel handling bridge
- Auxiliary fuel handling bridge (abandoned)
- Jib cranes
- Polar crane

These components are being addressed since their potential failure when lifting or carrying heavy loads may impact safety-related components. Therefore, their structural integrity must be maintained. The main and auxiliary fuel handling bridges and control rod drive crane are supported on the secondary shield walls. The fuel tilt machine travels through the transfer tube cast in the cylindrical wall. The fuel tilt machine permits movement of fuel in and out of the reactor building. The control rod drive crane and fuel tilt machine are category 2 structures. Their failure would not impact safety-related components. Therefore, they are not in-scope. The transfer tube is designed to isolate reactor building internals during plant operations with a blank flange on the reactor building internals side and a gate valve on the spent fuel pool side.

Control Rod Drive Service Structure

The control rod drive service structure located on top of the reactor vessel, supports the control rod drive mechanisms from excessive lateral motion to ensure that the control rods can drop into the core under design basis loading condition. The control rod drive service structure consists of five major assemblies.

- Lower control rod drive service structure skirt: A slotted carbon steel cylinder is welded to the upper surface of the reactor vessel closure head. A mating flange is welded to the skirt, providing a seating surface to which the upper control rod drive service structure is bolted.
- Upper control rod drive service structure skirt: A carbon steel cylindrical shell, with a lower flange, connects to the lower control rod drive service structure skirt. An upper flange connects to the closure head service structure shell flange.

- Closure Head Service Structure Shell: A carbon steel cylinder, attached to the upper control rod drive service structure skirt, supports the control rod drive service structure platform assembly.
- Control Rod Drive Service Structure Strut Support Assembly: Horizontal carbon steel beams, oriented in a radial direction, are welded to the closure head service structure shell on one end and supported on the other by angled beams.
- Control Rod Drive Service Structure Platform Assembly: A horizontal platform made of carbon steel beams is attached to the top of the closure head service structure shell and the control rod drive service structure strut support assembly. The control rod drive service structure platform assembly restrains the top ends of the control rod drive mechanisms from lateral movement during design basis loading.

Reactor Vessel Support Skirt

The reactor vessel supports include a support skirt and support flange. The reactor vessel support skirt is a cylindrical structure that supports the reactor vessel. The support skirt rests on a sole plate. The sole plate is fixed to a supporting, reinforced concrete pedestal through a steel flange bolted to the pedestal. The evaluation boundary of the reactor vessel support skirt begins at the weld of the skirt to the reactor vessel transition forging and terminates at the bottom of the skirt flange. The evaluation boundary also includes the exposed surface of the anchor bolts and shear pins.

The reactor vessel support skirt was designed, fabricated, tested, and inspected in accordance with ASME Section III (Reference 2.4-5). The support skirt consists of two carbon steel semi-circular rings welded together to form a cylinder. This cylinder is welded to the bottom of the reactor vessel transition forging. The cylinder has holes for ventilation of the reactor vessel cavity. The anchor bolts are prestressed to accommodate the loads of a design basis seismic event.

2.4.2.2 Concrete

The reactor building internals are supported by the reactor building. Structural reinforced concrete forms the basement floor slab (cover over the liner plate); columns; the walls surrounding the steam generators, reactor, and pressurizer; valve pits and pipe chases; other interior walls; the slabs of the valve pits and pipe chases; missile shields; fuel transfer canal; and the removable concrete hatches and covers. The concrete utilized in the reactor building internals meets the requirements of ANO-1 specifications.

2.4.3 Auxiliary Building

The ANO-1 auxiliary building is structurally independent from other structures. It is located adjacent to the ANO-1 reactor building and the turbine building, but is seismically separated from them by one-inch wide joints filled with an elastic resilient material. It is a conventionally designed, reinforced concrete structure founded on bedrock. The auxiliary building has reinforced concrete foundation mats at elevation 317' and elevation 335'. Reinforced concrete floor slabs are at elevations 335', 354', 372', 386', and 404'.

The concrete substructure and concrete walls and floors are designed for dead, live, and lateral loads. Seismic, wind, and other loads are carried to the foundation by diaphragm action in slabs and shear wall action in walls. The building is partly above grade (i.e., grade is at elevation 354') and partly below grade. Exterior concrete construction joints contain waterstops at joints below the plant's design flood level of elevation 361'.

The auxiliary building houses various systems that support normal operation, shutdown, and accident conditions of ANO-1. Most areas within the auxiliary building are classified and designed as seismic category 1 structures. Seismic category 1 structures are those whose failure could cause the uncontrolled release of radioactivity or are essential for safe reactor shutdown and the immediate and long-term operation following a loss of coolant accident. Some of the category 1 areas significant to plant safety include the control room, spent and new fuel shipment and storage facilities, and the cable spreading room. Category 1 areas have been determined to be within the scope of 10CFR54.4(a)(1).

Several internal boundaries within the auxiliary building are classified as seismic category 2 structures. Seismic category 2 structures are those whose failure may interrupt power generation, but will not prevent a safe reactor shutdown. However, failure of some of the category 2 areas within the auxiliary building (i.e., the liner plate within the spent fuel pool area and the small pipe chase at elevation 341', which is structurally connected to the category 1 wall along column line H) could possibly affect the function of a safety-related structure, system, or component. These category 2 areas are within the scope of 10CFR54.4(a)(2).

In addition, some areas within the auxiliary building (i.e., areas with 10CFR50.48-required fire barriers) are within the scope of 10CFR54.4(a)(3).

The boron holdup tank vault is located below grade, and is structurally connected to the auxiliary building,. The borated water storage tank sits on top of the vault. The post-accident sampling system building is anchored to the top of the ANO-1 and ANO-2 tank vaults (elevation 355'-4"). It contains category 2 equipment, but was designed to category 1 criteria to avoid potential interaction with safety-related equipment.

The auxiliary building comprises various structural components and commodities, which support or protect in-scope system components or equipment. The auxiliary building and its structural components and commodities fulfill the following intended functions.

- Provide essentially leak tight barriers to prevent uncontrolled release of radioactivity

- Provide structural support or functional support to safety-related equipment
- Provide shelter or protection to safety-related equipment (including radiation shielding)
- Provide rated fire barriers to confine or retard a fire from spreading to or from adjacent areas
- Serve as missile (internal or external) barriers
- Provide structural or functional support to nonsafety-related equipment, failure of which could directly prevent satisfactory accomplishment of required safety-related functions
- Provide protective barriers for internal flood event
- Provide protective barriers for external flood event
- Provide for storage of spent fuel assemblies

A listing of the auxiliary building passive, long-lived components and unique commodities, subject to an aging management review, is provided in Table 3.6-4. Components and commodities are grouped based on materials of construction, with sub-materials indicated as appropriate. Structural intended functions by component and commodity are also listed. The construction materials for the auxiliary building components and commodities are steel, threaded fasteners, and concrete (excluding prestressed concrete), and elastomers. In the material group fire barriers, fire doors are grouped as steel components, while fire walls and slabs are concrete components. Although the turbine building itself is not within the scope of license renewal, some fire doors and fire walls and slabs (10CFR50.48-required) within the turbine building are in-scope and subject to aging management review. These are addressed along with those for the auxiliary building. For the material group elastomers, none of the components or unique commodities are subject to an aging management review. There are no components or unique commodities associated with the material groups earthen structures or Teflon.

Commodities considered common to the auxiliary building and other in-scope structures are discussed in Section 2.4.6.2.

2.4.4 Intake Structure

The intake structure houses the circulating water, fire, and service water pumps, motor control centers, and traveling screens. It is a conventionally designed reinforced concrete structure founded on bedrock and located at the termination of the intake canal. The structure can be divided into two major sections. The first section is the portion of the building above grade elevation (El. 353'-3"). The remaining section is the pump bay area located below grade and partially submerged in water.

The above grade section contains pump motors, valve motor actuators, and related equipment. This section of the building has three predominant elevations which are elevation 354', elevation 366', and elevation 378'. HVAC equipment is located in the penthouse at elevation 378'. Pump motors and valve motor actuators required for plant protection (i.e., fire water and service water) are located on elevation 366', which is above the plant design flood level of elevation 361'. Generally, the remaining pump motors required for normal plant operation, such as the circulating water and screen wash pumps, are located on elevation 354'. System components related to plant protection, which are not adversely affected by flood waters or which would not be required during a flooding event (i.e., the lake/emergency cooling pond sluice gate actuators), are also located on elevation 354'.

The below grade portion of the intake structure contains the pump bays for various plant systems. The four circulating water system bays take their suction directly from Lake Dardanelle. Three service water system bays are located directly behind the circulating water bays. Depending on plant conditions, sluice gates in the service water bays can be aligned so that the fire water and service water pumps can take suction directly from Lake Dardanelle or from the emergency cooling pond.

The ANO-1 intake structure is integrally connected to the ANO-2 intake structure with a shear key and a row of reinforcing bars near the elevation 354' slabs. The intake structure gantry crane (L-007) is shared between the ANO-1 and ANO-2 intake structures. The gantry crane is supported by steel crane rails and girders on reinforced concrete piers. Procedurally, it is parked at a safe distance from the intake structure.

The portions of the intake structure required to be seismic category 1 are those that provide support to service water system components. The remainder of the building is seismic category 2. Thus, the portion of the intake structure housing seismic category 1 system components has been designed to category 1 standards to ensure the safe operation of the service water pumps. Category 1 building areas have been determined to be within the scope of 10CFR54.4(a)(1). Category 2 building areas in the intake structure are not within the scope of 10CFR54.4(a)(2).

The intake structure comprises various structural components and commodities, which support or protect in-scope system components or equipment. The intake structure and its structural components and commodities fulfill the following intended functions.

- Provide structural support or functional support to safety-related equipment

- Provide shelter or protection to safety-related equipment
- Serve as missile (internal or external) barriers
- Provide structural or functional support to nonsafety-related equipment, failure of which could directly prevent satisfactory accomplishment of required safety-related functions
- Provide protective barriers for external flood event

A listing of the intake structure passive, long-lived components and unique commodities, subject to an aging management review, is provided in Table 3.6-5. Components and commodities are grouped based on materials of construction, with sub-materials indicated as appropriate. Structural intended functions by component and commodity are also listed. The construction materials for the intake structure components and commodities are steel, threaded fasteners, and concrete (excluding prestressed concrete). For the material group fire barriers, there are no components or commodities within the scope of license renewal (i.e., none are 10CFR50.48-required). None of the components or unique commodities for the material group elastomers are subject to an aging management review and there are no components or commodities associated with the material groups earthen structures or Teflon.

Commodities considered common to the intake structure and other in-scope structures are discussed in Section 2.4.6.2.

2.4.5 Earthen Embankments

The structures included within this group for aging management review are earthen embankments submerged partially, or totally, in Lake Dardanelle and contained within their own boundaries. The following seismic category 1 structures are evaluated within this group.

- Emergency cooling pond
- Intake and discharge canals

The ANO-1 earthen embankments provide a heat sink during DBA or station blackout conditions. A listing of the earthen embankments components subject to an aging management review and their intended function, is provided in Table 3.6-6.

2.4.5.1 Emergency Cooling Pond

The emergency cooling pond is a seismic category 1, 14-acre, kidney-shaped pond located northwest of the plant. A general layout of the ECP is shown in SAR Figure 9-32. The bottom of the pond is at elevation 341 feet, with normal water level between five and six feet. The maximum ECP level of six feet is maintained by a spillway that discharges back to Lake Dardanelle. Plant discharge (ECP inlet) flows into a structure that is surrounded by a 100 foot long weir that peaks at elevation 346 feet. The purpose of the weir is to promote a uniform flow distribution in the ECP and direct the hot discharge to the surface. This maximizes the surface temperature, which maximizes heat rejection. The plant intake piping is at the lowest point of the ECP, with the pipe centerline at elevation 339.5 feet. The location of supply and return lines at opposite extremes serves to prevent a hydraulic short circuit.

The ECP is excavated in impervious clay strata with the bottom of the pond about 4 to 16 feet above rock. To preclude undercutting by water flow over the spillway, the downstream sections of the spillway crest voids and the adjacent embankment voids are pumped with an elastic type of grout. Also, a filter fabric material has been placed below, and an impervious membrane fabric placed above, the articulated concrete slabs to deter erosion. The pond side slopes are protected against wave action by 18 inches of riprap placed on the north side of the pond, to eliminate leakage through the existing filter material to the underdrain. A series of weirs assists in elimination of silt problems.

The ECP serves as a heat sink in the unlikely event of a loss of Lake Dardanelle water inventory. Under controlled conditions, with Lake Dardanelle available, the ECP may provide service water or auxiliary cooling water to ANO-1 with ANO-2 or ANO-1 providing normal makeup as necessary to preserve ECP inventory.

2.4.5.2 Intake and Discharge Canals

The intake canal supplies the reservoir water to ANO-1 for once-through cooling. It is also used to supply cooling tower makeup water and service water for ANO-2. The intake canal conveys water from the Illinois Bayou portion of Lake Dardanelle to the intake structure. It is approximately 4,000 feet long and, at normal pool elevation of 338 feet, the width varies from 80 feet at the mouth to 135 feet at the intake structure, with an average depth of 14 feet.

The intake canal deepens and widens before reaching the intake structure. This feature causes debris entrained in the lake water to drop out of suspension and collect in this area. This prevents excessive amounts of silt from entering the service water system. The discharge canal returns the cooling water to the reservoir. It is approximately 600 feet long and has an average width of 165 feet and with an average depth of 11 feet at normal pool elevation of 338 feet. Both canals are completely excavated and contain no sections formed by dikes or in-fill. Bank slopes are planted with grass or protected by rip-rap to prevent erosion. These canals are within the scope of license renewal and subject to aging management review.

2.4.6 Other Structures and Structural Components

2.4.6.1 Aboveground/Underground Yard Structures

The ANO-1 aboveground/underground structures and trenches requiring an aging management review are the following.

- Q-Condensate storage tank foundation
- Emergency diesel fuel oil storage tank vault
- Bulk fuel oil storage tank foundation
- Alternate AC diesel generator building foundation
- Electrical manholes
- Borated water storage tank foundation

Most of the above structures are seismic category 1 and their structural function is to provide support or protection to seismic category 1 (Q, safety-related) and/or seismic category 2 (non-Q, non-safety-related) equipment and components.

<u>Q-Condensate Storage Tank Foundation</u>

The Q-CST, including the valve pit and pipe trench, is a seismic category 1 structure located on the west side of the ANO-1 reactor building. The Q-CST is supported on a 2'-6" thick by 52 foot reinforced concrete octagon-shaped mat foundation. The foundation is supported on 42" diameter drilled concrete piers (caissons) embedded in bedrock. Two 11'-6" by 12'-6" by 8'-6" valve pits are located partially underneath and on opposite (i.e., north and south) sides of the mat foundation. The south valve pit is for ANO-1 and the north valve pit serves ANO-2. A 5'-0" high reinforced concrete wall surrounds the lower portion of the Q-CST to protect against loss due to external missile, assuring an adequate water supply will be available for the emergency feedwater system until transfer to the assured source, which is service water. The 1'-6" thick missile wall is keyed and integral to the Q-CST foundation mat. Category 1 structures have been determined to be within the scope of 10CFR54.4(a)(1).

<u>Emergency Diesel Fuel Oil Storage Tank Vault</u>

The emergency diesel fuel oil storage tank vault is a rigid reinforced concrete box structure located on the northwest side of the reactor building. It contains four diesel fuel storage tanks partitioned into separate rooms to provide protection against fire or flooding. The walls are designed to withstand hydrostatic loading over their full height. The structure has a mat foundation founded on rock. The vault is anchored to rock and has ventilation openings above flood elevation. The outside door is of watertight construction. The diesel fuel vault is a category 1 structure that is within the scope of 10CFR54.4(a)(1).

<u>Bulk Fuel Oil Storage Tank Foundation</u>

The bulk fuel oil storage tank is a 180,000 gallon, common storage tank for the on-site emergency AC power system and other systems such as the startup boiler, as discussed in SAR Section 9.5.4.2.

It has a non-Q, category 2 foundation and is not part of the safety-related fuel oil supply/storage associated with the on-site emergency AC power system. Therefore, the required independence of the AAC power system is maintained. The fuel level in the bulk fuel oil storage tank is administratively controlled to maintain a minimum of 4-1/2 days of fuel for the AAC generator at all times. The required minimum level is established considering other users that could be operable at the same time. However, the foundation is required to support the tank; therefore, the bulk fuel oil storage tank foundation has been determined to be within the scope of 10CFR54.4(a)(2).

Alternate AC Diesel Generator Building Foundation

The alternate AC diesel generator building is north of, and adjacent to, the north side berm of the bulk fuel oil storage tank. It is a seismic category 2 structure designed and built to the UBC (Reference 2.4-7). With the exception of the power distribution switchgear, major components of the AAC diesel generator are located in this building. It is a reinforced concrete slab, founded on grade beams supported by drilled in piers (caissons). The building is divided into an electrical equipment area, which also serves as the local operations station, and an engine room. This building houses the engine generator set, fuel oil transfer pump, fuel oil day tank, air start system, engine generator control cabinets, HVAC, and fire protection systems. The AAC system is designed, constructed, tested and maintained as a non-Q system conforming to augmented QA requirements based on NRC Regulatory Guide 1.155, "Station Blackout." Since the building foundation is required to support the alternate AC diesel generator, it has been determined to be within the scope of 10CFR54.4(a)(2).

Category 1 Electrical Manholes

The seismic category 1 electrical manholes are placed at various locations within the plant site. They are relatively small reinforced concrete structures founded either on natural soil or backfill materials. They are surrounded by backfill material and located partially underground. An access opening in the top slab, at grade level, is provided with a missile resistant reinforced concrete or carbon steel cover. The reinforced concrete foundations of these structures provide a fixed support for the walls that transmits loads to them. Since these structures are relatively small, relatively light loads are transmitted to the natural soils or backfill materials. Their foundations are completely independent of each other and the foundations of other structures. The category 1 electrical manholes are within the scope of 10CFR54.4(a)(1).

Borated Water Storage Tank Foundation

The BWST rests on a concrete slab that is part of the category 1 auxiliary building. The BWST is located on the roof of the boron holdup tank vault, which is subdivided by concrete walls into four separate areas. The roof slab requires approximately two feet in thickness to meet the tank structural requirements.

However, the biological shielding requirements governed the tank vault ceiling and final construction determined that a 4-foot thickness would be appropriate. The slab was designed as a two-way slab and the steel was sized appropriately for the application. A small ring wall, filled with oiled sand, was placed on the top of the concrete slab to separate the tank bottom from the top of the concrete and to provide a small slope for tank drainage purposes. Since the concrete foundation width and thickness is much larger than required for tank support, the slab is actually closer to a rigid mat foundation than to a conventional ring wall. The BWST foundation is within the scope of 10CFR54.4(a)(1).

The intended functions of the above structures components were determined by reviewing information contained in the SAR and ANO-1 engineering documents, as well as NEI 95-10 (Reference 2.4-1). The aboveground/underground yard structures fulfill the following intended functions.

- Provide structural support or functional support to safety-related equipment
- Provide shelter or protection to safety-related equipment
- Provide rated fire barriers to confine or retard a fire from spreading to or from adjacent areas
- Serve as missile (internal or external) barriers
- Provide structural or functional support to nonsafety-related equipment, failure of which could directly prevent satisfactory accomplishment of required safety-related functions
- Provide protective barriers for internal flood event

A listing of the aboveground/underground yard structures components and unique commodities subject to an aging management review, and their intended functions, is provided in Table 3.6-7.

2.4.6.2 Bulk Commodities

Bulk commodities are structural members or items that support or protect various in-scope system components or equipment, and are common to two or more structures. Bulk commodities meet the criteria of 10CFR54.4(a)(1), 10CFR54.4(a)(2), or 10CFR54.4(a)(3). In-scope structures with common bulk commodities include the reactor building (including reactor building internals), auxiliary building, intake structure, diesel fuel vault, BWST foundation, Q-CST foundation, and pipe trenches. Although the turbine building itself is not within the scope of license renewal, some bulk commodities (fire wrap banding, fire damper mountings, fire hose reels, fire wraps, and fire stops) within the turbine building are in-scope. These are addressed along with those for the in-scope structures' bulk commodities. Bulk commodities fulfill the following intended functions.

- Provide structural support and functional support to safety-related equipment
- Provide shelter or protection to safety-related equipment (including radiation shielding)
- Provide rated fire barriers to confine or retard a fire from spreading to or from adjacent areas
- Serve as missile (internal or external) barriers

- Provide structural or functional support to nonsafety-related equipment, failure of which could directly prevent satisfactory accomplishment of required safety-related functions
- Provide protective barriers for internal flood event
- Provide protective barriers for external flood event

A listing of passive, long-lived, bulk commodities subject to an aging management review is provided in Table 3.6-8. Bulk commodities are grouped based on materials of construction, with sub-materials indicated as appropriate. Structural intended functions by commodity are also listed. The construction materials for bulk commodities are steel, threaded fasteners, concrete (excluding prestressed concrete), fire barriers, elastomers, and Teflon. There are no bulk commodities associated with the material group earthen structures.

2.4.7 References for Section 2.4

2.4-1 NEI 95-10, "*Industry Guidelines for Implementing the Requirements of 10CFR Part 54 – The License Renewal Rule,*" Revision 0, Nuclear Energy Institute, March 1996.

2.4-2 ASTM A36, "*Standard Specification for Carbon Structural Steel,*" American Society for Testing and Materials.

2.4-3 ASTM A516, "*Standard Specification for Pressure Vessel Plates, Carbon Steel, for Moderate-and Lower-Temperature Service,*" American Society for Testing and Materials.

2.4-4 ASME *Boiler and Pressure Vessel Code*, American Society of Mechanical Engineers.

2.4-5 ASME *Boiler and Pressure Vessel Code*, Section III, "Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers.

2.4-6 ASTM A300, "*Specification for Notch Toughness Requirements for Normalized Steel Plates for Pressure Vessels*," American Society for Testing and Materials.

2.4-7 Uniform Building Code, 1967 Edition.

2.5 ELECTRICAL AND INSTRUMENTATION AND CONTROLS SYSTEM SCOPING AND SCREENING RESULTS

2.5.1 Purpose and Scope

The purpose of this review is to identify electrical SSCs at ANO-1 that are subject to an aging management review in accordance with 10CFR Part 54 in order to address aging concerns and manage aging effects during the period of extended operation. This process consists of two steps. The first is scoping to determine which structures and systems must be included in the license renewal process. The second step is screening to determine which components and structures of the included SSCs require an aging management review. The scoping process is discussed in Section 2.5.2 and the screening is discussed in Section 2.5.3.

2.5.2 Scoping of Electrical SSCs

The 10CFR Part 54 regulation requires a scoping of SSCs. SSCs are to be included in the license renewal process if they are.

- safety-related,
- non safety-related, but whose failure could prevent satisfactory accomplishment of a safety function, or
- relied on for compliance with Appendix R, EQ, PTS, ATWS, or station blackout.

Section 2.1 provides a discussion of the scoping process at a system level and Table 2.2-1 indicates those systems containing electrical components that are in the scope of license renewal. Additional information on electrical systems required for certain regulated events is provided herein.

2.5.2.1 EQ SSCs

Safety-related components that must continue to operate following accidents and HELBs and that are located in harsh environments resulting from that accident or HELB, are controlled as part of the EQ Program. The EQ Program tracks both components with individual equipment numbers and generic components used throughout the plant (such as cables). Because the EQ Program addresses aging of components in its scope, it is an important program to support license renewal. The harsh environments analyzed for ANO-1 are in the reactor building and in a number of rooms in the auxiliary building. Components included in the EQ Program are in the scope of license renewal per 10CFR Part 54. The detailed discussion of the EQ Program and the components covered by the EQ Program is contained in Section 4.4.

2.5.2.2 ATWS Electrical SSCs

In 1990, ANO-1 installed a DROPS/DSS for a diverse reactor trip and a DROPS/AMSAC for a backup actuation of EFW and a diverse main turbine trip. These systems place ANO-1 in compliance with 10CFR50.62 and provide plant protection in the event the reactor protection system fails to perform its function during an ATWS event. These are small, non-Q, self-contained microprocessor based systems with signal isolators connected to RCS pressure, nuclear instrumentation reactor power, and main feedwater flow signals. Trip relays are provided for interfacing with the plant components.

The electronics are considered active and the entire system is periodically calibrated and tested to verify proper operation. The electrical components in the DROPS/DSS and the DROPS/AMSAC are in the scope of license renewal. The aging management review also includes the cabling associated with the field sensors (pressure, flow, and reactor power) that supply inputs to the DROPS/DSS and the DROPS/AMSAC through signal isolators.

2.5.2.3 Station Blackout Electrical SSCs

In order to meet the requirements of 10CFR50.63, Entergy Operations installed a 4400-kW diesel generator in a separate structure that is totally independent of the other emergency power sources and their auxiliaries. The system is referred to as the alternate AC diesel generator or as the station blackout diesel. The diesel generator can be manually started within 10 minutes of a station blackout and used to power the class 1E electrical busses of both ANO units. The AAC diesel generator has its own air start system, fuel oil transfer pump, day tank, cooling, and lube oil subsystems. The AAC diesel generator is an active component, and is periodically tested to verify operability. The electrical components of the AAC diesel generator that supply the Class 1E busses are included in the scope of license renewal.

2.5.3 Screening of Electrical SSCs

As part of the integrated plant assessment for license renewal, only systems, structures, and components that are classified as long-lived and passive and which are within the scope of 10CFR Part 54 are subject to an aging management review. Active SSCs and those identified as having a periodic replacement schedule are not subject to an aging management review. ANO-1 participated in an industry wide initiative coordinated by NEI that developed a generic commodity evaluation methodology. The passive long-lived electrical components at ANO-1 were categorized using NEI 95-10 (Reference 2.5-1), Appendix B as a guide. In accordance with the NEI categorization, this screening identifies the following passive electrical components that are generic to ANO-1 systems: splices, connectors, terminal blocks, and cables. Splices, connectors, and terminal blocks are sometimes lumped together in the category of electrical connections.

The passive electrical components included in the aging management review are the separate electrical components that are not sub-components of a larger complex assembly. For example, the wiring, terminal blocks, and connectors located internal to a breaker cubicle are sub-components of the breaker. Because the breaker is an active component not subject to an aging management review, the subcomponents are not within the scope of this review.

2.5.3.1 Connectors

A connector is usually considered to be the plug and socket arrangement that allows an easy disconnection and reconnection of the electrical component. For the purpose of the ANO-1 LRA, cable splices, cable couplers, and insulating tape used in splices are also included in the scope of review since these are passive components utilized in the connection of cables that may not be evaluated as part of a larger component.

2.5.3.2 Terminal Blocks

The terminal blocks at ANO-1 are solid section blocks that are phenolic block molded and are capable of withstanding considerable temperature and radiation exposures. Terminal blocks that are not sub-components of larger active assemblies are in the scope of this review.

2.5.3.3 Cables

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 having overall coverings. Cable connections are used to connect the cable conductors to other cables or electrical devices and include connectors, splices, and terminal blocks. Cables in the scope of this review are those that are separate components and not part of some larger complex assembly.

2.5.4 Electrical Components Not in the Scope of License Renewal or Not Subject to Aging Management Review

A brief discussion is provided for certain electrical components at ANO-1 which are not in the scope of license renewal or are not subject to an aging management review.

2.5.4.1 Electrical Bus

Electrical buses at ANO-1 are not in the scope of license renewal or are not subject to an aging management review due to the fact that they are either part of a larger complex assembly or they are not safety-related. The isolated-phase bus that connects the main generator to the main transformers is not safety-related. The switchyard bus is likewise not safety-related. Some safety-related bus is contained within the safety-related 4.16kV switchgear, however, this bus is considered to be part of a larger complex assembly containing bus, breakers, relays, wiring and controls. Switchgear is classified as being active by 10CFR Part 54 and NEI 95-10.

2.5.4.2 Insulators

Electrical insulators associated with the ANO-1 switchyard are not in the scope of license renewal since they are not safety-related. Other insulators found in the plant are either not safety-related or are part of a larger complex assembly.

2.5.4.3 Transmission Conductor

Transmission conductors at ANO-1 are used in non safety-related applications. Therefore, they are not in the scope of license renewal.

2.5.5 References for Section 2.5

2.5-1 NEI 95-10, "*Industry Guidelines for Implementing the Requirements of 10CFR Part 54 – The License Renewal Rule,*" Revision 0, Nuclear Energy Institute, March 1996.

3.0 AGING MANAGEMENT REVIEW RESULTS

This section describes the results of the aging management reviews of the components and structures, identified in Section 2.0, that require aging management reviews. Specifically, this section:

- provides references to the descriptions of common aging management programs (Section 3.1),
- identifies the components and structures subject to aging management review and their intended functions,
- describes, or references, the processes used to identify aging effects requiring management (Appendix C describes the process for identifying aging effects associated with non-Class 1 mechanical components, which encompasses engineered safeguards, auxiliary system, and steam and power conversion system components),
- discusses the materials and environments which result in aging effects,
- identifies the aging effects requiring management,
- describes industry and operating experience with respect to the applicable aging effects, and
- lists the aging management programs that will manage the identified aging effects.

This section does not describe the aging management programs nor discuss how the programs will manage the identified aging effects. Instead, Appendix B describes these programs and provides the information necessary to demonstrate that the identified aging effects will be adequately managed.

3.1 COMMON AGING MANAGEMENT PROGRAMS

3.1.1 Chemistry Monitoring

This information is contained in Section 4.7 of Appendix B.

3.1.2 Quality Assurance

This information is contained in Section 2.0 of Appendix B.

3.1.3 Structure and System Walkdowns

This information is contained in Sections 4.16 of Appendix B.

3.2 REACTOR COOLANT SYSTEM

3.2.1 Description of the Process to Identify the Aging Effects Requiring Management for Reactor Coolant System Components

RCS components within the scope of license renewal that require aging management review are identified in Section 2.3. Their intended functions are identified in Table 3.2-1. Mechanical and structural components of the RCS include:

- RCS piping and letdown coolers
- Pressurizer
- Reactor vessel
- Reactor vessel internals
- Once-through steam generators
- Reactor coolant pumps
- Control rod drive mechanism pressure boundary

Entergy Operations, representing ANO-1, participated in a BWOG effort that developed a series of topical reports to demonstrate that the aging effects for RCS components are adequately managed for the period of extended operation under a renewed license. The following BWOG topical reports applicable to the ANO-1 RCS have been approved by the NRC.

- BAW-2243A , Reactor Coolant System Piping (Reference 3.2-1)
- BAW-2244A, Pressurizer (Reference 3.2-2)
- BAW-2251A, Reactor Vessel (Reference 3.2-3)
- BAW-2248A, Reactor Vessel Internals (Reference 3.2-4)

NRC-approved reports may be incorporated by reference provided the conditions of approval contained in the safety evaluation of the specific report are met. These reports have been incorporated by reference into the ANO-1 LRA as discussed in Section 2.3.1.2. Time-limited aging analyses associated with components of the RCS are discussed in Section 4.0.

Determination of the aging effects applicable to RCS components begins with identification of the potential aging effects defined in industry literature. From this set of potential aging effects, the materials, operating environment, and operating stresses define the aging effects requiring management for each component that is subject to aging management review. Aging effects requiring management are then validated by a review of industry and ANO-1 operating experience, to provide assurance that all aging effects requiring management are identified.

The review to identify the aging effects requiring management for RCS components considers the following potential aging effects that have been identified by reviewing industry literature.

- Loss of material
- Cracking (initiation and growth)—Cracking due to fatigue is a time-limited aging analysis and is addressed in Section 4.3.
- Reduction of fracture toughness—Reduction of fracture toughness of the reactor vessel beltline region due to neutron embrittlement is a time-limited aging analysis and is addressed in Section 4.2.
- Loss of mechanical closure integrity of bolted closures
- Dimensional changes by void swelling
- Mechanical distortion
- Fouling

The determination of aging effects requiring management considers the materials, environment, and stresses of ANO-1 components. The aging effects requiring management for the RCS components are discussed in the following sections.

3.2.2 RCS Piping and Letdown Coolers

Reactor coolant system piping subject to aging management review is identified in Section 2.3.1.3. As described in Section 2.3.1.3, ANO-1 is bounded by BAW-2243A (Reference 3.2-1) with regard to RCS piping and associated materials of construction. The approach for identifying aging effects requiring management for the RCS piping is described in Section 3.2.1. RCS items not within the scope of BAW-2243A that are evaluated in this section include instrumentation tubing, RTE thermowells, and the letdown coolers.

<u>Environment and Stress</u>

The operating environment of the ANO-1 RCS piping is consistent with that described in BAW-2243A, Section 3.1.1. The operating environment for the instrumentation tubing and the RTE thermowells is the same as for the RCS piping. The Primary Chemistry Monitoring Program includes specifications to periodically monitor the primary coolant. Limitations are established on dissolved oxygen, halides and other impurities. Corrective actions are taken in the event the primary coolant parameters are out of specification. RCS chemistry is maintained in accordance with the Primary Chemistry Monitoring Program.

Reactor coolant system piping is designed to accommodate service loadings (i.e., levels A through D); however, operation under level A and B service conditions contribute to normal aging stresses for the piping. ANO-1 has not been subjected to a level C or D event. Therefore, the ANO-1 RCS piping is bounded by BAW-2243A with respect to the qualitative assessment of stress.

The operating environments for the letdown coolers include primary water chemistry on the tube side and treated water from the intermediate cooling water on the shell side. The letdown coolers are designed to accommodate service loads defined by ASME Section III-C on the tube side and ASME Section VIII on the shell side.

<u>Aging Effects Requiring Management</u>

The aging effects applicable to the ANO-1 RCS piping are consistent with those described in Section 3 of BAW-2243A since ANO-1 is bounded by the generic report with respect to materials of construction, operating environment, level A and B service conditions, and operating experience. The results of the aging effects review are contained in BAW-2243A. The aging effects requiring management for the RCS piping are summarized in Table 3.2-1.

Reduction of fracture toughness for RCS valves fabricated from CASS was identified as an applicable aging effect in BAW-2243A. No Class 1 piping at ANO-1 is fabricated from CASS. The assessment of reduction of fracture toughness of RCS valves required evaluation of the plant-specific material forms and temperature environment of the valves that form the Class 1 boundary at ANO-1. Class 1 CASS valves were identified and an assessment was made to determine if the valves are exposed to elevated temperatures during power operation. A threshold temperature of 482°F (350°C) is a conservative limit below which reduction of fracture toughness is not an applicable aging effect. The

ANO-1 evaluation found only three Class 1 valves fabricated from CASS that are exposed to operating temperatures above 482°F and susceptible to reduction of fracture toughness: two 2 1/2-inch valves in the letdown line and the 2 1/2-inch pressurizer spray line block valve. The saturated lower bound fracture toughness of these valve bodies is approximately equal to the fracture toughness of austenitic weldments in pipes made using the submerged arc welding process (BAW-2243A, Section 4.2). Therefore, volumetric inspections of stainless steel piping welded joints between 1-inch and 4-inch NPS in the letdown piping or the pressurizer spray line piping, as defined by ANO-1 risk-informed ISI program for Examination Category B-J, will bound the subject valves since valve bodies have thicker walls and lower stresses than adjacent piping.

Items not within the scope of BAW-2243A are evaluated as follows. Instrumentation tubing and RTE thermowells are fabricated from austenitic stainless steel and are subject to the same environment as the other stainless steel piping in the RCS. Aging effects requiring management for the instrumentation tubing and RTE thermowells will be consistent with aging effects identified for stainless steel piping.

Aging effects requiring management for the letdown coolers include cracking of the tubes by fatigue and stress corrosion cracking on the tube surface exposed to intermediate cooling water, loss of material of tubes due to vibration, loss of material from the interior of the shell by corrosion, loss of material of the external surface of the heat exchanger by boric acid wastage, and loss of mechanical closure integrity.

<u>Industry Experience</u>

The industry experience for RCS piping is described in Section 3 of BAW-2243A.

<u>ANO-1 Operating Experience</u>

ANO-1 operating experience was reviewed to validate the identified aging effects requiring management for the RCS piping. This review included the station information management system, condition reporting system, and licensing event database. The review of ANO-1 operating experience identified five leaks associated with RCS small bore piping (1½" drain off RCP 32B in 1989-90, a leak at HPI/MU vent line in 1989, pinhole leaks at coupling adjacent to a decay heat valve in 1993, core flood tank drain in 1996 and a circumferential crack on a HPI/MU drain in 1998). All of these leaks and cracks were caused by vibrational fatigue. This cracking was due to design problems, which have since been corrected. The corrective actions taken resolved the design problem and ensured this problem would be addressed in subsequent design work.

Loss of material by boric acid wastage has previously been identified on the discharge cold leg HPI nozzle region. This condition was corrected. The ANO-1 Boric Acid Corrosion Prevention Program will monitor this type of aging effect and ensure appropriate corrective action.

Industry operating experience indicated that additional elements of bolting maintenance practices should be considered, such as personnel training, installation and maintenance procedures, plant-specific bolting degradation history, and corrective measures. The NRC captured the lessons from this experience in IE Bulletin 82-02, “Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants,” and Generic Letter 91-17, "Bolting Degradation or Failure in Nuclear Power Plants," and directed each licensee to assure that these lessons were being incorporated at their plant.

ANO-1 has taken actions to improve bolting practices in accordance with NRC Bulletin 82-02 and Generic Letter 91-17. Guidelines developed for preparing, installing and tightening threaded fasteners are applied to maintenance activities that involve threaded fasteners. In response to GL 91-17, training based on review of the EPRI bolting reports was provided to the appropriate departments. The training implemented as a result of GL 91-17 is captured in the "Bolting and Torquing Activities," program described in Appendix B. This program ensures proper bolt material selection and bolting preload control as well as proper maintenance procedures and practices. These actions resolve the bolting concerns of NRC Bulletin 82-02 and Generic Letter 91-17 for ANO-1.

ANO-1 Technical Specification leakage limits, ASME Section XI bolting examination (Examination Categories B-G-1 and B-G-2), the Boric Acid Corrosion Prevention Program, and continuation of routine ANO-1 maintenance practices reviewed under IE Bulletin 82-02 and GL 91-17 will assure management of loss of mechanical closure integrity for all bolted closures in the RCS.

Conclusion

During the review of industry information, NRC generic communications, and ANO-1 operating experience, no additional aging effects beyond those discussed in this section were identified. The aging effects requiring management for the RCS piping and the letdown coolers are summarized in Table 3.2-1. These aging effects must be adequately managed so that the intended functions listed in Table 3.2-1 will be maintained consistent with the current licensing basis for the period of extended operation. These aging effects requiring management will be adequately managed by the following existing programs, which are described in Appendix B.

- Boric Acid Corrosion Prevention Program
- Primary Chemistry Monitoring Program
- ASME Section XI Inservice Inspection Program, Subsection IWB (as supplemented by Code Case N-560 for Examination Category B-J welds)
- Leakage Detection in Reactor Building
- Alloy 600 Aging Management Program
- Bolting and Torquing Activities
- Small Bore Piping and Small Bore Nozzles Inspections

3.2.3 Pressurizer

The pressurizer is within the scope of license renewal as discussed in Section 2.3.1.4. The following pressurizer items are subject to aging management review.

- Pressurizer vessel
- Nozzles
- Other pressure retaining items
- Bolted closures
- Integral attachments
- Immersion heaters

As described in Section 2.3.1.4, ANO-1 is bounded by the BWOG topical report with regard to the pressurizer.

Environment and Stress

The operating environment of the ANO-1 pressurizer is consistent with that described in Section 3 of BAW-2244A (Reference 3.2-2). The ANO-1 Primary Chemistry Monitoring Program includes specifications to periodically monitor the primary coolant. Limitations are established on dissolved oxygen, halides and other impurities. Corrective actions are taken in the event primary coolant parameters are out of specification. ANO-1 RCS chemistry is maintained in accordance with the ANO-1 Primary Chemistry Monitoring Program.

The pressurizer is designed to accommodate service loadings (i.e., levels A through D); however, operation under level A and B service conditions contribute to the normal aging stresses for the pressurizer. ANO-1 has not been subjected to a level C or D event. Therefore, the ANO-1 pressurizer is bounded by BAW-2244A with respect to the qualitative assessment of stress.

Aging Effects Requiring Management

The aging effects requiring management for the ANO-1 pressurizer are consistent with those described in Section 3 of BAW-2244A. The results of the aging effects review are contained in BAW-2244A. The aging effects requiring management for the pressurizer are summarized in Table 3.2-1.

Industry Experience

The industry experience for the pressurizer is described in Section 3 of BAW-2244A.

ANO-1 Operating Experience

ANO-1 operating experience was reviewed to validate the identified aging effects requiring management for the pressurizer. This review included a search for instances of pressurizer aging at ANO-1 using the station information management system, condition reporting system, and licensing event database. From this review, no aging effects requiring management were identified beyond those identified in BAW-2244A.

Conclusion

During the review of industry information, NRC generic communications, and ANO-1 operating experience, no additional aging effects beyond those discussed in this section were identified. The aging effects requiring management for the pressurizer are summarized in Table 3.2-1. These aging effects must be adequately managed so that the intended function of the pressurizer listed in Table 3.2-1 will be maintained consistent with the current licensing basis for the period of extended operation. The aging effects requiring management will be adequately managed by the following existing programs that are described in Appendix B:

- Boric Acid Corrosion Prevention Program
- Primary Chemistry Monitoring Program
- ASME Section XI Inservice Inspection Program, Subsection IWB (as supplemented by Code Case N-560 for Examination Category B-J welds)
- Leakage Detection in Reactor Building
- Alloy 600 Aging Management Program
- Bolting and Torquing Activities

In addition to the above, the following new activity has been identified for license renewal and is described in Appendix B.

- Pressurizer Examinations

3.2.4 Reactor Vessel

The reactor vessel is within the scope of license renewal as discussed in Section 2.3.1.5. The following reactor vessel items are subject to aging management review.

- Shell and closure head
- Nozzles
- Interior attachments
- Bolted closures

As described in Section 2.3.1.5, ANO-1 is bounded by BAW-2251A (Reference 3.2-3) with regard to reactor vessel items and associated materials of construction. The approach for identifying the aging effects requiring management is described in Section 3.2.1.

<u>Environment and Stress</u>

The operating environment of the ANO-1 reactor vessel is consistent with that described in Section 3 of BAW-2251A. The ANO-1 Primary Chemistry Monitoring Program includes specifications to periodically monitor the primary coolant. Limitations are established on dissolved oxygen, halides and other impurities. Corrective actions are taken in the event primary coolant parameters are out of specification. RCS chemistry is maintained in accordance with the ANO-1 Primary Chemistry Monitoring Program.

The reactor vessel is designed to accommodate service loadings (i.e., levels A through D); however, operation under level A and B service conditions contribute to the normal aging stresses for the reactor vessel. ANO-1 has not been subjected to a level C or D event. Therefore, the ANO-1 reactor vessel is bounded by BAW-2251A with respect to the qualitative assessment of stress.

<u>Aging Effects Requiring Aging Management</u>

The aging effects applicable to the ANO-1 reactor vessel are consistent with those described in BAW-2251A since ANO-1 is bounded by the generic report with respect to materials of construction, operating environment, level A and B service conditions, and operating experience. The aging effects requiring management for the reactor vessel are summarized in Table 3.2-1.

<u>Industry Experience</u>

The industry experience for the reactor vessel is described in Section 3 of BAW-2251A.

<u>ANO-1 Operating Experience</u>

ANO-1 operating experience was reviewed to validate the identified aging effects requiring management for the reactor vessel. This review included a search for instances of aging of the reactor vessel at ANO-1 using the station information management system, condition reporting system, and licensing event database. From this review, no aging effects requiring management were identified beyond those identified in BAW-2251A.

Conclusion

During the review of industry information, NRC generic communications, and operating experience, no additional aging effects beyond those discussed in this section were identified for ANO-1. The aging effects requiring management for the reactor vessel are summarized in Table 3.2-1. These aging effects must be adequately managed so that the intended functions listed in Table 3.2-1 will be maintained consistent with the current licensing basis for the period of extended operation. The aging effects requiring management will be adequately managed by the following existing programs, which are described in Appendix B.

- Primary Chemistry Monitoring Program
- CRDM Nozzle and Other Vessel Closure Penetrations Inspection Program
- ASME Section XI Inservice Inspection Program, Subsection IWB
- Leakage Detection in Reactor Building
- Reactor Vessel Integrity Program
- Boric Acid Corrosion Prevention Program
- Alloy 600 Aging Management Program
- Bolting and Torquing Activities

3.2.5 Reactor Vessel Internals

The reactor vessel internals are within the scope of license renewal as discussed in Section 2.3.1.6. The following reactor vessel internal items are subject to aging management review.

- Plenum assembly
- Core support shield assembly
- Core barrel assembly
- Lower internals assembly
- Reactor vessel level monitoring system probe supports
- Remaining portions of the surveillance specimen holder tubes
- Thermal shield and thermal shield upper restraint

As described in Section 2.3.1.6, ANO-1 is bounded by BAW-2248A (Reference 3.2-4) with regard to reactor vessel internal items within the first four groups defined above. The RVLMS probe supports, surveillance specimen holder tubes, and thermal shield and thermal shield upper restraint are not within the scope of BAW-2248A but are within the scope of license renewal and subject to aging management review for ANO-1.

<u>Environment and Stress</u>

The operating environment, or chemistry of the fluid in contact with the ANO-1 reactor vessel internals, is maintained consistent with that described in Section 3 of BAW-2248A. The ANO-1 Primary Chemistry Monitoring Program includes specifications to periodically monitor the primary coolant. Limitations are established on dissolved oxygen, halides, and other impurities. Corrective actions are taken in the event primary coolant parameters are out of specification. ANO-1 RCS chemistry is maintained in accordance with the ANO-1 Primary Chemistry Monitoring Program.

The reactor vessel internals are designed to accommodate service loadings (i.e., levels A through D); however, operation under level A and B service conditions contribute to the normal aging stresses for the reactor vessel internals. ANO-1 has not been subjected to a level C or D event. Therefore, ANO-1 is bounded by BAW-2248A with respect to the qualitative assessment of stress.

<u>Aging Effects Requiring Management</u>

The aging effects requiring management of the ANO-1 reactor vessel internals are consistent with those described in Section 3 of BAW-2248A for reactor vessel internal items within the first four groups listed above. The results of the aging effects review are contained in BAW-2248A. Items specific to ANO-1 and not within the scope of BAW-2248A are evaluated below.

The reactor vessel internals items that support the RVLMS probes are constructed from austenitic stainless steel and the aging effects requiring management for these items are consistent with other stainless steel items in the plenum assembly (i.e., cracking and loss of mechanical closure integrity caused by stress relaxation of the RVLMS brazement guide assembly j-bolt and nut).

The thermal shield that surrounds the core barrel and the thermal shield upper restraint are constructed from austenitic stainless steel. The aging effects requiring management for these items, consistent with those identified in BAW-2248A for the core barrel assembly, are cracking and reduction of fracture toughness. The upper surveillance specimen holder tube assembly and some of the brackets and their bolts are still installed at ANO-1. These items are fabricated from austenitic stainless steel and the aging effects requiring management include cracking and loss of mechanical closure integrity by stress relaxation of the bolting.

The aging effects requiring management for the reactor vessel internals are summarized in Table 3.2-1.

Industry Experience

The industry experience for the reactor vessel internals is described in Section 3 of BAW-2248A. Subsequent to the issuance of BAW-2248A, NRC issued Information Notice 98-11, "Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants," on March 25, 1998. Information Notice 98-11 discusses cracking of reactor vessel internal baffle former bolts found at several foreign pressurized water reactors and includes, among other information, a brief discussion of the current and planned activities of the BWOG to address the potential for cracking of baffle bolts in domestic B&W plants.

ANO-1 Operating Experience

ANO-1 operating experience was reviewed to validate the identified aging effects requiring management for reactor vessel internals. This review included a search for instances of reactor vessel internals aging at ANO-1 using the station information management system, condition reporting system, and licensing event database. One issue identified in this review was cracking of the thermal shield bolting and core barrel bolting as discussed in Section 3.5.4 of BAW-2248A. The only other event of note was the surveillance specimen holder tube failures. This early failure was due to a design flaw and was not related to aging. From this review, no aging effects requiring management were identified beyond those identified in BAW-2248A.

Conclusion

During the review of industry information, NRC generic communications, and ANO-1 operating experience, no additional aging effects beyond those discussed in this section were identified. The aging effects requiring management for the reactor vessel internals are summarized in Table 3.2-1. These aging effects must be adequately managed so that the intended functions listed in Table 3.2-1 will be maintained consistent with the current licensing basis for the period of extended operation. The aging effects requiring management will be managed by the following existing programs, which are described in Appendix B.

- ASME Section XI Inservice Inspection Program, Subsection IWB
- Primary Chemistry Monitoring Program

In addition to the above, the following new program has been identified for license renewal and is described in Appendix B.

- Reactor Vessel Internals Aging Management Program

3.2.6 Once-Through Steam Generators

The once-through steam generators are within the scope of license renewal as discussed in Section 2.3.1.7. The following once-through steam generator items are subject to aging management review.

- Primary pressure boundary: hemispherical heads, support skirt transition ring, primary nozzles, Alloy 600 drain nozzle, bolted closures, tubesheets, tubes, plugs, sleeves, bolted closures, and integral attachments inspected in accordance with ASME Section XI, Subsection IWB.
- Secondary pressure boundary: shell, tubesheets, and integral attachments, steam outlet nozzles, main feedwater nozzles, emergency feedwater nozzles, feedwater header and riser piping, instrumentation nozzles, vent nozzles, drain nozzles, temperature sensing connections, bolted closures and integral attachments inspected in accordance with ASME Section XI, Subsection IWC.

The following is a description of the aging effects applicable to the once-through steam generators. The approach for identifying the aging effects requiring management is described in Section 3.2.1.

<u>Environment and Stress</u>

The ANO-1 Chemistry Control Programs include specifications to periodically monitor the primary coolant and the secondary coolant. Limitations are established for the primary coolant on dissolved oxygen, halides and other impurities. Limitations are established on specific impurities in the secondary coolant. ANO-1 primary side chemistry and secondary side chemistry are maintained by the ANO-1 Chemistry Control Programs.

The once-through steam generator is designed to accommodate all service loadings (i.e., levels A through D); however, operation under level A and B service conditions contribute to the normal aging stresses for the once-through steam generator items. ANO-1 has not been subjected to a level C or D event.

<u>Aging Effects Requiring Aging Management for Primary Pressure Boundary Items</u>

Potential aging effects that may be applicable to the items that support the primary pressure boundary include loss of material, cracking, mechanical distortion of tubes, and loss of mechanical closure integrity. Aging mechanisms that may lead to reduction of fracture toughness of once-through steam generator items include various forms of embrittlement (e.g., neutron and thermal). Neutron embrittlement is limited to the direct neutron flux of the reactor vessel beltline region and is not a concern for the once-through steam generator. Thermal embrittlement is negligible for all pressurized water reactor materials except cast austenitic stainless steel, and once-through steam generators at ANO-1 have no cast austenitic stainless steel parts.

<u>*Primary* - Loss of Material Assessment</u>

Loss of material may be due to intergranular attack, pitting, wear, erosion/corrosion, and wastage, as further discussed in the following paragraphs.

The Alloy-600 steam generator tubes are subject to loss of material due to intergranular attack, pitting, wear, or fretting, and erosion or erosion/corrosion. The plugs and sleeves installed inside the tubes are made of Alloy-600 or Alloy-690 and are less susceptible to loss of material.

Intergranular attack of steam generator tubes is characterized by a relatively uniform attack at grain boundaries over a portion of the tubing surface. Intergranular attack is caused by impurities that concentrate in steam generator secondary side crevices and sludge piles, where boiling occurs and circulation is poor. Once-through steam generator tubes are roll expanded over only a portion of the tubesheet thickness. Consequently, a crevice exists between the tubes and the tube bore hole through the tubesheets that provides the environment for intergranular attack.

Pitting is a localized corrosion mechanism that produces small holes in the metal. Low fluid velocity or stagnation is usually associated with the development of pitting. Pitting has occurred in once-through steam generator tubes.

Fretting and sliding wear of steam generator tubes at tube support locations has occurred in the industry. The forces imposed on the tubes by the secondary fluid cause high frequency vibration of the tubes and interaction with the tube support structures.

Erosion is the loss of surface metal due to the mechanical action of flowing fluid. Erosion/corrosion is the loss of material due to the combined actions of erosion by the flowing fluid and corrosion of the newly exposed base material by chemicals in the flowing fluid. Once-through steam generator tube damage has occurred due to erosion/corrosion near the fourteenth tube support plate at another B&W operating plant but has not been observed at ANO.

The external surfaces of the primary pressure boundary components are subject to loss of material due to boric acid wastage. The leakage of primary coolant through adjacent bolted closures, and the subsequent evaporation and concentration of boric acid, could lead to the presence of a boric acid slurry on the bolting and external surfaces of the vessel. The boric acid slurry could cause loss of material of the external surfaces. Therefore, loss of material is an aging effect requiring management for the external surfaces of the once-through steam generators at ANO-1.

Primary - Cracking Assessment

Because of the consequences of a breach of the primary system pressure boundary, cracking at welded joints is considered an aging effect requiring management for items fabricated from carbon steel and low-alloy steel. Welded joints are the more susceptible locations due to the various constituent zones within the joint, resulting in slight variations in residual stresses and mechanical properties. Cracking at welded joints is an aging effect requiring management for clad low-alloy steel heads, clad low-alloy tubesheets, and clad carbon steel nozzle forgings. In addition, cracking of the Alloy-600 and Alloy-690 tubes, plugs, sleeves, and drain nozzle by primary water stress corrosion cracking is an aging effect requiring management.

Primary - Mechanical Distortion Assessment

Steam generator tubes have been found to suffer a form of distortion called denting. Denting is the mechanical deformation of tubes due to corrosion of the tube support structures. The corrosion product is mostly magnetite. Because magnetite is less dense than the support structure, the corrosion products occupy more volume than the original base metal. As more magnetite forms, it expands into the crevice between the tube and the support structure. Eventually the crevice becomes completely filled and any further corrosion causes the tube to deform or the support structure to fracture. Therefore, mechanical distortion is an aging effect requiring management for the once-through steam generator tubes at ANO-1.

Primary - Loss of Mechanical Closure Integrity Assessment

Stress relaxation and corresponding loss of preload may lead to localized leakage of reactor coolant and a loss of mechanical closure integrity. This localized leakage of borated coolant may cause corrosive attack and loss of material of bolting, adjacent flange surfaces, and surfaces below the bolted connection leak source. Loss of mechanical closure integrity is associated with the condition of the closure bolting and bolting surfaces. Aging effects relevant to the mechanical closure bolting are cracking, loss of bolt preload due to stress relaxation, and loss of material for low-alloy steel bolting due to boric acid wastage. Loss of mechanical closure integrity is an aging effect requiring management for the once-through steam generators at ANO-1.

Aging Effects Requiring Management for Secondary Pressure Boundary Items

Potential aging effects applicable to the items that support the secondary pressure boundary include loss of material, cracking, and loss of mechanical closure integrity. Reduction of fracture toughness is not an aging effect requiring management for the secondary pressure boundary items.

Secondary - Loss of Material Assessment

The external surfaces of the secondary pressure boundary components are subject to loss of material due to boric acid wastage. The leakage of primary coolant through adjacent bolted closures, and the subsequent evaporation and concentration of boric acid, could lead to the presence of a boric acid slurry on the bolting and external surfaces of the secondary side of the steam generator. The boric acid slurry could cause loss of material of the external surfaces.

Erosion is the loss of surface metal due to the mechanical action of flowing fluid. Erosion/corrosion is the loss of material due to the combined actions of erosion by the flowing fluid and corrosion of the newly exposed base material by chemicals in the flowing fluid. The steam outlet nozzles may be affected due to the high fluid velocity through the nozzles. Erosion of secondary manways and inspection ports may occur as a result of leakage through bolted closures. Therefore, loss of material is an aging effect requiring management for the once-through steam generators at ANO-1.

<u>*Secondary* - Cracking Assessment</u>

Similar to the discussion above for primary pressure boundary welded joints, cracking at welded joints of the secondary pressure boundary is considered an aging effect requiring management for license renewal at ANO-1.

<u>*Secondary* - Loss of Mechanical Closure Integrity Assessment</u>

Similar to the discussion above for primary bolted connections, loss of mechanical closure integrity of secondary manways, inspection ports, and feedwater pipe flanges may occur and is considered an aging effect requiring management for license renewal. Therefore, loss of mechanical closure integrity is an aging effect requiring management for the once-through steam generators at ANO-1.

<u>Industry Experience</u>

In order to validate the identified aging effects requiring management, a survey of industry experience was performed. This survey included NRC generic communications, licensee event reports from nuclear power plants other than ANO-1, and NRC NUREGs. The following documents were included in this review.

- Numerous Information Notices
- IE Bulletin 79-13, "Cracking in Feedwater System Piping"
- IE Bulletin 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants"
- IE Bulletin 87-01, "Thinning of Pipe Walls in Nuclear Power Plants"
- GL 79-20, "Information Requested on PWR Feedwater Lines"
- GL 85-02, "Staff Recommended Actions Stemming from NRC Integrated Program for the Resolution of Unresolved Safety Issues Regarding Steam Generator Tube Integrity"
- GL 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants"
- GL 89-08, "Erosion/Corrosion Induced Pipe Wall Thinning"
- GL 91-17, "Generic Safety Issue 29, Bolting Degradation or Failure in Nuclear Power Plants"
- GL 95-03, "Circumferential Cracking of Steam Generator Tubes"

<u>ANO-1 Operating Experience</u>

ANO-1 operating experience was reviewed to validate the identified aging effects requiring management for the once-through steam generators. This review included a search for instances of aging at ANO-1 using the station information management system, condition reporting system, and licensing event database. No additional aging effects were identified from this review.

Conclusion

During the review of industry information, NRC generic communications, and ANO-1 operating experience, no additional aging effects beyond those discussed in this section were identified. The aging effects requiring management for the once-through steam generator are summarized in Table 3.2-1.

These aging effects must be adequately managed so that the intended functions listed in Table 3.2-1 will be maintained consistent with the current licensing basis for the period of extended operation. The aging effects requiring management will be adequately managed by the following existing programs, which are described in Appendix B:

- Boric Acid Corrosion Prevention Program
- Primary Chemistry Monitoring Program
- Secondary Chemistry Monitoring Program
- ASME Section XI Inservice Inspection Program, Subsections IWB and IWC
- Leakage Detection in Reactor Building
- Steam Generator Integrity Program
- Alloy 600 Aging Management Program
- Bolting and Torquing Activities

3.2.7 Reactor Coolant Pumps

Reactor coolant pumps are in the scope of license renewal as discussed in Section 2.3.1.8. The following reactor coolant pump items are subject to aging management review.

- Casing
- Cover
- Seal water heat exchanger
- Pressure-retaining bolting

The approach for identifying the aging effects requiring management on the reactor coolant pumps is described in Section 3.2.1.

Environment and Stress

The materials and operating environment of the reactor coolant pumps, including the bolted closures and connections, are similar to that evaluated in the RCS piping reviews (see Section 3.2.2).

The seal water heat exchangers are a double coil tube-in-tube design. The inner tube carries primary water and the outer tube carries treated water from the intermediate cooling water system.

Aging Effects Requiring Aging Management

The pump casings are constructed of cast austenitic stainless steel similar to the valve bodies evaluated in Section 3.2.2. The aging effects requiring management for the casing are cracking at welded joints and reduction of fracture toughness. The aging effects requiring management for the cover are cracking by fatigue and reduction of fracture toughness. Aging effects requiring management for the seal water heat exchangers include cracking and loss of material of the inner tube. The bolted closures and connections of the reactor coolant pump casings are made of the same material as RCS piping bolted closures and connections evaluated in Section 3.2.2 and the aging effects requiring management are cracking of the bolting material, loss of mechanical closure integrity, and loss of material.

Industry Experience

In order to validate the identified aging effects requiring management, a survey of industry experience was performed. This survey included NRC generic communications, licensee event reports from nuclear power plants other than ANO-1, and NRC NUREGs. The following documents were reviewed.

- IE Bulletin 82-02, “Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants” and
- GL 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants”

The results of the review of NRC generic communications for the RCS piping report (Reference 3.2-1) are also applicable to the reactor coolant pump. In addition, the aging effects requiring management for the reactor coolant pump casings, covers, and associated bolted closures and connections are further validated by the reviews performed and documented in the PWR RCS License Renewal Industry Report (Reference 3.2-5).

ANO-1 Operating Experience

ANO-1 operating experience was reviewed to validate the identified aging effects requiring management for reactor coolant pumps. This review included a search for instances of reactor coolant pump aging at ANO-1 using the station information management system, condition reporting system, and licensing event database. From this review, no aging effects requiring management were identified beyond those identified previously in this section.

Conclusion

During the review of industry information, NRC generic communications, and ANO-1 operating experience, no additional aging effects beyond those discussed in this section were identified. The aging effects requiring management for the reactor coolant pumps are consistent with those previously identified for RCS piping (see Section 3.2.2). The aging effects requiring management for the reactor coolant pumps are summarized in Table 3.2-1. These aging effects must be managed so that the intended functions listed in Table 3.2-1 will be maintained consistent with the current licensing basis for the period of extended operation. The aging effects requiring management will be managed by the following programs, which are described in Appendix B.

- ASME Section XI Inservice Inspection Program Subsection IWB (as supplemented by Code Case N-481 to manage reduction of fracture toughness of the pump casing) including an augmented inspection of a reactor coolant pump (visual inspection of the pressure retaining surfaces, including the cover, prior to entering the period of extended operation). The flaw tolerance evaluation used to comply with Code Case N-481 is a time-limited aging analysis and is addressed in Section 4.3.6.
- Primary Chemistry Monitoring Program
- Auxiliary Systems Chemistry Monitoring Program
- Leakage Detection in Reactor Building
- Boric Acid Corrosion Prevention Program
- Bolting and Torquing Activities

3.2.8 Control Rod Drive Mechanism Pressure Boundary

Control rod drive tube motor housings are identified in Section 2.3.1.9 as being subject to aging management review. The CRDM items subject to aging management review include.

- CRDM Motor Tube Assembly
- CRDM Closure Insert and Vent Assemblies

In addition, the RVLMS adapter flange/closure assembly components were identified as subject to aging management review in Section 2.3.1.9. The following is a description of the aging effects applicable to the control rod drive tube motor housings and RVLMS adapter flange/closure assembly. The approach for identifying the aging effects requiring management for the control rod drive tube motor housings is described in Section 3.2.1.

<u>Environment and Stress</u>

The chemistry of the fluid in contact with the control rod drive tube motor housings is maintained in a manner consistent with other RCS components. The ANO-1 Primary Chemistry Monitoring Program is maintained and includes specifications to periodically monitor the primary coolant. Limitations are established on dissolved oxygen, halides and other impurities. Corrective actions are taken in the event primary coolant parameters are out of specification.

The control rod drive tube motor housings are designed to accommodate service loadings (i.e., levels A through D). Operation under level A and B service conditions contribute only to the normal aging stresses for the control rod drive tube motor housings. ANO-1 has been subjected to level A and B service loadings, and not to level C or D events.

<u>Aging Effects Requiring Aging Management</u>

The control rod drive tube motor housings are fabricated from austenitic and martensitic stainless steels with the exception of the center section of the type B drive, which is a low-alloy steel forging clad with Alloy 82/182. The closure insert and vent assemblies and the RVLMS adapter flange assembly are fabricated from austenitic stainless steel. The aging effect requiring management for the closure insert and vent assemblies is loss of mechanical closure integrity. Aging effects requiring management for the control rod drive tube motor housings include cracking at welded joints and loss of mechanical closure integrity for the bolted connections. The aging effect requiring management for the RVLMS adapter flange assembly is cracking at welded joints. These effects are consistent with those previously identified in Section 3.2.2 for RCS piping.

<u>Industry Experience</u>

In order to validate the set of aging effects requiring management, a survey of industry experience was performed. This survey included NRC generic communications, licensee event reports from nuclear power plants other than ANO-1, and NRC NUREGs. From this review, no aging effects requiring management were identified beyond those identified in this section.

ANO-1 Operating Experience

ANO-1 operating experience was reviewed to validate the identified aging effects requiring management for control rod drive tube motor housings. This review included a search for instances of control rod drive mechanism aging at ANO-1 using the station information management system, condition reporting system, and licensing event database. From this review, no aging effects requiring management were identified beyond those identified in this section.

Conclusion

During the review of industry information, NRC generic communications, and ANO-1 operating experience, no additional aging effects beyond those discussed in this section were identified. The aging effects requiring management for the control rod drive tube motor housings, closure insert and vent assemblies, and RVLMS items are consistent with those previously identified for RCS piping (see Section 3.2.2). The aging effects requiring management for the control rod drive tube items subject to aging management review are summarized in Table 3.2-1. These aging effects must be managed so that the intended functions listed in Table 3.2-1 will be maintained consistent with the current licensing basis of the period of extended operation. The aging effects requiring management will be managed by the following existing programs, which are described in Appendix B.

- ASME Section XI Inservice Inspection Program, Subsection IWB
- Leakage Detection in Reactor Building
- Primary Chemistry Monitoring Program
- Bolting and Torquing Activities

3.2.9 References for Section 3.2

3.2-1 BAW-2243A, "*Demonstration of the Management of Aging Effects for the Reactor Coolant System Piping*," The B&W Owners Group Generic License Renewal Program, June 1996.

3.2-2 BAW-2244A, "*Demonstration of the Management of Aging Effects for the Pressurizer*," The B&W Owners Group Generic License Renewal Program, December 1997.

3.2-3 BAW-2251A, "*Demonstration of the Management of Aging Effects for the Reactor Vessel*," The B&W Owners Group Generic License Renewal Program, June 1996.

3.2-4 BAW-2248A, "*Demonstration of the Management of Aging Effects for the Reactor Vessel Internals*," The B&W Owners Group Generic License Renewal Program, December 1999.

3.2-5 NUMARC 90-07-01, "*PWR Reactor Coolant System License Renewal Industry Report*," Revision 1, Nuclear Management and Resource Council, May 1992.

Table 3.2-1 Aging Effects Requiring Aging Management for Reactor Coolant System Components					
Component/ Item Grouping	**Intended Function**	**Material**	**Environment**	**Aging Effect**	**Program/Activity**
RCS piping, pressurizer, RV, and OTSG	Pressure boundary	Low-alloy and clad carbon steel pressure retaining items	External-Ambient	Loss of material by boric acid wastage	Boric Acid Corrosion Prevention
Reactor Coolant System Piping					
Reactor coolant system piping , NPS ≥ 4 inches	Pressure boundary	Stainless steel clad carbon steel piping Stainless steel piping	Primary water	Cracking at welded joints	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-J as modified by Code Case N-560 • Examination Category B-P Leakage Detection in Reactor Building
Hot leg flowmeter assembly	Pressure boundary	Stainless steel clad (alloy 82/182)	Primary water	Loss of material (carbon steel) due to potential for cracking of alloy 82/182 cladding	Primary Chemistry Monitoring Alloy 600 Aging Management
Reactor coolant system piping, NPS ≥ 4 inches	Pressure boundary	Stainless steel clad carbon steel branch connections with alloy 82/182 weld build-up	Primary water	Cracking at welded joints	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-J as modified by Code Case N-560 • Examination Category B-P Leakage Detection in Reactor Building Alloy 600 Aging Management

Table 3.2-1 Aging Effects Requiring Aging Management for Reactor Coolant System Components

Component/ Item Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Reactor coolant system piping, 1 inch < NPS < 4 inches	Pressure boundary	Stainless steel piping and stainless steel branch connections	Primary water	Cracking at welded joints	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-J as modified by Code Case N-560 • Examination Category B-P • Small Bore Piping and Small Bore Nozzles Inspections Leakage Detection in Reactor Building
Reactor coolant system piping, 1 inch < NPS < 4 inches	Pressure boundary	2½-inch SS clad carbon steel HPI branch connections and safe ends with thermal sleeves	Primary water	Displacement or cracking of HPI/MU thermal sleeves Cracking at welded joints	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-J as modified by Code Case N-560—augmented examination of thermal sleeve • Examination Category B-P • Small Bore Piping and Small Bore Nozzles Inspections Leakage Detection in Reactor Building
Reactor coolant system piping, 1 inch < NPS < 4 inches	Pressure boundary	Stainless steel clad carbon steel branch connections with Alloy 600 safe ends	Primary water	Cracking at or near welded joints	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-J as modified by Code Case N-560 • Examination Category B-P • Small Bore Piping and Small Bore Nozzles Inspections Leakage Detection in Reactor Building Alloy 600 Aging Management

Table 3.2-1 Aging Effects Requiring Aging Management for Reactor Coolant System Components

Component/ Item Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Reactor coolant system piping, 1 inch < NPS < 4 inches	Pressure boundary	Alloy 600 branch connections	Primary water	Cracking at or near welded joints	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-J as modified by Code Case N-560 • Examination Category B-P Leakage Detection in Reactor Building Alloy 600 Aging Management
Reactor coolant system piping, NPS ≤ 1 inch	Pressure boundary	Stainless steel pipe and stainless steel branch connections	Primary water	Cracking at welded joints	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-P • Small Bore Piping and Small Bore Nozzles Inspections Leakage Detection in Reactor Building
Reactor coolant system piping, NPS ≤ 1 inch	Pressure boundary	Alloy 600 branch connections	Primary water	Cracking at or near welded joints	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-P Leakage Detection in Reactor Building Alloy 600 Aging Management
Reactor coolant system valves, NPS ≥ 4 inches	Pressure boundary	Stainless steel valve bodies and bonnets	Primary water	Cracking at welded joints	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-M-1 • Examination Category B-P Leakage Detection in Reactor Building

Table 3.2-1 Aging Effects Requiring Aging Management for Reactor Coolant System Components					
Component/ Item Grouping	**Intended Function**	**Material**	**Environment**	**Aging Effect**	**Program/Activity**
Reactor coolant system valves, NPS < 4 inches	Pressure boundary	Cast austenitic stainless steel valve bodies and bonnets	Primary water	Reduction of fracture toughness	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-P Leakage Detection in Reactor Building
Reactor coolant system bolting ≤ 2-inches diameter	Pressure boundary	Low-alloy steel valve bolting	Primary water	Cracking Loss of mechanical closure integrity Loss of material	ASME Section XI ISI-IWB • Examination Category B-G-2 • Examination Category B-P Leakage Detection in Reactor Building Bolting and Torquing Activities
Reactor coolant system valve bolting ≤ 2-inches diameter	Pressure boundary	Stainless steel valve bolting	Primary water	Cracking Loss of mechanical closure integrity	ASME Section XI ISI-IWB • Examination Category B-G-2 • Examination Category B-P Leakage Detection in Reactor Building Bolting and Torquing Activities
Letdown coolers tubes and manifold	Pressure boundary	Stainless steel	Primary water	Cracking Loss of material Loss of mechanical closure integrity	ASME Section XI ISI-IWB • Examination Category B-P Leakage Detection in Reactor Building
Letdown coolers shell and end plates	Pressure boundary	Carbon steel	Treated water	Loss of material	ASME Section XI ISI-IWB • Examination Category B-P Leakage Detection in Reactor Building
Pressurizer					

Table 3.2-1 Aging Effects Requiring Aging Management for Reactor Coolant System Components					
Component/ Item Grouping	**Intended Function**	**Material**	**Environment**	**Aging Effect**	**Program/Activity**
Pressurizer vessel	Pressure boundary	Stainless steel clad carbon steel	Primary water	Cracking at welded joints Cracking of cladding	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-B • Examination Category B-P Leakage Detection in Reactor Building Pressurizer Examinations
Pressurizer, full penetration welded nozzles, NPS > 1-inch	Pressure boundary	Stainless steel clad carbon steel	Primary water	Cracking at welded joints	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-D • Examination Category B-P Leakage Detection in Reactor Building
Pressurizer, safe ends of full penetration welded nozzles, NPS > 1-inch	Pressure boundary	Alloy 600	Primary water	Cracking at or near welded joints	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-F and B-J as modified by Code Case N-560 • Examination Category B-P Leakage Detection in Reactor Building Alloy 600 Aging Management
Pressurizer, pressure retaining partial penetration welds, NPS≤1-inch	Pressure boundary	Alloy 600	Primary water	Cracking at or near welded joints	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-P • Augmented inspection of repaired nozzle and VT-2 of other nozzles Leakage Detection in Reactor Building Alloy 600 Aging Management

Table 3.2-1 Aging Effects Requiring Aging Management for Reactor Coolant System Components

Component/ Item Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Pressurizer, dissimilar metal welds, NPS > 1-inch	Pressure boundary	Alloy 82/182	Primary water	Cracking at welded joints	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-F and B-J as modified by Code Case N-560 • Examination Category B-P Leakage Detection in Reactor Building Alloy 600 Aging Management
Pressurizer, immersion heaters (sheaths, end plugs, and welds) and the weld that connects the immersion heaters to the diaphragm plate	Pressure boundary	Stainless steel	Primary water	Cracking	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-P Leakage Detection in Reactor Building Pressurizer Examinations
Pressurizer integral attachments	Pressure boundary	Carbon steel	External-reactor building	Cracking at welded joints	ASME Section XI ISI-IWB • Examination Category B-H
Pressurizer bolting > 2-inches diameter	Pressure boundary	Low-alloy manway studs	External-reactor building	Cracking Loss of material Loss of mechanical closure integrity	ASME Section XI ISI-IWB • Examination Category B-G-1 • Examination Category B-P Leakage Detection in Reactor Building Boric Acid Corrosion Prevention Bolting and Torquing Activities
Pressurizer bolting ≤ 2-inches diameter	Pressure boundary	Low-alloy steel heater bundle studs	External-reactor building	Cracking Loss of material Loss of mechanical closure integrity	ASME Section XI ISI-IWB • Examination Category B-G-1 • Examination Category B-P Leakage Detection in Reactor Building Boric Acid Corrosion Prevention Bolting and Torquing Activities

Table 3.2-1 Aging Effects Requiring Aging Management for Reactor Coolant System Components					
Component/ Item Grouping	**Intended Function**	**Material**	**Environment**	**Aging Effect**	**Program/Activity**
Reactor Vessel					
Reactor vessel shell and closure head	Pressure boundary	SS clad low-alloy steel	Primary water	Cracking at welded joints Reduction of fracture toughness Cracking of 508 Class 2 forgings due to intergranular separation	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-A Reactor Vessel Integrity TLAA-See Section 4.2
	Support and orientation for the core	SS clad low-alloy steel	Primary water	Loss of material (internals support shelf)	ASME Section XI ISI-IWB • Examination Category B-N-1
Reactor vessel nozzles, full penetration welds	Pressure boundary	SS clad low-alloy steel	Primary water	Cracking at welded joints Cracking at nozzle inside radius	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-D
Reactor vessel, pressure retaining partial penetration welded nozzles	Pressure boundary	Alloy 600 CRDM nozzles and incore instrumentation nozzles	Primary water	Cracking at or near welded joints	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-E • Examination Category B-P Leakage Detection in Reactor Building CRDM Nozzle and Other Vessel Closure Penetrations Inspection (CRDM penetrations only) Alloy 600 Aging Management (incore nozzles)

Table 3.2-1 Aging Effects Requiring Aging Management for Reactor Coolant System Components

Component/ Item Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Reactor vessel interior attachment	Pressure boundary	Alloy 600 core guide lugs	Primary water	Cracking at or near attachment welds to cladding	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-N-2
Reactor vessel bolted closures, pressure retaining seating surfaces	Pressure boundary	SS clad low-alloy steel	Primary water	Loss of material	ASME Section XI ISI-IWB • Examination Category B-G-1 • Examination Category B-N-2 • Examination Category B-P Leakage Detection in Reactor Building
Reactor vessel bolting > 2 inches in diameter	Pressure boundary	Low-alloy steel	External	Cracking Loss of material Loss of mechanical closure integrity	ASME Section XI ISI-IWB • Examination Category B-G-1 • Examination Category B-P Leakage Detection in Reactor Building Bolting and Torquing Activities Boric Acid Corrosion Prevention
Reactor vessel bolting ≤ 2-inches in diameter (CRDM nozzles)	Pressure boundary	Stainless steel	External	Cracking Loss of mechanical closure integrity	ASME Section XI ISI-, IWB • Examination Category B-G-2 Leakage Detection in Reactor Building Boric Acid Corrosion Prevention Bolting and Torquing Activities

Table 3.2-1 Aging Effects Requiring Aging Management for Reactor Coolant System Components

Component/ Item Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Reactor Vessel Internals					
Core support shield assembly, core barrel assembly, lower internals assembly, plenum assembly, RVLMS support, thermal shield and associated upper restraint, SSHT	Provide support, orientation, guidance, and protection Provide a passageway for the distribution of the reactor coolant flow to the reactor core Provide gamma and neutron shielding	Austenitic stainless steel Martensitic stainless steel Ni-Cr alloy	Primary water	Loss of material Cracking of base metal, welds, and bolting Reduction of fracture toughness of base metal, welds, and bolting by irradiation embrittlement Reduction of fracture toughness of CASS items Dimensional changes by void swelling Loss of mechanical closure integrity (by stress relaxation and cracking)	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination category B-N-3 Reactor Vessel Internals Aging Management

Table 3.2-1 Aging Effects Requiring Aging Management for Reactor Coolant System Components

Component/ Item Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
OTSG					
OTSG hemispherical heads	Primary pressure boundary	SS clad carbon steel	Primary water	Cracking at welded joints	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-B • Examination Category B-P Leakage Detection in Reactor Building
OTSG support skirt transition ring	Primary pressure boundary	Low-alloy steel	External	Cracking at welded joints	ASME Section XI ISI-IWB • Examination Category B-H
OTSG primary nozzles, NPS > 1-inch	Primary pressure boundary	SS clad carbon steel	Primary water	Cracking at welded joints	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-D • Examination Category B-P Leakage Detection in Reactor Building
OTSG drain nozzle	Primary pressure boundary	Alloy-600	Primary water	Cracking at or near the welded joint	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-P Leakage Detection in Reactor Building Alloy 600 Aging Management Program
OTSG tubesheets	Primary pressure boundary	Inconel clad low-alloy steel	Primary water	Cracking at welded joints	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-B • Examination Category B-P Leakage Detection in Reactor Building

Table 3.2-1 Aging Effects Requiring Aging Management for Reactor Coolant System Components

Component/ Item Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
OTSG manways and inspection ports	Primary pressure boundary	SS clad carbon steel	Primary water	Cracking at welded joints Loss of mechanical closure integrity	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-P Leakage Detection in Reactor Building
OTSG tubes, plugs, and sleeves	Primary pressure boundary	Alloy-600 Alloy-690	Primary water	Cracking Loss of material Mechanical distortion	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-Q • Examination Category B-P Leakage Detection in Reactor Building Steam Generator Integrity
	Heat transfer from primary fluid to secondary fluid	Alloy-600 Alloy-690	Primary water	Fouling	Primary Chemistry Monitoring Steam Generator Integrity
OTSG shell, tubesheets, integral attachments	Secondary pressure boundary	Carbon steel	Secondary water	Cracking at welded joints Loss of material	Secondary Chemistry Monitoring ASME Section XI ISI-IWC • Examination Category C-A • Examination Category C-C • Examination Category C-H Boric Acid Corrosion Prevention
OTSG steam outlet nozzles, main feedwater nozzles, EFW nozzles, instrumentation nozzles, temperature sensing nozzles	Secondary pressure boundary	Carbon steel and Alloy- 600	Secondary water	Cracking Loss of material	Secondary Chemistry Monitoring ASME Section XI ISI-IWC • Examination Category C-B • Examination Category C-H

Table 3.2-1 Aging Effects Requiring Aging Management for Reactor Coolant System Components					
Component/ Item Grouping	**Intended Function**	**Material**	**Environment**	**Aging Effect**	**Program/Activity**
OTSG main and emergency feedwater header and piping	Secondary pressure boundary	Carbon steel	Secondary water	Cracking Loss of material	Secondary Chemistry Monitoring ASME Section XI-ISI-IWC • Examination Category C-F-2 • Examination Category C-H •
OTSG secondary manways and inspection ports	Secondary pressure boundary	Carbon steel	Secondary water	Cracking at welded joints Loss of mechanical closure integrity	Secondary Chemistry Monitoring ASME Section XI ISI-IWC • Examination Category C-H
OTSG bolting ≤ 2-inches diameter	Primary and secondary pressure boundary	Low-alloy steel manway studs and inspection openings	External-reactor building	Cracking at welded joints Loss of mechanical closure integrity Loss of material	ASME Section XI ISI-IWB and IWC • Examination Category B-G-2 • Examination Category B-P • Examination Category C-H Leakage Detection in Reactor Building Bolting and Torquing Activities
Reactor Coolant Pump					
RCP casing	Pressure boundary	Cast austenitic stainless steel	Primary water	Cracking at welded joints Reduction of fracture toughness	Primary Chemistry Monitoring ASME Section XI ISI-Augmented, IWB • Examination category B-P • Code case N-481(see Section 4.3.6 of the ANO-1 LRA) Leakage Detection in Reactor Building
			External-ambient	None	None

Table 3.2-1 Aging Effects Requiring Aging Management for Reactor Coolant System Components					
Component/ Item Grouping	**Intended Function**	**Material**	**Environment**	**Aging Effect**	**Program/Activity**
RCP cover	Pressure boundary	Cast austenitic stainless steel	Primary water	Cracking Reduction of fracture toughness	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination category B-P • Examination Category B-L-2—when RCP disassembled Leakage Detection in Reactor Building
			External-ambient	None	None
Seal water heat exchanger	Pressure boundary	Alloy 600	Primary water	Cracking Loss of material	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination category B-P Leakage Detection in Reactor Building
RCP bolting > 2 inches in diameter	Pressure boundary	Alloy steel	External	Cracking Loss of mechanical closure integrity Loss of material	ASME Section XI ISI-IWB • Examination Category B-G-1 • Examination Category B-P Leakage Detection in Reactor Building Bolting and Torquing Activities Boric Acid Corrosion Prevention
RCP bolting ≤ 2 inches in diameter	Pressure boundary	Alloy steel	External	Cracking Loss of mechanical closure integrity Loss of material	ASME Section XI ISI-IWB • Examination Category B-G-2 Leakage Detection in Reactor Building Bolting and Torquing Activities Boric Acid Corrosion Prevention

Table 3.2-1 Aging Effects Requiring Aging Management for Reactor Coolant System Components					
Component/ Item Grouping	**Intended Function**	**Material**	**Environment**	**Aging Effect**	**Program/Activity**
Control Rod Drive Mechanism					
Motor tube	Pressure boundary	Austenitic stainless steel Martensitic stainless steel Low-alloy steel clad with alloy 82/182	Primary water	Cracking at welded joints	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-O • Examination Category B-P Leakage Detection in Reactor Building
Closure insert and vent assembly and associated bolting	Pressure boundary	Austenitic stainless steel	Primary water	Loss of mechanical closure integrity	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-G-2 • Examination Category B-P Leakage Detection in Reactor Building Bolting and Torquing Activities
RVLMS adapter flange/closure assembly	Pressure boundary	Austenitic stainless steel	Primary water	Cracking	Primary Chemistry Monitoring ASME Section XI ISI-IWB • Examination Category B-G-2 • Examination Category B-P Leakage Detection in Reactor Building

3.3 ENGINEERED SAFEGUARDS

ANO-1 refers to the engineered safety features as the engineered safeguards. The following systems are included in this section:

- Core Flood
- Low Pressure Injection/Decay Heat Removal
- High Pressure Injection/Makeup and Purification
- Reactor Building Spray
- Reactor Building Cooling and Purge
- Sodium Hydroxide
- Reactor Building Isolation
- Hydrogen Control

Section 2.3.2 provides a description of these systems and identifies the components requiring aging management review for license renewal. The aging effects of specific material and environment combinations are described in Appendix C while the programs utilized to manage the aging effects are described in Appendix B. For the engineered safeguards systems, the specific materials and environments, the resulting aging effects, and the specific programs to manage those effects are listed in Table 3.3-1 through Table 3.3-8.

3.3.1 Materials and Environments

The engineered safeguards systems are exposed to borated water, treated water, raw water, nitrogen, air, sodium hydroxide, lube oil and ambient atmosphere. Most of the piping and piping components in the engineered safeguards systems are stainless steel. Carbon steel piping and piping components are used in the reactor building isolation system and carbon steel tanks are used in the NaOH system and the core flood system. Carbon steel bolting is used externally on some valves and flanges. In addition to stainless and carbon steel, brass/bronze is used for some lube oil piping components, and 90/10 CuNi is used for cooling coils in the reactor building cooling and purge systems.

The components, their intended functions, as well as the materials and environments for each engineered safeguard system are summarized in Table 3.3-1 though Table 3.3-8. The aging effects requiring management and the programs and activities that manage the aging effects for each applicable environment and material combination for each engineered safeguard system are discussed in the following paragraphs and are summarized in the tables at the end of this section.

3.3.2 Aging Effects Requiring Management

The aging effects requiring management for carbon steel external parts and bolting are loss of material due to leakage of borated water and loss of mechanical closure integrity. This effect is applicable to all subsystems in the engineered safeguards that contain carbon steel.

The stainless steel piping and valve bodies in the LPI/DHR system that are wetted by the sump water are susceptible to cracking and loss of material. Fouling is an aging effect requiring management for the lube oil side of the LPI/DHR pump lube oil coolers and the borated water side of the decay heat coolers. Aging effects for the service water side of the lube oil and decay heat coolers are addressed in Section 3.4.

Fouling is an aging effect requiring management for the pump lube oil coolers in the MUP/HPI system. Cracking is an aging effect for stainless steel in the MUP/HPI.

Cracking and fouling are aging effects for stainless steel in the reactor building spray system.

A specific aging effect applicable to the reactor building cooling and purge system is a loss of material in the reactor building cooler housing and the carbon steel ductwork and dampers. Fouling, loss of material, and loss of mechanical closure integrity are aging effects requiring management for the air cooler coils.

The stainless steel NaOH piping and components could experience cracking and loss of material. The carbon steel sodium hydroxide tank is susceptible to loss of material and loss of mechanical closure integrity.

The aging effects requiring management for the reactor building isolation system piping and valves are loss of mechanical closure integrity, loss of material from the internal surfaces of the carbon steel components, and cracking of stainless steel components of stainless steel components.

Minor fouling on the outside of the air-cooled heat exchangers due to dust and loss of material are identified as aging effects requiring management in the hydrogen control system.

3.3.3 Programs and Activities that Manage Aging Effects

The programs listed below will manage the aging effects identified in Section 3.3.2. Additionally, the Chemistry Control Programs, Oil Analysis Program, and the Maintenance Rule Program will prevent other aging effects from occurring.

- ASME Section XI ISI Program (IWC or IWD) is applicable to all engineered safeguards systems except the reactor building cooling and purge, and hydrogen control systems.
- ASME Inservice Inspection Program Augmented Inspections manage the aging effects of piping and components in the LPI/DHR, and reactor building isolation.
- The Wall Thinning Inspection Program is applicable to the sodium hydroxide tank and the reactor building isolation system.
- Boric Acid Corrosion Prevention Program is applicable to components in the core flood, LPI/DHR, HPI/MUP, reactor building spray, and reactor building isolation systems.
- Reactor Building Leak Rate Testing Program is credited for managing the aging effects of the reactor building penetrations.
- NaOH Tank Level Monitoring is applicable to the sodium hydroxide system.
- Reactor Building Sump Closeout Inspection is credited for the sump inspection for the LPI/DHR system.
- Core Flood Tank Monitoring is applicable to the core flood system.
- Preventive Maintenance Program activities include periodic inspection and cleaning of RB cooler components, interior and exterior inspections of the BWST, and inspections and cleaning of the heat exchanger external surfaces in the hydrogen control system.
- The Maintenance Rule Program is applicable to the core flood, LPI/DHR, HPI/MUP, and reactor building cooling and purge systems.
- The Heat Exchanger Monitoring Program is applicable to coolers in the LPI/DHR, HPI/MUP, and reactor building cooling and purge systems.
- ASME Inservice Inspection Program (IWD) manages the aging effects of piping and components for sodium hydroxide systems.

3.3.4 Operating Experience

A review of operating history using NPRDS and a review of NRC generic communications was performed to validate the set of aging effects requiring management as described in Appendix C. The industry correspondence found to be applicable to the systems in this section includes the following:

- NRC Bulletin 79-17, "Pipe Cracks in Stagnant Borated Water Systems at PWR Plants"
- NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems"
- NRC Bulletin 89-02, "Stress Corrosion Cracking of High-Hardness Type 410 Stainless Steel Internal Preloaded Bolting in Anchor Darling Model S350W Swing Check Valves or Valves of Similar Design"
- NRC Information Notice 79-19, "Pipe Cracks in Stagnant Borated Water Systems at Power Plants"
- NRC Information Notice 80-05, "Chloride Contamination of Safety-Related Piping and Components"
- NRC Information Notice 80-15, "Axial (Longitudinal) Oriented Cracking in Piping"
- NRC Information Notice 84-18, "Stress Corrosion Cracking in Pressurized Water Reactor Systems"
- NRC Information Notice 91-05, "Intergranular Stress Corrosion Cracking in Pressurized Water Reactor Safety Injection Accumulator Nozzles"
- NRC IE Circular 76-06, "Stress Corrosion Cracks in Stagnant, Low Pressure Stainless Piping Containing Boric Acid Solution at PWR's."

In addition, a review was performed at ANO to validate the accuracy and completeness of the list of applicable aging effects based on site experience. From these reviews, no aging effects requiring management were identified beyond those identified in this section.

Table 3.3-1 Core Flood System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Bolting External valve parts	Pressure boundary	Carbon steel	Borated water[2]	Loss of material Loss of mechanical closure integrity	ASME Section XI ISI- IWC (pressure tests) Core Flood Tank Monitoring Boric Acid Corrosion Prevention
			External-ambient	Loss of material[1]	Maintenance Rule
Piping Tubing Valves Orifices	Pressure boundary	Stainless Steel	Borated water	Cracking[1]	Primary Chemistry Monitoring Secondary Chemistry Monitoring
			External-ambient	None	None
Tanks	Pressure boundary	Carbon steel	Borated water[2]	Loss of material Loss of mechanical closure integrity	ASME Section XI ISI-IWC (pressure tests) Boric Acid Corrosion Prevention Maintenance Rule Core Flood Tank Monitoring
			External-ambient	Loss of material [1]	Maintenance Rule
		Stainless steel cladding Inconel	Borated water	Cracking[1]	Primary Chemistry Monitoring Secondary Chemistry Monitoring

1) Aging effect prevented by referenced program/activity
2) Component/system leakage

Table 3.3-2 Low Pressure Injection/Decay Heat Removal System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Piping Tubing Valves Flow elements Separators Heaters Pumps	Pressure boundary	Stainless steel	Borated water[(3)]	Cracking[(1)]	Primary Chemistry Monitoring
			External-ambient	None	None
Bolting External valve parts	Pressure boundary	Carbon steel	Borated water[(3)]	Loss of material Loss of mechanical closure integrity	ASME Section XI ISI-IWC Inspections Reactor Building Leak Rate Testing Boric Acid Corrosion Prevention Maintenance Rule
Piping Valves Appurtenances (wetted by sump water)	Pressure boundary	Stainless Steel	Raw water Borated water	Loss of material	Reactor Building Sump Closeout Inspection
				Cracking	ASME Section XI ISI -IWC Inspections Augmented Inspections
		Carbon steel	Raw water Borated water	Loss of material Loss of mechanical closure integrity	Reactor Building Sump Closeout Inspection Maintenance Rule

Table 3.3-2 Low Pressure Injection/Decay Heat Removal System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Heat exchangers (DH Coolers)	Pressure boundary	Stainless steel	Borated water [(2)]	Cracking[(1)]	Primary Chemistry Monitoring
				Loss of material[(1)]	ASME Section XI ISI-IWC (pressure tests) Heat Exchanger Monitoring
			External-ambient	None	None
	Heat transfer	Stainless steel	Borated water[(2)]	Fouling	ASME Section XI ISI-IWC (pressure tests) Heat Exchanger Monitoring
				Fouling	Service Water Integrity
Heat exchangers (Lube Oil Coolers)	Pressure boundary	Stainless steel	Lube oil	Loss of material[(1)]	Oil Analysis
			External-ambient	None	None
	Heat transfer	Stainless steel	Lube oil[(1)]	Fouling[(1)]	Oil Analysis
BWST	Pressure boundary	Carbon steel	Borated water	Loss of material[(1)]	Preventive Maintenance
			External-ambient	Loss of material Loss of mechanical closure integrity	Preventive Maintenance

1) Aging effect prevented by referenced program/activity
2) Outside of tubes
3) Component/system leakage

Table 3.3-3 Makeup and Purification/High Pressure Injection System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Piping Flow elements Valves Filters Separators Pumps casings	Pressure boundary	Stainless steel	Borated water	Cracking(1)	Primary Chemistry Monitoring
			External-ambient	None	None
Bolting External valve parts	Pressure boundary	Carbon steel	Borated water(3)	Loss of material Loss of mechanical closure integrity	ASME Section XI ISI –IWC (pressure tests) Reactor Building Leak Rate Testing Boric Acid Corrosion Prevention Maintenance Rule
Piping Valves Filters Tanks	Pressure boundary	Carbon steel Cast iron	Borated water(3)	Loss of material Loss of mechanical closure integrity	ASME Section XI ISI –IWC (pressure tests) Boric Acid Corrosion Prevention Maintenance Rule
			Lube oil	Loss of material(1)	Oil Analysis
Valves	Pressure boundary	Brass Bronze	Lube oil	Loss of material(1)	Oil Analysis
			External-ambient	None	None
Heat exchangers (lube oil coolers)	Pressure boundary	Stainless steel	Lube oil	Cracking(1)	Oil Analysis
			External-ambient	None	None
	Heat transfer	Stainless steel	Lube oil(2)	Fouling(1)	Oil Analysis

1) Aging effect prevented by referenced program/activity
2) Outside of tubes
3) Component/system leakage

Table 3.3-4 Reactor Building Spray System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Bolting External valve parts	Pressure boundary	Carbon steel	Borated water[2]	Loss of material Loss of mechanical closure integrity	ASME Section XI ISI-IWC (pressure tests) Boric Acid Corrosion Prevention
Piping Tubing Valves Separators Pump casings	Pressure boundary	Stainless steel	Borated water	Cracking[1]	Primary Chemistry Monitoring Secondary Chemistry Monitoring
			External-ambient	None	None
Heat exchanger (lube oil coolers)	Heat transfer	Stainless steel	Lube oil[2]	Fouling	Oil Analysis

1) Aging effect prevented by referenced program/activity
2) Component/system leakage

Table 3.3-5 Reactor Building Cooling and Purge System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Duct Dampers Pipe Valves Fan and cooler housings	Pressure boundary	Carbon steel	Gas-air	Loss of material[1]	Preventive Maintenance Maintenance Rule
			External-ambient	Loss of material[3]	Maintenance Rule
Heat exchangers	Pressure boundary	90/10 CuNi Stainless steel	Gas-air [2]	None	None
		Carbon steel	Gas-air	Loss of material[1] Loss of mechanical closure integrity[1]	Preventive Maintenance Maintenance Rule
			External-ambient	Loss of material [3]	Maintenance Rule
	Heat transfer	90/10 CuNi	Gas-air[2]	Fouling	Preventive Maintenance

1) On surfaces wetted by condensation
2) Outside of tubes
3) Prevented by program

Table 3.3-6 Sodium Hydroxide System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Piping Valves	Pressure boundary	Stainless steel	Sodium hydroxide	Loss of material	ASME Section XI ISI-IWD (pressure tests) NaOH Tank Level Monitoring
				Cracking	ASME Section XI ISI-IWD (pressure tests)
Bolting External valve parts	Pressure boundary	Carbon Steel	Sodium hydroxide[2]	Loss of material Loss of mechanical closure integrity	ASME Section XI ISI-IWD (pressure tests) NaOH Tank Level Monitoring
Tank	Pressure boundary	Carbon steel	Sodium hydroxide	Loss of material	ASME Section XI ISI-IWD (pressure tests) NaOH Tank Level Monitoring Wall Thinning Inspection
			External-ambient	Loss of material[1]	Maintenance Rule

1) Aging effect prevented by referenced program/activity
2) Component/system leakage

Table 3.3-7 Reactor Building Isolation System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Piping Valves	Pressure boundary	Carbon Steel	Treated water	Loss of Material	ASME Section XI ISI-IWC Inspections Reactor Building Leak Rate Testing Wall Thinning Inspection Secondary Chemistry Monitoring Auxiliary Systems Chemistry Monitoring
			Gas-nitrogen	None	None
			Gas-air	Loss of material	Reactor Building Leak Rate Testing Wall Thinning Inspection
			External-ambient	Loss of material[1]	Maintenance Rule
Bolting	Pressure boundary	Carbon steel	Borated water[2]	Loss of material Loss of mechanical closure integrity	ASME Section XI ISI-IWC Inspections Reactor Building Leak Rate Testing Boric Acid Corrosion Prevention
Piping Valves	Pressure boundary	Stainless steel	Treated water Borated water	Cracking	ASME Section XI ISI IWC Inspections Augmented Inspections Reactor Building Leak Rate Testing Primary Chemistry Monitoring Secondary Chemistry Monitoring Auxiliary Systems Chemistry Monitoring
			Gas-nitrogen	None	None
			External-ambient	None	None

1) Aging effect prevented by referenced program/activity
2) Component/system leakage

Table 3.3-8 Hydrogen Control System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Piping Valves	Pressure boundary	Carbon steel	Gas-air	None	None
			External-ambient	Loss of material(1)	Maintenance Rule
Valves	Pressure boundary	Stainless steel	Gas-air External-ambient	None	None
Recombiners	Pressure boundary	Stainless steel Incoloy-800	Gas-air External-ambient	None	None
Heat exchangers	Pressure boundary	Stainless steel	Gas-air	None	None
			External-ambient	None	None
	Heat transfer	Stainless steel	Gas-air(2)	None	None
			Gas-air(3)	Fouling	Preventive Maintenance
Sample stations	Pressure boundary	Stainless steel	Gas-air External-ambient	None	None
		Carbon steel	Gas-air	None	None
			External-ambient	Loss of material	Maintenance Rule

1) Aging effect prevented by referenced program/activity
2) Inside of tubes
3) Outside of tubes

3.4 AUXILIARY SYSTEMS

The following systems are included in this section:

- Spent Fuel
- Fire Protection
- Emergency Diesel Generator
- Auxiliary Building Sump and Reactor Building Drains
- Alternate AC Diesel Generator
- Halon
- Fuel Oil
- Instrument Air
- Chilled Water
- Service Water
- Penetration Room Ventilation
- Auxiliary Building Heating and Ventilation
- Control Room Ventilation

Section 2.3.3 provides a description of these systems and identifies the components requiring aging management review for license renewal. The aging effects of specific material and environment combinations are described in Appendix C while the programs utilized to manage the aging effects are described in Appendix B. For the auxiliary systems, the specific materials and environments, the resulting aging effects, and the specific programs to manage those effects are listed in Table 3.4-1 through Table 3.4-13.

3.4.1 Materials and Environments

The auxiliary systems are exposed to borated water, raw water, treated water, air, halon, freon, nitrogen, carbon dioxide, fuel oil, lube oil, concrete, soil and groundwater, and ambient atmosphere. Pressure boundary components are constructed of steel, stainless steel, carbon steel, cast iron, brass, bronze, copper, aluminum, glass and admiralty. Components whose intended function is heat transfer are constructed of 90/10 copper-nickel, brass, bronze, admiralty, copper, and stainless steel. In addition, the spent fuel pool racks that provide structural support, and the reactor building sump anti-vortex device, which has the safety function of vortex elimination as well as the reactor building sump floor drains screens, which have a safety function of debris screening, are all constructed of stainless steel. The Boraflex neutron absorber material used in the spent fuel pool is a boron carbide dispersion in an elastomeric silicone.

The components and their intended functions, as well as the materials and environments for each auxiliary system, are summarized in Table 3.4-1 through Table 3.4-13. The aging effects requiring management and the programs and activities that manage the aging effects for each applicable environment/material combination are discussed in the following paragraphs and are summarized in the tables at the end of this section.

3.4.2 Aging Effects Requiring Management

The aging effects for carbon steel external parts (bolts, nuts, etc.) in the spent fuel system are a potential loss of material and loss of mechanical closure integrity if leakage of borated water occurs. Cracking is a potential aging effect in the pool liner. Boraflex is addressed as a time-limited aging analysis in Section 4.7.

The aging effect for cast iron or carbon steel components in the fire protection system is a loss of material. The stainless steel, brass and bronze components in the system are subject to a loss of material and cracking. Loss of mechanical closure integrity is an aging effect for the diesel fire pump intake and exhaust subsystem components and lubrication subsystem components. Diesel fire pump exhaust subsystem components are susceptible to a loss of material and cracking, and the heat exchangers are susceptible to a loss of material and fouling. Loss of material and loss of mechanical closure integrity are aging effects for the diesel fire pump cooling water subsystem.

Loss of material is an aging effect for the carbon steel components in the EDG starting air system, cooling water carbon steel components, the unpainted carbon steel internal surfaces, and the outer portion of the intake and exhaust components that could be exposed to rain. Loss of material and fouling are aging effects for the EDG intake air aftercoolers, the lube oil coolers, and the cooling water heat exchangers. The stainless steel components in the EDG system are susceptible to cracking. Since portions of the engine subsystems are exposed to high vibration, loss of mechanical closure integrity is an aging effect for the skid mounted and connected components.

The aging effects for the auxiliary building sump and reactor building drains include a loss of material, cracking of the stainless steel and brass portions of the system that are potentially exposed to chlorides, fluorides, or sulfates, and loss of mechanical closure integrity.

Loss of material and loss of mechanical closure integrity are aging effects for carbon steel in the intake and exhaust system components, the cooling water, and the starting air components in the alternate AC generator system. Loss of material, fouling, and loss of mechanical closure integrity are aging effects for the intake air aftercooler and lube oil cooler, and loss of material, cracking and loss of mechanical closure integrity are aging effects for the exhaust subsystem carbon steel and stainless steel components. Loss of mechanical closure integrity affects the lube oil subsystem components. Fouling and a loss of material are aging effects for the AAC radiator. The aging effect for the AAC building ventilation subsystem components is a loss of material from wetted portions of the exhaust fan housings.

The halon system discharge tube assembly and pilot header flexible tubing and fittings are susceptible to a loss of material or cracking due to frequent disconnecting of equipment.

The fuel oil tanks in the fuel oil system are susceptible to a loss of material from the inside bottom surface. The bulk fuel oil storage tank could also experience a loss of material from the outside surface. Loss of material is an aging effect for the external surface of the underground piping. Loss of mechanical closure integrity is an aging effect for components subjected to engine vibration from the EDGs, AAC diesel, or fire diesel. Fouling is an aging effect for the AAC diesel fuel oil return cooler.

No aging effects were identified for the passive components in the instrument air system due to the exposure to dry gases and internal building environments.

The auxiliary building chiller components are susceptible to a loss of material from carbon steel internal surfaces. The auxiliary building chiller evaporators and condensers are susceptible to a loss of material from the external tube surfaces. Fouling is an aging effect for the heat exchanger tubes. The reactor building penetrations in the chilled water system are susceptible to a loss of material from the internal carbon steel surfaces and from the uninsulated external carbon steel surfaces, as well as a loss of mechanical closure integrity, and cracking of the stainless steel components.

The aging effects for the service water side of the heat exchangers in the service water system as well as the stainless steel, brass and bronze components in this system are loss of material, cracking and fouling. Carbon steel components are susceptible to a loss of material and fouling. The sluice gates are susceptible to a loss of material for the cast iron sub-components and loss of material and cracking of the stainless steel sub-components. The aging effects for the SW pump casings is loss of material from the carbon steel sub-components and loss of material and cracking of stainless steel sub-components.

A loss of material is an aging effect for portions of the penetration room ventilation system exhaust stack exposed to the weather.

Fouling is an aging effect for the external heat transfer surfaces, and for the external surfaces of the tubes and fins of the cooling subsystems in the auxiliary building electrical rooms, switchgear rooms, and decay heat pump rooms. The EDG ventilation subsystem is susceptible to a loss of material from the external surface of the exhaust penthouse assembly and portions of the intake exposed to rain. The decay heat pump room cooling and make up pump room cooling subsystems are also susceptible to a loss of material from the carbon steel components that are adjacent to the cooling coils.

Loss of material is an aging effect for the wet portions of the cooling coils and areas adjacent to or below the cooling coils in the housings for the control room coolers and fouling is an aging effect for the external coil surface of these coolers.

3.4.3 Programs and Activities that Manage Aging Effects

- The Chemistry Control Programs, the Oil Analysis Program, the Instrument Air Quality Program, and Maintenance Rule Program will prevent aging effects from occurring.

Additionally, the programs and activities listed below will manage the aging effects identified in Section 3.4.2.

- Maintenance Rule Program will ensure piping and component integrity is maintained in the fuel oil, chill water, penetration room ventilation and auxiliary building heating and ventilation systems, the alternate AC generator system and the wetted external portions of the EDG exhaust.

- Boric Acid Corrosion Prevention Program is applicable to components in the spent fuel system, and the auxiliary building sump and reactor building drain systems.
- Reactor Building Leak Rate Testing Program is credited for managing the aging effects of the reactor building penetrations.
- ASME Section XI Inservice Inspection Program (IWC or IWD) is credited for managing aging effects in the spent fuel, chilled water, and service water systems.
- The ASME Section XI ISI Augmented Inspection will manage aging effects of components wetted by sump water.
- The Wall Thinning Inspection Program is applicable for the chilled water system.
- The Heat Exchanger Monitoring Program is applicable to the chilled water system for seismic qualification of heat exchangers, and to the EDG heat exchanger testing per GL 89-13.
- Service Water Integrity Program applies to service water testing, cleaning, and inspections per GL 89-13 commitments.
- The Fire Protection Program monitors fire protection system leakage, piping thickness and preventive maintenance and includes sprinkler testing or replacement.
- The Reactor Building Sump Closeout Inspection is credited with inspection of the reactor building sump screens and floor drain screens.
- Preventive Maintenance Program applies to the auxiliary building ventilation system, the fuel oil system, and the control room ventilation system.
- Spent Fuel Pool Monitoring and Spent Fuel Pool Level Monitoring Programs apply to the spent fuel pool liner plate.
- Buried Pipe Inspection Program applies to the service water and fuel oil systems.
- RCP Oil Collection System Inspection applies to the RCP oil collection system
- Alternate AC Diesel Generator Testing and Inspections Program applies to AAC diesel generator system.
- Control Room Ventilation Testing applies to the control room ventilation system.
- EDG Testing and Inspections applies to the EDG system.
- The Control Room Halon Fire System Inspection applies to the halon system.
- Diesel Fuel Monitoring Program applies to the fuel oil system.

3.4.4 Operating Experience

A review of operating history using NPRDS and a review of NRC generic communications was performed to validate the set of aging effects requiring management as described in Appendix C. The industry correspondence that was found applicable to the systems in this section includes the following:

- NRC Bulletin 79-17, "Pipe Cracks in Stagnant Borated Water Systems at PWR Plants"
- NRC Bulletin 81-03, "Flow Blockage of Cooling Water to Safety System components by Corbicula Sp. (Asiatic Clam) and Mytilus Sp. (Mussel)"
- NRC Bulletin 89-02, "Stress Corrosion Cracking of High-Hardness Type 410 Stainless Steel Internal Preloaded Bolting in Anchor Darling Model S350W Swing Check Valves or Valves of Similar Design"
- NRC Information Notice 79-19, "Pipe Cracks in Stagnant Borated Water Systems at Power Plants"
- NRC Information Notice 79-23, "Emergency Diesel Generator Lube Oil Coolers"
- NRC Information Notice 80-05, "Chloride Contamination of Safety-Related Piping and Components"
- NRC Information Notice 80-15, "Axial (Longitudinal) Oriented Cracking in Piping"
- NRC Information Notice 81-21, "Potential Loss of Direct Access to Ultimate Heat Sink"
- NRC Information Notice 83-51, "Diesel Generator Events"
- NRC Information Notice 84-18, "Stress Corrosion Cracking in Pressurized Water Reactor Systems"
- NRC Information Notice 84-71, "Graphitic Corrosion of Cast Iron in Salt Water"
- NRC Information Notice 85-24, "Failures of Protective Coatings in Pipes and Heat Exchangers"
- NRC Information Notice 85-30, "Microbiologically Induced Corrosion of Containment Service Water System"
- NRC Information Notice 85-56, "Inadequate Environment Control for Components and Systems in Extended Storage or Layup"
- NRC Information Notice 86-73, "Recent Emergency Diesel Generator Problems"
- NRC Information Notice 86-96, "Heat Exchanger Fouling Can Cause Inadequate Operability of Service Water Systems"
- NRC Information Notice 87-43, "Gaps in Neutron-Absorbing Material in High-Density Spent Fuel Storage Racks"

- NRC Information Notice 88-37, “Flow Blockage of Cooling Water to Safety System Components”
- NRC Information Notice 88-92, “Potential for Spent Fuel Pool Draindown”
- NRC Information Notice 89-01, “Valve Body Erosion”
- NRC Information Notice 89-07, “Failures of Small-Diameter Tubing in Control Air, Fuel Oil, and Lube Oil Systems Which Render Emergency Diesel Generators Inoperable”
- NRC Information Notice 89-76, “Biofouling Agent: Zebra Mussel”
- NRC Information Notice 90-26, “Inadequate Flow of Essential Service Water to Room Coolers and Heat Exchangers for Engineered Safety-Feature Systems”
- NRC Information Notice 90-39, “Recent Problems with Service Water Systems”
- NRC Information Notice 93-70, “Degradation of Boraflex Neutron Absorber Coupons”
- NRC Information Notice 94-59, “Accelerated Dealloying of Cast Aluminum-Bronze Valves Caused by Microbiologically Induced Corrosion”
- NRC Information Notice 94-79, “Microbiologically Influenced Corrosion of Emergency Diesel Generator Service Water Piping”
- NRC Information Notice 95-38, “Degradation of Boraflex Neutron Absorber in Spent Fuel Storage Racks”
- NRC Generic Letter 88-14, “Instrument Air Supply System Problems Affecting Safety-Related Equipment”
- NRC Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment”
- NRC IE Circular 76-06, “Stress Corrosion Cracks in Stagnant, Low Pressure Stainless Piping Containing Boric Acid Solution at PWRs.”

In addition, a review was performed at ANO to validate the accuracy and completeness of the list of aging effects requiring management based on site experience. From these reviews, no aging effects requiring management were identified beyond those discussed in this section.

Table 3.4-1 Spent Fuel System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Liner plate	Pressure boundary	Stainless steel	Borated water	Cracking	Primary Chemistry Monitoring
			External-Concrete	Cracking	Spent Fuel Pool Level Monitoring
Gates Racks	Structural support	Stainless steel	Borated water	Cracking(1)	Primary Chemistry Monitoring
Piping Valves	Pressure boundary	Stainless steel	Borated water	Cracking(1)	Primary Chemistry Monitoring
Fuel transfer tube blind flanges	Pressure boundary	Stainless steel	Borated water	Cracking(1)	Primary Chemistry Monitoring
Bolting External valve parts	Pressure boundary	Carbon steel	Borated water(2)	Loss of material Loss of mechanical closure integrity	ASME Section XI ISI-IWD (pressure tests) Boric Acid Corrosion Prevention Reactor Building Leak Rate Testing

1) Aging effect prevented by referenced program/activity
2) Component/system leakage

Table 3.4-2 Fire Protection System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Pumps Piping Valves	Pressure boundary	Cast iron Carbon steel	Raw water	Loss of material	Fire Suppression Water Supply Surveillance Fire Water Piping Thickness Evaluation Reactor Building Leak Rate Testing Service Water Chemical Control
			External-ambient	Loss of material(1)	Maintenance Rule Fire Suppression Water Supply System Surveillance
Piping Valves	Pressure boundary	Stainless steel	Raw water	Loss of material(1) Cracking	Service Water Chemical Control Fire Suppression Water Supply System Surveillance
			External-ambient	None	None
Piping Valves	Pressure boundary	Brass Bronze	Raw water	Loss of material(1) Cracking	Service Water Chemical Control Fire Suppression Water Supply System Surveillance
			External-ambient	None	None
Diesel Fire Pump Subsystems and Components					
Intake air	Pressure boundary	Carbon steel Aluminum	Gas-air	Loss of mechanical closure integrity	Preventive Maintenance Fire Suppression Water Supply System Surveillance
			External-ambient	Loss of material(1)	Maintenance Rule
Exhaust air	Pressure boundary	Carbon steel	Gas-air	Loss of mechanical Closure integrity Cracking Loss of material	Preventive Maintenance Fire Suppression Water Supply System Surveillance
			External-ambient	None	None

Table 3.4-2 Fire Protection System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Lube oil	Pressure boundary	Carbon steel Cast iron	Lube oil	Loss of mechanical closure integrity	Preventive Maintenance Oil Analysis Fire Suppression Water Supply System Surveillance
			External-ambient	None	None
Cooling water	Pressure boundary	Carbon steel Cast iron	Treated water	Loss of mechanical closure integrity Loss of material(1)	Fire Suppression Water Supply System Surveillance
			External-ambient	Loss of material(1)	Maintenance Rule
Heat exchangers	Pressure boundary	Cast iron, Brass	Lube oil	Loss of material	Fire Suppression Water Supply System Surveillance Oil Analysis
			External-ambient	Loss of material	Preventive Maintenance
	Heat transfer	90/10 Cu-Ni Copper	Treated water(2) Lube oil(3)	Fouling	Fire Suppression Water Supply System Surveillance Oil Analysis

1) Aging effect prevented by referenced program/activity
2) Inside of tubes
3) Outside of tubes

Table 3.4-3 Emergency Diesel Generator System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Starting Air Subsystem					
Valves Bolting External valve parts	Pressure boundary	Carbon steel	Gas-air	Loss of material	Emergency Diesel Generator Testing and Inspections
			External-ambient	Loss of material Loss of mechanical closure integrity	Emergency Diesel Generator Testing and Inspections Maintenance Rule
Piping Valves Tanks Strainers	Pressure boundary	Stainless steel	Gas-air	None	None
			External-ambient	None	None
Tubing	Pressure boundary	Copper	Gas-air	None	None
			External-ambient	None	None
Air Intake and Exhaust Subsystems					
Piping Filters Expansion joints Turbochargers	Pressure boundary	Carbon steel	Gas-air	Loss of material Loss of mechanical closure integrity	Emergency Diesel Generator Testing and Inspections
			External-ambient	Loss of material(1)	Maintenance Rule
Valves	Pressure boundary	Brass, bronze	Gas-air	Cracking	Emergency Diesel Generator Testing and Inspections
			External-ambient	None	None
Heat exchangers	Pressure boundary	Carbon steel	Gas-air	Loss of material Loss of mechanical closure integrity	Emergency Diesel Generator Testing and Inspections
			External-ambient	Loss of material(1)	Maintenance Rule

Table 3.4-3 Emergency Diesel Generator System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
	Heat transfer	Copper with aluminum fins	Treated water[(2))]	Fouling	Emergency Diesel Generator Testing and Inspections Auxiliary Systems Water Chemistry Monitoring
			Gas-air	Fouling	Emergency Diesel Generator Testing and Inspections Auxiliary Systems Water Chemistry Monitoring
Lube Oil Subsystem					
Piping Valves Filters Pumps	Pressure boundary	Carbon steel	Lube oil	Loss of material[(1)]	Oil Analysis
			External-ambient	Loss of mechanical closure integrity	Emergency Diesel Generator Testing and Inspections Maintenance Rule
				Loss of material[(1)]	Maintenance Rule
Valves	Pressure boundary	Stainless steel	Lube oil	Cracking[(1)]	Oil Analysis
			External-ambient	Loss of mechanical closure integrity	Emergency Diesel Generator Testing and Inspections
Filters	Pressure boundary	Aluminum	Lube oil	Loss of material[(1)]	Oil Analysis
			External-ambient	Loss of mechanical closure integrity	Emergency Diesel Generator Testing and Inspections
Valves	Pressure boundary	Brass, Bronze, admiralty	Lube oil	Loss of material[(1)]	Oil Analysis
			External-ambient	Loss of mechanical closure integrity	Emergency Diesel Generator Testing and Inspections
Sight glasses	Pressure boundary	Glass	Lube oil	None	None
			External-ambient	Loss of mechanical closure integrity	Emergency Diesel Generator Testing and Inspections

Table 3.4-3 Emergency Diesel Generator System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Strainer	Pressure boundary	Cast iron	Lube oil	Loss of material(1)	Oil Analysis
			External-ambient	Loss of mechanical closure integrity Loss of material(1)	Emergency Diesel Generator Testing and Inspections Maintenance Rule
Heat exchangers	Pressure boundary	Carbon steel	Treated water	Loss of material(1) Fouling	Emergency Diesel Generator Testing and Inspections Auxiliary Systems Chemistry Monitoring
	Heat transfer	Brass	Lube oil	Fouling	Emergency Diesel Generator Testing and Inspections Auxiliary Systems Chemistry Monitoring
Cooling Water Subsystem					
Pipe Pumps Valves Tanks	Pressure boundary	Carbon steel	Treated water	Loss of material(1)	Emergency Diesel Generator Testing and Inspections Auxiliary Systems Chemistry Monitoring
				Loss of mechanical closure integrity	Emergency Diesel Generator Testing and Inspections
			External-ambient	Loss of material(1)	Maintenance Rule
Valves Thermowells	Pressure boundary	Stainless steel	Treated water	Cracking	Emergency Diesel Generator Testing and Inspections Auxiliary Systems Chemistry Monitoring
			External-ambient	None	None
Level Glass	Pressure boundary	Glass	Treated water	None	None
			External-ambient	None	None

Table 3.4-3 Emergency Diesel Generator System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Heat exchangers	Pressure boundary	Carbon steel	Treated water	Loss of material[1]	Emergency Diesel Generator Testing and Inspections Auxiliary Systems Chemistry Monitoring
	Heat transfer	Admiralty	Treated water[2]	Fouling	Auxiliary Systems Chemistry Monitoring Emergency Diesel Generator Testing and Inspections
			Lube Oil[3]	Fouling Loss of material[1]	Oil Analysis
	Pressure Boundary	Carbon Steel	Lube Oil[3]	Loss of material[1] Loss of mechanical closure integrity	Emergency Diesel Generator Testing and Inspections Maintenance Rule Oil analysis

1) Aging effect prevented by referenced program/activity
2) Inside of tubes
3) Outside of tubes

Table 3.4-4 Auxiliary Building Sump and Reactor Building Drain System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Piping Valves Bolting External valve parts	Pressure boundary	Carbon steel	Treated water, borated water	Loss of material Loss of mechanical closure integrity	Local Leak Rate Testing Primary Chemistry Monitoring Secondary Chemistry Monitoring
			External-ambient	None	None
Piping Valves Tanks Bolting	Pressure Boundary	Carbon Steel	Borated water[1] Oil	Loss of material Loss of mechanical closure integrity	Boric Acid Corrosion Prevention RCP Oil Leakage Collection System Inspection
Piping Valves	Pressure boundary	Stainless steel	Borated water Raw water Treated water	Loss of material Cracking Loss of mechanical closure integrity	ASME Section XI ISI-Augmented Inspections Reactor Building Leak Rate Testing
			External-ambient	None	None
Valves	Pressure boundary	Brass Bronze admiralty	Raw water, treated water, borated water	Loss of material Cracking	ASME Section XI ISI-Augmented Inspections
			External-ambient	None	None
Anti-vortex device	Vortex elimination	Stainless steel	External-ambient	None	None
Screens	Debris screening	Stainless steel	Borated water Treated water Raw water Oil	Loss of material Cracking	Reactor Building Sump Closeout Inspection

1) Component/system leakage

Table 3.4-5 Alternate AC Diesel Generator System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Starting Air Subsystem					
Valves Tanks Filters Motor casing	Pressure boundary	Carbon steel	Gas-air	Loss of material	Alternate AC Diesel Generator Testing and Inspection
			External-ambient	Loss of mechanical closure integrity Loss of material[(1)]	Alternate AC Diesel Generator Testing and Inspection Maintenance Rule
Piping Valves	Pressure boundary	Stainless steel	Gas-air	None	None
			External-ambient	Loss of mechanical closure integrity	Alternate AC Diesel Generator Testing and Inspection
Valves	Pressure boundary	Brass, bronze or admiralty	Gas-air	None	None
			External-ambient	Loss of mechanical closure integrity	Alternate AC Diesel Generator Testing and Inspection
Valves Filter	Pressure boundary	Aluminum	Gas-air	None	None
			External-ambient	Loss of mechanical closure integrity	Alternate AC Diesel Generator Testing and Inspection
Air Intake and Exhaust Subsystems					
Piping Valves Muffler Turbocharger	Pressure boundary	Carbon steel	Gas-air	Loss of material Loss of mechanical closure integrity	Alternate AC Diesel Generator Testing and Inspection Maintenance Rule
			External-ambient	Loss of material [(1)]	Maintenance Rule
Valves Expansion joints	Pressure boundary	Stainless steel	Gas-air	Cracking[(1)]	Alternate AC Diesel Generator Testing and Inspection
			External-ambient	None	None
Filters	Pressure boundary	Aluminum	Gas-air External-ambient	None	None

Table 3.4-5 Alternate AC Diesel Generator System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Heat exchanger	Pressure boundary	Cast iron	Gas-air	Loss of material Loss of mechanical closure integrity	Alternate AC Diesel Generator Testing and Inspection Maintenance Rule
			External-ambient	Loss of material [1]	Maintenance Rule
	Heat transfer	Copper admiralty	Treated water[2]	Fouling	Alternate AC Diesel Generator Testing and Inspection Auxiliary Systems Chemistry Monitoring
			Gas-air[3]	Fouling	Alternate AC Diesel Generator Testing and Inspection
Lube Oil Subsystem					
Valves Pumps Heater	Pressure boundary	Carbon steel	Lube oil	Loss of material[1]	Oil Analysis
			External-ambient	Loss of mechanical closure integrity	Alternate AC Diesel Generator Testing and Inspection
				Loss of material[1]	Maintenance Rule
Valves	Pressure boundary	Stainless steel	Lube oil	Cracking[1]	Oil Analysis
			External-ambient	Loss of mechanical closure integrity	Alternate AC Diesel Generator Testing and Inspection
Valves	Pressure boundary	Brass Bronze Admiralty	Lube oil	Loss of material[1]	Oil Analysis
			External-ambient	Loss of mechanical closure integrity	Alternate AC Diesel Generator Testing and Inspection

Table 3.4-5 Alternate AC Diesel Generator System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Heat exchanger	Pressure boundary	Carbon steel	Treated water	Loss of material Loss of mechanical closure integrity	Alternate AC Diesel Generator Testing and Inspection Auxiliary Systems Chemistry Monitoring
			External-ambient	Loss of material[1]	Maintenance Rule
	Heat transfer	Copper	Lube oil[2]	Loss of material[1] Fouling	Oil Analysis
			Treated water[3]	Fouling	Alternate AC Diesel Generator Testing and Inspection Auxiliary Systems Chemistry Monitoring
Cooling Water Subsystem					
Piping Valves Pumps Tanks Heaters Orifices Filters	Pressure boundary	Carbon steel	Treated water	Loss of mechanical closure integrity	Alternate AC Diesel Generator Testing and Inspection Auxiliary Systems Chemistry Monitoring
				Loss of material[1]	Auxiliary Systems Chemistry Monitoring
			External-ambient	Loss of material[1]	Maintenance Rule
Valves Thermowells Expansion joints	Pressure boundary	Stainless steel	Treated water	Cracking[1]	Auxiliary Systems Chemistry Monitoring Alternate AC diesel Generator Testing and Inspection
			External-ambient	None	None
Valves	Pressure boundary	Brass Bronze Admiralty	Treated water	Loss of material[1]	Auxiliary Systems Chemistry Monitoring
			External-ambient	None	None

Table 3.4-5 Alternate AC Diesel Generator System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Heat exchanger	Pressure boundary	Carbon steel	Gas-air	Loss of material[1]	Maintenance Rule
			External-ambient	Loss of material	Alternate AC Diesel Generator Testing and Inspection
	Heat transfer	Brass Bronze admiralty	Treated water[2] Gas-air [3]	Fouling	Alternate AC Diesel Generator Testing and Inspection Auxiliary Systems Chemistry Monitoring
	Pressure boundary	Brass Bronze admiralty	Treated water[2] Gas-air [3]	Loss of material	Alternate AC Diesel Generator Testing and Inspection Auxiliary Systems Chemistry Monitoring
Alternate AC Building Ventilation					
Fans	Pressure boundary	Carbon steel	Gas-air	None	None
			External-ambient	Loss of material[1]	Alternate AC Diesel Generator Testing and Inspection Maintenance Rule
Dampers/louvers	Pressure boundary	Aluminum	Gas-air External-ambient	None	None

1) Aging effect prevented by referenced program/activity
2) Inside of tubes
3) Outside of tubes

Table 3.4-6 Halon System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Valves	Pressure boundary	Brass	Gas-halon Gas-nitrogen	None	None
			External-ambient	None	None
Pipe	Pressure boundary	Steel	Gas-halon Gas-nitrogen	None	None
			External-ambient	Loss of material(1)	Maintenance Rule
Tanks	Pressure boundary	Carbon steel	Gas-halon Gas-nitrogen	None	None
			External-ambient	Loss of material(1)	Maintenance Rule
Discharge nozzles	Pressure boundary	Aluminum	Gas-halon Gas-nitrogen	Loss of material	Control Room Halon Fire System Inspection
			External-ambient	None	None
Discharge tube	Pressure boundary	Steel	Gas-halon Gas-nitrogen	Loss of material	Control Room Halon Fire System Inspection
			External-ambient	Loss of material(1) Cracking	Maintenance Rule
Pilot header discharge tube flexible connectors	Pressure boundary	Stainless steel	Gas-halon Gas-nitrogen	Loss of material	Control Room Halon Fire System Inspection
			External-ambient	Loss of material(1) Cracking	Maintenance Rule

1) Aging effect prevented by referenced program/activity

Table 3.4-7 Fuel Oil System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Piping Valves Filters Pumps Tubing	Pressure boundary	Carbon steel	Fuel oil	Loss of material Loss of mechanical closure integrity	EDG Testing and Inspections Diesel Fuel Monitoring
			External-ambient	Loss of material[1]	Maintenance Rule
			External-buried	Loss of material	Buried Pipe Inspection
Valves Filters Thermowells	Pressure boundary	Stainless steel	Fuel oil	Loss of mechanical closure integrity	EDG Testing and Inspections Diesel Fuel Monitoring
			External-ambient	None	None
Tubing Valves	Pressure boundary	Brass Bronze Copper admiralty	Fuel oil	Loss of mechanical closure integrity	EDG Testing and Inspections Diesel Fuel Monitoring
			External-ambient	None	None
Pumps Strainers	Pressure boundary	Cast iron	Fuel oil	Loss of mechanical closure integrity	EDG Testing and Inspections Diesel Fuel Monitoring
			External-ambient	None	None
Tanks	Pressure boundary	Carbon steel	Fuel oil	Loss of material	EDG Testing and Inspections Diesel Fuel Monitoring
			External-ambient	Loss of material[1]	Maintenance Rule
Heat exchanger	Pressure boundary Heat transfer	Stainless steel	Fuel oil[2]	Fouling	Alternate AC Diesel Generator Testing and Inspections Diesel Fuel Monitoring
			Gas-air[3]	Fouling	Alternate AC Diesel Generator Testing and Inspections

1) Aging effect prevented by referenced program/activity
2) Inside of tubes
3) Outside of tubes

Table 3.4-8 Instrument Air System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Tubing	Pressure boundary	Stainless steel	Gas-air	Cracking(1)	Instrument Air Quality
			External-ambient	None	None
Tubing	Pressure boundary	Copper	Gas-air	Loss of material(1)	Instrument Air Quality
			External-ambient	None	None
Valves	Pressure boundary	Stainless steel	Gas-air	Cracking(1)	Instrument Air Quality
			External-ambient	None	None
Valves	Pressure boundary	Carbon steel	Gas-air	Loss of material(1)	Instrument Air Quality
			External-ambient	Loss of material(1)	Maintenance Rule
Valves	Pressure boundary	Brass Bronze admiralty	Gas-air	Loss of material(1)	Instrument Air Quality
			External-ambient	None	None
Piping Flanges	Pressure boundary	Carbon steel	Gas-air	Loss of material(1)	Instrument Air Quality
			External-ambient	Loss of material(1)	Maintenance Rule
Tanks	Pressure boundary	Carbon steel	Gas-air	Loss of material(1)	Instrument Air Quality
			External-ambient	Loss of material(1)	Maintenance Rule
Regulators	Pressure boundary	Aluminum	Gas-air	None	None
			External-ambient	None	None

1) Aging effect prevented by referenced program/activity

Table 3.4-9 Chilled Water System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Piping Valves Thermowells Tanks Pumps	Pressure boundary	Carbon steel	Treated water	Loss of material	Reactor Building Leak Rate Testing ASME Section XI ISI –IWC (pressure tests) Auxiliary Systems Chemistry Monitoring Wall Thinning Inspection
			External-ambient	Loss of mechanical closure integrity	Maintenance Rule
Valves Tubing	Pressure boundary	Stainless steel	Treated water	Cracking[(1)]	Auxiliary Systems Chemistry Monitoring
			External-ambient	None	None
Valves Coils	Pressure boundary	Brass Bronze Copper	Treated water	Loss of material[(1)]	Auxiliary Systems Water Chemistry Monitoring
			External-ambient	None	None
Sight glasses	Pressure boundary	Glass	Treated Water	None	None
			External-ambient	None	None
Valves Filters Sight glasses Compressors Mufflers	Pressure boundary	Stainless steel Brass/bronze Carbon steel Glass	Gas-freon External-ambient	None	None
Valves	Pressure boundary	Stainless steel	Lube oil	Cracking[(1)]	Oil Analysis
			External-ambient	None	None

Table 3.4-9 Chilled Water System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Heat exchangers (evaporators)	Pressure boundary	Carbon steel	Treated water	Loss of material	Heat Exchanger Monitoring Auxiliary Systems Chemistry Monitoring
			External-ambient	Loss of material(1)	Maintenance Rule
	Heat transfer	Copper Brass Bronze	Gas-freon(2) Treated water(3)	Fouling	Heat Exchanger Monitoring Auxiliary Systems Chemistry Monitoring
	Pressure boundary	Copper Brass Bronze	Gas-freon(2) Treated water(3)	Loss of material	Auxiliary Systems Chemistry Monitoring
Heat exchangers (condensers)	Pressure boundary	Carbon steel	External-ambient	Loss of material(1)	Maintenance Rule
	Heat transfer	Copper Brass Bronze	Gas-freon(2)	Fouling	
	Pressure boundary	Copper Brass Bronze	Gas-freon(2)	Loss of material	Heat Exchanger Monitoring

1) Aging effect prevented by referenced program/activity
2) Inside of tubes
3) Outside of tubes

Table 3.4-10 Service Water System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Piping Pumps Strainers Valves	Pressure boundary	Carbon steel	Raw water	Loss of material	Service Water Integrity ASME Section XI ISI -IWC & IWD (pressure tests)
			External-buried	Loss of material	Buried Pipe Inspection
			External-ambient	Loss of material[(1)]	Maintenance Rule
Piping Pumps Valves Flow elements Thermowells	Pressure boundary	Stainless steel	Raw water	Loss of material Cracking	Service Water Integrity ASME Section XI ISI –IWC & IWD (pressure tests)
			External-ambient	None	None
Valves	Pressure boundary	Brass, bronze	Raw water	Loss of material Cracking	Service Water Integrity ASME Section XI ISI –IWC & IWD (pressure tests)
			External-ambient	None	None
Sluice gates	Pressure boundary	Cast iron	Raw water	Loss of material	Service Water Integrity
			External-ambient	Loss of material[(1)]	Maintenance Rule
		Stainless steel	Raw water	Loss of material Cracking	Service Water Integrity
			External-ambient	None	None

Table 3.4-10 Service Water System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Heat exchangers	Pressure boundary	Carbon steel	Raw water	Loss of material	Service Water Integrity ASME Section XI ISI –IWC & IWD (pressure tests) Heat Exchanger Monitoring
			External-ambient	Loss of material[1]	Maintenance Rule
		Stainless steel	Raw water	Loss of material Cracking	Service Water Integrity ASME Section XI ISI-IWD (pressure tests) Heat Exchanger Monitoring
			External-ambient	None	None
		Copper Stainless steel 90/10 Cu-Ni, admiralty	Raw water[2,3]	Loss of material	Service Water Integrity Heat Exchanger Monitoring
	Heat Transfer	Copper Stainless steel 90/10 Cu-Ni, admiralty	Raw water	Fouling	Service Water Integrity Heat Exchanger Monitoring

1) Aging effect prevented by referenced program/activity
2) Inside of tubes
3) Outside of tubes

Table 3.4-11 Penetration Room Ventilation System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Duct Dampers Valves Expansion joints	Pressure boundary	Carbon steel	Gas-air	None	None
			External-ambient	Loss of material[1]	Maintenance Rule
Exhaust stack	Pressure boundary	Carbon steel	Gas-air	None	None
			External-ambient	Loss of material[1]	Penetration Room Ventilation System Testing Maintenance Rule
Tubing	Pressure boundary	Copper Brass	Gas-air External-ambient	None	None
Flow elements	Pressure boundary	Stainless steel	Gas-air External-ambient	None	None
Blowers Filters	Pressure boundary	Carbon steel	Gas-air	None	None
			External-ambient	Loss of material[1]	Maintenance Rule

1) Aging effect prevented by referenced program/activity

Table 3.4-12 Auxiliary Building Heating and Ventilation System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Exterior ductwork, Louvers Fans	Pressure boundary	Carbon steel	Gas-air	None	None
			External-ambient	Loss of material(3)	EDG Testing and Inspections Preventive Maintenance Maintenance Rule
Ductwork Dampers	Pressure boundary	Carbon steel	Gas-air	None	None
			External-ambient	Loss of material(1)	Maintenance Rule
Heat exchangers (Switchgear room coolers)	Pressure boundary	Carbon steel	Gas-air	None	None
			External-ambient	Loss of material(1)	Maintenance Rule
		Copper	Gas-air	None	None
	Heat transfer	Copper	Gas-air(2)	Fouling	Preventive Maintenance
Heat exchangers (Decay heat room coolers)	Pressure boundary	Carbon steel	Gas-air	Loss of material(3)	Preventive Maintenance
			External-ambient	Loss of material(1)	Maintenance Rule
		Copper	Gas-air	None	None
	Heat transfer	Copper	Gas-air(2)	Fouling	Service Water Integrity Preventive Maintenance
Heat exchangers (Make up pump room coolers)	Pressure boundary	Carbon steel	Gas-air	Loss of material(3)	Preventive Maintenance
			External-ambient	Loss of material(1)	Maintenance Rule
		90/10 CuNi	Gas-air	None	None

1) Aging effect prevented by referenced program/activity
2) Outside of tubes
3) On surfaces wetted by condensation

Table 3.4-13 Control Room Ventilation System					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Ductwork Dampers Heat exchangers Fans Filters	Pressure boundary	Carbon steel	Gas-air	None	None
			External-ambient	Loss of material(1)	Maintenance Rule
Tubing Valves	Pressure boundary	Copper Brass admiralty	Gas-air & CO_2 External-ambient	None	None
Heat exchangers (Evaporators)	Pressure boundary	Copper	Gas-freon External-ambient	None	None
		Carbon steel	Gas-air & CO_2	Loss of material(4)	Control Room Ventilation Testing Preventive Maintenance
			External-ambient	None	None
	Heat transfer	Copper	Freon(2)	None	None
			Gas-air	Fouling	Control Room Ventilation Testing Preventive Maintenance
Heat Exchanger (Condenser)	Pressure boundary	Carbon steel	External-ambient	Loss of material(1)	Maintenance Rule
			Lube oil	Loss of material(1)	Oil Analysis
	Heat transfer	90/10 CuNi	Freon(2)	None	None
Compressor	Pressure boundary	Carbon steel	Lube Oil	Loss of material(1)	Oil Analysis
			Freon	None	None
			External-ambient	Loss of material(1)	Maintenance Rule

1) Aging effect prevented by referenced program/activity
2) Inside of tubes
3) Outside of tubes
4) On surfaces wetted by condensation

3.5 STEAM AND POWER CONVERSION SYSTEMS

The following systems are included in this section.

- Main steam
- Main feedwater
- Emergency feedwater
- Condensate storage and transfer

Section 2.3.4 provides a description of these systems and identifies the components requiring aging management review for license renewal. The aging effects of specific material and environment combinations are described in Appendix C while the programs utilized to manage the aging effects are described in Appendix B. For the steam and power conversion systems, the specific materials and environments, the resulting aging effects and the specific programs to manage those effects are listed in Table 3.5-1 through Table 3.5-4.

3.5.1 Materials and Environment

The steam and power conversion systems are exposed to treated water, lube oil, nitrogen, air, and ambient atmosphere. The piping, tubing and associated components are constructed of carbon steel, stainless steel, and cast iron. Carbon steel bolting is used externally on valves, flanges, and tank manways. There is brass, bronze, copper and aluminum present in the steam supply and exhaust piping and the lube oil coolers in the EFW system. This is in addition to cast iron, carbon steel, stainless steel, and chrome-moly steel. The potential aging effects for the applicable environment and material combinations are discussed in the following sections.

The components, and their intended functions, as well as the materials and environments for each steam and power conversion system, are summarized in Table 3.5-1 through Table 3.5-4. The aging effects requiring management and the programs and activities that manage the aging effects for each applicable environment and material combination are discussed in the following paragraphs and are summarized in the tables at the end of this section.

3.5.2 Aging Effects Requiring Management

The carbon steel and stainless steel piping and components in the main steam system are subject to cracking, loss of mechanical closure integrity, and loss of material.

The carbon steel piping and valves in the main feedwater system can be affected by a loss of material and cracking. A loss of material and subsequent loss of mechanical closure integrity is a potential aging effect for the bolted carbon steel bolting and closures.

Loss of material is an aging effect requiring management for the EFW pumps, discharge piping and valves, the EFW turbine steam supply and exhaust piping, the cooler housings, and the cooling and seal water piping and valves, which are primarily constructed from carbon steel. Fouling is an aging effect requiring management for the coolers. Loss of mechanical closure integrity is an aging effect for the EFW turbine steam supply piping.

The aging effect requiring management for the carbon steel piping and valves in the condensate storage system is loss of material. Additionally, the carbon steel bolting on the condensate storage tank manways can experience loss of material resulting in a loss of mechanical closure integrity.

3.5.3 Programs and Activities that Manage Aging Effects

The programs listed below will manage the aging effects identified in Section 3.5.2. Additionally, the Secondary Chemistry Monitoring Program, the Oil Analysis Program, and Maintenance Rule Program will prevent other aging effects from occurring.

- The ASME Section XI Inservice Inspection Program (IWC and IWD pressure testing and inspections) is applicable to the steam and power conversion systems.
- Flow Accelerated Corrosion Prevention Program is applicable to components in the main feedwater and main steam systems.
- The Wall Thinning Inspection Program is applicable to carbon steel piping in the main steam and condensate storage and transfer systems. This program will also provide verification of the EFW pressure boundary.
- The ASME Section XI Inservice Inspection Program (Augmented Inspections) will provide special inspection of safety-related stainless steel and carbon steel piping in the Main Steam System and safety-related carbon steel piping in the Main Feedwater System.
- The Heat Exchanger Monitoring Program will provide monitoring of coolers in the EFW tank.
- The Maintenance Rule Program is applicable to the exterior of the safety-related condensate storage tank.
- Leakage Detection in Reactor Building monitors leakage in the reactor building sump per ANO-1 Technical Specifications for components in the reactor building including main feedwater and main steam systems.
- EFW pump testing is applicable to components in the EFW system.

3.5.4 Operating Experience

A review of operating history using NPRDS and a review of NRC generic communications was performed to validate the set of aging effects that require management. The industry correspondence that was found applicable to the systems in this section includes the following:

- NRC Bulletin 79-13, “Cracking in Feedwater System Piping”
- NRC Bulletin 87-01, “Thinning of Pipe Walls in Nuclear Power Plants”
- NRC Information Notice 80-29, “Broken Studs on Terry Turbine Steam Inlet Flanges”
- NRC Information Notice 81-04, “Cracking in Main Steam Lines”
- NRC Information Notice 84-87, “Piping Thermal Deflection Induced by Stratified Flow”
- NRC Information Notice 86-106, “Feedwater Line Break”
- NRC Information Notice 87-36, “Significant Unexpected Erosion of Feedwater Lines"
- NRC Information Notice 88-17, “Summary of Responses to NRC Bulletin 87-01, Thinning of Pipe Walls in Nuclear Power Plants”
- NRC Information Notice 91-18, “High-Energy Piping Failures Caused by Wall Thinning”
- NRC Information Notice 91-19, “Steam Generators Feedwater Distribution Piping Damage”
- NRC Information Notice 91-28, “Cracking in Feedwater System Piping”
- NRC Information Notice 91-38, “Thermal Stratification in Feedwater System Piping”
- NRC Information Notice 92-07, “Rapid Flow-Induced Erosion/Corrosion of Feedwater Piping”
- NRC Generic Letter 79-20, “Information Requested on PWR Feedwater Lines”
- NRC Generic Letter 89-08, “Erosion/Corrosion-Induced Pipe Wall Thinning.”

In addition, a review was performed at ANO to validate the accuracy and completeness of the list of applicable aging effects based on site experience. From these reviews, no aging effects requiring management were identified beyond those discussed in this section.

Table 3.5-1 Main Steam					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Piping Tubing Valves Steam traps	Pressure boundary	Carbon steel	Treated water	Loss of material Loss of mechanical closure integrity Cracking	ASME Section XI ISI -IWC Inspections Leakage Detection in Reactor Building Flow Accelerated Corrosion Prevention Wall-Thinning Inspection Secondary Chemistry Monitoring
			Gas-nitrogen	None	None
			External-ambient	Loss of material(1)	Maintenance Rule
Piping Valves Tubing	Pressure boundary	Stainless steel	Treated water	Loss of material Loss of mechanical closure integrity Cracking	ASME Section XI ISI -IWC Inspections Leakage Detection in Reactor Building Secondary Chemistry Monitoring Augmented ISI-special inspection of Q stainless steel piping
			Gas-nitrogen	None	None
			External-ambient	None	None

1) Aging effect prevented by referenced program/activity

Table 3.5-2 Main Feedwater					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Piping Tubing Valves	Pressure boundary	Carbon steel	Treated water	Loss of material Cracking	ASME Section XI ISI -IWC Inspections Leakage Detection in Reactor Building Augmented Inspections Flow Accelerated Corrosion Prevention Secondary Chemistry Monitoring
			Treated water[(2)]	Loss of material Loss of mechanical closure integrity	ASME Section XI ISI –IWC Inspections
			External-ambient	Loss of material[(1)]	Maintenance Rule

1) Aging effect prevented by referenced program/activity
2) Component/system leakage

Table 3.5-3 Emergency Feedwater

Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Pumps and discharge piping					
Piping Tubing Valves Pump casings	Pressure boundary	Carbon steel	Treated water	Loss of material	ASME Section XI ISI -IWC Inspections Wall Thinning Inspection Emergency Feedwater Pump Testing Secondary Chemistry Monitoring
			External-ambient	Loss of material(1)	Maintenance Rule
Orifice plates Valves	Pressure boundary	Stainless steel	Treated water	Cracking(1)	Secondary Chemistry Monitoring
			External-ambient	None	None
Steam supply and exhaust					
Piping Tubing Valves Steam traps Orifice plates Turbine casing	Pressure boundary	Carbon steel Chrome-moly	Treated water	Loss of material Loss of mechanical closure integrity	ASME Section XI –IWC & IWD (pressure tests) Wall Thinning Inspection Emergency Feedwater Pump Testing Secondary Chemistry Monitoring
			External-ambient	Loss of material(1)	Maintenance Rule
Orifice plates Valves Expansion joints	Pressure boundary	Stainless steel	Treated water	Cracking(1)	Secondary Chemistry Monitoring
			External-ambient	None	None
Tubing Valves	Pressure boundary	Brass, bronze, copper	Treated water	Loss of material(1)	Secondary Chemistry Monitoring
			External-ambient	None	None

Table 3.5-3 Emergency Feedwater					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Lubricating oil					
Piping Filter Pump casing	Pressure boundary	Carbon steel	Lube oil	Loss of material(1)	Oil Analysis
			External-ambient	Loss of material(1)	Maintenance Rule
Tubing	Pressure boundary	Stainless steel	Lube oil	Cracking(1)	Oil Analysis
			External-ambient	None	None
Filter	Pressure boundary	Aluminum	Lube oil	Loss of material(1)	Oil Analysis
			External-ambient	None	None
Valve	Pressure boundary	Cast iron	Lube oil	Loss of material(1)	Oil Analysis
			External-ambient	None	None
Heat exchanger (Turbine lube oil)	Pressure boundary (shell, head)	Carbon steel	Lube oil, treated water	Loss of material	Heat Exchanger Monitoring Emergency Feedwater Pump Testing Oil Analysis
			External-ambient	Loss of material(1)	Maintenance Rule
	Pressure boundary (tubes)	Copper	Treated water(2)	Loss of material(1)	Emergency Feedwater Pump Testing Secondary Chemistry Monitoring
			Lube oil(3)	Loss of material(1)	Oil Analysis
Cooling and seal water					
Piping Valves	Pressure boundary	Carbon steel	Treated water	Loss of material	Wall-Thinning Inspection Emergency Feedwater Pump Testing Secondary Chemistry Monitoring
			External-ambient	Loss of material(1)	Maintenance Rule

Table 3.5-3 Emergency Feedwater					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Piping Valves Orifices	Pressure boundary	Stainless steel	Treated water	Cracking[(1)]	Secondary Chemistry Monitoring
			External-ambient	None	None
Heat exchanger	Pressure boundary	Carbon steel	Treated water	Loss of material	Heat Exchanger Monitoring Emergency Feedwater Pump Testing Secondary Chemistry Monitoring
	Heat transfer	Carbon steel	Treated water[(2)]	Fouling	Emergency Feedwater Pump Testing Secondary Chemistry Monitoring
			Lube oil[(3)]	Fouling	Oil Analysis

1) Aging effect prevented by referenced program/activity
2) Inside of tubes
3) Outside of tubes

Table 3.5-4 Condensate Storage and Transfer					
Component Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Piping Tubing Valves Appurtenances	Pressure boundary	Carbon steel	Treated water	Loss of material	ASME Section XI ISI-IWD (pressure tests) Condensate Storage Tank Level Monitoring Secondary Chemistry Monitoring Wall-Thinning Inspection
			External-ambient	Loss of material(1)	Maintenance Rule
Piping Tubing Valves Appurtenances	Pressure boundary	Stainless steel	Treated water	Cracking(1)	Secondary Chemistry Monitoring
			External-ambient	None	None
Tank Tank heater	Pressure boundary	Stainless steel	Treated water	Cracking(1)	Secondary Chemistry Monitoring
			Gas-air	None	None
			External-ambient	None	None
		Carbon steel	External-ambient	Loss of material Loss of mechanical closure integrity	ASME Section XI ISI–IWD (pressure tests) Condensate Storage Tank Level Monitoring Maintenance Rule

1) Aging effect prevented by referenced program/activity

3.6 STRUCTURES AND STRUCTURAL COMPONENTS

Structures and their structural components and commodities that are within the scope of license renewal and subject to aging management reviews, are discussed in Section 2.4 and summarized in Table 3.6-2 through Table 3.6-8. Intended functions and general notes are provided in Table 3.6-1.

Determination of the aging effects applicable to structures and their structural components and commodities begins with identification of the aging effects defined in industry literature. From this set of aging effects, the component and commodity materials and operating environments define the aging effects for each structural component or commodity that is subject to an aging management review. These aging effects are then validated by a review of industry and ANO-1 operating experience to provide reasonable assurance that the full set of aging effects are established for the aging management review.

Structural components and commodities have been grouped into the following construction materials.

- Steel
- Concrete
- Prestressed concrete
- Threaded fasteners
- Fire barriers
- Earthen embankments
- Elastomers and Teflon

3.6.1 Steel

Steel components and commodities are exposed to various service environments depending on location. They are either protected from weather, exposed to weather, exposed to raw water, or exposed to borated water. High temperature, humidity, and radiation play a role in the degradation of the components and commodities located in these environments. Examples of environmental conditions include the following.

- The reactor building liner plate and other steel components are exposed to the internal environment of the reactor building
- Embedments encased in concrete are protected from the external environment and the highly alkaline environment of the concrete protects the steel from corrosion
- Steel inside the auxiliary building may be exposed to elevated temperatures and humidity. Steel outside of the auxiliary building is exposed seasonally to hot and cold temperatures and at times, elevated humidity
- Steel in the intake structure is exposed to raw water
- Steel in the auxiliary building spent fuel pool and the reactor building internals is exposed to borated water

The spent fuel pool steel gates and liner, and the spent fuel steel racks are addressed in Sections 2.3.3 and 3.4. The steel sluice gates are also addressed in these sections.

The review to identify the aging effects for steel, including sub-materials (welds and thermal shields associated with pipe supports) in air or fluid environments considers the following potential aging effects based on available industry literature.

- Loss of material (due to general corrosion, galvanic corrosion, crevice corrosion, pitting corrosion, erosion/erosion-corrosion, MIC and boric acid corrosion)
- Cracking (due to fatigue, stress corrosion, and intergranular attack)
- Change in material properties (due to radiation embrittlement and intermetallic embrittlement)

Other potential aging effects and aging mechanisms do not apply to ANO-1 steel components and commodities due to the absence of susceptible material and environmental conditions. Aging effects for steel associated with the various in-scope structures consider ANO-1 construction materials and environments, and are described in the following subsections. The types of steel components and commodities (i.e., made of carbon steel, stainless steel or galvanized steel) that are subject to an aging management review and the results for ANO-1 are summarized in Table 3.6-1 through Table 3.6-8.

There are no structural steel components or commodities subject to an aging management review made of low-alloy steel. Stainless steel components and commodities are either exposed to interior ambient conditions or borated water. Therefore, aging effects for stainless steel in other environments are not discussed in this section. Referring to Table 3.6-4, the new fuel racks are made of aluminum and have been grouped with steel components and commodities. However, no aging effects were identified for aluminum in an air environment as it is highly resistant to corrosion in atmospheric conditions.

3.6.1.1 Loss of Materials

Loss of material in steel may be caused by corrosion of the steel. This may be seen as material dissolution, corrosion product build-up, and pitting and it may be uniform or localized. General corrosion is the result of a chemical or electrochemical reaction between a material and an aggressive environment. Both oxygen and moisture must be present. Relative humidity, temperature, sulfur dioxide, and chloride concentrations are among important variables. Protective coatings help to prevent the onset of this aging effect. Exposed steel is normally coated for corrosion protection. Therefore, loss of material due to corrosion is not an aging effect as long as the coatings are maintained.

Coatings for the reactor building liner plate, attachments to the liner plate, reactor building penetrations, and reactor building hatches are identified in SAR Section 5.2.1.3.5 and SAR Table 5-2. For the reactor building liner plate, behind miscellaneous welded attachments, loss of material due to corrosion is an aging effect if the cavity formed between the attachment and the liner plate is not sealed to protect against moisture intrusion. For steel encased in concrete, loss of material due to corrosion is not an aging effect because the adjacent concrete provides an alkaline environment that is an effective corrosion inhibitor. An exception to this conclusion may be the reactor building liner plate below the floor if the expansion joint sealant is not maintained.

Due to high humidity and elevated temperatures of the exterior and interior environments of ANO-1 structures (i.e., intake structure, auxiliary building), a loss of material due to general corrosion is an aging effect for carbon steel components and commodities exposed to or protected from weather. Loss of material due to general corrosion is also an aging effect for galvanized steel components and commodities exposed to weather. However, it is not an aging effect for galvanized steel and stainless steel components and commodities protected from weather.

Similarly, since carbon steel is susceptible to corrosion in systems using raw water, a loss of material due to general corrosion and other types of corrosion, is an aging effect applicable to the intake structure carbon steel components and commodities submerged in or wetted by raw water. However, it is not an aging effect for galvanized steel in raw water.

Some stainless steel components within the reactor building are in a borated water environment. Chloride levels in stagnant and low flow areas affect the corrosion of such. However, chloride levels for borated water within the reactor building do not exceed the threshold limit. Therefore, loss of material due to pitting corrosion is not an aging effect requiring management for stainless steel in borated water.

An aging effect requiring management for the control rod drive service structure is loss of material of the lower control rod drive service structure skirt due to corrosion from exposure to boric acid.

An aging effect requiring management for the reactor vessel support skirt is a loss of material of the reactor vessel support skirt, flange, anchor bolts, and shear pins also due to boric acid corrosion.

3.6.1.2 Cracking

Cracking of steel may be caused by fatigue. Cracking due to fatigue has been identified as a time-limited aging analysis for the reactor building liner plate and has been evaluated for the period of extended operation. The results of this evaluation are presented in Section 4.6. For the reactor building liner plate and penetrations, this evaluation has determined that the original fatigue analysis remains valid for the period of extended operation. Furthermore, the design and operation of the steel components and commodities will not exceed the fatigue loading of $2x10^6$ cycles as specified by the AISC.

Dissimilar metal welds are used in certain reactor building penetrations. Where dissimilar metals are used, appropriate welding techniques have been utilized. Bellows are not used in ANO-1 reactor building penetrations. Cracking of dissimilar metal reactor building penetration welds due to fatigue and thermal stresses during plant operations is not likely because the welds are located in a non-aggressive environment.

Therefore, cracking due to fatigue is not an aging effect requiring management for ANO-1 steel components and commodities, except for welds associated with the control rod drive service structure and reactor vessel support skirt.

The reactor building internals contain small amounts of stainless steel in borated water which are utilized in locations that are not necessary for structural integrity. Therefore, cracking of stainless steel components and commodities due to stress corrosion and intergranular attack is not considered to be a significant issue to the structural integrity of the reactor building internals and thus, it is not an aging effect requiring management.

3.6.1.3 Change in Material Properties

A change in material properties driven by radiation embrittlement (for steel inside the primary shield wall), or intermetallic embrittlement (for galvanized steel exposed to temperatures at or above 400°F) is manifested in steel as a reduction or increase in yield strength, reduction in modulus of elasticity, reduction in ultimate tensile ductility, and an increase in ductile-to-brittle transition temperature.

Change in material properties due to radiation exposure is not an aging effect because steel at ANO-1 will not experience radiation exposure above the threshold necessary to cause embrittlement. For the reactor building, the primary shield wall and the concrete pedestal under the reactor vessel provide adequate shielding. The distance between the steel components and commodities and reactor core provides a further reduction in the radiation levels at the steel components and commodities. Other steel components and commodities are not subject to radiation exposure above the threshold necessary to cause embrittlement.

Change in material properties due to intermetallic embrittlement is not an aging effect since galvanized steel components and commodities do not support or protect high temperature piping.

3.6.1.4 Conclusion of Aging Effects for Steel

In conclusion, as a result of the review of industry information, NRC generic communications, and ANO-1 operating experience, no additional aging effects beyond those discussed in this section were identified.

The aging effect, loss of material, requires management for the following.

- the reactor building liner plate, hatches, and penetrations
- the control rod drive service structure and reactor vessel support skirt
- steel components and commodities

The aging effect, cracking, requires management for the following.

- the control rod drive service structure and reactor vessel support skirt

Management of Aging Effects

The above aging effects must be adequately managed so that the structural intended functions of steel components and commodities are maintained consistent with the current licensing basis for the period of extended operation.

These aging effects will be adequately managed by the following programs that are described in Appendix B.

- Maintenance Rule Program
- Reactor Building Leak Rate Testing Program
- ASME Section XI, Inservice Inspection Program (IWE and IWF)
- Inspection and Preventive Maintenance of the ANO-1 Polar Crane
- Service Water Chemical Control Program
- Battery Quarterly Surveillance
- Boric Acid Corrosion Prevention Program
- Fire Protection Program (Fire Barrier Inspections and Fire Hose Station Inspections)

3.6.2 Concrete

ANO-1 concrete components and commodities are designed in accordance with ACI 318-63 (Reference 3.6-1) and constructed in accordance with ACI 301 (Reference 3.6-2) using material designs conforming to ACI and ASTM standards which provide a good quality, dense, low permeability concrete.

The following discussion provides information on typical environments for concrete components and commodities at ANO-1.

Reactor Building

The reactor building and the reactor building internals concrete components and commodities are exposed to different service environments depending on their location. The top of the concrete floor of the reactor building is exposed to the internal environment of the reactor building. High temperature, humidity, and radiation play a role in the potential degradation of the components located within the reactor building. External surfaces of the reactor building dome and cylinder wall above grade are exposed to external atmospheric conditions. The cylinder wall above grade enclosed by adjacent buildings is exposed to a controlled environment that protects it from external weather. The reactor building concrete foundation slab and the portion of the external cylinder wall below grade are exposed to backfill and groundwater.

Auxiliary Building

Concrete within the auxiliary building may be exposed to elevated temperatures, humidity and radiation. Exterior concrete is exposed to hot and cold temperatures, and at times, elevated humidity. Above grade, exterior concrete is subjected to rainfall, snowfall, and wind. Concrete that is below grade is exposed to chemicals in the groundwater and may also be in direct contact with backfill materials.

Intake Structure

Concrete within the intake structure may be exposed to elevated temperatures and humidity. Exterior concrete is subjected to hot and cold temperatures and, at times, elevated humidity. Above grade, exterior concrete is subjected to rainfall, snowfall, and wind. Concrete that is below grade is exposed to chemicals in raw water (i.e., groundwater, Lake Dardanelle, emergency cooling pond) and may also be in direct contact with backfill materials.

Q-Condensate Storage Tank (Q-CST) Foundation

The foundation of the Q-CST is supported on 42” diameter drilled concrete piers embedded in bedrock. The drilled piers for this foundation are subject to the same, or similar environment as the auxiliary building concrete below grade.

Environments of the other structures, which have concrete components or commodities, are within the parameters of the above structures and, therefore, are not described.

The review to identify the aging effects requiring management for concrete, including sub-materials (non-shrink grout, epoxy grout, embedments, and reinforcement), considers the following potential aging effects based on available industry literature.

- Loss of material (due to abrasion and cavitation or elevated temperature)
- Cracking (due to elevated temperature or restraint against free contraction)
- Change in material properties (due to elevated temperature)

Other potential aging effects and aging mechanisms do not apply to ANO-1 concrete components and commodities due to the absence of susceptible material and environmental conditions. Aging effects for concrete associated with the various in-scope structures consider ANO-1 construction materials and environments and are described in the following subsections. The types of concrete components and commodities (i.e., reinforced concrete, masonry) that are subject to an aging management review and the results for ANO-1 are summarized in Table 3.6-1 through Table 3.6-8

3.6.2.1 Loss of Material

Loss of material is manifested in concrete as scaling, spalling, pitting, and erosion. It may be uniform or localized.

Loss of material due to abrasion and cavitation is limited to concrete that is continuously exposed to running water. The intake structure exterior concrete wall, at the lake level, is an example of concrete exposed to running water (i.e., wave action). Therefore, a loss of material due to abrasion is an aging effect requiring management for the intake structure. However, the highest average water velocity in the intake structure does not exceed the threshold for cavitation damage. Therefore, a loss of material due to cavitation is not an aging effect.

Loss of material due to elevated temperatures may cause concrete surface scaling and cracking. ACI 318-63 provided a maximum temperature limit of 150ºF and the ASME Code, Section III, Division 2, Subsection CC, indicates that aging is not significant for concrete temperatures less than 150ºF. ACI-349 (Reference 3.6-3) allows local area temperatures to reach 200ºF prior to requiring special provisions. Since plant controls maintain temperatures of concrete components and commodities below the established threshold temperature limits, a loss of material due to elevated temperature is not an aging effect.

3.6.2.2 Cracking

Cracking is manifested in concrete as a separation of the concrete into two or more parts. Cracking may occur in concrete as general cracking, map cracking, hairline cracking, pitting, and erosion. The aging mechanism that can lead to cracking of concrete components and commodities is elevated temperature.

As stated above, concrete components and commodities are not exposed to temperatures that exceed the established thresholds for degradation due to elevated temperature as identified in ACI 318-63. Therefore, cracking due to elevated temperature is not an aging effect for concrete components and commodities at ANO-1. However, based on past findings, cracking due to restraint against free contraction is considered an aging effect for masonry block walls. Since there are no exterior block walls associated with in-scope structures at ANO-1, the freezing of water in masonry wall joints does not apply.

3.6.2.3 Change in Material Properties

A change in material properties is manifested in concrete as increased permeability, increased porosity, reduction in pH, reduction in tensile strength, reduction in compressive strength, reduction in modulus of elasticity, and reduction in bond strength. The aging mechanism that could lead to change in material properties of concrete components and commodities is elevated temperature.

As stated above, concrete components and commodities are not exposed to temperatures that exceed the thresholds for degradation identified in ACI 318-63. Change in material properties due to elevated temperature is not an aging effect because concrete is not exposed to temperatures above the threshold limits.

3.6.2.4 Conclusion of Aging Effects for Concrete

In conclusion, as a result of the review of industry information, NRC generic communications, and ANO-1 operating experience, no additional aging effects beyond those discussed in this section have been identified. This review concludes that the following aging effects for concrete require management for the period of extended operation.

- a loss of material for the intake structure exterior concrete wall at the normal lake level
- cracking for masonry block walls

No aging effects were found to be applicable to the other in-scope concrete components and commodities at ANO-1.

Management of Aging Effects

The programs credited for the management of the identified aging effects for concrete components and commodities are the following.

- Maintenance Rule Program
- Fire Protection Program (Fire Barrier Inspections)

3.6.3 Prestressed Concrete

Reactor building structural components within the scope of license renewal that require aging management reviews and their intended functions are discussed in Section 2.4 and summarized in Table 3.6-2. The ANO-1 reactor building design incorporates a post-tensioning system that provides prestress forces to counteract forces resulting from design loads. Prestressed concrete components for the ANO-1 reactor building considered within the scope of license renewal and subject to aging management review are the following.

- Tendon wires
- Tendon anchorage
- Dome
- Cylinder walls

The prestressed concrete design places the cylinder and dome concrete in compression for normal loading conditions over the current and extended periods of operation. The compression minimizes the number and width of cracks induced by shrinkage, temperature, or load. The prestressed concrete components are designed in accordance with ACI 318-63 and mixed in accordance with ACI 301 (Reference 3.6-2) using ingredients conforming to ACI and ASTM standards, which provide a good quality, dense, low permeability concrete.

The post-tensioning system components are not exposed to external atmospheric conditions with the exception of the tendon end caps. The anchorage system on the bottom of the vertical tendons is exposed to the moist environment of the tendon gallery. The tendon wires are encased in bulk fill grease. The tendon anchors are enclosed in sealed end caps.

The codes and standards used for the reactor building design and fabrication, including the applicable edition, are identified in the ANO-1 SAR Section 5.2. The design of the reactor building post-tensioning buttresses and anchorage zone complies with ACI 318-63 (Reference 3.6-1). The review to identify the aging effects for prestressed concrete components considers the following potential aging effects based on industry literature.

- Loss of material (due to general corrosion)
- Cracking (due to fatigue)
- Change in material properties (due to elevated temperature or radiation embrittlement)
- Loss of prestress (due to material strain)

Other potential aging effects and aging mechanisms do not apply to ANO-1 prestressed concrete components due to the absence of susceptible material and environmental conditions. The aging effects for the reactor building post-tensioning system consider ANO-1 construction materials and environments and are discussed below. The results as they apply to the reactor building post-tensioning system are summarized in Table 3.6-2.

3.6.3.1 Loss of Material

The aging effect that could potentially result in loss of the ability of the post-tensioning system to impose compressive forces on the concrete reactor building structure is loss of material due to general corrosion. Corrosion must be considered for both the carbon steel tendon wires within the grease-filled conduits and for the carbon steel anchorage providing the tendon wire terminations. Stressed components of the post-tensioning system are normally well protected against corrosion. The tendon and the anchorage are enclosed within the ducts and end caps that are filled with bulk fill grease. Although the vertical tendon bottom anchorage is exposed to the moist environment of the tendon gallery, the end caps are coated for corrosion protection, the anchorage is encased in grease, and they are not in direct contact with water. Corrosion at the bottom anchorage of the vertical tendons has not been identified at ANO-1. However, potential grease leakage could occur and would most likely be at the tendon anchorage. Therefore, loss of material due to general corrosion is an aging effect requiring management for the tendon wires and anchorage.

3.6.3.2 Cracking

Cracking due to fatigue could lead to loss of the ability of the post-tensioning system to impose compressive forces on the reactor building structure. The post-tensioning system is not subjected to cyclical loads over the life of the plant. Therefore, cracking due to fatigue is not an aging effect.

However, minor concrete cracking has been observed on a few exposed concrete surfaces of the ANO-1 reactor building. Thus, cracking is an aging effect requiring management for the dome and cylinder wall. Engineering review of these locations has determined that the cracking will not challenge the intended functions of the reactor building prestressed concrete components under design basis loads.

3.6.3.3 Change in Material Properties

A change in material properties is manifested in concrete as increased permeability, increased porosity, reduction in pH, reduction in tensile strength, reduction in compressive strength, reduction in modulus of elasticity, and reduction in bond strength. Aging mechanisms that can lead to change in material properties include elevated temperature or radiation embrittlement.

Prestressed concrete components are not exposed to temperatures that exceed the thresholds for degradation identified in ACI 318-63. Therefore, change in material properties due to elevated temperature is not an aging effect. Change in material properties due to radiation exposure is also not an aging effect because the reactor building prestressed concrete components will not experience sufficient radiation to cause embrittlement.

However, based on ANO-1 operating experience, some leaching (i.e., change in material properties) has been observed on a few exposed concrete surfaces of the reactor building. Thus, change in material properties is an aging effect requiring management for the dome and cylinder wall. Engineering review of these locations has determined that the leaching will not challenge the intended functions of the reactor building prestressed concrete components under design basis loads.

3.6.3.4 Loss of Prestress

Loss of prestress due to material strain occurring under constant stress has been identified as a time-limited aging analysis and is evaluated in Section 4.5.

3.6.3.5 Conclusion of Aging Effects for Prestressed Concrete

As a result of the review of industry information and NRC generic communications, no additional aging effects requiring management beyond those discussed in this section have been identified. Loss of material due to general corrosion was determined to be an aging effect requiring management for the period of extended operation for the tendon wires and anchorage. Material loss at the tendon anchorage can ultimately lead to tendon failure if the corrosion progresses to the point of cracking of the tendon anchorage. Cracking and change in material properties are also aging effects requiring management.

<u>Management of Aging Effects</u>

The programs credited for the management of the identified and observed aging effects for prestressed concrete components are the following.

- ASME Section XI Inservice Inspection Program–IWL Inspections
- Maintenance Rule Program

3.6.4 Threaded Fasteners

As indicated in Table 3.6-1 through Table 3.6-8, in-scope threaded fasteners are made of the same materials as other steel components and commodities. The environmental conditions for threaded fasteners are the same as those for steel. Refer to the discussion in Section 3.6.1.

High strength fasteners used in seismic category 1 areas are ASTM-A325 (Reference 3.6-4) or ASTM-A490 (Reference 3.6-5). Other fasteners in seismic category 1 areas are ASTM-A307 (Reference 3.6-6). ASTM-A325 and ASTM-A307 fasteners are made of carbon steel. ASTM-A490 fasteners are made of alloy steel. However, threaded fastener materials, such as ASTM-A307, may be zinc-coated. Therefore, aging effects for galvanized steel threaded fasteners are also evaluated, in addition to those for stainless steel threaded fasteners.

Except for embedded bolts, fasteners associated with the spent fuel pool gates and liner, the spent fuel racks, and sluice gates are addressed in Sections 2.3.3 and 3.4. Embedded bolts are addressed in Section 3.6.2.

Based on available industry literature, the potential aging effects for threaded fasteners, including sub-materials (structural bolts, expansion anchors, and undercut anchors), are as follows.

- Loss of material (due to boric acid wastage, general corrosion and other forms of corrosion)
- Cracking of bolting material (due to stress corrosion and intergranular attack)
- Change in material properties (due to radiation embrittlement and intermetallic embrittlement)

Other potential aging effects and aging mechanisms do not apply to ANO-1 threaded fasteners due to the absence of susceptible material and environmental conditions. Aging effects for threaded fasteners associated with the various in-scope structures consider ANO-1 construction materials and environments and are described in the following subsections. The types of threaded fasteners subject to an aging management review and the results for ANO-1 are summarized in Table 3.6-1 through Table 3.6-8.

3.6.4.1 Loss of material

Loss of material due to boric acid wastage is generally found in closures of reactor coolant systems. Structural steel connections may be exposed to boric acid if the closures leak. Loss of material due to boric acid wastage is an aging effect for non-boron treated (i.e., those other than A325 low carbon martensitic) threaded fasteners in the vicinity of the spent fuel pool since it contains borated water.

Loss of material due to general corrosion is typically attributed to leaking joints. Loss of material due to general corrosion is an aging effect for carbon steel and low-alloy steel threaded fasteners protected from or exposed to weather, and in raw water. It is also an aging effect for galvanized steel threaded fasteners exposed to weather.

Stainless steel is susceptible to a loss of material due to general corrosion if chlorides are present. It is considered to be an aging effect for stainless steel threaded fasteners since ANO-1 raw water contains chlorides (i.e., applicable to the intake structure). Loss of material due to other forms of corrosion (i.e., crevice corrosion, pitting corrosion, MIC) is also an aging effect requiring management for stainless steel threaded fasteners in raw water. However, loss of material is not an aging effect requiring management for stainless steel threaded fasteners in borated water.

3.6.4.2 Cracking

Cracking of bolting material may be attributed to stress corrosion and intergranular attack. Reported failures have been limited to high strength or ultra-high strength bolting. However, since the specified yield strengths for ANO-1 high strength bolts (i.e., ASTM-A325 and ASTM-A490) do not exceed the threshold, cracking is not an aging effect.

Stress corrosion cracking is not an aging effect for stainless steel threaded fasteners protected from weather because the maximum yield strength for austenitic stainless steel is less than that for reported degradation. However, the chloride content of raw water affects the cracking of stainless steel. Therefore, stress corrosion cracking is an aging effect requiring management for stainless steel threaded fasteners in a raw water environment (i.e., those associated with the intake structure). Since materials susceptible to stress corrosion are also susceptible to intergranular attack, cracking due to intergranular attack is an aging effect requiring management for stainless steel threaded fasteners in raw water. However, cracking is not an aging effect for stainless steel threaded fasteners in borated water.

3.6.4.3 Change in Material Properties

For a general description of a change in material properties, refer to Section 3.6.1.

As discussed for steel components and commodities in Section 3.6.1.3, a change in material properties due to radiation embrittlement is not an aging effect for threaded fasteners at ANO-1. A change in material properties due to intermetallic embrittlement is also not an aging effect since galvanized steel fasteners are not used on high temperature piping systems.

3.6.4.4 Conclusion of Aging Effects For Threaded Fasteners

In order to validate the aging effects, industry literature and ANO-1 operating experience were also reviewed. These documents did not identify other age-related degradation issues.

Similarly, the results of ANO-1 structural-related walkdowns, performed in response to NRC Bulletins 79-02 and 79-14, indicated less than a one percent deficiency rate at support anchorage points. During a more recent walkdown, no deteriorated threaded fasteners were found. Additionally, since the ANO-1 design considers adequate preload of bolted connections and adequate installation, self-loosening by vibration is not an aging effect.

In conclusion, as a result of the review of industry information, NRC generic communications, and ANO-1 operating experience, no additional aging effects beyond those discussed in this section have been identified.

Management of Aging Effects

Considering that the materials, environments, and aging effects requiring management for threaded fasteners are similar to or the same as those for steel components and commodities, the same programs are credited for the management of their aging effects. Refer to Section 3.6.1.

3.6.5 Fire Barriers

Fire barrier commodities include fire wraps and fire stops associated with 10CFR50.48-required fire barrier walls and floors and are located within the reactor building, auxiliary building, diesel fuel vault, and some areas of the turbine building. Fire wraps may be spray applied, trowelled on, or wrapped onto members. Fire stops are materials used to close gaps in penetrations.

Fire barriers comprise a variety of materials and are exposed to internal ambient environments. Based on available information, the following are potential aging effects for fire barrier commodities.

- Loss of material (due to flaking and abrasion)
- Cracking, delamination, and separation (due to vibration, movement and shrinkage)
- Change in material properties (due to radiation)

Other potential aging effects and aging mechanisms do not apply to ANO-1 fire barriers due to the absence of susceptible material and environmental conditions. Aging effects for fire barrier commodities associated with the various in-scope structures consider ANO-1 construction materials and environments and are described in the following subsections. The types of fire barrier commodities subject to an aging management review and the results for ANO-1 are summarized in Table 3.6-8.

3.6.5.1 Loss of Material

Loss of material decreases the material's fire rating over time. Flaking may occur as fire wrap fibers become free due to force, gravity, airflow, and vibration. Since spray-applied fireproofing may become loose or airborne, loss of material due to flaking is an aging effect for fire wraps (excluding rigid types). Abrasion may occur if the fire barrier experiences continuous movement or interfaces with a moving item. Considering that there is vibrating equipment within the various structures, loss of material may also occur due to abrasion and is an aging effect for both fire wraps and fire stops.

3.6.5.2 Cracking, Delamination, and Separation

Cracking and delamination may occur in fire barrier commodities due to vibrations. As a result of vibrating equipment within the various structures, cracking and delamination are aging effects for ANO-1 fire wraps and fire stops. In addition, vibration may destroy the adhesion between a fire stop and an adjacent surface resulting in separation. Separation of fire stops, as well as cracking and delamination, may also be due to movement and shrinkage. Although movement is not expected to increase during the period of extended operation, ANO-1 fire stops may be exposed to differential movement. Therefore, cracking, delamination, and separation due to movement are aging effects requiring management for fire stops. Cracking, delamination, and separation due to shrinkage are also aging effects requiring management for sealant used at floor and wall piping penetrations, in contact with piping.

3.6.5.3 Change in Material Properties

Change in material properties due to gamma radiation may occur above a radiation exposure of 10^6 rads. Radiation exposure in the reactor building and in some rooms of the auxiliary building may exceed the established threshold limit. However, the threshold is not exceeded in other structures that have fire barriers. Therefore, a change in material properties due to radiation is an aging effect requiring management for ANO-1 fire barrier commodities within the reactor building and some rooms of the auxiliary building.

3.6.5.4 Conclusion of Aging Effects For Fire Barriers

In conclusion, as a result of the review of industry information, NRC generic communications, and ANO-1 operating experience, no additional aging effects beyond those discussed in this section have been observed. These reviews support that loss of material, cracking, delamination, separation, and change in material properties are aging effects requiring management for fire barrier commodities.

Management of Aging Effects

These aging effects must be managed during the period of extended operation so that the structural intended function of fire barriers is maintained. The Fire Protection Program-Fire Barrier Inspections is credited for managing these aging effects and is described in Appendix B.

3.6.6 Earthen Embankments

The emergency cooling pond and intake and discharge canals were evaluated for the aging effects of loss of material, loss of form, and loss of material properties. The design of the emergency cooling pond and intake and discharge canals minimizes the potential for occurrence of detrimental aging effects. Loss of form is the aging effect requiring management for the period of extended operation for the emergency cooling pond as reflected in Table 3.6-6. Frost action, wind erosion, and change in material properties due to desiccation are precluded based on the original design and the absence of an aggressive environment.

Loss of form due to sedimentation of the emergency cooling pond and intake and discharge canals is an aging effect. However, loss of form due to sedimentation effects for the intake and discharge canals is limited because of the engineered features for maintaining maximum flow. This design feature is controlled through power operations to ensure that sediment build-up does not affect safety systems. As the likelihood of buildup is minimized, an aging management program is not required for the intake and discharge canals. The aging management program being credited with managing the potential loss of form for the emergency cooling pond is the Annual Emergency Cooling Pond Sounding.

3.6.7 Elastomers and Teflon

Elastomers associated with ANO-1 structures and their components and commodities are made of rubber or neoprene, or have similar properties. Elastomers are exposed to various air conditions, fluids, and radiation. The use of Teflon (or polytetrafluoroethylene materials) is typically used at sliding surfaces; however, its use is limited at ANO-1. Based on application, Teflon is exposed to the interior environments of structures.

The following are potential aging effects for elastomers.

- Cracking (due to ultraviolet radiation [rubber only], thermal exposure and ionizing radiation)
- Change in material properties (due to ultraviolet radiation [rubber only], thermal exposure and ionizing radiation)

The potential aging effect for Teflon is a change in material properties due to radiation exposure.

Other potential aging effects and aging mechanisms do not apply to ANO-1 elastomeric and Teflon components and commodities due to the absence of susceptible material and environmental conditions. Aging effects for elastomers and Teflon consider ANO-1 construction materials and environments and are described in the following subsections. The types of elastomers (i.e., waterstops) and Teflon, which are subject to an aging management review, and the results for ANO-1 are summarized in Table 3.6-8.

3.6.7.1 Cracking

Ultraviolet radiation (i.e., sunlight) can cause cracking in rubber elastomers. Cracking, due to ultraviolet radiation is not an aging effect for rubber elastomers (i.e., rubber waterstops) since they are concealed in the exterior walls of ANO-1 structures and not exposed to sunlight.

Cracking due to thermal exposure may occur if the ambient temperature is 95ºF or above. Cracking due to thermal exposure is not an aging effect for elastomers associated with the exterior environment (i.e., waterstops at construction joints within exterior walls, most of which are located below grade) since the threshold temperature will not be exceeded.

Cracking due to ionizing radiation may alter the molecular structure of elastomers if exposed to 10^6 rads or greater. However, cracking due to ionizing radiation is not an aging effect for elastomers in exterior walls (waterstops) since the radiation threshold will not be exceeded.

3.6.7.2 Change in Material Properties

As for cracking, change in material properties due to ultraviolet radiation, thermal exposure, and ionizing radiation is not an aging effect for elastomers in exterior walls (waterstops).

Change in material properties for Teflon results in a decrease in tensile strength, ultimate elongation, and Young’s modulus. The flexure modulus increases. Exposure to a radiation dose of 10^4 rads may cause scission (i.e., breaking of chemical bonds). Since radiation exposures within ANO-1 structures containing Teflon materials (i.e., the reactor building and auxiliary building) may exceed the radiation threshold dose, this is an aging effect requiring management.

3.6.7.3 Conclusion on Aging Effects for Elastomers and Teflon

No additional aging effects beyond those discussed above were identified for elastomers or Teflon in the Entergy Operations' review of industry correspondence and ANO-1 operating experience. Therefore, although no aging effects requiring management were identified for elastomeric components, change in material properties is an aging effect requiring management for Teflon.

Management of Aging Effects

The management of aging effects for Teflon will be performed under the following programs.

- Maintenance Rule Program
- ASME Section XI Inservice Inspection Program-IWF

3.6.8 References for Section 3.6

3.6-1 ACI 318-63, "*Building Code Requirements for Reinforced Concrete*," American Concrete Institute.

3.6-2 ACI 301, "*Specifications for Structural Concrete for Buildings*," American Concrete Institute.

3.6-3 ACI 349, "*Code Requirements for Nuclear Safety-related Concrete Structures*," American Concrete Institute.

3.6-4 ASTM A325, "*Standard Specification for Structural Bolts, Steel, Heat Treated, 120/105 ksi Minimum Tensile Strength*," American Society for Testing and Materials.

3.6-5 ASTM A490, "*Standard Specification for Heat-Treated Steel Structural Bolts, 150 ksi Minimum Tensile Strength*," American Society for Testing and Materials.

3.6-6 ASTM A307, "Standard Specification for Carbon Steel Bolts and Studs, 60,000 PSI Tensile Strength," American Society for Testing and Materials.

Table 3.6-1 Intended Functions and General Notes for Tables 3.6-2 through 3.6-8

LIST OF INTENDED FUNCTIONS:

1) Provide essentially leak tight barriers to prevent uncontrolled release of radioactivity.
2) Provide structural support or functional support to safety-related equipment.
3) Provide shelter or protection to safety-related equipment (including radiation shielding).
4) Provide rated fire barriers to confine or retard a fire from spreading to, or from, adjacent areas.
5) Serve as missile (internal or external) barriers.
6) Provide structural or functional support to nonsafety-related equipment, failure of which could directly prevent satisfactory accomplishment of required safety-related functions.
7) Provide protective barriers for internal flood event.
8) Provide protective barriers for external flood event.
9) Provide for storage of spent fuel assemblies.
10) Provide a heat sink during design basis accidents or station blackout.

GENERAL NOTES:

* Denotes commodity

A) The spent fuel pool steel gates and liner, the spent fuel steel racks, and their associated fasteners (excluding embedded bolts) are addressed in Sections 2.3.3 and 3.4. Some of the threaded fasteners within the auxiliary building may be galvanized steel; however, no aging effects were identified for such.
B) Embedments include plates and bolts below the concrete surface. Reinforcement includes embedded bars, wires, and strands.
C) The steel sluice gates and their associated fasteners (excluding embedded bolts) are addressed in Sections 2.3.3 and 3.4.
D) For steel or threaded fasteners associated with the intake structure that are normally submerged in water, Maintenance Rule walkdowns will be coordinated with the inspection and cleaning of the service water and circulating water bays.
E) For steel or threaded fasteners normally submerged in water, the Service Water Chemical Control Program supplements the management of aging effects.
F) Aging effect applies to wall at the normal lake level (approx. El. 338’).
G) Includes mounting brackets for snubbers, but excludes snubbers since they are active commodities and not subject to an aging management review.

Table 3.6-1 Intended Functions and General Notes for Tables 3.6-2 through 3.6-8
H) Fire damper curtains and trap doors are active commodities and not subject to an aging management review. I) Although the turbine building is not a structure within the scope of license renewal, there are 10CFR50.48-required fire components and commodities within it that are in-scope and subject to aging management review. Environmental conditions for the turbine building are within the parameters of the auxiliary building. J) Associated with 10CFR50.48-required fire walls or floors. Penetration sealant used as a fire stop is fire rated. K) 10CFR50.48-required floors and walls provide a fire barrier function. L) For fireproof hatches, those associated with 10CFR50.48-required fire barrier floors are within the scope of license renewal and subject to aging management review. M) Bulk commodities support or protect various in-scope system components or equipment, and are common to two or more structures. N) For seismic category 1 areas, low-alloy threaded fasteners (ASTM-A490) may have been used in addition to carbon steel threaded fasteners in association with carbon steel components and commodities. Since the aging effects for the operating environments are the same for carbon steel and low-alloy steel fasteners, only carbon steel is noted as the fastener material. O) Banding by itself is not considered a fire barrier. However, in conjunction with 10CFR50.48-required fire wraps, it also fulfills intended function number 4. P) Includes reactor building internals. Q) Except piping and equipment integral attachments. R) Applies to the decay heat vaults. S) Applies to the intake structure. T) Refer to associated steel commodity grouping for applicable structure, environment, and program or activity. U) Applies to the reactor building and auxiliary building.

Table 3.6-2 Reactor Building					
Component/ Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
STEEL (Sub-materials: Welds)					
Liner plate Anchorage/embedment/ attachment Threaded fasteners[N] Personnel hatch Emergency personnel hatch Equipment hatch Mechanical penetrations Electrical penetrations Fuel transfer tube	1,2,6	Carbon steel	Protected from weather	Loss of material	Maintenance Rule ASME Section XI ISI-IWE Reactor Building Leak Rate Testing
CONCRETE (Sub-materials: Non-Shrink Grout, Epoxy Grout, Embedments[B], and Reinforcement[B])					
Dome Cylinder wall	2,3,4,5,6, 10	Prestressed concrete	Exposed to weather	Cracking Change in material properties	Maintenance Rule ASME Section XI ISI-IWL
Floor	2,3,4,5,6, 10	Reinforced concrete	Protected from weather	None	None
Foundation	2,3,4,5,6, 10	Reinforced concrete	Exposed to weather	None	None
POST TENSIONING SYSTEM					
Tendon wires Tendon anchorage	2	Carbon steel	Protected from weather	Loss of material	ASME Section XI ISI-IWL

Table 3.6-3 Reactor Building Internals					
Component/ Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
STEEL (Sub-materials: Welds)					
Anchorage/embedment/ attachment Threaded fasteners[N]	2,6	Carbon steel	Protected from weather	Loss of material	Maintenance Rule ASME Section XI ISI-IWF[Q]
Structural shapes	5,6	Carbon steel	Protected from weather	Loss of material	Maintenance Rule ASME Section XI ISI-IWF[Q] Boric Acid Corrosion Prevention
		Stainless steel	Protected from weather Borated water	None	None
OTSG support steel Pressurizer support steel	5,6	Carbon steel	Protected from weather	Loss of material	Maintenance Rule ASME Section XI ISI-IWF[Q] Boric Acid Corrosion Prevention
Main fuel handling bridge Jib cranes ANO-1 polar crane	6	Carbon steel	Protected from weather	Loss of material	Maintenance Rule Inspection and Preventive Maintenance of ANO-1 Polar Crane
Control rod drive service structure Reactor vessel support skirt	2,3	Carbon steel	Protected from weather	Loss of material Cracking	Boric Acid Corrosion Prevention ASME Section XI ISI-IWF[Q]
CONCRETE (Sub-materials: Non-shrink Grout, Epoxy Grout, Embedments[B] and Reinforcement[B])					
Primary and secondary shield walls	2 to 6	Reinforced concrete	Protected from weather	None	None
Reinforced concrete	2,3,4,5,6, 10		Protected from weather	None	None

Table 3.6-3 Reactor Building Internals					
Component/ Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Columns Other walls Hatches	2,6	Reinforced concrete	Protected from weather	None	None
Reactor missile shield	2,5	Reinforced concrete	Protected from weather	None	None
Fuel transfer canal	3	Reinforced concrete	Protected from weather	None	None

Table 3.6-4 Auxiliary Building					
Component/ Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
STEEL[A] (Sub-materials: Welds)					
Control room extension substructure	5	Carbon steel	Protected from weather	Loss of material	Maintenance Rule
Tornado missile shield wall, El. 354'	5	Carbon steel	Protected from weather	Loss of material	Maintenance Rule
Boron holdup tank vault beams (top of steel El. 353'-3 7/8")	2	Carbon steel	Protected from weather	Loss of material	Maintenance Rule
Spent fuel pool superstructure	2,6	Carbon steel	Protected from weather	Loss of material	Maintenance Rule
Fire doors (10CFR50.48-required)[I, J]	4	Carbon steel	Protected from weather	Loss of material	Fire Protection • Fire Barrier Inspections
Watertight/flood doors	7	Carbon steel	Protected from weather	Loss of material	Maintenance Rule
HELB doors	3	Carbon steel	Protected from weather	Loss of material	Maintenance Rule
Missile/impingement doors	3,5	Carbon steel	Protected from weather	Loss of material	Maintenance Rule
*New fuel racks	2	Aluminum	Protected from weather	None	None
*Main steam line support structure (El. 341' to El. 354')	2	Carbon steel	Protected from weather	Loss of material	Maintenance Rule
*Fuel storage bridge assembly (H3) framing	2,6	Carbon steel	Protected from weather	Loss of material	Maintenance Rule
*Battery racks (i.e., associated with Battery banks D06 and D07)	2	Carbon steel	Protected from weather	Loss of material	Battery Quarterly Surveillance
*Exterior louvers (i.e., EDG stack venting)	2	Carbon steel	Exposed to weather	Loss of material	Maintenance Rule
*Exhaust stack supports (i.e., EDGs and EFW turbine)	2	Carbon steel	Protected from weather Exposed to weather	Loss of material	Maintenance Rule

Table 3.6-4 Auxiliary Building					
Component/ Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
*Control room panel supports	2,6	Carbon steel	Protected from weather	Loss of material	Maintenance Rule
*Control room halon system supports and bottle racks	2,6	Carbon steel	Protected from weather	Loss of material	Maintenance Rule
THREADED FASTENERS[A,N] (Sub-materials: Structural Bolts, Expansion Anchors, and Undercut Anchors)					
Threaded fasteners for: Control room Tornado missile shield wall Boron holdup tank vault beams Spent fuel pool superstructure Watertight/flood doors HELB doors Missile/impingement doors	2,6	Carbon steel	Protected from weather	Loss of material	Maintenance Rule
Threaded fasteners for: Fire doors (10CFR50.48-required)	2,6	Carbon steel	Protected from weather	Loss of material	Fire Protection • Fire Barrier Inspections
* Threaded fasteners for: New fuel racks	2,6	Aluminum	Protected from weather	None	None
* Threaded fasteners for: Main steam line support structure Fuel storage bridge assembly (H3) framing Control room panel supports Control room halon system supports and bottle racks	2,6	Carbon steel	Protected from weather	Loss of material	Maintenance Rule

Table 3.6-4 Auxiliary Building					
Component/ Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
* Threaded fasteners for: Battery racks	2,6	Carbon steel	Protected from weather	Loss of material	Battery Quarterly Surveillance
* Threaded fasteners for: Exterior louvers (i.e., EDG stack venting)	2,6	Carbon steel	Exposed to weather	Loss of material	Maintenance Rule
* Threaded fasteners for: Exhaust stack supports (i.e., EDGs and EFW turbine)	2,6	Carbon steel	Protected from weather Exposed to weather	Loss of material	Maintenance Rule
CONCRETE (Sub-materials: Non-shrink Grout, Epoxy Grout, Embedments[B], and Reinforcement[B])					
PASS building sub-structure[K]	4,6,7	Reinforced concrete	Protected from weather	None	None
Building foundation mat (El. 317', El. 335')	2,6,8	Reinforced concrete	Exposed to weather	None	None
Floor slabs[I, K] (El. 335', El. 354', El. 372', El. 386', El. 404')	2,4,6	Reinforced concrete	Protected from weather	None	None
Exterior walls, below grade[K] (El. 317' to approx. El. 354')	2,4,6,8	Reinforced concrete	Exposed to weather	None	None
Exterior walls, above grade[K] (approx. El. 354' and above)	2,4,5,6	Reinforced concrete	Exposed to weather	None	None
Columns and beams (all floors)	2,6	Reinforced concrete	Protected from weather	None	None
Roof slabs	3	Reinforced concrete	Exposed to weather	None	None
Interior walls[I, K] (load bearing)	1[R],2,4,5,6,7	Reinforced concrete	Protected from weather	None	None
	2,4,5,6,7	Masonry blockwalls	Protected from weather	Cracking	Maintenance Rule Fire Protection: • Fire Barrier Inspections

Table 3.6-4 Auxiliary Building					
Component/ Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Boron holdup tank vault (slabs at El. 327’ and El. 355’-4” and vault room walls)	2	Reinforced concrete	Protected from weather	None	None
Main steam line tunnel (El. 341’ to El. 354’)	3	Reinforced concrete	Protected from weather	None	None
Spent fuel pool bottom slab and walls	9	Reinforced concrete	Protected from weather	None	None
Small pipe chase at approx. El. 341’	2	Reinforced Concrete	Protected from weather	None	None
Sump at El. 317’	2	Reinforced concrete	Protected from weather	None	None

Table 3.6-5 Intake Structure					
Component/ Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
STEEL[C] (Sub-materials: Welds)					
Beams in service water bays (top of steel El. 351'-7 ½")	2,6	Carbon steel	Protected from weather	Loss of material	Maintenance Rule
Louvered doors	2	Carbon steel	Protected from weather	Loss of material	Maintenance Rule
*Supports for roof hatch Nos. 75, 76, and 77	2,5,6	Carbon steel	Protected from weather Exposed to weather	Loss of material	Maintenance Rule
*Submerged pump shaft supports	2,6	Carbon steel	Raw water	Loss of material	Maintenance Rule [D] Service Water Chemical Control[E]
*Supports for fire pump diesel storage tank	2	Carbon steel	Protected from weather	Loss of material	Maintenance Rule
THREADED FASTENERS[C,N] (Sub-materials: Structural Bolts, Expansion Anchors, and Undercut Anchors)					
Threaded fasteners for steel beams in service water bays	2, 6	Carbon steel	Protected from weather	Loss of material	Maintenance Rule
Threaded fasteners for louvered doors	2,6	Carbon steel	Protected from weather	Loss of material	Maintenance Rule
* Threaded fasteners for above indicated roof hatch supports	2, 6	Carbon steel	Protected from weather Exposed to weather	Loss of material	Maintenance Rule
* Threaded fasteners for submerged pump shaft supports	2, 6	Carbon steel	Raw water	Loss of material	Maintenance Rule [D] Service Water Chemical Monitoring[E]
* Threaded fasteners for fire pump diesel storage tank supports	2, 6	Carbon steel	Protected from weather	Loss of material	Maintenance Rule

Table 3.6-5 Intake Structure					
Component/ Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
CONCRETE (Sub-materials: Non-shrink Grout, Epoxy Grout, Embedments[B], and Reinforcement[B])					
Building foundation (El. 322'-6")	2,6,8	Reinforced concrete	Exposed to weather	None	None
Floor slabs (El. 354' category 1 portion & El. 366')	2,6	Reinforced concrete	Protected from weather	None	None
Exterior walls, below grade (El. 322'-6" to approx. El. 354')	2,6,8	Reinforced concrete	Exposed to weather	None	None
Exterior walls, above grade (El. 354' to El. 378')	2,3,5,6	Reinforced concrete	Exposed to weather	Loss of material[F]	Maintenance Rule
Interior walls (El. 322'-6" to El. 378')	2,6	Reinforced concrete	Protected from weather	None	None
Columns and beams (El. 354' to El. 378')	2,6	Reinforced concrete	Protected from weather	None	None
Roof slab (El. 378')	3	Reinforced concrete	Exposed to weather	None	None
H&V equipment penthouse walls and roof slab (El. 378' to El. 387')	2,6	Reinforced concrete	Protected from weather Exposed to weather	None	None

Table 3.6-6 Earthen Embankments					
Component / Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Emergency cooling pond	10	Natural soils	Raw water	Loss of form	Annual Emergency Cooling Pond Sounding
Intake canal	10	Natural soils	Raw water	None	None
Discharge canal	10	Natural soils	Raw water	None	None

Table 3.6-7 Aboveground/Underground Yard Structures						
Component / Commodity Grouping	Structure	Intended Function	Material	Environment	Aging Effect	Program/Activity
STEEL (Sub-materials: Welds)						
Manhole covers including threaded fasteners[N]	Category 1 electrical manholes	2,5,6	Carbon steel	Exposed to weather	Loss of material	Maintenance Rule
CONCRETE (Submaterials: Non-shrink Grout, Epoxy Grout, Embedments[B] and Reinforcement[B])						
Slab/drilled piers	Q-Condensate storage tank foundation valve & pipe trench)	2,3,5,6	Reinforced concrete	Exposed to weather	None	None
Slab	Fuel oil tank foundation	6	Reinforced concrete	Exposed to weather	None	None
Walls Floor slab Columns	Emergency diesel fuel oil storage tank vault	2 to 7	Reinforced concrete	Protected from weather	None	None
Slab	BWST foundation	2,3	Reinforced concrete	Exposed to weather	None	None
Slab	AAC diesel generator building foundation	6	Reinforced concrete	Protected from weather	None	None
Manhole covers	Category 1 electrical manholes	2,5,6	Reinforced concrete	Exposed to weather	None	None
Walls Slab	Category 1 electrical manholes	2,4,6	Reinforced concrete	Protected from weather	None	None

Table 3.6-8 Bulk Commodities						
Component/ Commodity Grouping	Structure	Intended Function	Material	Environment	Aging Effect	Program/Activity
STEEL (Submaterials: Thermashield [associated with pipe supports] and Welds)						
*Piping and tubing supports[G]	Reactor building[P] Auxiliary building Intake structure Diesel fuel vault Pipe trenches	2,6	Carbon steel	Protected from weather Raw water[S]	Loss of material	Maintenance Rule [D] ASME Section XI ISI-IWF[Q] Service Water Chemical Control[E,S]
*Pipe whip restraints	Reactor building [P] Auxiliary building	2,6	Carbon steel	Protected from weather	Loss of material	Maintenance Rule ASME Section XI ISI-IWF[Q]
* Motor operated valve supports	Reactor building [P] Auxiliary building Intake structure Diesel fuel vault	2,6	Carbon steel	Protected from weather	Loss of material	Maintenance Rule ASME Section XI ISI-IWF[Q]

Table 3.6-8 Bulk Commodities						
Component/ Commodity Grouping	Structure	Intended Function	Material	Environment	Aging Effect	Program/Activity
* Hatch frames/covers[L] (i.e., associated with HELB, watertight/flood, missile, impingement, and/or fireproof hatches)	Auxiliary building Intake structure	3,4,5, and/or 7	Galvanized steel	Protected from weather Raw water[S]	None	None
	Auxiliary building	3,4,5, and/or 7	Carbon steel	Protected from weather	Loss of material	Maintenance Rule Fire Protection • Fire Barrier Inspections
	Intake structure	3,4,5, and/or 7	Carbon steel	Protected from weather Exposed to weather Raw water	Loss of material	Maintenance Rule [D] Service Water Chemical Control[E]
	Q-CST foundation (valve pit)	3,4,5, and/or 7	Carbon steel Galvanized steel	Exposed to weather	Loss of material	Maintenance Rule
* Conduit supports	Reactor building[P] Auxiliary building Intake structure Diesel fuel vault Pipe trenches	2,6	Carbon steel	Protected from weather Raw water[S]	Loss of material	Maintenance Rule [D] Service Water Chemical Control[E,S]

Table 3.6-8 Bulk Commodities						
Component/ Commodity Grouping	Structure	Intended Function	Material	Environment	Aging Effect	Program/Activity
* Cable trays and supports	Reactor building[P] Auxiliary building Intake structure	2,3,6	Galvanized steel	Protected from weather Raw water[S]	None	None
	Reactor building[P] Auxiliary building Intake structure	2,3,6	Carbon steel	Protected from weather Raw water[S]	Loss of material	Maintenance Rule Service Water Chemical Control[E,S]
* H&V duct supports	Reactor building[P] Auxiliary building	2,6	Carbon steel	Protected from weather	Loss of material	Maintenance Rule
* Cabinets, electrical panels and supports	Reactor building[P] Auxiliary building Intake structure	2,3,6	Carbon steel	Protected from weather Raw water[S]	Loss of material	Maintenance Rule [D] Service Water Chemical Control[E,S]
* Equipment supports	Reactor building [P] Auxiliary building Intake structure	2,6	Carbon steel	Protected from weather Raw water[S]	Loss of material	Maintenance Rule [D] ASME Section XI ISI-IWF[Q] Service Water Chemical Control[E,S]

Table 3.6-8 Bulk Commodities						
Component/ Commodity Grouping	Structure	Intended Function	Material	Environment	Aging Effect	Program/Activity
* Hazard barrier curbs	Auxiliary building Intake structure	7	Carbon steel	Protected from weather	Loss of material	Maintenance Rule
* Banding for 10CFR50.48-required fire wraps[O]	Reactor building [P] Auxiliary building Turbine building [I]	2,6	Carbon steel	Protected from weather	Loss of material	Fire Protection: • Fire Barrier Inspections
	Reactor building[P] Auxiliary building Turbine building [I]	2,6	Galvanized steel	Protected from weather	None	None
* Fire damper mountings[H, J] (10CFR50.48-required)	Auxiliary building Intake structure Diesel fuel vault Turbine building[I]	4	Galvanized steel	Protected from weather	None	None

Table 3.6-8 Bulk Commodities						
Component/ Commodity Grouping	Structure	Intended Function	Material	Environment	Aging Effect	Program/Activity
* Fire hose reels (10CFR50.48-required)	Reactor building[P] Auxiliary building Intake structure Turbine building[I]	2,6	Carbon steel	Protected from weather	Loss of material	Fire Protection : • Fire Hose Station Inspections
	Reactor building[P] Auxiliary building Intake structure Turbine building[I]	2,6	Galvanized steel	Protected from weather	None	None
THREADED FASTENERS [N](Sub-materials: Structural Bolts, Expansion Anchors, and Undercut Anchors)						
*Threaded fasteners[T] on: Piping and tubing supports Pipe whip restraints MOV supports Conduit supports H&V duct supports Cabinets, electrical panels and supports Equipment supports Hazard barrier curbs	Reactor building[P] Auxiliary building Intake structure Diesel fuel vault Pipe trenches	2,6	Carbon steel	Protected from weather Raw water[S]	Loss of material	Maintenance Rule[D] Service Water Chemical Control[E,S] ASME Section XI ISI-IWF[Q]

Table 3.6-8 Bulk Commodities						
Component/ Commodity Grouping	Structure	Intended Function	Material	Environment	Aging Effect	Program/Activity
*Threaded fasteners[T] for: hatch frames/covers	Auxiliary building Intake structure Q-CST foundation (valve pit)	2,6	Carbon steel	Protected from weather Exposed to weather Raw water[S]	Loss of material	Maintenance Rule[D] Fire Protection : • Fire Barrier Inspections Service Water Chemical Control[E,S]
			Galvanized steel	Protected from weather Raw water[S]	None	None
				Exposed to weather	Loss of material	Maintenance Rule
* Pipe lugs	Reactor building[P] Auxiliary building Intake structure Diesel fuel vault Pipe trenches	2,6	Stainless steel	Protected from weather	None	None
	Intake structure	2,6	Stainless steel	Raw water	Loss of material Cracking	Maintenance Rule[D] ASME Section XI ISI-IWF[Q] Service Water Chemical Control[E]

Table 3.6-8 Bulk Commodities						
Component/ Commodity Grouping	Structure	Intended Function	Material	Environment	Aging Effect	Program/Activity
* Tubing clips	Reactor building[P] Auxiliary building Intake structure Diesel fuel vault Pipe trenches	2,6	Stainless steel	Protected from weather	None	None
	Intake structure	2,6	Stainless steel	Raw water	Loss of material Cracking	Maintenance Rule [D] ASME Section XI ISI-IWF[Q] Service Water Chemical Control[E]
* Threaded fasteners for: Cable trays and supports	Reactor building[P] Auxiliary building Intake structure	2,6	Carbon steel	Protected from weather Raw water[S]	Loss of material	Maintenance Rule [D] Service Water Chemical Control[E,S]
			Galvanized steel	Protected from weather Raw water[S]	None	None
* Threaded fasteners for: Fire damper mountings	Auxiliary building Intake structure Diesel fuel vault Turbine building[I]	2,6	Galvanized steel	Protected from weather	None	None

Table 3.6-8 Bulk Commodities						
Component/ Commodity Grouping	Structure	Intended Function	Material	Environment	Aging Effect	Program/Activity
* Threaded fasteners for: Fire hose reels	Reactor building[P] Auxiliary building Intake structure Turbine building[I]	2,6	Carbon steel	Protected from weather	Loss of material	Fire Protection: • Fire Hose Station Inspections
			Galvanized steel	Protected from weather	None	None
CONCRETE (Sub-materials: Non-shrink Grout, Epoxy Grout, Embedments[B], and Reinforcement[B]						
* Equipment pads and foundations	Reactor building[P] Auxiliary building Intake structure	2,6	Reinforced concrete	Protected from weather	None	None
* Hatch covers/plugs[L] (i.e., associated with HELB, watertight/flood, missile, impingement, and/or fireproof hatches)	Auxiliary building Intake structure Diesel fuel vault BWST foundation	3,4,5 and/or 7	Reinforced concrete	Protected from weather Exposed to weather[S]	None	None

Table 3.6-8 Bulk Commodities						
Component/ Commodity Grouping	Structure	Intended Function	Material	Environment	Aging Effect	Program/Activity
FIRE BARRIERS						
* 10CFR50.48-required Fire wraps[J] (spray-applied, trowelled-on, or wrapped onto members)	Reactor building[P] Auxiliary building Turbine building[I]	4	3M-wrap Monokote coating Pyrocrete coating Metal lathe and plaster Ceramic fiber blanket	Protected from weather	Loss of material Cracking/ delamination Change in material properties[U]	Fire Protection: • Fire Barrier Inspections
* 10CFR50.48-required Fire stops[J] (penetration sealant)	Reactor building[P] Auxiliary building Diesel fuel vault Turbine building[I]	4	Silicone foam Grout Boot	Protected from weather	Loss of material Cracking/ delamination/ separation Change in Material properties[U]	Fire Protection : • Fire Barrier Inspections

Table 3.6-8 Bulk Commodities						
Component/ Commodity Grouping	Structure	Intended Function	Material	Environment	Aging Effect	Program/Activity
ELASTOMERS						
Waterstops at construction joints of exterior concrete walls	Reactor building Auxiliary building Diesel fuel vault Q-CST foundation (valve pit & pipe trench)	8	Rubber Polyvinyl-chloride Neoprene	Protected from weather	None	None
TEFLON						
* Piping support restraints	Reactor building[P] Auxiliary building	2,6	Polytetra-fluoroethylene	Protected from weather	Change in material properties	Maintenance Rule ASME Section XI ISI-IWF[Q]
* Equipment pad/ foundation plates	Reactor building[P] Auxiliary building	2,6	Polytetra-fluoroethylene	Protected from weather	Change in material properties	Maintenance Rule ASME Section XI ISI-IWF[Q]

3.7 ELECTRICAL AND INSTRUMENTATION AND CONTROLS

This section discusses the passive electrical component types that were initially scoped and met the screening criteria of Section 2.5. The potential aging effects are evaluated to determine their applicability to the components subject to aging management review. Table 3.7-1 lists the aging effects for passive electrical components that require aging management.

3.7.1 Aging Management Review Methodology

This review follows the "plant spaces" approach and methodology as described in Sandia Report SAND96-0344, "Aging Management Guideline for Commercial Nuclear Power Plants – Electrical Cable and Terminations" (Reference 3.7-1). As described in this report, Sandia evaluated aging mechanisms and consolidated historical maintenance and industry operating information into one source. The report contains the following primary conclusions.

- Cables and terminations that are included in the scope of aging management review are highly reliable and can be expected to perform their safety function during the initial license term and for license renewal.
- Based on the analysis of failure data, the number of cable and termination failures that have occurred throughout the industry is extremely low in proportion to the general population.
- The stressors and aging mechanisms affecting cables and terminations are generally well understood and characterized.
- Cable aging can be evaluated on an on-going basis, using theoretical techniques, measurement of physical properties, and periodic inspection.

The Sandia report concluded that detailed component level review was only required for those passive electrical components which are located near heat or radiation sources, are subject to continuous or near continuous loading at a significant percentage of the cable ampacity limits, are exposed to wetting (medium voltage only) or adverse chemical environments, or are subject to repeated or damaging mechanical stress. Low voltage instrument circuits that are sensitive to small variations in impedance were also determined to be potentially affected by oxidation of connector or termination contacts. The Sandia report also recommended that special consideration should be given to the possibility of installation damage to cables and electrical equipment.

Heat stress can cause accelerated aging for some passive electrical components and is primarily a concern for cable insulation. The Sandia report recommended the identification of a service-limiting temperature threshold that does not exceed the 60-year service-limiting temperature applicable to the materials of concern. For ANO-1, temperature thresholds of 105^0F outside of the reactor building and 120^0F inside the reactor building have been chosen in the ANO-1 EQ program for the ambient temperatures assumed in aging analyses. For the purposes of license renewal evaluations, ANO-1 has also chosen the service-limiting temperature thresholds of 105^0F outside of the reactor building and 120^0F inside the reactor building to be consistent with the ambient temperature assumptions in the ANO-1 EQ program.

The EQ system component evaluation worksheets represent a broad database of material evaluations and provide a qualified source for verification of material properties. Therefore, as long as the temperature of an area remains below these thresholds, further evaluation of the heat stress and thermal aging of the passive electrical components in that area is not required to support license renewal. As noted in the Sandia report, the service-limiting temperature threshold is not meant to be an absolute maximum since small, short term temperature excursions above the service-limiting temperature (such as during particularly hot summer days) will not significantly affect material aging.

Cables subject to aging management review that are not in the ANO EQ program are similar to those that are covered by the EQ program. The majority of EQ cables at ANO-1 have been tested and qualified to at least $2.0\text{x}10^8$ rads. For the license renewal evaluation, a total integrated dose of $1.0\text{x}10^8$ rads will be established as the radiation dose threshold level of concern for cables and cable terminations that are not in the EQ program scope.

Cables that are operated continuously near their current carrying capacity will operate closer to their temperature limits due to ohmic heating. This is an important concern for power cables only and does not affect instrumentation or low voltage control cables due to their low operating currents. For ANO-1, cables are sized with conservative margins for their current carrying capacities, insulation properties, and mechanical construction. The ANO-1 base capacity rating of cable is normally as conservative or more conservative than that established in published IPCEA standards. Only a small portion of the major “Q” equipment is continuously in service during normal plant operation. Cables that are operated continuously at a significant percentage of their current carrying capacity during normal plant operation have been identified and are evaluated in Section 3.7.6

The majority of the cables at ANO-1 are located in dry locations inside plant structures and are not exposed to an adverse chemical environment. ANO-1 is located on a fresh water inland lake and is not exposed to saltwater. The cables that are potentially exposed to wet conditions or chemical environments have been identified and are evaluated in Section 3.7.6.

The passive electrical components that are exposed to repeated mechanical stress are those cables and connectors that must be periodically moved or disconnected from equipment for plant outages or surveillance testing. These have been identified and are evaluated in Sections 3.7.4, 3.7.5, and 3.7.6.

Low-voltage instrument cables and connectors that operate at low currents or are otherwise sensitive to small variations in impedance have been identified and are evaluated in Sections 3.7.4 and 3.7.6.

3.7.2 Review of ANO-1 Plant Spaces

The main structures that house passive electrical equipment within the scope of license renewal are the reactor building, the auxiliary building, the turbine building, and the intake structure. There are also a limited number of passive electrical components in specialized structures such as the fuel oil storage vault and duct banks and manholes. The following discusses the ambient conditions in each of these structures.

3.7.2.1 ANO-1 Reactor Building

The "Q" equipment housed in the reactor building that is required to function under accident conditions includes the reactor, the reactor coolant system, the reactor building coolers, and hydrogen recombiners. A wide variety of instrumentation systems have sensors inside the reactor building which are required to be available under accident conditions. During normal plant operation the reactor building is cooled by recirculating air through coolers located inside the reactor building.

In response to elevated reactor building temperatures identified at ANO-1 in 1987, a thermal aging assessment was completed for the electrical equipment in the reactor building. This evaluation documented the impact of increased reactor building temperatures at various elevations and locations. The reactor building temperature has since decreased with the installation of additional insulation on the RCS and a new reactor building cooling unit. The corrective actions taken have returned reactor building general area temperatures back to 120^0F, or less.

Since ambient temperature in portions of the reactor building are above the 120^0F threshold value chosen for evaluation of thermal aging, passive electrical components in those areas of the reactor building have been identified and are evaluated in Sections 3.7.4 through 3.7.6.

3.7.2.2 ANO-1 Auxiliary Building

The auxiliary building houses nearly all of the major "Q" electrical switchgear and electrical "Q" components (station batteries, inverters, transformers, chargers, etc.) as well as the majority of the "Q" pumps and valves. During normal plant operation, the majority of the auxiliary building is supplied with outside air from the supply fans that is cooled or heated as necessary and exhausted by the area exhaust fans. Separate ventilation systems are provided for potentially radioactive areas. A number of individual room exhaust fans and cooling units are provided. This equipment is designed to maintain the general auxiliary building areas (not including the EDG room, SFP/fuel handling area, or control room) less than 105^0F during the summer.

The EDG rooms are equipped with exhaust fans that provide once through cooling to ensure the room temperature does not become excessive when the diesel generators are running. The service water system cools the EDG jacket water system. The room cooling only has to remove the EDG heat load. Since the EDGs are normally not running, room temperatures are normally low.

The spent fuel pool and fuel handling area has its own ventilation equipment that supplies outside air for cooling as required. The SFP area air temperature is not expected to increase significantly above the outdoor temperature, therefore excessive temperatures do not occur in this area.

The control room cooling units maintain a comfortable environment for the operators and cool the control room equipment. The temperature in the control room is normally maintained well below 105^0F, therefore this is a very mild environment for passive electrical equipment.

Walkdowns were performed in the ANO-1 auxiliary building to determine if the general area temperatures in the auxiliary building are normally maintained at less than 105^0F. Thermography surveys were completed for the rooms in the auxiliary building to collect additional data on the ambient temperatures and to detect areas with elevated temperatures. Cable trays were also surveyed. Components located in areas where the temperature exceeds 105^0F have been identified and are evaluated in Sections 3.7.4 through 3.7.6.

3.7.2.3 ANO-1 Turbine Building

The turbine building primarily houses non-Q equipment associated with electrical generation and not necessary for plant safety. The turbine building is cooled by supply and exhaust fans to provide once through cooling by the outside air. The lower levels of the turbine building are relatively cool. The upper portions of the turbine building and portions near the steam and feedwater piping can have elevated temperatures. In a few areas, Q cables in conduit are mounted in the turbine building or pass through the lower levels of the turbine building in their routing to the intake structure.

Walkdowns were performed in the ANO-1 turbine building to determine if the general area temperatures in the turbine building are normally maintained at less than 105^0F. Thermography surveys were completed for the turbine building to collect additional data on the ambient temperatures and to detect areas with elevated temperatures. Cable trays were also surveyed. Components in ambient temperatures above the 105^0F threshold have been identified and are evaluated in Sections 3.7.4 through 3.7.6.

3.7.2.4 ANO-1 Intake Structure

The intake structure houses the service water pumps and motors, the disconnect switchgear for the swing SW pump, the SW discharge header valves, several “Q” motor control centers, the fire pumps and their associated controls, a non-Q lighting panel, a non-Q power panel, and a non-Q cathodic protection rectifier. The intake structure is cooled by exhaust fans during normal plant operation to provide once through cooling by the outside air. The lower level of the intake structure has few heat loads, and is expected to remain at essentially the outdoor ambient temperature during summer operation due to the high flow rate of outside air through this level.

The intake structure ventilation system has been sized to maintain the maximum upper level area temperature at less than 105^0F with an outside air temperature of 95^0F. Although, under accident conditions, the exhaust fan could be lost and temperatures could be temporarily elevated (natural convection has been shown to provide adequate cooling), this accident condition does not impact the normal equipment aging environmental conditions. The outside air temperature can also exceed 95^0F during summer operation, but the duration is not significant when evaluating the average yearly conditions.

Walkdowns were performed in the ANO-1 intake structure to determine if the general area temperatures in the intake structure are normally maintained at less than 105^0F. Thermography surveys were completed for areas within the intake structure to collect additional data on the ambient temperatures and to detect areas with elevated temperatures. Cable trays were also surveyed. Components that have ambient temperatures above the 105^0F threshold have been identified and are evaluated in Sections 3.7.4 through 3.7.6.

3.7.2.5 Fuel Oil Storage Vaults

The underground fuel oil storage vaults contain the safety grade fuel oil storage tanks and the fuel oil transfer pumps for the emergency diesel generators. The vaults have no significant heat loads and are provided with exhaust fans to provide fresh air ventilation. The vault temperature remains cool year round due to the heat transfer to the ground and the lack of significant heat loads. Walkdowns have been performed in the fuel oil storage vaults to determine if the general area temperatures are maintained at less than 105^0F. Thermography surveys confirmed this to be the case. Therefore, the fuel oil storage vaults do not expose equipment to an excessive temperature and are not expected to have any passive electrical components in the scope of license renewal that are above the 105^0F threshold value chosen for evaluation of thermal aging.

3.7.2.6 ANO-1 Main Steam Line Area (Penthouse)

The main steam line area is cooled by two exhaust fans to provide once through cooling by outside air. The main steam line area (Room 170), located on elevation 404’, comprises the main steam isolation valve room and the main steam safety valve room. Within the main steam safety valve room, higher equipment elevations are accessible only by ladder. Thermography surveys were conducted in these areas during plant walkdowns. For the components in these areas that are exposed to temperatures that exceed 105^0F, further evaluation is provided in Sections 3.7.4 through 3.7.6.

3.7.2.7 AAC Diesel Generator Building

The equipment purchased for the AAC diesel generator was specified to have a design life of 40 years, which will envelope the extended license period. Nevertheless, walkdowns of the AAC generator building were performed and thermography surveys of the area were conducted. No areas with elevated temperatures were detected in the AAC diesel generator building.

3.7.3 Summary of the Plant Spaces Screening

Passive electrical components in the portions of the reactor building, where the ambient temperature is above the 120^0F license renewal threshold value, or the high radiation threshold value is exceeded, are evaluated. The auxiliary building and intake structure are controlled environments that are mostly benign spaces for the passive electrical equipment in the scope of license renewal. For the few areas in the auxiliary building that have equipment temperatures above the 105^0F threshold value, the passive electrical components are evaluated. The passive electrical components in the turbine building that are above the 105^0F threshold value are evaluated. Therefore, using the plant spaces approach, the Sandia selection of significant stressors, and NEI 95-10 (Reference 3.7-2) passive component categorization, the following subsets of passive electrical components within the scope of license renewal and subject to aging management review require component specific evaluation.

- Passive electrical equipment in elevated temperature locations
- Passive electrical equipment in areas of the reactor building that are above the threshold radiation level
- Wetted medium voltage power cables and cables exposed to potentially hazardous chemicals
- Power cables loaded above a significant fraction of their ampacity rating (significant fraction is defined as 50% of the ampacity rating)
- Cables and connectors subject to frequent manipulation (frequent is defined as being disconnected/reconnected more than once per refueling cycle)
- Low voltage instrument cables and connectors that operate at low currents or are otherwise sensitive to small variations in impedance

3.7.4 Connectors

Potential aging mechanisms that were considered for ANO-1 connectors include corrosion of metals, electrical stresses, water or humidity effects, mechanical stresses (including wear), and thermal or radiation aging of the organic components. During normal plant operation, the plant connectors are exposed primarily to dry conditions, and significant corrosion is not expected. The connectors do not provide any substantial mechanical support, and therefore mechanical stresses are not significant. The electrical stress of most connectors is insignificant, since the connectors are large in relation to the actual current carrying capacity. The organic portions of connectors (such as insulating materials, O-rings, filler materials, cases, etc.) are susceptible to aging from both thermal and radiation exposure. Frequent manipulation can cause wear on the surfaces in contact and loosen the sealing material or the cases.

A small number of splices have been identified as being subject to the aging effects discussed in the Sandia report. Moisture and elevated temperatures are the stressors for the splices identified. In order to manage the aging effects on these splices and demonstrate acceptable performance during the extended license term, an Electrical Component Inspection Program will be established to inspect and monitor the condition of the identified splices. This program is described in detail in Appendix B.

Connectors that are subject to frequent manipulation have been identified. The connector types identified consist of terminal blocks, multi-pin connectors, screw terminals, and battery terminal posts. Terminal blocks are addressed in the next section. For the other connector types, ANO-1 will rely on good maintenance practices to ensure that frequent manipulation does not unacceptably degrade connectors during the extended license term. Inspections of connectors are completed during the reconnection of connectors following any maintenance that required the connector to be disconnected. The effects of frequent manipulation (wear, loose fittings, cracking, etc.) are easily detected by visual inspections. Electrical checks of many vital functions are performed after reconnection of connectors to verify continuity. Therefore, no additional measures are necessary to manage aging for these connectors.

Connectors that are terminating impedance sensitive circuits have been identified. This group consists of coaxial and triaxial connectors. Oxidation or corrosion of the connector pins could interfere with the operation of these circuits. In order to ensure this does not happen, an Electrical Component Inspection Program will be established to periodically inspect these connectors. This program will ensure the proper operation of these circuits during the period of extended operation. This program is described in Appendix B.

3.7.5 Terminal Blocks

Potential aging mechanisms that were considered for ANO-1 terminal blocks include corrosion of exposed metal surfaces, electrical stresses, mechanical stresses, and thermal or radiation aging. Corrosion of an exposed metal surface is possible, especially in a high humidity environment, at the terminal lug to cable interface and on the terminal block. At ANO-1, no terminal blocks subject to aging management review are located in a high humidity environment and thus corrosion of terminal blocks is not a significant aging effect. No deterioration from electrical stresses is expected in terminal blocks since the current carrying capability is typically much larger that the actual current. The terminal blocks are not utilized as structural members and the mechanical stresses are too low to cause significant aging effects. Organic portions of the terminal blocks (such as insulating materials, filler materials, etc.) can be susceptible to aging from both thermal and radiation exposure. Phenolic material is very resistant to aging from elevated temperature or radiation. For example, terminal block manufacturer, Buchanan, lists 302^0F as a maximum continuous service temperature. The EPRI Data Bank qualifications for radiation for the General Electric or the Buchanan terminal blocks are over 10^8 rads. It can be concluded that thermal or radiation aging is not a significant aging effect for the phenolic terminal blocks at ANO-1.

Numerous terminal blocks subject to aging management review and subject to various aging effects have been identified. One terminal block was found in an area where it would be exposed to elevated temperatures. The aging effect for the remaining terminal blocks is damage due to frequent manipulation of the wires connected to the blocks.

As noted previously, terminal blocks are highly resistant to aging from elevated temperature. Testing of terminal blocks of the type used at ANO-1 demonstrates a qualified life significantly longer than the license renewal term. The terminal block identified above is only exposed to elevated temperature and not exposed to moisture or hazardous chemicals. Therefore, no additional actions are necessary to demonstrate the integrity of this terminal block through the period of license renewal.

The terminal blocks that are subject to frequent manipulation have been identified. Periodic maintenance and surveillance procedures call for the lifting of leads from terminal blocks for testing purposes. These same procedures require that the leads are reconnected and an independent verification performed. Good maintenance practices are relied upon at ANO-1 to ensure that frequent manipulation of the connections does not degrade any terminal block during the license renewal term. Therefore, no additional measures are necessary to manage the aging of terminal blocks.

3.7.6 Cables

Cable aging effects considered include corrosion of conductors, electrical stresses, water and humidity effects, thermal and radiation aging, and mechanical stresses. Corrosion of conductors is not expected since the conductors are covered by insulation. Deterioration of low voltage cables from electrical stress is not expected since the electrical stresses on the insulation are very low with respect to the material capabilities.

Ohmic heating can be significant for those cables that are routinely or continuously operated with high currents relative to their ampacity limits. The majority of the emergency equipment loads are only placed in operation following the initiation of an accident, therefore the "Q" switchgear cables, load center transformers, load centers and motor control centers are normally loaded to a small percentage of their rating. The essential 4.16 kV switchgear is normally energized from a non-Q supply and the "Q" supply from the emergency diesel generators is not normally in service.

Exposure to a wetted environment can be a significant aging effect for medium or high voltage cables. Water and humidity effects are not a concern for the majority of the cables at ANO-1 because they are located in dry environments. The cables that are buried underground could be exposed to a wetted environment. The aging effects of moisture intrusion, water treeing, or contamination could result in the effect of a reduced insulation resistance to ground and a potential electrical failure.

Chemical attack of organic materials used in cables may occur due to the exposure to hydraulic fluids, fuel oils, and lubricating oils or other chemicals. Significant differences exist between the chemical resistance of different types of insulation. Details on the compatibility of cables with chemical agents can be found in Table 4-6 of the Sandia Report. The cables are most likely to be exposed to the chemicals through leaks or spills. The effects of exposure to chemicals could be a softening, flowing, cracking, or discoloration of the insulation or jacket that could result in the effect of a reduced insulation resistance to ground and a potential electrical failure of the circuit.

Thermal aging of cables affects the organic materials in the insulation and can cause embrittlement, cracking, or discoloration of the insulation through thermal oxidation. This could result in the effect of a reduced insulation resistance to ground and a potential electrical failure.

Radiation stress is only significant for cables exposed to cumulative exposures of greater than the selected threshold of $1x10^8$ rads. Radiation exposure of cables will affect the organic materials in the insulation and can cause embrittlement, cracking, swelling, or discoloration of the insulation through thermal oxidation. This could result in the effect of a reduced insulation resistance to ground and a potential electrical failure. Cables subject to aging management review will not reach the radiation threshold, and thus, radiation aging is not a significant aging effect for ANO-1 cables.

Mechanical stresses in the cable systems will not change significantly during the license renewal period. Installation practices at ANO-1 and operating experience indicates that insulation cut through is not a concern. The Bechtel cable installation procedures in place during initial plant construction and the current ANO cable installation procedures should prevent cable damage during installation. Therefore, mechanical stress is not considered an aging effect requiring management for ANO-1 except for those cables that are frequently manipulated. Cutting, chafing abrasion, or splitting could occur and result in the effect of a reduced insulation resistance to ground and a potential electrical failure. In order to determine the cables and connectors that are frequently manipulated, a review of the maintenance procedures and interviews with maintenance personnel were completed. The cables that are frequently manipulated are primarily those with connectors, although some terminate on terminal blocks.

Low voltage control and instrument circuits degrade primarily because of environmental influences. Circuit electrical loading is not an aging effect in these applications due to the extremely low currents. Oxidation of connector or termination contacts may have an impact on the impedance sensitive circuits. The effect would be an increase in the circuit resistance and a reduction in the signal strength.

Cables that are in the scope of license renewal and are exposed to the various aging effects have been identified. To ensure that the aging effects due to these various conditions do not degrade the ability of these cables to function, an Electrical Component Inspection Program will be established in order to monitor the condition of the cables for the period of extended operation. The program will periodically inspect the cables and document the results for trending purposes. This program is described in detail in Appendix B.

Cables that are in the scope of license renewal and subject to frequent manipulation have been identified. The connectors and terminal blocks to which these cables are terminated are discussed in previous sections. This discussion on managing aging effects for connectors also applies to the cables that are frequently manipulated. Inspections of these cables are completed during the reconnection process. The effects of frequent manipulation are easily detected by visual inspections. Therefore, no additional measures are necessary to manage aging of these cables.

The cables used in impedance sensitive circuits have been identified. The aging effect that needs to be considered for impedance sensitive circuits is corrosion of the connectors. The connectors in this category were discussed in a previous section. The cables do not have an aging effect that can impact the operation of the circuit in terms of impedance changes. Therefore, there is no need to consider these cables further.

3.7.7 Industry and Operating Experience

Industry and operating experience have been reviewed to ensure no additional aging effects exist beyond those discussed herein. This review was performed for cables, connectors, splices, and terminal blocks. No unique aging effects for these components were identified from this review.

3.7.8 References for Section 3.7

3.7-1 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.

3.7-2 NEI 95-10, "*Industry Guidelines for Implementing the Requirements of 10CFR Part 54 – The License Renewal Rule*," Revision 0, Nuclear Energy Institute, March 1996.

Table 3.7-1 Passive Electrical Components					
Component / Commodity Grouping	Intended Function	Material	Environment	Aging Effect	Program/Activity
Splices	Connect cable conductors to other cables.	Raychem splice kit, Raychem cable splices, Okonite insulating tape, Raychem heat shrink tape sealant	Moisture, Elevated Temperature	Reduced insulation resistance, embrittlement, and cracking	Electrical Component Inspection
Connectors	Connect cable conductors to other cables or electrical devices.	Polysufone, Kapton, EPDM, CSPE	Moisture, Elevated Temperature	Reduced insulation resistance, embrittlement, cracking, and electrical failure	Electrical Component Inspection
Terminal Block	Connect cable conductors to electrical devices.	Phenolic	Elevated temperature	NA	NA
Cables	Provide electrical connection between two sections of an electrical circuit.	XLPE, EPR, FR-EP, CPE, CSPE, FR-EPDM, Okoprene, Okoguard, Okolon, Okonite-FMR, SR	Moisture, Elevated Temperature, Ohmic Heating, Corrosive Chemicals	Reduced insulation resistance, embrittlement, cracking, and electrical failure	Electrical Component Inspection

4.0 TIME LIMITED AGING ANALYSES

As discussed in Section 1.0, two areas of technical reviews are required to support an application for a renewed operating license. The first area of technical review is the ANO-1 Integrated Plant Assessment, which is described in Sections 2.0 and 3.0. The second area of technical review required for license renewal is the identification and evaluation of plant-specific time-limited aging analyses and exemptions, which are provided in this section. The identification and evaluations contained in this section meet the requirements contained in 10CFR54.21(c) and allow the NRC to make the finding contained in 10CFR54.29(a)(2).

4.1 IDENTIFICATION OF TIME-LIMITED AGING ANALYSES

4.1.1 Process Overview

10CFR54.21(c) requires an evaluation of time-limited aging analyses be provided as part of the application for a renewed license. Time-limited aging analyses are defined in 10CFR54.3 as those licensee calculations and analyses that meet six specific criteria. The process used to identify the ANO-1 specific time-limited aging analyses is consistent with the guidance provided in NEI 95-10 (Reference 4.1-1

To assure that the ANO-1 time-limited aging analyses were identified, several document sets were searched. ANO-1 specific documents that were reviewed for time-limited aging analyses include the ANO-1 licensing correspondence file, the ANO-1 SAR, B&W topical reports referenced in correspondence and in the SAR, and ASME Section XI Summary Reports.

The identified calculations and analyses were reviewed to determine those that meet the six criteria of 10CFR54.3. The analyses and calculations that meet the six criteria are the ANO-1 specific time-limited aging analyses, which are listed in Table 4.1-1. The reactor vessel neutron embrittlement analyses are evaluated in BAW-2251A and is discussed in Section 4.2. Metal fatigue is applicable to Class 1 components and is discussed in Section 4.3. Environmental qualification is discussed in Section 4.4. Reactor building tendon prestress is discussed in Section 4.5. Fatigue of the reactor building liner plate and reactor building penetrations is addressed in Section 4.6. Boraflex is discussed in Section 4.7. Other time-limited aging analyses are discussed in Section 4.8.

As required by 10CFR54.21(c)(1), an evaluation of ANO-1 specific time-limited aging analyses must be performed to demonstrate that

- the analyses remain valid for the period of extended operation, or
- 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.

4.1.2 Identification of Exemptions

10CFR Part 54 also requires that the application for a renewed license include a list of current plant-specific exemptions granted pursuant to 10CFR50.12 that are based on time limited aging analyses as defined in 10CFR54.3. No 10CFR50.12 exemptions involving a time-limited aging analysis as defined in 10CFR54.3 are required during the period of extended operation.

4.1.3 References for Section 4.1

4.1-1. NEI 95-10, "Industry Guidelines for Implementing the Requirements of 10CFR Part 54 – The License Renewal Rule," Revision 0, Nuclear Energy Institute, March 1996.

Table 4.1-1 List of ANO-1 Time-limited Aging Analyses		
Component/Subject	**TLAA**	**Section of ANO-1 LRA where TLAA is addressed**
Reactor Vessel	Metal Fatigue	Section 4.3
	Analytical Evaluation of Flaws	Section 4.3
	BAW-1511 (USE)	Section 4.2
	BAW-1895 (RT_{PTS})	Section 4.2
	BAW-2143P (USE & RT_{PTS})	Section 4.2
	BAW-2148 (USE)	Section 4.2
	BAW-2166 (USE & RT_{PTS})	Section 4.2
	BAW-2178 (USE)	Section 4.2
	BAW-2192 (USE)	Section 4.2
	BAW-2222 (USE & RT_{PTS})	Section 4.2
	BAW-2257 (USE)	Section 4.2
	BAW-10013 (Intergranular Separations)	Section 4.8
	BAW-10018 (Thermal Shock)	Section 4.2 No Longer Applicable—See BAW-2251A
	BAW-10051 (FIV Analysis)	Section 4.8
RCS Piping	Metal Fatigue	Section 4.3
	Analytical Evaluation of Flaws	Section 4.3
	BAW-1847 (Leak Before Break Analysis)	Section 4.8
	BAW-2127 (final evaluation of surge line thermal stratification)	Section 4.3
	BAW-2085 (preliminary evaluation of surge line thermal stratification)	Section 4.3
Reactor Vessel Internals	Metal Fatigue	Section 4.3
	Analytical Evaluation of Flaws	Section 4.3
	BAW-10051 (FIV Analysis)	Section 4.8
	BAW-10008 (Stress and Deflection Analyses)	Section 3.5, Appendix B
Reactor Coolant Pumps	Metal Fatigue	Section 4.3
	Analytical Evaluation of Flaws	Section 4.3

Table 4.1-1 List of ANO-1 Time-limited Aging Analyses		
Component/Subject	**TLAA**	**Section of ANO-1 LRA where TLAA is addressed**
Steam Generator	Metal Fatigue	Section 4.3
	Analytical Evaluation of Flaws	Section 4.3
	BAW-1823 (for sleeves) related to metal fatigue	Section 4.3
	BAW-2120 (for sleeves) related to metal fatigue	Section 4.3
	BAW-2005 (for sleeves) related to metal fatigue	Section 4.3
	BAW-10146 (originally BAW-1588) (for tube thickness) related to metal fatigue	Section 4.3
	BAW-10027--related to metal fatigue	Section 4.3
Control Rod Drive	Metal Fatigue	Section 4.3
	Analytical Evaluation of Flaws	Section 4.3
	BAW-10047 (Sandvik flaw evaluation)	Section 4.3
Pressurizer	Metal Fatigue	Section 4.3
	Analytical Evaluation of Flaws	Section 4.3
Concrete Reactor Building Tendon Prestress	ACI 318-63 ANO-1 SAR, Section 5.2.4.2.1.	Section 4.5
Reactor Building Liner Plate and Penetrations-Fatigue	The ANO-1 SAR Section 5.2.1.4.7.3 describes the fatigue review that was performed for the ANO-1 reactor building liner plate and penetrations	Section 4.6
Spent Fuel Racks Boraflex	Boraflex Degradation	Section 4.7
Environmental Qualification of Electrical Equipment	Qualified Life	Section 4.4
Reactor Coolant Pump Motor Flywheels	Analytical Evaluations of Flaws	Section 4.8

4.2 REACTOR VESSEL NEUTRON EMBRITTLEMENT

Entergy Operations participated in a BWOG effort that developed a series of topical reports to demonstrate that the aging effects for reactor coolant system components are adequately managed for the period of extended operation. One of the BWOG topical reports is BAW-2251A, "Demonstration of the Management of Aging Effects for the Reactor Vessel" (Reference 4.2-1). This topical report addresses the reactor vessel. Time-limited aging analyses applicable to the reactor vessel are addressed within BAW-2251A.

The process used by Entergy Operations to incorporate BAW-2251A by reference is discussed in Section 2.3.1.2. In addition, Entergy Operations' responses to the Applicant Action Items listed in the NRC SER of BAW-2251A are provided in Table 2.3-4 of the ANO-1 LRA.

The following time-limited aging analyses are applicable to the reactor vessel.

- Thermal fatigue evaluation addressed in Section 4.3.4;
- Flaw growth acceptance under ASME Boiler and Pressure Code Section XI (Reference 4.2-2) addressed in Section 4.3.6 ;
- Neutron embrittlement of the beltline region, including pressurized thermal shock and Charpy upper-shelf energy reduction addressed in this section; and
- Intergranular separation in the heat affected zone of low alloy steel under austenitic stainless steel weld cladding addressed in Section 4.8.

The ANO-1 Reactor Vessel Integrity Program as described in Appendix B is being utilized to ensure that the time dependent parameters used in the time-limited aging analysis evaluations reported in BAW-2251A are tracked such that the time-limited aging analysis remains valid through the period of extended operation for ANO-1.

4.2.1 Pressurized Thermal Shock

10CFR50.61(b)(1) provides rules for protection against pressurized thermal shock for pressurized water reactors. Licensees are required to perform an assessment of the projected values of reference temperature whenever a significant change occurs in projected values of RT_{PTS}, or upon request for a change in the expiration date for the operation of the facility. For license renewal, RT_{PTS} values are calculated for 48 EFPY for ANO-1.

10CFR50.61(c) provides two methods for determining RT_{PTS}: (Position 1) for material that does not have surveillance data available, and (Position 2) for material that has surveillance data. Availability of surveillance data is not the only measure of whether Position 2 may be used. The data must also meet tests of sufficiency and credibility.

RT_{PTS} is the sum of the initial reference temperature (IRT_{NDT}), the shift in reference temperature caused by neutron irradiation (ΔRT_{NDT}), and a margin term (M) to account for uncertainties.

IRT_{NDT} is determined using the method of Section III of the ASME Boiler and Pressure Vessel Code (Reference 4.2-3). That is, IRT_{NDT} is the greater of the drop weight nil-ductility transition temperature or the temperature that is 60°F below that at which the material exhibits Charpy test values of 50 ft-lbs. and 35 mils lateral expansion. For a material for which test data is unavailable, generic values may be used if there are sufficient test results for that class of material. For Linde 80 weld material with the exception of WF-70, the IRT_{NDT} is taken to be the currently NRC accepted values of -7°F or -5°F. The ANO-1 reactor vessel does not contain any Linde 80 WF-70 weld material. For forgings and plate material, measured values are used where appropriate data is available. Where not available, the generic value of +3°F is used for forgings and +1°F is used for plate material.

For Position 1 material (surveillance data not available), ΔRT_{NDT} is defined as the product of the chemistry factor (chemistry factor) and the fluence factor. Chemistry factor is a function of the material's copper and nickel content expressed as weight percent. "Best estimate" copper and nickel contents are used which are the means of measured values for the materials. For ANO-1, best estimate values were obtained from BAW-2251A. The value of chemistry factor is directly obtained from tables in 10CFR50.61. The fluence factor value is calculated using end-of-life peak fluence at the inner surface at the material's location. Fluence values were obtained by extrapolation to 48 EFPY from the current 32 EFPY values.

For Position 2 material (surveillance data available), the discussion above for Position 1 applies except for determination of chemistry factor, which in this instance is a material-specific value calculated as follows.

- Multiply each ΔRT_{NDT} value by its corresponding fluence factor
- Sum these products
- Divide this sum by the sum of the squares of the fluence factors

The margin term (M) is generally determined as follows.

$$M = 2\,(\sigma_I^2 + \sigma_\Delta^2)^{0.5}$$

where σ_I is the standard deviation for IRT_{NDT}

and σ_Δ is the standard deviation for ΔRT_{NDT}.

For Position 1, $\sigma_I = 0$ if measured values are used. If generic values are used, σ_I is the standard deviation of the set of values used to obtain the mean value. For ΔRT_{NDT}, σ_Δ=28°F for welds and 17°F for base metal (plate and forgings), except that σ_Δ need not exceed one-half of the mean value of ΔRT_{NDT}. For Position 2, the same method for determining the σ values are used except that σ_Δ values are halved (14°F for welds and 8.5°F for base metal).

10CFR50.61(b)(2) establishes screening criteria for RT_{PTS}: 270°F for plates, forgings, and axial welds and 300°F for circumferential welds. The values for RT_{PTS} at 48 EFPY for ANO-1 are provided in Appendix A, Table A-1, of BAW-2251A. The projected RT_{PTS} values are within the established screening criteria for 48 EFPY.

By letter dated July 1, 1998 [1CAN079801 (Reference 4.2-4)], Entergy Operations submitted a response to a request for additional information for ANO-1 regarding Supplement 1 to Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity." The information was also contained in the BWOG topical report BAW-2325 (Reference 4.2-5). After reviewing BAW-2325, the NRC noted changes in transition temperature shift data for certain surveillance capsules and issued the BWOG several requests for additional information. Subsequent interaction between the BWOG and the staff resulted in the publication of Revision 1 to BAW-2325 (Reference 4.2-6) in February of 1999. Since BAW-2251A was completed prior to the staff's approval of BAW-2325, Revision 1, an assessment was performed relative to the staff's findings regarding chemistry factors reported in BAW-2251A. The chemistry factors reported in BAW-2251A are equivalent to or exceed the chemistry factors reported in BAW-2325, Revision 1, for the limiting beltline welds at ANO-1. In addition, ANO-1 has recalculated 48 EFPY fluence for the beltline region using the methodology described in BAW-2251A, Appendix D, and BAW-2241AP (Reference 4.2-7) and has determined that the 48 EFPY fluence estimates reported in BAW-2251A remain conservative. Therefore, the 48 EFPY RT_{PTS} values for the limiting beltline welds reported in BAW-2251A, Table A-1, remain conservative for ANO-1 since both the chemistry factors and fluence estimates remain conservative.

In order to avoid exceeding the pressurized thermal shock screening criteria at ANO-1 during the period of extended operation, Entergy Operations utilizes low leakage core designs. In addition, Entergy Operations is involved in various industry activities that provide new information or new analysis techniques associated with the reactor vessel beltline region.

4.2.2 Charpy Upper-Shelf Energy

Appendix G of 10CFR Part 50 requires that reactor vessel beltline materials "have Charpy upper-shelf energy … of no less than 75 ft. lb. initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft. lb.…." The BWOG position on upper-shelf energy for 32 EFPY is documented in responses to NRC Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity," as reported in BAW-2166 (Reference 4.2-8) and BAW-2222 (Reference 4.2-9). The BWOG position on upper-shelf energy for 48 EFPY is documented in BAW-2275 (Reference 4.2-10), which is included in BAW-2251A as Appendix B.

Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," provides two methods for determining Charpy upper-shelf energy (C_VUSE). Position 1 for material that does not have surveillance data available and Position 2 for material that does have surveillance data. For Position 1, the percent drop in C_VUSE, for a stated copper content and neutron fluence, is determined by reference to Figure 2 of Regulatory Guide 1.99, Revision 2. This percentage drop is applied to the initial C_VUSE to obtain the adjusted C_VUSE. For Position 2, the percent drop in C_VUSE is determined by plotting the available data on Figure 2 and fitting the data with a line drawn parallel to the existing lines that upper bounds all the plotted points.

The 48 EFPY C_VUSE values determined for the reactor vessel beltline materials for ANO-1 are reported in Table 4-4 of BAW-2251A. The T/4 fluence values reported in this table were calculated in accordance with the ratio of inner surface to T/4 values (i.e., neutron fluence lead factors at T/4) determined in the latest Reactor Vessel Surveillance Program report. As shown in this table, the C_VUSE is maintained above 50 ft-lb for base metal (plates and forgings); however, the C_VUSE for weld metal drops below the required 50 ft-lb. at 48 EFPY. Appendix G of 10CFR Part 50 provides for this by allowing operation with lower values of C_VUSE if "… it is demonstrated … that the lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code."

This equivalent margin analysis was performed for 48 EFPY and the results are reported in Appendix B to BAW-2251A for service levels A, B, C, and D. The analysis used conservative materials models and load combinations, e.g., treating thermal gradient stress as a primary stress. For service levels A and B, the analytical results demonstrate that there is sufficient margin beyond that required by the acceptance criteria of Appendix K of the ASME Code (1995 Edition). For service levels C and D, the most limiting transient was evaluated, and again the analytical results demonstrate that there is sufficient margin beyond that required by the acceptance criteria of Appendix K of the ASME Code. The evaluations for all service levels conclusively demonstrate the adequacy of safety against fracture for the ANO-1 reactor vessel for 48 EFPY.

As discussed in the previous section, Entergy Operations submitted a response to a request for additional information for ANO-1 regarding Supplement 1 to Generic Letter 92-01, Revision 1 (Ref. 4.2-6). The copper composition reported in BAW-2251A is equivalent to or exceeds the copper content reported in BAW-2325, Revision 1. In addition, the 48 EFPY fluence estimates were redone. It was determined the fluence estimates listed in BAW-2251A remain conservative. Therefore, the C_VUSE in Table 4-4 of BAW-2251A remains conservative.

4.2.3 References for Section 4.2

4.2-1 BAW-2251A, "*Demonstration of the Management of Aging Effects for the Reactor Vessel*," The B&W Owners Group Generic License Renewal Program, June 1996.

4.2-2 ASME Boiler and Pressure Vessel Code, Section XI, "*Rules for In-Service Inspection of Nuclear Power Plant Components*," American Society of Mechanical Engineers.

4.2-3 ASME Boiler and Pressure Vessel Code, Section III, "*Rules for Construction of Nuclear Power Plant Components*," American Society of Mechanical Engineers.

4.2-4 1CAN079801, Letter from D. James (ANO) to the NRC, "*Generic Letter 92-01, Supplement 1, Reactor Vessel Structural Integrity, Request for Additional Information*," dated July 1, 1998.

4.2-5 BAW-2325, "*Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity – Generic Letter 92-01, Revision 1, Supplement 1*," B&W Owners Group, May 1998.

4.2-6 BAW-2325, "*Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity*," Revision 1, B&W Owners Group, January 1999.

4.2-7 BAW-2241AP, "*Fluence and Uncertainty Methodologies*," The B&W Owners Group, April 1997.

4.2-8 BAW-2166, "*Response to Generic Letter 92-01*," B&W Nuclear Service Company, June 1992.

4.2-9 BAW-2222, "*Response to Closure Letters to Generic Letter 92-01, Revision 1*," B&W Nuclear Technologies, June 1994.

4.2-10 BAW-2275, T. Wiger and D. Killian, "*Low Upper-Shelf Toughness Fracture Mechanics Analysis of B&W Designed Reactor Vessels for 48 EFPY*," Framatome Technologies, Inc.

4.3 METAL FATIGUE

4.3.1 Background

The thermal fatigue analysis of reactor coolant components has been identified as a time-limited aging analysis for ANO-1. Specific RCS components have been designed considering transient cycle assumptions, as listed in vendor specifications and the ANO-1 SAR. For ANO-1, B&W designed the main reactor coolant system and piping. Bechtel designed the ASME Class 1 portions of the ancillary systems attached to the B&W scope of supply. The evaluation of each vendor's piping design is performed separately.

4.3.2 B&W Scope of Supply

The B&W scope of supply includes major components in the RCS and the associated interconnecting piping. Vessels were designed in accordance with ASME Section III (Reference 4.3-1), 1965 Edition, with Addenda through Summer 1967.

Reactor coolant pumps were designed in accordance with ASME Section III, 1968 edition. RCS piping supplied by B&W was designed to Nuclear Piping Code, USAS B31.7 Class 1 (Reference 4.3-2).

4.3.3 Bechtel Scope of Supply

Bechtel supplied piping includes the Class 1 portions of the ancillary systems attached to the B&W scope of supply and miscellaneous vents, drains and instrumentation lines. Ancillary systems include core flood, low pressure injection/decay heat removal, high pressure injection/makeup and purification, pressurizer auxiliary spray, pressurizer surge line low point drain, reactor coolant letdown, reactor coolant high point vent and other miscellaneous vents, drains and instrumentation piping. Bechtel supplied piping was designed to Class 1 standards of Nuclear Piping Code USAS B31.7, dated February 1968, and as corrected for Errata under date of June 1968, or later appropriate ASME Section III Code sections, provided they have been reconciled.

4.3.4 Thermal Fatigue

10CFR54.21(c) requires an evaluation of fatigue to demonstrate that either the analyses remain valid for the period of extended operation, have been projected to the end of the extended operation period, or the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. ANO-1 addresses fatigue by ensuring that its effects are adequately managed for the period of extended operation. This method comprises the following activities:

- Design documentation defines the transient cycles to be monitored and the number of allowable cycles. Available documentation reflects the current plant configuration.
- Appropriate actions are taken to address significant industry fatigue concerns that were not considered in the original design. This includes issues like thermal stratification fatigue and environmentally-assisted fatigue. The ANO-1 specific review of industry experience is provided in this section.

- The actual plant transient cycles are tracked and documented to ensure the allowable number of cycles is not exceeded. The ANO-1 transient cycle monitoring program is reviewed in Section 4.3.5.

4.3.4.1 ANO-1 Fatigue Design Documentation

Fatigue evaluations were required in the design of the ANO-1 Class 1 components in accordance with the requirements specified in the applicable design codes (i.e., ASME Section III and USAS B31.7). For Class 1 components, separate discussions of fatigue are provided for the B&W-supplied components and for the Bechtel-supplied components.

4.3.4.2 B&W-Supplied Components

Design cyclic loadings and thermal conditions for the B&W-designed reactor coolant system Class 1 components are defined by the component design specifications. The RCS functional specification provides the set of transients that were used in the design of the B&W-supplied components and are included as part of each component design specification. The component design specification defines the transient cycle assumptions used in the fatigue evaluations for each component.

As a part of the GLRP effort, a review was performed to determine which Class 1 components were more sensitive to fatigue (environmentally-assisted fatigue effects were not considered in this study) and which transients caused the greatest impact in terms of fatigue stress on those components. Based on the fatigue usage factor-design transient matrices, the transients used to calculate fatigue usage factors for the B&W-supplied Class 1 components were identified. For this set of design transients, the number of transients accrued was compiled and a conservative projection was made to determine if the number of design cycles would be exceeded in the period of extended operation. In no instance, for ANO-1, did the extrapolation exceed the allowable number of design cycles prior to 60 years of operation. The existing usage factor calculations remain valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

4.3.4.3 Bechtel-Supplied Piping

Design cyclic loading and thermal conditions for the Bechtel-supplied piping are defined in a Bechtel Class 1 piping design specification. Existing cumulative usage factors and analyzed thermal transients documented in thermal fatigue calculations for the Bechtel-supplied piping have been reviewed. Based on the number of transient cycles accrued for ANO-1 and the rate these cycles have been accumulated, the number of transient cycles that were originally projected for the current license of 40 years envelopes the number of cycles projected for the end of the 60-year operation.

As allowed by ASME NB-3630, for the Class 1 piping of pipe size of 1” NPS or less, ASME Subsection NC-3600 rules have generally been applied for Code qualification, using the stress range reduction factor of 1.0 for thermal fatigue stress allowables, as long as the location does not exceed 7000 full temperature thermal cycles during its operation. In order to identify the specific locations where extended operation could invalidate the existing stress range reduction factor in the piping analysis, the design temperatures and operating conditions of the ANO-1 mechanical systems were considered. These mechanical systems were reviewed to determine those likely to see 7,000 equivalent full temperature thermal cycles during plant operation. This review determined that the assumption of less than 7,000 equivalent full temperature thermal cycles is valid for the period of extended operation. The Bechtel-supplied piping has been Code qualified for the original number of design cycles which envelopes the predicted number of cycles for 60 year operation. The existing fatigue calculations remain valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i)

4.3.4.4 ANO-1 Response to Industry Experience on Fatigue

Industry experience has identified thermal conditions that would result in thermal fatigue stresses that were not considered in the original design evaluations. The following section discusses industry-wide thermal fatigue issues and how they have been addressed for ANO-1.

<u>Environmentally Assisted Fatigue (NRC GSI-190)</u>

Historically, the ASME fatigue design curves have been obtained from the results of fatigue tests on small specimens, conducted under a laboratory (air) environment. Recent laboratory test results indicate that the effects of light water reactor environments can reduce the fatigue life of RCS components. Interim fatigue curves based on this test data have been published in NUREG/CR-5999.

In a study funded by the NRC, the impacts of these interim fatigue curves on various nuclear power plant components (including ANO-1) were assessed. The results of the assessment were published in NUREG/CR-6260, “Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components.” Based on the results of NUREG/CR-6260, the NRC initiated GSI-190, to address concerns regarding the environmental effects on operation beyond the current license term (i.e. from 40 years to 60 years).

In December 1999, in closing GSI-190 (Ref.43), the NRC concluded:

“The results of the probabilistic analyses, along with the sensitivity studies performed, the interactions with the industry (NEI and EPRI), and the different approaches available to the licensees to manage the effects of aging, lead to the conclusion that no generic regulatory action is required, and that GSI-190 is resolved. This conclusion is based primarily on the negligible calculated increases in core damage frequency in going from 40 year to 60 year lives. However, the calculations supporting resolution of this issue, which included consideration of environmental effects, and the nature of age-related degradation indicate the potential for an increase in the frequency of pipe leaks as plants continue to operate. Thus, the staff concludes that, consistent with existing requirements in 10CFR54.21, licensees should address the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal."

In summary, environmental effects have negligible impact on core damage frequency, and no generic regulatory action is required. However, environmental effects can increase the frequency of pipe leaks and licensees who apply for license renewal should address the effects of coolant environment on component fatigue life as part of their aging management programs.

An effective approach to manage this issue is to identify ANO-1 specific locations that are the most susceptible to failure from thermal fatigue, and other degradation mechanisms, and include these locations in the augmented inservice inspection program.

Section 5.3 of NUREG/CR-6260 provides the following critical component locations in B&W plants that are applicable to ANO-1.

1) Reactor vessel shell and lower head
2) Reactor vessel inlet and outlet nozzles
3) Pressurizer surge line
4) Makeup/high pressure injection (HPI) nozzles
5) Reactor vessel core flood nozzle
6) Decay heat removal system Class 1 piping

An environmentally assisted fatigue analysis for the reactor vessel was completed by the BWOG and documented in BAW-2251A (Reference 4.3-3). This study derived environmental fatigue factors based on the model described in NUREG/CR-6335, “Fatigue Strain-Life Behavior of Carbon and Low-Alloy Steels, Austenitic Stainless Steels, and Alloy 600 in LWR Environments.” These factors were applied to the reactor vessel items studied in NUREG/CR-6260 (the reactor vessel shell and lower head, the vessel inlet and outlet nozzles, and the core flood nozzles). The conclusion of this analysis was that after accounting for environmentally assisted fatigue, the reactor vessel fatigue usage factors remain acceptable for the period of extended operation. Therefore, locations 1, 2, and 5 above have been resolved by analysis.

The remaining three locations (locations 3, 4, and 6) are included in the risk-informed inservice inspection (RI-ISI) program which has been recently approved by the NRC as an alternative to requirements of ASME Section XI Inservice Inspection (Reference 4.3-4). This RI-ISI program was a pilot program developed using methodology consistent with ASME Code Case N-560, “Alternative Examination Requirements for Class 1, Category B-J Piping Welds Section XI, Division 1.”

The primary objective of this program was to identify the “risk important” piping segments for inspection based on analysis of the probability and the consequences of piping failures. Implementing the RI-ISI program will ensure inspections take place where degradation mechanisms (including thermal fatigue) would be most likely at ANO-1.

In summary, three of the six locations listed in NUREG/CR-6260 as most susceptible to environmentally assisted fatigue, are resolved by the BWOG engineering report BAW-2251A. The potential environmental assisted fatigue damage to the remaining three locations (pressurizer surge line, HPI nozzles and decay heat removal suction line) will be managed by the RI-ISI program through the extended operation period. Note that the ANO-1 decay heat removal suction line has been shown in NUREG/CR-6260 to be acceptable with the environmental factors applied.

NRC Bulletin 88-08, Thermal Stresses in Piping Connected to Reactor Coolant Systems

In response to NRC Bulletin 88-08, “Thermal Stresses in Piping Connected to Reactor Coolant Systems,” and Supplements 1 and 2 to this bulletin, ANO-1 reviewed 23 different piping configurations connected to the RCS. Each line was reviewed to determine susceptibility to thermal stresses. It was determined that three HPI lines and the pressurizer auxiliary spray line met the criteria of the Bulletin. During refueling outage 1R8, portions of the pressurizer auxiliary spray line were modified or replaced to eliminate the condition of thermal stratification. Additionally, a combination of dye penetrant testing, ultrasonic testing, and visual examinations of the relevant welds from the check valve to the RCS nozzle were conducted for the three HPI lines, in refueling outage 1R8. No detrimental effects of thermal stratification were found.

In addressing Supplement 3 to NRC Bulletin 88-08, ANO-1 participated with the BWOG to fund EPRI Program 3153, “Thermal Stratification in Horizontal Lines.” Additional evaluation of the 23 lines was performed to consider flow from the RCS. Based on this evaluation, ANO-1 included the fourth HPI line in the NDE scope. This fourth HPI line was examined in 1R9 using the enhanced ultrasonic testing method from Supplement 2 of the bulletin. No detrimental conditions were found**.** A design change package was implemented to add temperature-monitoring instrumentation on the HPI lines. The decay heat system suction line from the RCS also required monitoring and evaluation due to packing leaks on an isolation valve. These actions meet the requirements of Bulletin 88-08.

Because of stratified flows in lines monitored on ANO-2, the ANO-1 systems were reviewed again, to identify systems with attributes similar to the ANO-2 stratified lines. Four lines were found to require monitoring and evaluation. They were pressurizer main spray, decay heat drop leg, reactor coolant system drains, and reactor coolant system letdown drains. Temperature monitoring and evaluations have demonstrated that these ANO-1 lines are qualified for their service conditions.

In response to Bulletin 88-08, ANO-1 committed to performing enhanced ultrasonic examinations of 17 HPI welds and visual inspection of two segments of HPI piping, as part of the ANO-1 10-year interval Inservice Inspection Plan. Subsequently, the scope of ISI inspections for the HPI lines and pressurizer surge line was modified based on an ANO-1 risk analysis performed consistent with the requirements of ASME Code Case N-560, "Alternative Examination Requirements for Class 1, Category B-J Piping Welds Section XI, Division 1". This commitment will continue through the extended period of operation of ANO-1. This issue has therefore been resolved for the extended period of operation.

<u>NRC Bulletin 88-11, Pressurizer Surge Line Thermal Stratification</u>

The review of the pressurizer surge line thermal stratification concern was performed in several stages. Entergy Operations participated in a BWOG effort to review data from a B&W plant in Germany and an extensive data collection program at Oconee Unit 1. Inspections of the ANO-1 surge line determined that only snubbers were used and no rigid or whip restraints existed on the pipe that could restrict the pipe thermal movement. An elevation survey of the pressurizer surge line was performed in 1989 and the survey results indicated no bowing. Visual inspection of a vertical snubber found no damage or defect, and there was no evidence of thermal movements exceeding the travel of the snubber. A BWOG report was prepared to justify continued plant operation and present a bounding analysis. The final report was later submitted to complete the actions requested in NRC Bulletin 88-11. Based on this BWOG report, revisions of the RCS functional specification were completed to add restrictions on operating conditions for the surge line. A reanalysis was completed of the stress in the surge line drain.

The NRC has reviewed the ANO-1 responses and determined the actions taken met the requirements of the bulletin. The identified thermal stresses are now considered in the appropriate stress and fatigue calculations and this concern has been resolved.

Originally, ANO-1 committed to performing enhanced ultrasonic examinations of two elbows of the pressurizer surge line as part of ANO-1 10-year interval Inservice Inspection Plan in response to Bulletin 88-11. Subsequently, the scope of ISI inspections of the pressurizer surge line has been modified based on a ANO-1 risk analysis performed consistent with the requirements of ASME Code Case N-560, "Alternative

Examination Requirements for Class 1, Category B-J Piping Welds Section XI, Division 1". This commitment will continue through the extended period of operation of ANO-1. This issue has therefore been resolved for the extended period of operation.

HPI/MU Nozzle Cracking in B&W Plants

(NRC Information Notice 82-09, Cracking in Piping of Makeup Coolant Lines at B&W Plants; NRC Generic Letter 85-20, Resolution of Generic Issue 69: High Pressure Injection/Make-Up Nozzle Cracking in Babcock and Wilcox Plants; and NRC Information Notice 97-46, Unisolable Crack in High-Pressure Injection Piping)

A BWOG task force addressed potential high pressure injection/makeup nozzle cracking problems which affected B&W plants. The task force determined that the root cause of the nozzle cracking was loose thermal sleeves and that the cracks were propagated by thermal fatigue. The cracked safe-ends were MU nozzles with loose thermal sleeves. Based on the recommendations made by the task force, actions taken by ANO-1 included repair of nozzles with loose of damaged thermal sleeves, maintenance of adequate minimum flow, institution of an augmented inspection program for the nozzles, and performance of stress analysis with the modified thermal sleeves. The augmented ISI inspection of the HPI/MU nozzles is consistent with the methodology and scope of inspection recommended by the BWOG Safe-End Task Force. Ultrasonic testing of the knuckle region of the HPI nozzles every fifth refueling cycle, and radiography of the thermal sleeves will continue through the period of extended operation. These issues have, therefore, been resolved for the period of extended operation.

4.3.5 Review of ANO-1 Transient Cycle Logging Program

ANO-1 has a process to log transient history and operating transient cycles. Applicable site procedures contain the responsibilities, logging requirements, reporting requirements and transient type definitions. Guidance is provided for collection of the necessary plant data and for projection of the number of transient cycles to the end of plant life. The ANO-1 operating transient cycle logs are retained for the duration of the license, per site procedures and ANO-1 Technical Specifications.

The operational history of ANO-1 is consistent with the assumption in the GLRP reports that only evaluated level A and B events. Entergy Operations will maintain the ANO-1 Transient Cycle Logging Program through the license renewal period.

4.3.6 Flaw Growth Acceptance For ASME Section XI Inservice Inspection Program

Inservice inspections at ANO-1 have, in some cases, identified crack-like indications, primarily in welds. Indications detected during ISI that exceed acceptance criteria are either repaired, replaced or analytically evaluated to demonstrate the indications can be accepted as they are. Analytical evaluations include a crack growth analysis, based on fracture mechanics techniques, to determine the service life of the affected component. Indications that are determined not to grow beyond an acceptable limit are justified for continued operation. These crack growth analyses consider the same design transient cycle assumptions used in the original design. Since these crack growth analyses are performed using the full number of design transients cycles, which have been demonstrated to be applicable for 60 years of operation, these flaw growth calculations remain valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

A relief request for successive inspection of the RCP weld flaws was written by ANO. It relied on the evaluation conducted with Code Case N-481 to provide justification. The NRC contracted the Idaho National Engineering Laboratory to evaluate the request and granted relief on January 4, 1995.

The Code Case N-481 flaw tolerance evaluation was reviewed to determine if the evaluation is acceptable for the period of extended operation. The evaluation demonstrated that the ANO-1 reactor coolant pump casings meet the safety and serviceability requirements of ASME Code Case N-481. Fatigue flaw growth and thermal embrittlement were considered in the assessment. A fatigue crack growth calculation was performed, which included an assumption of 240 heatup and cooldown cycles. Since ANO-1 has not increased the number of design transients for license renewal, the flaw growth evaluation for the pump casing is acceptable for the period of extended operation.

It has, therefore, been demonstrated that the existing results for flaw evaluations performed under the ASME Section XI Inservice Inspection program remain valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

4.3.7 References for Section 4.3

4.3-1 ASME Boiler and Pressure Vessel Code, Section III, "*Rules for Construction of Nuclear Power Plant Components,*" American Society of Mechanical Engineers.

4.3-2 USAS B31.7, "*Nuclear Power Piping,*" USA Standards Institute.

4.3-3 BAW-2251A, "*Demonstration of the Management of Aging Effects for the Reactor Vessel,*" The B&W Owners Group Generic License Renewal Program, June 1996.

4.3-4 1CNA089904, Letter from the NRC to C. Randy Hutchinson, "*Risk-Informed Alternative to Certain Requirements of ASME XI, Table IWB-2500-1 at Arkansas One, Unit 1 (TAC MA 2023*)", Docket No. 50313 , dated August 25, 1999.

4.3-5 ASME Boiler and Pressure Vessel Code, Section XI, "*Rules for In-Service Inspection of Nuclear Power Plant Components,*" American Society of Mechanical Engineers.

4.3-6 W.D. Travers memorandum to A.C. Thadani, closeout of Generic Safety Issue 190, "*Fatigue Evaluation of Metal Components for 60-year plant life,*" dated December 26, 1999

4.4 ENVIRONMENTAL QUALIFICATION

Originally installed electrical equipment at ANO-1 was required to be environmentally qualified to the requirements of NRC IE Bulletin 79-01B, “Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors,” commonly referred to as the Division of Operating Reactors Guidelines. ANO's Environmental Qualification Program complies with the scope of 10CFR50.49 (the EQ rule) requirements. ANO was “grandfathered” by 10CFR50.49 to allow qualification in accordance with DOR Guidelines. Therefore, the current licensing basis for the ANO EQ Program is the DOR Guidelines, as "grandfathered" by 10CFR50.49. The ANO EQ Program includes three main elements identifying applicable equipment and environmental requirements, establishing the qualification, and maintaining (or preserving) that qualification.

The first element involves establishment and control of the EQ Master List of equipment and the service conditions for the harsh environment plant areas. The second element involves establishment and control of the equipment’s EQ documentation, including vendor test reports, vendor correspondence, calculations, evaluations of equipment tested conditions to plant required conditions, and determinations of configuration and maintenance requirements. The third element includes preventive maintenance processes (for replacing parts and the equipment at specified intervals), design control processes (ensuring changes to the plant are evaluated for impact to the EQ Program), procurement processes (ensuring new and replacement equipment is purchased to applicable EQ requirements), and corrective action processes (to identify and correct problems).

As a normal part of the ANO EQ process, when the EQ documentation process establishes that equipment or parts thereof, have a limited life, the preventive maintenance process ensures the equipment or parts are replaced prior to the expiration of the qualified life. If excess conservatism exists in the original qualified life determination, then reanalysis could be performed to extend the qualified life. The reanalysis utilizes standard EQ techniques (such as Arrhenius methodology), and it becomes part of the EQ documentation. Parameter conservatisms may exist in the ambient temperature of the equipment, in unrealistically low activation energy, and in the application of the equipment. The primary method used for reanalysis is reducing excess conservatism in the equipment service temperatures by using temperature values closer to an actual temperature measured in the area around the equipment being reanalyzed.

For EQ equipment with a qualified life less than the required design life of the plant, “ongoing qualification” is a method of long-term qualification involving additional testing. Ongoing qualification or retesting, as described in IEEE Std. 323-1974 (Reference 4.4-1), Section 6.6(1) or (2), is not currently considered by Entergy Operations to be a viable option, and there are no plans to implement such an option. If this option becomes viable in the future, ongoing qualification or re-testing would be performed in accordance with accepted industry and regulatory standards.

ANO-1 has some equipment that was qualified to the DOR Guidelines and other equipment that was qualified in accordance with 10CFR50.49.

The license renewal review preserves the CLB for the component. That is, if the device was originally qualified to the DOR Guidelines, the review will evaluate the extension into the license renewal period utilizing the DOR Guidelines. Each generic equipment EQ documentation file currently lists the qualification criteria applicable to the device, and this qualification criteria is assumed to be the basis of review for the evaluation of the license renewal period.

Pursuant to 10CFR54.21, methods of managing identified aging mechanisms need to be developed and implemented to provide reasonable assurance of preserving intended function(s) that may be degraded by aging. The EQ regulations from the DOR Guidelines to 10CFR50.49 contain requirements for addressing significant aging mechanisms. Therefore, EQ programs meeting these regulations are adequate for managing significant aging mechanisms for the equipment governed by the program. The ANO EQ Program addresses significant aging mechanisms.

Equipment covered by the ANO EQ Program has been evaluated to determine if existing EQ aging analyses can be projected to the end of the period of extended operation by re-analysis or additional analysis. Qualification into the license renewal period will be treated the same as equipment currently qualified at ANO for 40 years or less. When aging analysis cannot justify a qualified life into the license renewal period, then the component or parts will be replaced prior to exceeding its qualified life in accordance with the ANO EQ Program.

EQ equipment has been reevaluated for the environmental service conditions that are applicable to the equipment (i.e., 60 years of exposure versus 40 years). The environmental service conditions are divided into two basic areas; normal and accident. For electrical equipment exposed to a harsh environment, 10CFR50.49 requires consideration of all significant effects from normal service conditions. This would include the expected thermal aging effects from the temperature to which the device is normally exposed and any radiation effects during normal plant operation. 10CFR50.49 also requires an evaluation of the effects from any harsh environments to which the equipment could be exposed under accident conditions. In general, the harsh environments that have been analyzed as part of the EQ Program were initiated by a LOCA, HELBs inside of the reactor building, and HELBs outside of the reactor building.

The evaluation of the environmental service conditions for the license renewal period requires a reevaluation of only the normal aging effects only**.** Therefore, the normal effects from operation for a 60-year period instead of a 40-year period will be evaluated. Radiation effects, thermal aging effects, and wear/cycle aging effects, as applicable, are analyzed for the period of extended operation. The following sections describe each of these considerations in more detail.

Radiation Considerations:

The evaluation of the license renewal period addresses radiation aging qualification. An assumption is made that the normal dose for the license renewal period will be 1.5 times (i.e., 60 years/40 years) the established dose for the 40-year period. The dose values used are based on equipment locations, applying either an inside reactor building value, outside reactor building value or application specific value. The total integrated dose for the 60-year period is determined by adding the established accident dose to the newly determined 60-year normal dose for the device. If the device was qualified for this total integrated dose, no additional review is required.

If the increase in the normal dose by this methodology results in a total integrated dose above the qualified dose for the component, a location specific review may be required to determine the specific component's total dose. Other options include requiring component or part replacement prior to exceeding the qualified total dose or performing radiation surveys to determine actual operating dose and evaluating against this value.

Some components have been installed under a plant modification, and will not experience 60 years of radiation aging by the end of the license renewal period. In some of these cases, credit may be taken for less than 60 years of aging.

Thermal Considerations:

The design temperature for the auxiliary building is 105°F except where equipment walkdowns or temperature monitoring have determined localized elevated temperatures exist. Many components in the auxiliary building are exposed to a much lower temperature than 105°F, for example below 90°F. Temperature monitoring can be utilized to confirm lower than design temperatures exist in these areas, and on that basis, extend qualified lives into the license renewal period.

The ANO-1 reactor building has experienced elevated temperatures in the past and the EQ reviews at one time used temperatures based on limited monitoring and analysis that identified increasing temperatures at the higher elevations. These temperature values have been refined by the collection of additional operating temperature data. Engineering reports are utilized to summarize this data and provide the basis for refining temperature values and equipment service lives.

If extension of the current 40-year life were chosen rather than component replacement, it would be based upon re-evaluation of current aging analysis. The aging analysis will be revised, as applicable, to identify the maximum service life based on traditional EQ techniques such as Arrhenius methodology as determined by the ANO EQ Program.

Some components have been installed under a plant modification, and will not experience 60 years of thermal aging by the end of the license renewal period. In some of these cases, credit may be taken for less than 60 years of aging.

Wear/Cycles Considerations:

The EQ evaluation of the license renewal period will address wear/cycle aging qualification. For most components, this aging does not apply; however, for electromechanical equipment like solenoid valves there would be an associated number of cycles over 40 years. An assumption is made that the number of cycles for the license renewal period would be 1.5 times (i.e., 60 years/40 years) the established number for the 40-year period.

Some components have been installed under a plant modification, and will not experience 60 years of cycling by the end of the license renewal period. In some of these cases, credit may be taken for less than 60 years of aging.

4.4.1 Allis Chalmers Motors

Four small Allis Chalmers motors are installed on the DH room coolers, located in the decay heat pump rooms.

4.4.1.1 Radiation Considerations

The post accident dose of 1.49×10^{7} rads plus the normal aging 60-year life dose of 5.25×10^{6} rads results in a 2.015×10^{7} rads TID, which is still much less than the qualified value of 1×10^{8} rads.

4.4.1.2 Thermal Considerations

The motors only experience temperatures that produce significant aging when they are in operation. Per thermal endurance analysis, the motors were shown to be qualified for 20 years of continuous operation (runtime). This is greater than the expected maximum of 12 accumulated years of operation and 28 years of non-operation at 105^{0}F. The 12 years of operation is sufficiently conservative to allow the units to be extended through the license renewal period since they are only operated for extended periods of time during plant shutdowns. The motors are qualified for greater than 12 years of operation (runtime) and 48 years of shutdown (non-runtime).

4.4.1.3 Wear/Cycles Considerations

The original assumption for the total number of cycles is adequately conservative to allow the device to extend through the license renewal period.

4.4.1.4 Conclusion

Allis Chalmers motor EQ aging analyses have been projected to the end of the period of extended operation.

4.4.2 Anaconda Instrumentation Cable, FR-EP Insulation

Anaconda type FR-EP instrumentation, control and power cable is used in various applications at ANO.

4.4.2.1 Radiation Considerations

The post accident dose of $4x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $2.0x10^8$ rads.

4.4.2.2 Thermal Considerations

The thermal aging test conditions qualify the cable for 40 years at 160°F or 60 years at 154°F. ANO-1's high point vent valves utilize this cable with an average service temperature of 153.6°F (maximum average for these applications), while other EQ equipment using this cable is at 120°F, or below. The high point vent valves were installed in 1982 (less than 53 years needed through the end of the license renewal period).

4.4.2.3 Wear/Cycles Considerations

Not applicable to cable.

4.4.2.4 Conclusion

Anaconda instrumentation cable with FR-EP insulation EQ aging analyses have been projected to the end of the period of extended operation.

4.4.3 Anaconda Control and Power Cable, EP Insulation

Anaconda type EP low and medium voltage control and power cable is used in various applications at ANO.

4.4.3.1 Radiation Considerations

The post accident dose of $4x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $2.0x10^8$ rads.

4.4.3.2 Thermal Considerations

The thermal aging test conditions qualify the cable for 40 years at 188°F or 60 years at 183°F. For ANO-1, EQ equipment using this cable is at 120°F, or below.

4.4.3.3 Wear/Cycles Considerations

Not applicable to cable.

4.4.3.4 Conclusion

Anaconda, low and medium voltage, control and power cable with EP insulation EQ aging analyses have been projected to the end of the period of extended operation.

4.4.4 Anaconda EPR Insulated Instrumentation, Control/Power Cable

Anaconda type EPR, instrumentation, control and power cable is used in various applications at ANO.

4.4.4.1 Radiation Considerations

The post accident dose of $4x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $2.0x10^8$ rads.

4.4.4.2 Thermal Considerations

The thermal aging test conditions qualify the cable for 40 years at 188°F or 60 years at 183°F. For ANO-1, EQ equipment using this cable is at 120°F, or below.

4.4.4.3 Wear/Cycles Considerations

Not applicable to cable.

4.4.4.4 Conclusion

Anaconda, instrumentation, control and power cable with EPR insulation EQ aging analyses have been projected to the end of the period of extended operation.

4.4.5 ASCO Solenoid Valves, Outside Reactor Building

ASCO solenoid valves outside the reactor building are used on the air supplies to the MSIVs, the decay heat cooler bypass valves, and the service water supply valves to the decay heat and reactor building spray pump coolers.

4.4.5.1 Radiation Considerations

The accident dose for the ANO-1 applications of these components is no higher than 1.3695×10^7 rads. Adding to that the normal aging 60-year life dose of 7.95×10^3 rads results in a 1.3703×10^7 rads TID, which is still less than the qualified value of 2×10^7 rads.

4.4.5.2 Thermal Considerations

These solenoids are normally de-energized. The solenoids that are in the decay heat rooms are qualified for greater than 60 years at their normal temperature of 105°F. The solenoids for the MSIVs have a life shorter than 40 years at their temperature of 125°F, and will be replaced prior to exceeding the established life as a normal course of the ANO EQ Program.

4.4.5.3 Wear/Cycles Considerations

The devices are qualified for 40,000 cycles, which is much greater than the 3432 cycles expected over 60 years.

4.4.5.4 Conclusion

ASCO solenoid valves outside reactor building EQ aging analyses have been projected to the end of the period of extended operation for some applications while others are scheduled for replacement prior to exceeding their qualified life. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.6 ASCO Solenoid Valves, Inside Reactor Building

ASCO solenoid valves inside reactor building are used on the air supplies to the building purge valves.

4.4.6.1 Radiation Considerations

The accident dose for the ANO-1 applications of these components is 4.0×10^7 rads. Since these were just installed in 1996 we conservatively add to that the normal aging 40-year life dose of 1.0×10^7 rads which results in a 5.0×10^7 rads TID and this is still less than the qualified value of 2×10^8 rads.

4.4.6.2 Thermal Considerations

These solenoids are normally de-energized and have a life shorter than 40 years at 120°F; therefore, they will be replaced prior to exceeding the established life as a normal course of the ANO EQ Program.

4.4.6.3 Wear/Cycles Considerations

The devices are qualified for 40,000 cycles, which is much greater than the 3432 cycles expected over a 60-year period.

4.4.6.4 Conclusion

The ASCO solenoid valves inside reactor building are scheduled to be replaced prior to exceeding their qualified life; therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.7 Boston Insulated Wire, Instrumentation, Control, and Power cable

Boston insulated wire, type Bostrad7 instrumentation, control, and power cable is used in various applications at ANO.

4.4.7.1 Radiation Considerations

The post accident dose of $4x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $1.0x10^8$ rads.

4.4.7.2 Thermal Considerations

The EQ documentation shows that this cable is qualified for greater than 40 years at 140°F based on separate effects tests and analysis. These cables will be replaced prior to exceeding their qualified life as a normal course of the ANO EQ Program or will be re-evaluated using typical EQ techniques. ANO-1 EQ equipment using this cable is at 140°F, or below.

4.4.7.3 Wear/Cycles Considerations

Not applicable to cable.

4.4.7.4 Conclusion

The Boston insulated wire cables will be replaced prior to exceeding their qualified life or re-evaluated using standard EQ techniques. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.8 Buchanan Terminal Blocks, Outside Reactor Building

Buchanan terminal blocks are installed in various applications outside the reactor building.

4.4.8.1 Radiation Considerations

The post accident dose of 1.48×10^7 rads plus the normal aging 60-year life dose of 5.25×10^6 rads results in a 2.005×10^7 rads TID, which is much less than the qualified value of 2.0×10^8 rads.

4.4.8.2 Thermal Considerations

These terminal blocks have a qualified life of 60 years at 120°F. The ANO-1 applications are at, or below, 120°F.

4.4.8.3 Wear/Cycles Considerations

Not applicable to terminal blocks.

4.4.8.4 Conclusion

Buchanan terminal blocks, outside reactor building, EQ aging analyses have been projected to the end of the period of extended operation.

4.4.9 Buchanan Terminal Blocks, Inside Reactor Building

Buchanan terminal blocks are installed in various applications inside the reactor building, typically inside Limitorque motor operated valve actuators.

4.4.9.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $2.04x10^8$ rads.

4.4.9.2 Thermal Considerations

These terminal blocks have a qualified life of 62 years at 120°F. The ANO-1 applications are at, or below, 120°F.

4.4.9.3 Wear/Cycles Considerations

Not applicable to terminal blocks.

4.4.9.4 Conclusion

Buchanan terminal blocks, inside reactor building, EQ aging analyses have been projected to the end of the period of extended operation.

4.4.10 Conax Thermocouples

Conax thermocouples are installed on the decay heat system to monitor decay heat return temperatures as NRC Regulatory Guide (RG) 1.97, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident,” Category 2 variables.

4.4.10.1 Radiation Considerations

The post accident dose of 1.49×10^7 rads plus the normal aging 60-year life dose of 5.25×10^6 rads results in a 2.015×10^7 rads TID, which is much less than the qualified value of 2.27×10^8 rads.

4.4.10.2 Thermal Considerations

The EQ documentation shows that the thermocouples are qualified for 40 years at a maximum service temperature of 179°F. However, using the 1.27 eV activation energy, 300-302°F test temperatures and 470.25 hour test duration, the equivalent qualified life at 105°F service temperature, for the ANO-1 specific applications, is well in excess of 60 years.

4.4.10.3 Wear/Cycles Considerations

Not applicable to Conax thermocouples.

4.4.10.4 Conclusion

Conax thermocouples EQ aging analyses have been projected to the end of the period of extended operation.

4.4.11 Conax Resistance Temperature Detectors

Conax Resistance Temperature Detectors monitor temperatures for a RG 1.97 Category 1 variable.

4.4.11.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $2.2x10^8$ rads.

4.4.11.2 Thermal Considerations

The EQ documentation shows that the resistance temperature detectors are qualified for 40 years at a maximum service temperature of 250°F. However, using the 3.91 eV activation energy, 302°F test temperatures and 168 hour test duration, the equivalent qualified life at 180°F service temperature, for the ANO-1 specific applications, is well in excess of 60 years.

4.4.11.3 Wear/Cycles Considerations

Not applicable to Conax Resistance Temperature Detectors.

4.4.11.4 Conclusion

Conax Resistance Temperature Detectors EQ aging analyses have been projected to the end of the period of extended operation.

4.4.12 Conax Multipin Connector

Conax multipin connectors have been used in limited applications.

4.4.12.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID, which is much less than the qualified value of 2.25×10^8 rads.

4.4.12.2 Thermal Considerations

The EQ documentation shows that the connectors are qualified for 40 years at a maximum service temperature of 227°F. However, using the 3.81 eV activation energy, 276°F test temperatures and 168 hour test duration, the equivalent qualified life at 180°F service temperature, for the ANO-1 specific applications, is in excess of 60 years.

4.4.12.3 Wear/Cycles Considerations

These devices were installed in 1986 and refueling outages are once every 18 months. Assuming they are mated and unmated every refueling outage, the total number of mating and unmating operations is (2034 - 1986)/1.5 = 32 cycles. The devices were tested for 50 mating and unmating cycles. This is adequate to allow the devices to extend through the license renewal period.

4.4.12.4 Conclusion

Conax multipin connectors EQ aging analyses have been projected to the end of the period of extended operation.

4.4.13 Conax Electrical Penetration Assemblies

ANO-1 has a total of 59 Conax electrical canister type penetrations of which, currently, 19 are spares and 40 are in use.

4.4.13.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $8.7x10^7$ rads.

4.4.13.2 Thermal Considerations

The EQ documentation shows that the assemblies are qualified for 43 years at 140°F. However, using the 1.04 eV activation energy, 302°F test temperatures and 171 hour test duration, the equivalent qualified life at 120°F service temperature, for the ANO-1 applications, is in excess of 60 years.

4.4.13.3 Wear/Cycles Considerations

Not applicable to Conax Electrical Penetration Assemblies.

4.4.13.4 Conclusion

Conax Electrical Penetration Assemblies EQ aging analyses have been projected to the end of the period of extended operation.

4.4.14 Conax Electrical Connection Seal Assembly

ANO-1 utilizes Conax electrical connection seal assemblies in various applications to provide a sealed feedthrough of circuits to the end device, such as Rosemount transmitters.

4.4.14.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID, which is much less than the qualified value of 2.28×10^8 rads.

4.4.14.2 Thermal Considerations

The EQ documentation shows that the Conax electrical connection assemblies are qualified for 60 years at 180°F. The ANO-1 applications are at, or below, 180°F.

4.4.14.3 Wear/Cycles Considerations

Not applicable to Conax Electrical Connection Seal Assembly.

4.4.14.4 Conclusion

Conax Electrical Connection Seal Assembly EQ aging analyses have been projected to the end of the period of extended operation.

4.4.15 Conax Electrical Feedthrough Adapters

ANO-1 utilizes these Conax feedthrough adapters in limited applications.

4.4.15.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $2.0x10^8$ rads.

4.4.15.2 Thermal Considerations

The EQ documentation shows that the adapters are qualified for 60 years at 180°F. The ANO-1 applications are at, or below, 180°F.

4.4.15.3 Wear/Cycles Considerations

Not applicable to Conax feedthrough adapters.

4.4.15.4 Conclusion

Conax feedthrough adapters EQ aging analyses have been projected to the end of the period of extended operation.

4.4.16 Eaton Flame Retardant Ethylene Propylene Diene Monomer Insulated Cable

ANO-1 utilizes Eaton (Samuel-Moore) cable with insulation and a Hypalon jacket (Dekoron).

4.4.16.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID, which is much less than the qualified value of 2.0×10^8 rads.

4.4.16.2 Thermal Considerations

The thermal aging test conditions qualify the cable for 40 years at 194°F or 60 years at 186°F. For ANO-1, EQ equipment using this cable is at 180°F, or below.

4.4.16.3 Wear/Cycles Considerations

Not applicable to Eaton cable.

4.4.16.4 Conclusion

Eaton cable EQ aging analyses have been projected to the end of the period of extended operation.

4.4.17 Gems DeLaval Level Sensors

For ANO-1, these are the reactor building sump and flood level detectors that are RG 1.97 qualified. They are located inside of the reactor building.

4.4.17.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $2.0x10^8$ rads.

4.4.17.2 Thermal Considerations

The EQ documentation shows that the sensors are qualified for 40 years at 120°F. These devices are located at the bottom level of the reactor building. Actual operating temperature in this area is typically less than 90° F. Using the 0.78 eV activation energy, 120°C test temperatures and 2161.3 hour test duration, the equivalent qualified life, at a conservative 110°F service temperature, is in excess of 60 years.

4.4.17.3 Wear/Cycles Considerations

Not applicable to Gems/DeLaval level sensors.

4.4.17.4 Conclusion

Gems/DeLaval level sensors EQ aging analyses have been projected to the end of the period of extended operation.

4.4.18 General Atomic Radiation Detectors

The reactor building high range radiation detectors that are RG 1.97 qualified. These devices are located inside of the reactor building.

4.4.18.1 Radiation Considerations

The radiation detectors have no organic materials and are not affected by radiation.

4.4.18.2 Thermal Considerations

The radiation detectors have no organic materials and are not affected by thermal aging.

4.4.18.3 Wear/Cycles Considerations

Not applicable to General Atomic radiation detectors.

4.4.18.4 Conclusion

General Atomic radiation detectors EQ aging analyses remain valid for the period of extended operation.

4.4.19 General Electric Terminal Blocks

General Electric terminal blocks have been qualified for use inside the reactor building for ANO-1, typically in Limitorque motor operated valve actuators.

4.4.19.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID, which is much less than the qualified value of 2.59×10^8 rads.

4.4.19.2 Thermal Considerations

The EQ documentation shows that the terminal blocks are qualified for 60 years at 180°F. The ANO-1 applications are at, or below, 180°F.

4.4.19.3 Wear/Cycles Considerations

Not applicable to General Electric terminal blocks.

4.4.19.4 Conclusion

General Electric terminal blocks EQ aging analyses have been projected to the end of the period of extended operation.

4.4.20 ITT/General Controls Electro-hydraulic Actuators

These are ITT/General Controls electro-hydraulic actuators installed on the penetration room ventilation dampers located outside the reactor building.

4.4.20.1 Radiation Considerations

The post accident dose of 3.3×10^5 rads plus the normal aging 60-year life dose of 1.32×10^3 rads results in a 3.3132×10^5 rads TID, which is much less than the qualified value of 2.0×10^6 rads.

4.4.20.2 Thermal Considerations

The EQ documentation shows that the actuators are qualified for 40 years at 105°F with specific part replacements at intervals less than 40 years. These will be replaced prior to exceeding their qualified life as a normal course of the ANO EQ Program. The ANO-1 applications are at, or below, 105°F.

4.4.20.3 Wear/Cycles Considerations

These ITT/General Controls electro-hydraulic actuators are cycled once a month, and over 60 years, would be exposed to 792 cycles. The devices were tested for 1250 cycles, which exceeds the required cycles with significant margin.

4.4.20.4 Conclusion

The ITT/General Controls electro-hydraulic actuator parts are scheduled to be replaced prior to exceeding their qualified life; therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.21 Limitorque Motor Operated Valve Actuators; Alternating Current/Inside Reactor Building

Limitorque type SMB motor operated valve actuators with alternating current powered motors are installed inside the reactor building in various applications.

4.4.21.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $2.04x10^8$ rads.

4.4.21.2 Thermal Considerations

The EQ documentation shows that the actuators are qualified for greater than 60 years at 140°F. Most applications are at, or below, 120°F. Applications at temperatures above 140°F have been evaluated separately and replacements are identified as a normal part of the ANO EQ Process.

4.4.21.3 Wear/Cycles Considerations

Most of these motor operated valves will be exposed to an expected 792 cycles over 60 years, while a few (like the sampling valves) are conservatively assumed to be cycled once a week over the 60 years (+ 10%) for a total of 3432 cycles. Limitorque motor operated valves have been tested to 4195 cycles which exceeds the required with significant margin.

4.4.21.4 Conclusion

Limitorque motor operated valves alternating current applications inside reactor building EQ aging analyses have been projected to the end of the period of extended operation for most applications while others have scheduled replacements prior to exceeding their qualified life. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.22 Limitorque Motor Operated Valve Actuators; Alternating Current/Outside Reactor Building

Limitorque type SMB motor operated valve actuators with alternating current powered motors are installed outside the reactor building in various applications.

4.4.22.1 Radiation Considerations

The highest combination of post accident dose and normal aging 60-year life dose for the areas involved is $1.37x10^7$ rads TID, which is less than the qualified value of $2.0x10^7$ rads.

4.4.22.2 Thermal Considerations

The EQ documentation shows that the actuators are qualified for greater than 60 years at 140°F. The ANO-1 alternating current applications outside reactor building are at, or below, 140°F.

4.4.22.3 Wear/Cycles Considerations

These Limitorque motor operated valves are cycled once a month, or less, and over 60 years would be exposed to 792 cycles. The devices were tested for at least 1993 cycles, which exceeds the required with significant margin.

4.4.22.4 Conclusion

Limitorque motor operated valves alternating current applications outside reactor building EQ aging analyses have been projected to the end of the period of extended.

4.4.23 Limitorque Motor Operated Valve Actuators; Direct Current/Outside Reactor Building

Limitorque type SMB motor operated valve actuators with direct current powered motors are installed outside the reactor building in various applications. The current EQ documentation conservatively evaluates these devices for inside reactor building applications.

4.4.23.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID, which is much less than the qualified value of 2.04×10^8 rads.

4.4.23.2 Thermal Considerations

The EQ documentation shows that the actuators are qualified for greater than 60 years at 140°F. The ANO-1 applications are at, or below, 120°F.

4.4.23.3 Wear/Cycles Considerations

These Limitorque motor operated valves are cycled once a month, or less, and over 60 years would be exposed to 792 cycles. The devices were tested for 2004 cycles, which exceeds the required with significant margin.

4.4.23.4 Conclusion

Limitorque motor operated valves direct current applications outside reactor building EQ aging analyses have been projected to the end of the period of extended operation.

4.4.24 NAMCO EA-170 Limit Switches

NAMCO EA-170 Series limit switches are used to provide position indication.

4.4.24.1 Radiation Considerations

The post accident dose of 1.48×10^7 rads plus the normal aging 60-year life dose of 5.25×10^6 rads results in a 2.005×10^7 rads TID, which is much less than the qualified value of 2.04×10^8 rads.

4.4.24.2 Thermal Considerations

These switches have a life shorter than 40 years at their service temperature of 105°F. They will be replaced prior to exceeding the established life as a normal course of the ANO EQ Program.

4.4.24.3 Wear/Cycles Considerations

The devices are qualified for 100,000 cycles, which is much greater than the 792 cycles expected over a 60-year period.

4.4.24.4 Conclusion

The NAMCO EA-170 Series limit switches are scheduled to be replaced prior to exceeding their qualified life; therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.25 NAMCO EA-180 Limit Switches

NAMCO EA-180 Series limit switches are used to provide position indication.

4.4.25.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID, which is much less than the qualified value of 2.2×10^{10} rads.

4.4.25.2 Thermal Considerations

These switches have a life shorter than 40 years at their service temperature of 120°F. They will be replaced prior to exceeding the established life as a normal course of the ANO EQ Program.

4.4.25.3 Wear/Cycles Considerations

The devices are qualified for 100,300 cycles, which is much greater than the 3432 cycles expected over a 60-year period.

4.4.25.4 Conclusion

The NAMCO EA-180 Series limit switches are scheduled to be replaced prior to exceeding their qualified life; therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.26 NAMCO EA-740 Limit Switches with NAMCO Connectors

NAMCO EA-740 Series limit switches with NAMCO connectors are used to provide position indication. The switch and connector are qualified by the same report. The ANO-1 applications of these grouped devices are currently limited to outside the reactor building.

4.4.26.1 Radiation Considerations

The post accident dose of 1.48×10^7 rads plus the normal aging 60-year life dose of 5.25×10^6 rads results in a 2.005×10^7 rads TID, which is much less than the qualified value of 2.04×10^8 rads.

4.4.26.2 Thermal Considerations

These switches have a life of 47.1 years at their service temperature of 105°F. The current applications were installed in 1986, therefore, a life of 48 years is needed (2034 - 1986 = 48) to extend through the renewal period. Therefore, they will be replaced prior to exceeding the established life as a normal course of the ANO EQ Program, or will be re-evaluated using typical EQ techniques such as reducing the assumed service temperature by using actual plant operating temperature data.

4.4.26.3 Wear/Cycles Considerations

The devices are qualified for 100,000 cycles, which is much greater than the 3,432 cycles expected over a 60-year period.

4.4.26.4 Conclusion

The NAMCO EA-740 Series limit switches with NAMCO connectors will be replaced prior to exceeding their qualified life, or re-evaluated using standard EQ techniques. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.27 NAMCO EA-740 Limit Switches

NAMCO EA-740 Series limit switches are used to provide position indication.

4.4.27.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $2.04x10^8$ rads.

4.4.27.2 Thermal Considerations

The ANO-1 switches have a life of 53 years at their service temperature of 106°F. The applications were installed no earlier than December 1984 (2034 – 1984 = 50 years needed). Therefore, EQ Documentation shows that the life will extend through the renewal period.

4.4.27.3 Wear/Cycles Considerations

The devices are qualified for 100,322 cycles, which is much greater than the 792 cycles expected over a 60-year period.

4.4.27.4 Conclusion

The NAMCO EA-740 Series limit switches EQ aging analyses have been projected to the end of the period of extended operation.

4.4.28 NAMCO Quick Connectors

NAMCO connectors are used in various applications to provide a sealed feedthrough of circuits to the end device, such as Rosemount transmitters.

4.4.28.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $2.04x10^8$ rads.

4.4.28.2 Thermal Considerations

These connectors have a life greater than 60 years at 120°F. The ANO-1 applications are at or below 120°F.

4.4.28.3 Wear/Cycles Considerations

Not applicable to NAMCO connectors.

4.4.28.4 Conclusion

The NAMCO connectors EQ aging analyses have been projected to the end of the period of extended operation.

4.4.29 Okonite 5 kV Power Cable with EPR insulation and an Okolon jacket

ANO-1 utilizes Okonite 5 kV power cable in limited EQ applications.

4.4.29.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID, which is much less than the qualified value of 2.0×10^8 rads.

4.4.29.2 Thermal Considerations

The thermal aging test conditions qualify the cable for 60 years at 150°F. EQ equipment using this cable is at 120°F, or below.

4.4.29.3 Wear/Cycles Considerations

Not applicable to Okonite cable.

4.4.29.4 Conclusion

The Okonite 5 kV power cable EQ aging analyses have been projected to the end of the period of extended operation.

4.4.30 Okonite 2 kV Power and Control Cable with Okonite or Okoguard insulation and Okoprene or Okolon jackets

ANO-1 utilizes Okonite 2 kV Power and Control cable in various applications.

4.4.30.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID, which is much less than the qualified value of 2.0×10^8 rads.

4.4.30.2 Thermal Considerations

The thermal aging test conditions qualify the cable for 60 years at 150°F. EQ equipment using this cable is at 120°F, or below.

4.4.30.3 Wear/Cycles Considerations

Not applicable to Okonite cable.

4.4.30.4 Conclusion

The Okonite 2 kV power cable EQ aging analyses have been projected to the end of the period of extended operation.

4.4.31 Okonite 600V power cable with Okonite insulation and an Okolon jacket

ANO-1 utilizes Okonite 600V power cable with Okonite insulation and an Okolon jacket in various applications.

4.4.31.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $2.0x10^8$ rads.

4.4.31.2 Thermal Considerations

The thermal aging test conditions qualify the cable for 40 years at 158°F or 60 years at 150°F. EQ equipment using this cable is at 156°F, or below. Cables currently identified at temperatures above 150°F will be replaced prior to exceeding their qualified life as a normal course of the ANO EQ Program or re-evaluated using standard EQ techniques.

4.4.31.3 Wear/Cycles Considerations

Not applicable to Okonite cable.

4.4.31.4 Conclusion

For most applications of this Okonite 600V cable, the EQ aging analyses have been projected to the end of the period of extended operation. Specific applications will be replaced prior to exceeding their qualified life, or re-evaluated using standard EQ techniques. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.32 Okonite 600V power cable with FMR insulation

ANO-1 utilizes Okonite 600V power cable with FMR insulation in various applications.

4.4.32.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $2.0x10^8$ rads.

4.4.32.2 Thermal Considerations

The thermal aging test conditions qualify the cable for 40 years at 158°F or 60 years at 150°F. Additional analyses are used to compute a life in excess of 40 years at 180°F. Most EQ equipment using this cable is at 120°F, or below. Specific applications of this cable are at an average temperature of 165°F. Cables currently identified at temperatures above 150°F will be replaced prior to exceeding their qualified life as a normal course of the ANO EQ Program, or re-evaluated using standard EQ techniques.

4.4.32.3 Wear/Cycles Considerations

Not applicable to Okonite cable.

4.4.32.4 Conclusion

For most applications of this Okonite 600V cable, the EQ aging analyses have been projected to the end of the period of extended operation. Specific applications will be replaced prior to exceeding their qualified life, or re-evaluated using standard EQ techniques. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.33 Okonite T-95 and No. 35 Splicing Tapes

ANO-1 utilizes Okonite tape splices made of the T-95 and No. 35 tapes in various applications.

4.4.33.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID, which is much less than the qualified value of 2.0×10^8 rads.

4.4.33.2 Thermal Considerations

Thermal aging test conditions qualify the tape splices for 40 years at 165°F, or 60 years at 158°F. Additional analyses are used to establish a life of 40 years at 90°C (194°F). Most applications are at, or below, 150°F. For those above 158°F, replacement is required prior to exceeding the qualified life.

4.4.33.3 Wear/Cycles Considerations

Not applicable to Okonite tape splices.

4.4.33.4 Conclusion

The Okonite tape splice EQ aging analyses have been projected to the end of the period of extended operation, for most applications, while others have scheduled replacements prior to exceeding their qualified life. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.34 Raychem 600V Flamtrol XLPE Cable

ANO-1 utilizes Raychem 600V power, control and instrumentation cable in various applications.

4.4.34.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $2.0x10^8$ rads.

4.4.34.2 Thermal Considerations

The thermal aging test conditions qualify the cable for greater than 40 years at 150°F or 60 years at 143°F. EQ equipment using this cable is at 120°F, or below.

4.4.34.3 Wear/Cycles Considerations

Not applicable to Raychem cable.

4.4.34.4 Conclusion

The Raychem cable EQ aging analyses have been projected to the end of the period of extended operation.

4.4.35 Raychem Cable Splice and Jacket Repair Tape (type NJRT)

ANO-1 utilizes Raychem cable splice and jacket repair tape (type NJRT) in limited applications.

4.4.35.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID, which is much less than the qualified value of 2.15×10^8 rads.

4.4.35.2 Thermal Considerations

The EQ documentation shows that the tapes are qualified for 40 years at 194°F. This is equivalent to greater than 60 years at 180 °F using standard EQ techniques. The ANO-1 applications are at, or below, 180 °F.

4.4.35.3 Wear/Cycles Considerations

Not applicable to Raychem repair tapes.

4.4.35.4 Conclusion

The Raychem repair tapes EQ aging analyses have been projected to the end of the period of extended operation.

4.4.36 Raychem Cable Splices (types WCSF-N, NPK, NMCK, ANK, etc.)

ANO-1 utilizes Raychem cable splices in various applications, mostly in instrumentation circuits.

4.4.36.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $2.0x10^8$ rads.

4.4.36.2 Thermal Considerations

The EQ documentation shows that the splices are qualified for 40 years at 194°F. This is equivalent to greater than 60 years at 180 °F using standard EQ techniques. The ANO-1 applications are at, or below, 180 °F.

4.4.36.3 Wear/Cycles Considerations

Not applicable to Raychem splices.

4.4.36.4 Conclusion

The Raychem splices EQ aging analyses have been projected to the end of the period of extended operation.

4.4.37 Reliance Electric, Electric Motors

ANO-1 utilizes Reliance Electric, electric motors in limited applications. Four large motors are installed on the reactor building coolers and four other smaller motors are installed in the reactor building coolers dropout dampers.

4.4.37.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID, which is much less than the qualified value of 1.0×10^9 rads.

4.4.37.2 Thermal Considerations

The EQ documentation shows that the motors are qualified for greater than 60 years at 120 °F. The ANO-1 applications are at, or below 120 °F.

4.4.37.3 Wear/Cycles Considerations

The devices are qualified for 1024 cycles, which is much greater than 704 cycles expected over a 60-year period.

4.4.37.4 Conclusion

The Reliance Electric, electric motors EQ aging analyses have been projected to the end of the period of extended operation.

4.4.38 Rockbestos Coaxial Cable

ANO-1 utilizes Rockbestos coaxial cable in various applications.

4.4.38.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $2.0x10^8$ rads.

4.4.38.2 Thermal Considerations

The thermal aging test conditions qualify the cable for 40 years at 155°F or 60 years at 150°F. Additional analyses are used to compute a life in excess of 40 years at 180°F. Most EQ equipment using this cable is at 120°F, or below. Specific applications of this cable are at 180°F, or below. Cables currently identified at temperatures above 150°F will be replaced prior to exceeding their qualified life as a normal course of the ANO EQ Program, or re-evaluated using standard EQ techniques.

4.4.38.3 Wear/Cycles Considerations

Not applicable to Rockbestos cable.

4.4.38.4 Conclusion

For most applications of this Rockbestos coaxial cable, the EQ aging analyses have been projected to the end of the period of extended operation. Specific applications will be replaced prior to exceeding their qualified life, or re-evaluated using standard EQ techniques. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.39 Rockbestos Firewall III Irradiation Cross-Linked Polyethylene Cable

ANO-1 utilizes Rockbestos Firewall III irradiation cross-linked polyethylene cable in various applications.

4.4.39.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $2.0x10^8$ rads.

4.4.39.2 Thermal Considerations

The EQ documentation shows that the cables are qualified for 60 years at 190°F. The ANO-1 applications are at, or below, 120°F.

4.4.39.3 Wear/Cycles Considerations

Not applicable to Rockbestos cable.

4.4.39.4 Conclusion

The Rockbestos Firewall III irradiation cross-linked polyethylene cable EQ aging analyses have been projected to the end of the period of extended operation.

4.4.40 Rockbestos Firezone R Silicone Rubber High Temperature Cable

ANO-1 utilizes Rockbestos Firezone R silicone rubber high temperature cable in limited applications.

4.4.40.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID. This cable was tested to 5.0×10^7 rads and additional analysis shows it can withstand up to 1.5×10^8 rads.

4.4.40.2 Thermal Considerations

The EQ documentation shows that the cables are qualified for 40 years at 181°F, or 60 years at 176°F. The ANO-1 applications are at, or below, 180°F. Cables currently identified at temperatures above 176°F will be replaced prior to exceeding their qualified life as a normal course of the ANO EQ Program, or re-evaluated using standard EQ techniques.

4.4.40.3 Wear/Cycles Considerations

Not applicable to Rockbestos cable.

4.4.40.4 Conclusion

For some applications of this Rockbestos Firezone R silicone rubber high-temperature cable, the EQ aging analyses have been projected to the end of the period of extended operation. Other applications will be replaced prior to exceeding their qualified life or re-evaluated using standard EQ techniques. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.41 Rockbestos Firewall III Chemically Cross-Linked Polyethylene Cable

ANO-1 utilizes Rockbestos Firewall III chemically cross-linked polyethylene cable in various applications.

4.4.41.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $2.0x10^8$ rads.

4.4.41.2 Thermal Considerations

The EQ documentation shows that the cables are qualified for 60 years at 190°F. The ANO-1 applications are at, or below, 120°F.

4.4.41.3 Wear/Cycles Considerations

Not applicable to Rockbestos cable.

4.4.41.4 Conclusion

The Rockbestos Firewall III chemically cross-linked polyethylene cable EQ aging analyses have been projected to the end of the period of extended operation.

4.4.42 Rosemount Model 1153 Series D Pressure Transmitters

Rosemount Model 1153 Series D pressure transmitters are used in various applications.

4.4.42.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID. Output code R transmitters (improved radiation performance) are qualified for 1.1×10^8 rads and standard transmitters are qualified for 5.2×10^7 rads. The transmitters inside the reactor building are Type R, which are qualified for the high radiation. The radiation outside the reactor building is lower than the qualification of the standard transmitters, therefore, they are qualified.

4.4.42.2 Thermal Considerations

The transmitter, circuit boards, and O-rings are already identified in the ANO EQ Program as requiring replacement at specific intervals based on the service temperature of the specific transmitter.

4.4.42.3 Wear/Cycles Considerations

Not applicable to Rosemount transmitters.

4.4.42.4 Conclusion

The ANO EQ Program requires replacements prior to exceeding the qualified life; therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.43 Rosemount Model 1154 Pressure Transmitters

Rosemount model 1154 pressure transmitters are used in various applications.

4.4.43.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $1.1x10^8$ rads.

4.4.43.2 Thermal Considerations

The transmitter, circuit boards, and O-rings are already identified in the ANO EQ Program as requiring replacement at specific intervals based on the service temperature of the specific transmitter.

4.4.43.3 Wear/Cycles Considerations

Not applicable to Rosemount transmitters.

4.4.43.4 Conclusion

The ANO EQ Program requires replacements prior to exceeding the qualified life; therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.44 Rotork Motor Operated Valve Actuators, Model NA1

Rotork motor operated valve actuators are used in specific applications.

4.4.44.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is less than the minimum qualified value of $6.0x10^7$ rads.

4.4.44.2 Thermal Considerations

The EQ documentation shows that the actuators are qualified for greater than 60 years at 120°F. The ANO-1 applications are at, or below, 120°F.

4.4.44.3 Wear/Cycles Considerations

The devices are qualified for 2,000 cycles, which is greater than the 792 cycles expected over a 60-year period.

4.4.44.4 Conclusion

The Rotork motor operated valve actuators EQ aging analyses have been projected to the end of the period of extended operation.

4.4.45 Target Rock Solenoid Operated Valves (Report 2375)

Target Rock solenoid operated valves are used in various applications.

4.4.45.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $1.35x10^8$ rads.

4.4.45.2 Thermal Considerations

The EQ documentation shows that the valves are qualified for greater than 60 years at 180°F. The ANO-1 applications are at, or below, 180°F.

4.4.45.3 Wear/Cycles Considerations

The devices are qualified for 18,000 cycles, which is greater than the conservatively assumed one cycle per week (+ 10%), or a total of 3,432 cycles expected over a 60-year period.

4.4.45.4 Conclusion

The Target Rock solenoid operated valves EQ aging analyses have been projected to the end of the period of extended operation.

4.4.46 Target Rock Solenoid Operated Valves (Reports 2375 and 3996)

Target Rock solenoid operated valves are used in specific applications.

4.4.46.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID, which is much less than the qualified value of 1.35×10^8 rads.

4.4.46.2 Thermal Considerations

The solenoid operated valves are already identified in the ANO EQ Program as requiring replacement at specific intervals based on the service temperature of the specific valve.

4.4.46.3 Wear/Cycles Considerations

The devices are qualified for 18,000 cycles, which is greater than the conservatively assumed one cycle per week (+ 10%), or total of 3,432 cycles, expected over a 60-year period.

4.4.46.4 Conclusion

The ANO EQ Program requires replacements prior to exceeding the qualified life; therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.47 Target Rock Modulating Solenoid Operated Valves (Report 3414)

Target Rock modulating solenoid operated valves are used in only one EQ application, emergency feedwater flow control. The valve was installed in 1984 and needs a 50 to 51-year life (2034 - 1984 = 50 years) for the period of extended operation.

4.4.47.1 Radiation Considerations

The post accident dose of $4.0\text{x}10^7$ rads plus the normal aging 60-year life dose of $1.5\text{x}10^7$ rads results in a $5.5\text{x}10^7$ rads TID, which is less than the qualified value of $7.07\text{x}10^7$ rads.

4.4.47.2 Thermal Considerations

The EQ documentation shows that the valves are qualified for 53 years at 105°F. The ANO-1 application is outside reactor building and is at or below 105°F.

4.4.47.3 Wear/Cycles Considerations

The devices are qualified for 2,000 full cycles and 99,565 partial cycles, which is greater than the 1,144 cycles expected over a 60-year period.

4.4.47.4 Conclusion

The Target Rock solenoid operated valve EQ aging analyses have been projected to the end of the period of extended operation.

4.4.48 Target Rock Solenoid Operated Valves (Reports 2375 and 1827)

Target Rock solenoid operated valves are used in specific applications.

4.4.48.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $1.35x10^8$ rads.

4.4.48.2 Thermal Considerations

The EQ documentation shows that the valves are qualified for greater than 60 years at 120°F. The ANO-1 applications are at, or below, 120°F.

4.4.48.3 Wear/Cycles Considerations

The devices are qualified for 18,000 cycles, which is greater than the conservatively assumed one cycle per week (+ 10%), or total of 3,432 cycles expected over a 60-year period.

4.4.48.4 Conclusion

The Target Rock solenoid operated valve EQ aging analyses have been projected to the end of the period of extended operation.

4.4.49 TEC Valve Flow Monitoring System

These are the pressurizer safety valve and PORV flow monitoring system, which consists of accelerometers, hardline cable, cable couplers, pre-amplifiers, and transient shields.

4.4.49.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID, which is much less than the qualified value of 2.02×10^8 rads.

4.4.49.2 Thermal Considerations

The EQ documentation supports qualified lives greater than 60 years at 180°F for several subcomponents. Other subcomponents have a qualified life of less than 40 years and will be replaced as a normal course of the ANO EQ Program or re-evaluated using standard EQ techniques. One subcomponent has been determined to be qualified for 49 years at 160°F and will be replaced prior to exceeding the life as a normal course of the ANO EQ Program, or re-evaluated using standard EQ techniques.

4.4.49.3 Wear/Cycles Considerations

Not applicable to the valve flow monitoring system.

4.4.49.4 Conclusion

For some subcomponents, the EQ aging analyses have been projected to the end of the period of extended operation. Others will be replaced prior to exceeding their qualified life or re-evaluated using standard EQ techniques. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.50 TEC Reactor Vessel Level Monitoring System

The reactor vessel level monitoring systems, consists of level sensors, cable, cable connectors, and splices.

4.4.50.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID, which is much less than the qualified value of 1.1×10^8 rads. The qualified life of the TEC Model 125 RLI probe is 35 years, based on the expected neutron flux inside of the reactor.

4.4.50.2 Thermal Considerations

The EQ documentation shows that most of the subcomponents are qualified for 40 years or less and are currently identified as requiring replacement as a normal course of the ANO EQ Program. Other subcomponents have a qualified life of greater than 60 years at their associated service temperature.

4.4.50.3 Wear/Cycles Considerations

The connectors have been qualified for 160 cycles which is greater than the 40 cycles expected over a 60 year period.

4.4.50.4 Conclusion

For some subcomponents, the EQ aging analyses have been projected to the end of the period of extended operation. Most subcomponents will be replaced prior to exceeding their qualified life, or re-evaluated using standard EQ techniques. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.51 Weed Resistance Temperature Detectors

Several Weed Resistance Temperature Detectors that are installed as a part of the reactor cooling system hot legs level monitoring system as a RG 1.97 Category 1 variable.

4.4.51.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $3.03x10^8$ rads.

4.4.51.2 Thermal Considerations

The ANO-1 applications have a life of 55.5 years at 180°F for 4 years and 155°F for the remainder and were installed in 1986 (2034 – 1986 = 48 years needed). Therefore, the EQ Documentation indicates the life will extend through the renewal period.

4.4.51.3 Wear/Cycles Considerations

Not applicable to resistance temperature detectors.

4.4.51.4 Conclusion

The Weed resistance temperature detectors EQ aging analyses have been projected to the end of the period of extended operation.

4.4.52 Dow-Corning 3145 Silicone Sealant

Dow-Corning silicone sealant is used in various applications.

4.4.52.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID, which is much less than the qualified value of 2.2×10^8 rads.

4.4.52.2 Thermal Considerations

The EQ documentation shows that this sealant is qualified for greater than 60 years at 180°F. The ANO-1 applications are at or below 180°F.

4.4.52.3 Wear/Cycles Considerations

Not applicable to this silicone sealant.

4.4.52.4 Conclusion

The Dow-Corning silicone sealant EQ aging analyses have been projected to the end of the period of extended operation.

4.4.53 Westinghouse Hydrogen Recombiners

Two hydrogen recombiners were added to the ANO-1 reactor building in 1986. These recombiners are primarily constructed of metallic materials, metal enclosed thermal insulation and metal clad ceramic insulated heater elements, which are considered non-age sensitive materials. The age sensitive materials are the Boston Insulated Wire heater power cable and the Firezone 101 heater connection cable. Based on information from Westinghouse, the Firezone 101 is not age or radiation sensitive except for the Teflon finisher, which is not required for the cable to perform its safety function.

4.4.53.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID, which is much less than the qualified value of 2.0×10^8 rads.

4.4.53.2 Thermal Considerations

The EQ documentation indicates the Boston Insulated Wire cable is qualified in this application for 44.4 years at 156°F for 5 years and 120°F for the remainder. This will not be enough life to allow it to extend through the license renewal period. This cable will be replaced prior to exceeding the qualified life as a normal course of the ANO EQ Program, or re-evaluated using standard EQ techniques.

4.4.53.3 Wear/Cycles Considerations

Not applicable to the hydrogen recombiners.

4.4.53.4 Conclusion

The Boston Insulated Wire cable to the recombiners will be replaced prior to exceeding the qualified life, or re-evaluated using standard EQ techniques. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.54 Westinghouse Motors, Model ABDP

Seven pumps utilize Westinghouse motors in the decay heat, reactor building spray, and makeup/HPI systems, which are located outside the reactor building.

4.4.54.1 Radiation Considerations

The post accident dose of 1.48×10^7 rads plus the normal aging 60-year life dose of 5.25×10^6 rads results in a 2.005×10^7 rads TID, which is much less than the qualified value of 1.4×10^8 rads.

4.4.54.2 Thermal Considerations

The EQ documentation shows that the Westinghouse motors are qualified in these applications for 40 years at 105°F. These motors will be replaced prior to exceeding the qualified life as a normal course of the ANO EQ Program, or re-evaluated using standard EQ techniques.

4.4.54.3 Wear/Cycles Considerations

The motors are qualified for 1,000 cycles, which is more than the 792 cycles expected over a 60-year period.

4.4.54.4 Conclusion

These Westinghouse motors will be replaced prior to exceeding their qualified life, or re-evaluated using standard EQ techniques. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.55 Westinghouse Motors, Models TBFC and SBDP

Applications of these motors are the penetration room ventilation fan motors, located outside reactor building.

4.4.55.1 Radiation Considerations

The post accident dose of $1.49x10^7$ rads plus the normal aging 60-year life dose of $5.25x10^6$ rads results in a $2.015x10^7$ rads TID, which is much less than the qualified value of $1.4x10^8$ rads.

4.4.55.2 Thermal Considerations

The EQ documentation shows that these Westinghouse motors are qualified for 78,840 hours of operation at their maximum operating temperature of 120°C. During plant operation, the motors do not run (i.e., only run during surveillances). The surveillances would, conservatively, run the motors 160 hours every 18 months, which would equal 6,400 hours in 60 years. Add a conservative 1-year post-accident operating time (8,760 hours) and a conservative total run-time would be 15,160 hours. This is significantly less than the 78,840 hours for which the motor is qualified and would conservatively allow it to extend through the license renewal period.

4.4.55.3 Wear/Cycles Considerations

The motors are qualified for 1,000 cycles, which is more than the 880 cycles expected over a 60-year period.

4.4.55.4 Conclusion

These Westinghouse motor EQ aging analyses have been projected to the end of the period of extended operation.

4.4.56 Babcock & Wilcox Core Exit Thermocouples

These are the 24 incore detectors with thermocouples that are RG 1.97 qualified.

4.4.56.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID, which is much less than the qualified value of 2.07×10^8 rads.

4.4.56.2 Thermal Considerations

The EQ documentation shows that some subcomponents of these devices have a qualified life of less than 40 years. The thermocouple, detector probe, socket connector, and O-ring have qualified lives less than 40 years at its service temperature of 120°F. The pin half connector with the mineral-insulated cable is qualified for greater than 60 years at their service temperature of 120^0 F. Therefore, the EQ program already identifies the necessary replacements that support plant operation in the license renewal period.

4.4.56.3 Wear/Cycles Considerations

The connectors are qualified for 50 mating/unmatings. These are unmated and pulled each refueling outage (i.e. once every 18 months over 60 years or 40 times).

4.4.56.4 Conclusion

For the pin half connector with the mineral-insulated cable, the EQ aging analyses have been projected to the end of the period of extended operation. Other subcomponents will be replaced prior to exceeding their qualified life as a normal course of the ANO EQ Program, or re-evaluated using standard EQ techniques. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.57 Gamma Metrics Neutron Detectors and Cable Assemblies

Two Gamma Metrics excore neutron detectors and cable assemblies that were installed in 1984 and are RG 1.97 qualified. The detector assemblies (detector and mineral-insulated cable) are located in wells in the concrete biological shield wall next to the reactor vessel. The cable assemblies (organic connecting cables) and junction boxes are located outside the D-ring walls in the general areas of the reactor building.

4.4.57.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $3.2x10^9$ rads (detector assembly and cable assemblies) and $6.0x10^7$ (junction box o-ring). Given their location, these detectors are also exposed to neutron radiation. The current ANO EQ documentation shows that the detector assemblies have a qualified life of just over 40 years, based on the expected 40-year neutron flux dose of $2.762x10^9$ rads to the detector assembly. These will be replaced prior to exceeding their qualified life, or re-evaluated using standard EQ techniques. The cable assemblies and junction boxes are qualified for 60 years.

4.4.57.2 Thermal Considerations

The EQ documentation shows that the detector assemblies and junction box O-ring have a qualified life of 40 years at their service temperature of 120°F. These will be replaced prior to exceeding their qualified life as a normal course of the ANO EQ Program, or re-evaluated using standard EQ techniques. Mineral-insulated cable extending from the detector is non-age sensitive. The organic cable is qualified for 50 years at 180°F while the application is at, or below 120°F. This 50 year life is enough to extend through the renewal period, given the installation date of the system.

4.4.57.3 Wear/Cycles Considerations

Not applicable to the Gamma Metrics components.

4.4.57.4 Conclusion

For the organic cable, the EQ aging analyses have been projected to the end of the period of extended operation. Other subcomponents will be replaced prior to exceeding their qualified life, or re-evaluated using standard EQ techniques. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.58 Brand Rex Cross-Linked Polyethylene Coaxial Cable

Brand Rex cross-linked polyethylene coaxial cable is used in various applications.

4.4.58.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $2.0x10^8$ rads.

4.4.58.2 Thermal Considerations

The EQ documentation indicates that this cable is qualified for 60 years at 150°F. The ANO-1 applications are at, or below, 120°F.

4.4.58.3 Wear/Cycles Considerations

Not applicable to Brand Rex cable.

4.4.58.4 Conclusion

The Brand Rex cross-linked polyethylene coaxial cable EQ aging analyses have been projected to the end of the period of extended operation.

4.4.59 Brand Rex Cross-Linked Polyethylene Power and Control Cable

Brand Rex cross-linked polyethylene power and control cable is used in various applications.

4.4.59.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $2.0x10^8$ rads.

4.4.59.2 Thermal Considerations

The EQ documentation shows that this cable is qualified for 60 years at 150°F. The ANO-1 applications are at, or below, 120°F.

4.4.59.3 Wear/Cycles Considerations

Not applicable to Brand Rex cable.

4.4.59.4 Conclusion

The Brand Rex cross-linked polyethylene power and control cable EQ aging analyses have been projected to the end of the period of extended operation.

4.4.60 NDT International Acoustic Sensor, Connector and Cable

These are the 16 main steam safety valve position indicators that were added to the main steam safety valves as RG1.97 Category 2 variables.

4.4.60.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID, which is much less than the qualified value of 2.0×10^8 rads.

4.4.60.2 Thermal Considerations

The EQ documentation shows that these devices are qualified for greater than 60 years at their maximum service temperature of 180°F.

4.4.60.3 Wear/Cycles Considerations

Not applicable to the sensors, connectors, and cable.

4.4.60.4 Conclusion

The NDT International acoustic sensor, connector and cable EQ aging analyses have been projected to the end of the period of extended operation.

4.4.61 American Insulated Wire 600V Instrumentation Cable

American Insulated Wire 600V cross-linked polyethylene insulation, Hypalon jacket, instrumentation cable is used in various applications.

4.4.61.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID, which is much less than the qualified value of 1.0×10^8 rads.

4.4.61.2 Thermal Considerations

The EQ documentation shows that this cable is qualified for 60 years at 125°F. The ANO-1 applications are at, or below, 120°F.

4.4.61.3 Wear/Cycles Considerations

Not applicable to American Insulated Wire cable.

4.4.61.4 Conclusion

The American Insulated Wire 600V cross-linked polyethylene insulation, Hypalon jacket, instrumentation cable EQ aging analyses have been projected to the end of the period of extended operation.

4.4.62 American Insulated Wire 600V Power and Control Cable

American Insulated Wire 600V EPR insulation, Hypalon jacket, power, and control cable is used in various applications.

4.4.62.1 Radiation Considerations

The post accident dose of 4.0×10^7 rads plus the normal aging 60-year life dose of 1.5×10^7 rads results in a 5.5×10^7 rads TID, which is much less than the qualified value of 2.06×10^8 rads.

4.4.62.2 Thermal Considerations

The EQ documentation shows that this cable is qualified for 60 years at 123°F. The ANO-1 applications are at, or below, 120°F.

4.4.62.3 Wear/Cycles Considerations

Not applicable to American Insulated Wire cable.

4.4.62.4 Conclusion

The American Insulated Wire 600V EPR insulation, Hypalon jacket, power, and control cable EQ aging analyses have been projected to the end of the period of extended operation.

4.4.63 AMP Pre-insulated Butt Splices

AMP pre-insulated butt splices are limited to outside reactor building in harsh radiation only applications.

4.4.63.1 Radiation Considerations

The post accident dose of 1.49×10^7 rads plus the normal aging 60-year life dose of 5.25×10^6 rads results in a 2.015×10^7 rads TID, which is much less than the qualified value of 2.59×10^8 rads.

4.4.63.2 Thermal Considerations

The EQ documentation shows that these are qualified for 40 years at 194°F. This is equivalent to much greater than 60 years at 105°F using standard EQ techniques. The ANO-1 applications are at, or below, 105°F.

4.4.63.3 Wear/Cycles Considerations

Not applicable to AMP butt splices.

4.4.63.4 Conclusion

The AMP pre-insulated butt splices EQ aging analyses have been projected to the end of the period of extended operation.

4.4.64 EGS Quick Disconnect Electrical Connectors

EGS quick disconnect electrical connectors are used in various applications.

4.4.64.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $2.0x10^8$ rads.

4.4.64.2 Thermal Considerations

The EQ documentation shows that the connectors are qualified for 40 years at 150°F. This is equivalent to much greater than 60 years at 120°F using standard EQ techniques. Most ANO-1 applications are at, or below, 120°F. Two applications are at, or below 150°F but were installed in 1995 (2034-1995=39) so only 39 years of life are needed through the renewal period. The connector o-ring is only qualified for 10 years at 150 ^{0}F. The EQ Documentation for the specific applications using these devices identifies this replacement as a normal course of the ANO EQ Program.

4.4.64.3 Wear/Cycles Considerations

The devices are qualified for 160 disconnect/connect cycles, which is greater than the conservative 60 expected cycles (once every year).

4.4.64.4 Conclusion

The EGS quick disconnect electrical connectors EQ aging analyses have been projected to the end of the period of extended operation. The connector o-rings will be replaced prior to exceeding their qualified life, or re-evaluated using standard EQ techniques. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.65 EGS Grayboot Electrical Connectors

EGS Grayboot electrical connectors are used in various applications.

4.4.65.1 Radiation Considerations

The post accident dose of $4.0x10^7$ rads plus the normal aging 60-year life dose of $1.5x10^7$ rads results in a $5.5x10^7$ rads TID, which is much less than the qualified value of $2.08x10^8$ rads.

4.4.65.2 Thermal Considerations

The EQ documentation shows that the connectors are qualified for 47.4 years at 130°F. This is equivalent to greater than 60 years at 120°F using standard EQ techniques and the ANO-1 applications are at, or below, 120°F.

4.4.65.3 Wear/Cycles Considerations

The devices are qualified for 160 disconnect/connect cycles, which is greater than the conservative 60 expected cycles (once every year).

4.4.65.4 Conclusion

The EGS Grayboot electrical connectors EQ aging analyses have been projected to the end of the period of extended operation.

4.4.66 Valcor Model V526-5683 Solenoid Operated Valve

Only one Valcor solenoid valve of this model is installed outside the reactor building, in the post-accident sampling system.

4.4.66.1 Radiation Considerations

The post accident dose of $1.49x10^7$ rads plus the normal aging 60-year life dose of $5.25x10^6$ rads results in a $2.015x10^7$ rads TID, which is much less than the qualified value of $5.9x10^7$ rads.

4.4.66.2 Thermal Considerations

The EQ documentation identifies the subcomponents of this valve (solenoid assembly, valve seat, coil housing O-rings, and bracket assembly) as being qualified for less than 40 years at a 120^0F ambient. These will be replaced prior to exceeding their qualified life as a normal course of the ANO EQ Program, or re-evaluated using standard EQ techniques.

4.4.66.3 Wear/Cycles Considerations

The devices are qualified for 60,000 cycles, which is greater than the 792 cycles expected over a 60-year period.

4.4.66.4 Conclusion

The ANO EQ Program requires replacements prior to exceeding the qualified life; therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.67 Valcor Model V526-5961-1 Solenoid Operated Valve

Valcor solenoid valves are installed in a limited number of applications, outside the reactor building.

4.4.67.1 Radiation Considerations

The post accident dose of $1.49x10^7$ rads plus the normal aging 60-year life dose of $5.25x10^6$ rads results in a $2.015x10^7$ rads TID, which is much less than the qualified value of $2.0x10^8$ rads.

4.4.67.2 Thermal Considerations

The EQ documentation shows that the valves are qualified for 40 years at 120°F. This is equivalent to greater than 60 years at 105°F using standard EQ techniques and the ANO-1 applications are at, or below, 105°F.

4.4.67.3 Wear/Cycles Considerations

The devices are qualified for 57,500 cycles, which is greater than the 792 cycles expected over a 60-year period.

4.4.67.4 Conclusion

The Valcor model V526-5961-1 solenoid operated valve EQ aging analyses have been projected to the end of the period of extended operation.

4.4.68 General Cable Corporation 5 kV Power Cable

General Cable Corporation 5 kV power cable is used in limited applications outside the reactor building such as decay heat, high pressure injection, and reactor building spray pump motors.

4.4.68.1 Radiation Considerations

The post accident dose of $1.02x10^7$ rads plus the normal aging 60-year life dose of $5.25x10^6$ rads results in a $1.545x10^7$ rads TID, which is much less than the qualified value of $2.0x10^8$ rads.

4.4.68.2 Thermal Considerations

The EQ documentation shows that this cable is qualified for 60 years at 145 °F. The ANO-1 applications are at, or below, 120°F.

4.4.68.3 Wear/Cycles Considerations

Not applicable to cable.

4.4.68.4 Conclusion

The General Cable Corporation 5 kV power cable EQ aging analyses have been projected to the end of the period of extended operation.

4.4.69 GSI-168 "EQ of Electrical Components"

As discussed in SECY-93-049, the staff reviewed significant license renewal issues and found that several were related to environmental qualification. An essential aspect of these issues was whether the licensing basis should be reassessed or enhanced in connection with license renewal, and whether this reassessment should be extended to the current license term. In late 1993, the Commissioners instructed the Staff that the current EQ licensing basis must be used in the license renewal period and that any EQ concerns identified by the staff during the review of EQ for license renewal should be evaluated for the effect on current licenses, independent of license renewal.

The NRC Staff's EQ Task Action Plan was initiated to address the adequacy of current EQ practices. Upon completion of the EQ-TAP review, the focus of Staff concerns was limited to issues related to the adequacy of accelerated aging practices in existing qualifications, and the lack of a "feedback mechanism" in EQ programs (i.e., programmatic requirements to determine the current condition of EQ equipment so that it can be evaluated against the assumptions and parameters for qualification). The EQ-TAP was subsequently closed and the remaining open issues were incorporated into GSI-168 for management tracking purposes. The EQ-TAP review did not identify any generic safety issues related to these open issues. NRC research on these topics is in progress. NRC guidance for addressing GSI-168 for license renewal is contained in a June 1998 letter to NEI (Reference 4.4-2). In this letter, the NRC states:

"With respect to addressing GSI-168 for license renewal, until completion of an ongoing research program and staff evaluations, the potential issues associated with GSI-168 and their scope have not been defined to the point that a license renewal applicant can reasonably be expected to address them at this time. Therefore, an acceptable approach described in the SOC is to provide a technical rationale demonstrating that the current licensing basis for EQ pursuant to 10CFR50.49 will be maintained in the period of extended operation. Although the SOC also indicates that an applicant should provide a brief description of one or more reasonable options that would be available to adequately manage the effects of aging, the staff does not expect an applicant to provide the options at this time."

Environmental qualification evaluations of electrical equipment are identified as time-limited aging analyses for ANO. The evaluations of these time-limited aging analyses are considered the technical rationale that the current licensing basis will be maintained during the period of extended operation. These evaluations are provided in this section of the ANO-1 LRA. Consistent with the above NRC guidance, no additional information is required to address GSI-168 in a renewal application at this time.

4.4.70 References for Section 4.4

4.4-1 IEEE Std 323-1974, "*Qualifying Class 1E Equipment for Nuclear Power Generating Stations*," The Institute of Electrical and Electronics Engineers, Inc., 1974.

4.4-2 C. I. Grimes (NRC) letter to D. Walters (NEI), "*Guidance on Addressing GSI 168 for License Renewal*," Project 690, dated June 2, 1998.

4.5 CONCRETE REACTOR BUILDING TENDON PRESTRESS

Loss of prestress in the post-tensioning system is due to material strain occurring under constant stress.

In accordance with ACI 318-63 (Reference 4.5-1), the design of the ANO-1 reactor building post-tensioning system provides for prestress losses caused by the following.

- Seating anchorage
- Elastic shortening of concrete
- Creep of concrete
- Shrinkage of concrete
- Relaxation of prestressing steel stress
- Frictional loss due to curvature in the tendons and contact with tendon conduit

By assuming an appropriate initial stress from tensile loading, and using appropriate prestress loss parameters, the magnitude of the design losses and the final effective prestress at the end of 40 years for typical dome, vertical, and hoop tendons was calculated at the time of initial licensing. This analysis, summarized in SAR Section 5.2.4.2.1, is a time-limited aging analysis requiring review for license renewal.

ASME Code Section XI, Subsection IWL, provides requirements for inservice inspection and repair and replacement activities of the post-tensioning systems of concrete reactor buildings. Subsection IWL requires visual examination of tendon wires and tendon anchorage hardware, including bearing plates, anchorheads, wedges, buttonheads, shims, and the adjacent concrete. Tendon force and elongation is required to be measured to evaluate the prestress force of the system. In addition, tendon wires or strand samples are required to be removed and tested to determine the yield strength, ultimate tensile strength and elongation.

ANO-1 is completing a calculation of the final effective tendon prestress based on additional information on concrete creep from existing creep tests and results of the tendon surveillance testing. This calculation will confirm projections on tendon relaxation that show the ANO-1 reactor building tendon elements will be acceptable for the period of extended operation. In addition, the ASME Section XI Inservice Inspection Program, IWL inspections will be adequate to manage the effects of aging on the intended function for the period of extended operation. Implementation of this program dispositions this time-limited aging analysis in accordance with 10CFR54.21(c)(1)(iii).

4.5.1 References for Section 4.5

4.5-1 ACI 318-63, "*Building Code Requirements for Reinforced Concrete*," American Concrete Institute.

4.6 Reactor Building Liner Plate Fatigue Analysis

The interior surface of the reactor building is lined with welded carbon steel plate to provide an essentially leak tight barrier. At the penetrations, the reactor building liner plate is thickened to reduce stress concentrations. Design criteria are applied to the liner plate to assure that the specified leak rate is not exceeded under design basis accident conditions. The following fatigue conditions, as described in SAR Section 5.2.1.4.7.3, were considered in the design of the liner plate:

1. Thermal cycling due to annual outdoor temperature variations. The number of cycles for this loading is 40 cycles for the plant life of 40 years.
2. Thermal cycling due to reactor building interior temperature varying during the startup and shutdown of the reactor coolant system. The number of cycles that is assumed for this loading condition is 500.
3. One thermal cycle is assumed due to DBA conditions .

The design analysis for the liner plate, which considers these fatigue conditions, is considered to be a time-limited aging analysis for the purposes of license renewal.

Each of the above fatigue conditions have been evaluated for continued operation for up to 60 years. For item (1), an increase in the number of thermal cycles due to annual outdoor temperature variations from 40 to 60 cycles is considered to be insignificant in comparison to the assumed 500 thermal cycles due to the reactor building interior temperature varying during heatup and cooldown of the reactor coolant system.

For item (2), ANO-1 operating experience indicates that the assumed 500 thermal cycles due to startup and shutdown of the reactor coolant system, is conservative. The projected number of thermal cycles for 60 years of operation, has been determined to be less than the original design 500 cycle design assumptions.

For item (3), the assumed value for thermal cycles due to DBA conditions remains valid. None have occurred and none are expected to occur.

Integrated leak rate tests are additional sources of load changes. These loads are considered within the set of design loads whose cumulative total was assumed to be 500 cycles. Due to the limited number of these tests, these additional load cycles on the liner plate are bounded by the 500 cycle assumption.

Finally, the design of the reactor building penetrations has been reviewed. The design meets the general requirements of ASME Section III for thermal cycling. By design, thermal load cycles in the piping systems are isolated from the liner plate penetrations by concentric sleeves between the pipe and the liner plate.

The high temperature lines penetrating the reactor building wall and liner plate are the feedwater and main steam lines. The design number of thermal load cycles in these two systems is greater than the design number of heatup and cooldown cycles of the reactor coolant system. The projected number of cycles for ANO-1 through 60 years of operation has been determined to be less than the original design assumptions.

In summary, the assumed fatigue conditions used in the reactor building liner plate fatigue analysis are bounding for 60 years of plant operation. Therefore, this time-limited aging analysis remains valid for the period of extended operation and meets the criteria of CFR54.21(c)(1)(i).

4.6.1 References for Section 4.6

4.6-1 ASME Boiler and Pressure Vessel Code, Section III, "*Rules for Construction of Nuclear Power Plant Components,*" American Society of Mechanical Engineers.

4.7 AGING OF BORAFLEX IN SPENT FUEL POOL RACKS

Boraflex is a boron carbide dispersion in an elastomeric silicone that is currently used in a portion (Region I) of the ANO-1 spent fuel storage racks as a neutron absorber. The Region II section of the spent fuel storage racks does not contain Boraflex. Potential stressors for the Boraflex in the pool include the chemical environment of borated water and gamma radiation, which changes the material characteristics of the base polymer. NRC Information Notices 87-43, "Gaps in Neutron Absorbing Material in High Density Spent Fuel Storage Racks," 93-70, "Degradation of Boraflex Neutron Absorber Coupons," and 95-38, "Degradation of Boraflex Neutron Absorber in Spent Fuel Storage Racks," identified the concern of aging of Boraflex neutron absorbing material. This concern resulted in NRC Generic Letter 96-04, "Boraflex Degradation in Spent Fuel Pool Storage Racks."

In the ANO response to Generic Letter 96-04 [0CAN109605 (Reference 4.7-1)], Entergy Operations committed to continue monitoring and analysis of the Boraflex degradation at ANO-1. In order to ensure that the 5 percent subcriticality margin can be maintained for the life of the spent fuel storage racks, Entergy Operations will continue the existing coupon monitoring program as required into the extended license period. Entergy Operations will also continue to monitor spent fuel pool silica levels and perform silica evaluations. These evaluations are based on the EPRI RACKLIFE system or its equivalent. Projected Boraflex performance will be assessed to confirm the 5 percent subcriticality margin will be maintained as required.

Degradation of Boraflex is treated as a time-limited aging analysis at ANO-1 since this analysis meets the six criteria of 10CFR54.3. The analysis meets 10CFR54.21(c)(1)(ii) and the sampling actions meet 10CFR54.21(c)(1)(iii). Therefore, this time limited aging analysis is valid for the period of extended operation.

4.7.1 References for Section 4.7

4.7-1 0CAN109605, Letter from D. Mims (ANO) to the NRC, "*120 Day Response to Generic Letter 96-04, Boraflex Degradation in Spent Fuel Pool Storage Racks*," dated October 24, 1996.

4.8 OTHER TIME-LIMITED AGING ANALYSES

4.8.1 Reactor Vessel Underclad Cracking

Intergranular separations (underclad cracking) in low alloy steel heat-affected zones under austenitic stainless steel weld cladding were detected in SA-508, Class-2 reactor vessel forgings manufactured to a coarse grain practice and clad by high-heat-input submerged arc processes. BAW-10013 (Reference 4.8-1) contains a fracture mechanics analysis that demonstrates the critical crack size required to initiate fast fracture is several orders of magnitude greater than the assumed maximum flaw size plus predicted flaw growth due to design fatigue cycles. The flaw growth analysis was performed for a 40-year cyclic loading, and an end-of-life assessment of radiation embrittlement (i.e., fluence at 32 EFPY) was used to determine fracture toughness properties. The report concluded that the intergranular separation found in B&W vessels would not lead to vessel failure. This conclusion was accepted by the AEC. To cover the period of extended operation, an analysis was performed using current ASME Code (Reference 4.8-2) requirements. This analysis is fully described in Appendix C of BAW-2251A (Reference 4.8-3).

In May 1973, the AEC issued Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components." The guide states that underclad cracking "… has been reported only in forgings and plate material of SA-508 Class-2 composition made to coarse grain practice when clad using high-deposition-rate welding processes identified as 'high-heat-input' processes such as the submerged-arc wide-strip and the submerged-arc six-wire processes. Cracking was not observed in clad SA-508 Class-2 materials clad by 'low-heat-input' processes controlled to minimize heating of the base metal. Further, cracking was not observed in clad SA-533 Grade B Class-1 plate material, which is produced to fine grain practice. Characteristically, the cracking occurs only in the grain-coarsened region of the base-metal heat-affected zone at the weld bead overlap." The guide also notes that the maximum observed dimensions of these subsurface cracks is 0.165-inch deep by 0.5-inch long.

The BAW-10013 fracture mechanics analysis is a flaw evaluation performed before the ASME Code requirements for flaw evaluation, the K_{1a} curve for ferritic steels as indexed against RT_{NDT}, and the ASME Code fatigue crack growth curves for carbon and low alloy ferritic steels were available. The revised analysis uses current fracture toughness information, applied stress intensity factor solutions, and fatigue crack growth correlations for SA-508 Class-2 material. The objective of the analysis is to determine the acceptability of the postulated flaws for 48 EFPY using ASME Code, Section XI (Reference 4.8-4), (1995 Edition), IWB-3612 acceptance criteria.

The revised analysis was applied to three relevant regions of the reactor vessel: the beltline, the nozzle belt, and the closure head/head flange. The analysis conservatively considered 360 cycles of 100°F/hr normal heatup and cooldown transients. For the power maneuvering transients, the range in applied stress intensity factors for the closure head region was assumed the same as that determined for the beltline region. This assumption is considered conservative since the closure head region is subject to a low flow condition while the beltline region is subject to a forced flow condition.

An initial flaw size of 0.353-inch deep by 2.12-inch long (6:1 aspect ratio) was conservatively assumed for each of the three regions. The flaw was further assumed to be an axially oriented, semi-elliptical surface flaw in contrast to the observed flaws which are subsurface with a maximum size of 0.165-inch deep by 0.5-inch long.

The maximum crack growth and applied stress intensity factor for the normal and upset conditions were found to occur in the nozzle belt region. The maximum crack growth considering all the normal and upset condition transients for 48 EFPY, was determined to be 0.180-inch, which results in a final flaw depth of 0.533-inch. The maximum applied stress intensity factor for the normal and upset condition results in a fracture toughness margin of 3.6 which is greater than the IWB-3612 acceptance criteria of 3.16.

The maximum applied stress intensity factor for the emergency and faulted conditions occurs in the closure head to head flange region and the fracture toughness margin was determined to be 2.24, which is greater than the IWB-3612 acceptance criterion of 1.41. It is therefore concluded that the postulated intergranular separations in the ANO-1 reactor vessel 508 Class-2 forgings are acceptable for continued safe operation through the period of extended operation. Therefore, this time limited aging analysis is acceptable per 10CFR 54.21(c)(1)(ii).

4.8.2 Reactor Vessel Incore Instrumentation Nozzles- Flow Induced Vibration Endurance Limit

ANO identified one additional time-limited aging analysis associated with the reactor vessel that was not specifically identified in the Reactor Vessel GLRP report BAW-2251A. Report BAW-10051, "Flow Induced Vibration Endurance Limit Assumptions" (Reference 4.8-7) is an analysis that calculated stress values for the reactor vessel incore nozzles and compared them to endurance limit (stress) values. These endurance limit values were based on an assumed value of 10^{12} cycles for 40 years of operation. The number of fatigue cycles was extended for 60 years, and the component item stress values were compared to the recalculated endurance limit values and shown to be acceptable. Therefore, this time limited aging analysis is acceptable per 10CFR54.21(c)(1)(ii).

4.8.3 Leak Before Break

The successful application of leak-before-break to the ANO-1 RCS main coolant piping is described in the BWOG Topical Report entitled, "The B&W Owners Group Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSSS," BAW-1847, Revision 1, September 1985 (Reference 4.8-5). This report provides the technical basis for evaluating postulated flaw growth in the main RCS piping under normal plus faulted loading conditions and was approved by the NRC for the current term of operation. The time-limited aging analyses in BAW-1847, Revision 1, include fatigue flaw growth and the qualitative assessment of thermal aging of cast austenitic stainless steel reactor coolant inlet and exit nozzles.

Fatigue Flaw Growth

The leak-before-break analysis reported in BAW-1847, Revision 1, was performed in accordance with the guidance provided in Section 5.2, Item (d), of NUREG 1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks." Specifically, a surface flaw was postulated at selected locations of the piping system (i.e., highest stress coincident with the lower bound of the material properties for base metal, weldments and safe ends), and a fatigue crack growth analysis for postulated flaws was then performed to demonstrate that the surface flaws are likely to propagate in the through-wall direction and develop leakage before they will propagate circumferentially around the pipe. Flaw growth calculations are reported in Section 4.3, Table 4-3, of BAW-1847, Revision 1, and are based on 240 heatup and cooldown cycles and 22 cycles of safe shutdown earthquake.

As described in Section 4.3.5 , the original transient cycles that were defined for 40 years of operation for the RCS components are being monitored by ANO-1. If a transient cycle count approaches or exceeds the allowable design limit, corrective actions are taken. Therefore, the flaw growth evaluation reported in BAW-1847, Revision 1, is applicable to 60 years of operation since ANO-1 has not revised the transients defined in the RCS design specification for license renewal.

<u>Thermal Aging of Cast Austenitic Stainless Steel Reactor Coolant Pump Suction and Discharge Nozzles</u>

The susceptibility of the RCS main coolant piping to thermal aging was qualitatively addressed in Section 3.3.4.3 of BAW-1847, Revision 1. As described in BAW-2243A (Reference 4.8-1), there are no RCS main coolant piping segments fabricated from CASS. However, the heat affected zone of the welded joint that connects the wrought austenitic stainless steel 28-inch pump transition piece to the CASS reactor coolant pump inlet and exit nozzles may be susceptible to thermal embrittlement. Limited data regarding thermal aging of CASS material was available at the time of the preparation of BAW-1847, Revision 1. In the BWOG report, the values of fracture toughness for aged cast austenitic stainless steel were assumed to be bounded by the ferritic piping and ferritic weldments. Since the publication of BAW-1847, Revision 1, a significant amount of data has been obtained regarding thermal aging of CASS materials.

Test data obtained by Argonne National Laboratory [O. K. Chopra and W. J. Shack, "Assessment of Thermal Embrittlement of Cast Stainless Steels," NUREG/CR-6177, U.S. Nuclear Regulatory Commission, Washington DC, May 1994], indicate that prolonged exposure of CASS to reactor coolant operating temperatures can lead to reduction of fracture toughness by thermal embrittlement. The fracture toughness curves for the ferritic base metal and ferritic weld metals used in the reactor coolant system piping leak-before-break analysis were compared to the lower-bound fracture toughness curves of ANO-1 reactor coolant pump CASS materials (i.e., statically cast CF8M) from the Argonne report. The fracture toughness curve of the lower-bound CASS material is below the fracture toughness curves used in the RCS piping leak-before-break analysis. Therefore, the assumption in BAW-1847, Revision 1, that the fracture toughness of the ferritic piping and ferritic weldments bounds the fracture toughness of CASS materials cannot be supported.

A flaw stability analysis was performed using the lower-bound CASS fracture toughness curves from the Argonne report cited above to show acceptability of leak-before-break for the reactor coolant system main coolant piping for the period of extended operation. The most limiting material and location used in the RCS piping leak-before-break analysis (i.e., BAW-1847) was determined to be the base metal material of the straight section of the 28-inch cold leg pipe. Both the suction and discharge nozzles of the reactor coolant pump casings are attached to the 28-inch cold leg pipes and have similar geometry and loading applied to them as the limiting location used for the leak-before-break analysis. The discharge and suction nozzles of the reactor coolant pump casings were evaluated for leak-before-break using lower-bound CASS fracture toughness properties.

Bounding 10 gpm leakage crack sizes (margin of 10 on the plant's leak detection capability) for the reactor coolant pump suction and discharge nozzle were determined using a method that is consistent with that reported in BAW-1847, Revision 1. In the revised analysis, the applied loadings were considered using the absolute sum load combination method. Therefore, in accordance with SRP 3.6.3, a margin of 1.0 on load was used. The leakage crack length (twice the leakage flaw size) for the suction nozzle was determined to be 8.62 inches and the leakage crack length for the discharge nozzle was determined to be 8.86 inches. In addition, a crack extension value of 0.6 inches was considered in the flaw stability analysis. A flaw stability analysis was performed for the reactor coolant pump inlet (suction) and exit (discharge) nozzles, and the discharge nozzle was found to be limiting. The maximum applied J value at the discharge nozzle, for the 10 gpm leakage flaw size, was determined to be 0.510 kips/in. The critical crack length was determined to be 21.6 inches. Therefore, the margin on flaw size was determined to be 2.4. This is greater than the required margin of 2 in accordance with SRP 3.6.3. Based on the results of this analysis, it is concluded that all the required margins for LBB per SRP 3.6.3 are met, even with consideration of the lower bound CASS fracture toughness properties for the suction and discharge nozzles.

In addition, thermal aging of the SMAW weldment that connects the stainless steel transition pieces to the RCP nozzles was considered. The J-R curve for aged and unaged stainless steel weldments was obtained from NUREG/CR-6428, "Effects of Thermal Aging on Fracture Toughness and Charpy-Impact Strength of Stainless Steel Pipe Welds." Based on the results of this work, a conservative estimate of J-R curve for aged stainless steel welds is given by $J = 40 + 83.5\Delta a^{0.643}$ in units of kJ/m^2. The J-R curve for aged CASS CF8M material used in the LBB analysis, reported above, is given by $J=167(\Delta a)^{0.31}$, which is also given in units of kJ/m^2. As noted above, the flaw stability analysis considered a maximum crack extension of 0.6 inches (15.24 mm). The results show that the J-R curve for aged CASS material used in the LBB analysis, bounds the J-R curve for aged stainless steel weld material given in NUREG/CR-6428 at a maximum crack extension of 15.24 mm.

Summary: Leak-Before-Break for the Period of Extended Operation

In summary, demonstration that the fatigue flaw growth analysis reported in BAW-1847, Revision 1, remains valid for the period of extended operation is addressed by the ANO-1 Transient Cycle Logging and Reporting discussion in Section 4.3.5. The remainder of the generic leak-before-break analysis for the B&W operating plants reported in BAW-1847, Revision 1, remains valid for the period of extended operation with the exception of the assessment of reduction of fracture toughness by thermal aging of cast austenitic stainless steel reactor coolant pump nozzles. Reduction of fracture toughness of the reactor coolant pump nozzles was determined to be acceptable for the period of extended operation through the flaw stability analysis described above.

4.8.4 Reactor Coolant Pump Motor Flywheels

Entergy Operations identified a time-limited aging analysis with the reactor coolant pump motor. The analysis included a fatigue crack growth evaluation to determine the growth of pre-existing cracks. The crack growth evaluation was performed for 4,000 startup/shutdown cycles, which exceeds the number of design cycles for the reactor coolant pump motor by a factor of 8. Therefore, the crack growth evaluation is a time-limited aging analysis and the existing analysis is adequate for the period of extended operation. This time-limited aging analysis is acceptable per 10CFR 54.21(c)(1)(i).

4.8.5 References for Section 4.8

4.8-1 BAW-10013, "Study of Intergranular Separations in Low-Alloy Steel Heat-Affected Zones Under Austenitic Stainless Steel Weld Cladding," B&W Nuclear Power Generation, December 1971.

4.8-2 ASME Boiler and Pressure Vessel Code, American Society of Mechanical Engineers.

4.8-3 BAW-2251A, "*Demonstration of the Management of Aging Effects for the Reactor Vessel*," The B&W Owners Group Generic License Renewal Program, June 1996.

4.8-4 ASME Boiler and Pressure Vessel Code, Section XI, "*Rules for In-Service Inspection of Nuclear Power Plant Components*," American Society of Mechanical Engineers.

4.8-5 BAW-1847, "The B&W Owners Group Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSSS," Revision 1, B&W Owners Group, September 1985.

4.8-6 BAW-2243A, "Demonstration of the Management of Aging Effects for the Reactor Coolant System Piping," The B&W Owners Group Generic License Renewal Program, June 1996.

Appendix A

Safety Analysis Report Supplement

INTRODUCTION

This appendix contains the SAR Supplement required by 10CFR54.21(d) for the ANO-1 License Renewal Application. The LRA contains the technical information required by 10CFR54.21(a) and 10CFR54.21(c). Section 3.0 and Appendix B of the ANO-1 LRA provide descriptions of the programs and activities that manage the effects of aging for the period of extended operation. Section 4.0 contains the evaluations of the time-limited aging analyses for the period of extended operation. These sections have been used to prepare the program and activity descriptions that are contained in the SAR Supplement. The SAR Supplement will be incorporated into the ANO-1 SAR following issuance of the renewed operating license for ANO-1. Upon inclusion of the SAR Supplement in the ANO-1 SAR, changes to the descriptions of the programs and activities will be made in accordance with 10CFR50.59.

CHAPTER 4 CHANGES

4.1.2.6 Service Lifetime

The original design service lifetime for the major RCS components ~~is~~ was 40 years. The number of cyclic system temperature, pressure, and operational changes (Table 4-8) ~~is~~ was based on operation for this design lifetime. The commencement date for the original design service life was the date of the Construction Permit which approved the PSAR for Unit 1, which is December 6, 1968. However, in 1990, per License Amendment 131, an extension was granted to allow the operating license term to be changed to start at the issuance of the operating license to allow a 40-year service life that does not include the construction time period and end on May 20, 2014. A new operating license has been granted to extend the licensed term an additional 20 years to May 20, 2034. This was justified based on design transient cycles. The reactor coolant system was originally qualified using a conservative estimate of design cycles for a 40 year life. The design life is not dependent on years of service. The design life is dependent on fatigue cycles. In evaluations performed by the NSSS vendor, the actual cycles were extrapolated to 60 years. For the major RCS components, the design cycles exceeded the estimated cycles for a 60-year life. The actual transient cycles are tracked and documented to ensure they are maintained below the allowable number of design cycles as further discussed in Section 16.2.21. Table 4-8 shows the complete listing of transients used in the design of components within the Reactor Coolant Pressure Boundary. Records of significant transients are available from the ~~daily~~ periodic reviews of the Shift Superintendent's log. A record of all significant transients is maintained by the technical support staff.

4.1.2.8 Vessel Radiation Exposure

The reactor vessel is the only RCS component exposed to a significant level of neutron irradiation and is therefore the only component subject to material radiation damage. The maximum exposure from fast neutrons (E > 1.0 MeV) has been computed to be less than 3.0×10^{19} neut/cm² over a 40-year life with an 80 percent load factor. Revised calculations, based on results of the Reactor Vessel Material Surveillance Program and reduced leakage fuel cycle configurations, indicate that the maximum exposures will be less than half of this value. The maximum inside surface fast neutron fluence at 48 EFPY is projected to be 1.44×10^{19} neut/cm². Reactor vessel irradiation calculations are described in Section 4.3.3.

4.3.3 REACTOR VESSEL

(Excerpted from Nil Ductility Transition Temperature (NDTT))

Revised calculations, based on results of the Reactor Vessel Material Surveillance Program and reduced leakage fuel cycle configurations, indicate that the maximum fluence value will be less than half of the originally estimated $3x10^{19}n/cm^2$. The maximum inside surface fast neutron fluence at 48 EFPY is projected to be $1.44x10^{19}$ n/cm^2. The corresponding EOL transition temperature is similarly reduced while the minimum upper shelf energy value is increased.

(Excerpted from Flux and Total Integrated Flux (nvt) at Reactor Vessel Wall)

Revised calculations, based on results of the Reactor Vessel Material Surveillance Program and reduced leakage fuel cycle configurations, indicate that the maximum fluence values will be less than the originally calculated values. The revised fluence value for 40 years at 80 percent load is 1.10×10^{19} n/cm^2. This value was used to respond to the Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock, 10CFR50.61. The projected fast fluence value has since been revised lower to 8.71×10^{18} n/cm^2, as reported to the NRC via 1CAN119608. The projected maximum fast fluence value for 48 EFPY has been determined to be 1.44×10^{19} n/cm^2.

(Excerpted from Expected NDTT Shift)

Revised values of calculated vessel exposure have been used to determine the increase in the NDTT in accordance with 10CFR50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock." The results have been reported to and accepted by the NRC (1CNA128603). For the 40-year exposure at 80 percent load, the calculated value of the NDTT is 264°F. The calculation and supporting documentation are described in BAW-1895 (Jan. 1986). "Pressurized Thermal Shock Evaluation in Accordance with 10CFR50.61 for B&W Owners Group Reactor Pressure Vessels." Since the submittal of BAW-1895, the fourth capsule report has been produced and submitted. This capsule report indicates the NDTT value is 257°F based on a further reduction in the projected fluence value.

BAW-2251-A shows the RT_{PTS} to be 278°F for 48 EFPY for a circumferential weld.

The NDTT shift is factored into the plant startup and shutdown procedures so that full operating pressure is not attained until the reactor vessel temperature is about DTT. The heatup and cooldown curves are given in Technical Specification 3.1.2, “Pressurization, Heatup and Cooldown Limitations. The total stress in the vessel wall due to both pressure and the associated heatup and cooldown transient is restricted to 5,000 - 10,000 psi, which is below the threshold of concern for safe operation. An adjusted 100°F per hour heatup rate can be maintained throughout life. An adjusted rate is one in which the pressure is held constant to maintain stresses at the desired low level while temperatures are at a level below DTT. A 100°F per hour temperature increase is maintained until DTT is passed and pressure can be raised to a new higher level. These operating restrictions are based on the NRL generalized fracture analysis diagram, which is a semi-empirical method of material selection and approximate analysis to prevent brittle fracture. This diagram plots failure stress (normalized to yield) as a function of temperature referenced to the NDTT for a family of finite flaw sizes. The parametric crack size curves were determined partially by fracture mechanics and partially by plotting actual failure data. The assumed flaw for this analysis was slightly greater than 24 inches.

CHAPTER 5 CHANGES

5.2.1.4.7.3 Loads

All load combinations are considered in the above analysis. The following fatigue loads are considered in liner design:

A. Thermal cycling due to annual outdoor temperature variations - Daily temperature variations will not penetrate a significant distance into the concrete shell to appreciably change the average temperature of the shell relative to the liner plate. The number of cycles for this loading is ~~40~~ 60 cycles for the plant life of ~~40~~ 60 years.

B. Thermal cycling due to reactor building interior temperature varying during the heatup and cooldown of the reactor system - The number of cycles for this loading is assumed to be 500.

C. One cycle was assumed for thermal cycling due to a DBA.

CHAPTER 6 CHANGES

6 ENGINEERED SAFEGUARDS

(Excerpted from Chapter 6)

Operability of engineered safeguards equipment is assured in several ways. Much of the equipment in these systems function during normal reactor operation thus providing a constant check on operational status. Where equipment is used for emergency functions only, such as in the Reactor Building Spray System, the systems have been designed to permit meaningful periodic tests. Operational reliability has been achieved by using proven component designs wherever possible and/or by conducting tests. Quality control and assurance requirements are implemented during the design, manufacture, and installation of the engineered safeguards components and systems to assure that a high quality level is maintained. The quality program is based upon the use of accepted industry codes and standards as well as supplementary test and inspections. The resultant high quality level of the components gives assurance that they will perform their intended function under the worst anticipated conditions following a LOCA. Materials for equipment required to operate under accident conditions are selected on the basis of the additional exposure received in the event of a Design Basis Accident (DBA). All equipment must ~~operate for the designed 40-year~~ remain functional throughout the life of the plant. Certain safety-related equipment must operate during the design plant life as well as function as required during and following a DBA at the end of plant life.

CHAPTER 11 CHANGES

11.2.1.2 Radiation Exposure of Materials and Components

~~No regulations similar to those established for the protection of individuals exist for materials and components. However, m~~Materials and components are selected on the basis that their design radiation exposure will not cause significant changes in their physical properties which adversely affect operation of equipment during the design life of the plant. Materials for equipment required to operate under accident conditions are selected on the basis of the additional exposure received in the event of a DBA. The approximate radiation damage threshold for various materials is shown in Table 11-13.

NEW CHAPTER 16

16.0 AGING MANAGEMENT PROGRAMS AND ACTIVITIES

The integrated plant assessment for license renewal identified several new programs and activities, modifications to existing programs, and existing programs, necessary to continue operation of ANO-1 during the additional 20 years beyond the initial license term. This chapter describes these programs and activities.

16.1 NEW ACTIVITIES

16.1.1 BURIED PIPE INSPECTION

Buried Pipe Inspections will be performed to ensure that a loss of material due to external surface corrosion of buried piping is adequately managed. The safety-related portions of underground carbon steel piping on the service water and fuel oil systems are within the scope of this inspection. The aging effect addressed by the Buried Pipe Inspection is a loss of material due to corrosion of the external surfaces of pipe caused by loss of the protective coating. This inspection will be initiated prior to the end of the initial 40-year license term.

16.1.2 ELECTRICAL COMPONENT INSPECTION

The Electrical Component Inspection Program will inspect splices, connectors, and cables located in areas that may be conducive to accelerated aging. The scope of this inspection includes cables exposed to elevated temperatures, wet environments, or corrosive chemicals. The scope also includes cables that can experience elevated temperatures due to the current they are carrying and connectors used in impedance-sensitive circuits and cable splices subject to aging-related stressors. The aging effect for cables and cable splices is a change of material properties, as evidenced by cracking or discoloration of the insulation. The aging effect for connectors in impedance-sensitive circuits is a change of material due to corrosion of connector pins. The Electrical Component Inspection Program will be formally implemented and the first inspection of in-scope cables, splices, and connectors completed prior to the expiration of the initial 40-year licensing term.

16.1.3 HEAT EXCHANGER MONITORING

The Heat Exchanger Monitoring Program will inspect heat exchangers to the extent required to ensure seismic qualification is maintained. The Heat Exchanger Monitoring Program manages aging effects on the following safety-related systems and components: reactor building coolers, emergency diesel generator jacket cooling water heat exchangers, make-up pump coolers, make-up pump room coolers, decay heat room coolers, decay heat system heat exchangers, electrical room chillers and coolers, control room chillers and coolers, and emergency feedwater turbine lube oil cooler. The aging effects addressed by the Heat Exchanger Monitoring Program are cracking or loss of material that could result in degradation in the seismic qualification of the heat exchangers. Inspection will be initiated prior to the end of the initial 40-year license term.

16.1.4 PRESSURIZER EXAMINATIONS

The Pressurizer Examinations include two specific examinations: the pressurizer cladding and the pressurizer heater penetration weld examination.

16.1.4.1 Pressurizer Cladding Examination

The pressurizer cladding examination will assess the condition of the pressurizer cladding. The scope of this activity will include the cladding and attachment welds to the cladding of the pressurizer. The aging effect is cracking of cladding by thermal fatigue, which may propagate to the underlying ferritic steel. These inspections are included in the ISI program and will be carried forward to the period of extended operation.

16.1.4.2 Pressurizer Heater Bundle Penetration Welds Examination

The pressurizer heater bundle penetration welds examination will assess the condition of the pressurizer heater penetration welds. This examination will be applicable to the heater sheath-to-diaphragm plate penetration welds inside the pressurizer. The aging effect is cracking at the heater bundle penetration welds, which may lead to reactor coolant leakage. The heater bundle examination may occur prior to entering the period of extended operation or during the period of extended operation.

16.1.5 REACTOR VESSEL INTERNALS AGING MANAGEMENT

Ongoing industry efforts are aimed at characterizing the aging effects requiring management associated with the reactor vessel internals. Further understanding of these aging effects will be developed by the industry over time and will provide additional bases for the inspections under this program. Entergy Operations will participate in the BWOG Reactor Vessel Internals Aging Management Program and other industry programs, as appropriate, to continue investigation of aging effects requiring management for the reactor vessel internals. These activities will assist in establishing appropriate monitoring and inspection programs for the reactor vessel internals. Entergy Operations will provide periodic updates after the completion of significant milestones in the preparation of the Reactor Vessel Internals Inspection, commencing within one year of the issuance of the renewed license. Entergy Operations will submit a report to the NRC, at or about, the end of the initial 40-year operating license term. This report will summarize the current understanding of aging effects applicable to the reactor vessel internals and will contain the Entergy Operations' inspection plan, including methods for each inspection. Entergy Operations will perform the Reactor Vessel Internals Inspection. Should data or evaluations indicate that this inspection can be modified or eliminated, Entergy Operations will provide plant-specific justification to demonstrate the basis for the modification or elimination. The purpose of the Reactor Vessel Internals Inspection is to inspect and examine the reactor vessel internals to assure the aging effects will not result in loss of the intended functions of the internals during the period of extended operation. The inspection applies to the reactor vessel internals. This inspection will begin during the period of extended operation. The aging effects for the reactor vessel internals include cracking due to either stress corrosion or irradiation assisted stress corrosion, reduction of fracture toughness due to irradiation embrittlement, dimensional changes due to void swelling, and loss of bolted closure integrity due to stress relaxation.

16.1.6 SPENT FUEL POOL MONITORING

The Spent Fuel Pool Monitoring Program will manage the aging effects requiring management of the ANO-1 spent fuel pool liner. Stress corrosion cracking is possible from the external surface of the liner in weld heat-affected zones since this was not verified to be chloride-free during construction. This program will be initiated before the end of the initial 40-year license term.

16.1.7 WALL THINNING INSPECTION

Wall Thinning Inspections will be performed to ensure wall thickness is above the minimum required to avoid leaks or failures under normal conditions and postulated transient and accident conditions, including seismic events. Wall Thinning Inspections cover the following safety-related systems and components: EFW pump casing and carbon steel discharge piping and valves, EFW steam supply components downstream of steam admission valves, EFW steam exhaust piping and valves, carbon steel cooling water, seal water, and instrument piping and valves, turbine lube oil cooler, carbon steel EFW supply header piping and valves (condensate supply), NaOH tank, carbon steel piping and components, and carbon steel components of penetrations 11, 42, 43, 48, 49, 51, 52, 54, 58, 59, 60, 62, and 64. The aging effect to be addressed by Wall Thinning Inspection is a loss of material due to corrosion of the internal surfaces of carbon steel piping and components.

16.2 <u>EXISTING ACTIVITIES</u>

16.2.1 ALLOY 600 AGING MANAGEMENT

The Alloy 600 Aging Management Program will manage cracking by PWSCC of Alloy 600 and Alloy 82/182 locations for the period of extended operation. The Alloy 600 Aging Management Program will be applicable to the Alloy 600 items and Alloy 82/182 weld material in the RCS, including the hot leg flow meter element. The aging effect managed by the Alloy 600 Aging Management Program is cracking of Alloy 600 items and Alloy 82/182 weld material in the RCS.

16.2.2 ALTERNATE AC DIESEL GENERATOR TESTING AND INSPECTIONS

The Alternate AC Diesel Generator Testing and Inspections ensures that the effects of aging are managed before the loss of the intended functions of the system. The Alternate AC Diesel Generator Testing and Inspections applies to the station blackout diesel and its components. The aging effects addressed by the Alternate AC Diesel Generator Testing and Inspections include: loss of material or loss of mechanical closure integrity for the starting air subsystem components; loss of material or loss of mechanical closure integrity for the intake combustion air subsystem components; loss of material, fouling, and loss of mechanical closure integrity for the intake air aftercooler; loss of material for carbon steel components, cracking of the stainless steel components, or loss of mechanical closure integrity for the exhaust subsystem components; loss of mechanical closure integrity for the lube oil subsystem components; fouling, loss of material from wear, and loss of mechanical closure integrity for the lube oil cooler; loss of material and loss of mechanical closure integrity for the cooling water subsystem components; fouling and a loss of material for the AAC radiator; loss of material from wetted portions of the exhaust fan housings; and fouling of the fuel oil heat exchanger.

16.2.3 ASME SECTION XI INSERVICE INSPECTION

16.2.3.1 IWB Inspections

The ASME Section XI, Subsection IWB Inspections under the scope of the Inservice Inspection Plan identifies and corrects degradation of ASME Class 1 pressure retaining components and their integral attachments. The scope of the ASME Section XI, Subsection IWB Inspections, credited for license renewal, is identified specifically for each component and for applicable component features in the ISI Plan. The aging effects managed as part of the ASME Section XI, Subsection IWB Inspections include cracking, loss of mechanical closure integrity at bolted connections, and loss of material. In addition, a one-time visual inspection of a reactor coolant pump casing will be performed prior to the end of the initial 40-year license term.

16.2.3.2 IWC Inspections

ASME Section XI, Subsection IWC Inspections under the scope of the ANO-1 Inservice Inspection Plan identify and correct degradation of ASME Class 2 pressure retaining components and their integral attachments. The scope of the ASME Section XI, Subsection IWC Inspections, credited for license renewal, includes components on the following systems: core flood, RBS, main feedwater, spent fuel, service water, HPI/makeup and purification, LPI/decay heat, EFW, main steam, reactor building isolation, and chilled water system. The aging effects managed as part of the ASME Section XI, Subsection IWC Inspections include cracking, loss of mechanical closure integrity, and loss of material.

16.2.3.3 IWD Inspections

ASME Section XI, Subsection IWD Inspections under the scope of the Inservice Inspection Plan identify and correct degradation of ASME Class 3 pressure-retaining components. The scope of the ASME Section XI, Subsection IWD Inspections, credited for license renewal, includes components on the following systems: service water, spent fuel, main steam, EFW, sodium hydroxide, and condensate storage. The aging effects managed as part of the ASME Section XI, Subsection IWD Inspections include cracking, loss of mechanical closure integrity, and loss of material.

16.2.3.4 IWE Inspections

ASME Section XI, Subsection IWE Inspections under the scope of the Inservice Inspection Plan identify and correct degradation of Class MC pressure retaining components, their integral attachments, the metallic shell and penetration liners of Class CC pressure retaining components, and their integral attachments. The scope of the ASME Section XI, Subsection IWE Inspections, credited for license renewal includes inspections of the reactor building liner plate. The aging effect managed as part of the ASME Section XI, Subsection IWE Inspections is a loss of material of the steel surfaces.

16.2.3.5 IWF Inspections

ASME Section XI Inservice Inspection Program, IWF Inspections identify and correct degradation of ASME Class 1, 2, 3, or MC component supports. The aging effects managed as part of the ASME Section XI, Subsection IWF include cracking, loss of material, and change in material properties.

16.2.3.6 IWL Inspections

ASME Section XI Inservice Inspection Program, IWL Inspections provides instructions and documentation requirements for assessing the quality and structural performance of the reactor building’s post-tensioning systems and concrete components. The scope includes the concrete reactor building's post-tensioning systems and reinforced concrete components. The aging effects are loss of material for tendon anchorage and cracking and change in material properties for concrete.

16.2.3.7 Augmented Inspections

The ASME Section XI, Augmented Inspections identify and correct degradation of components outside of the jurisdiction of ASME Section XI. Augmented periodic inspections are completed for several main feedwater and main steam system welds, not in the Class 2 piping, to support the high energy line break analysis. Augmented inspections are completed for the BWST header including the lines from the reactor building sump. Augmented inspections that will be added to the program because of license renewal include a special augmented inspection on the welds of the piping wetted by the reactor building sump water, some supplemental inspections of the "Q" stainless piping of the main steam system, at least a one-time inspection of the penetration 68 piping and components and the decay heat pump room drain valves, and special inspections of penetrations 10, 47, 58, and 64. The aging effects managed by these inspections are cracking and a loss of material. The new inspections will be initiated prior to the end of the initial 40-year license term.

16.2.3.8 Small Bore Piping and Small Bore Nozzles Inspections

The Small Bore Piping and Small Bore Nozzles Inspections identify aging effects on small bore piping and nozzles. The small bore piping and small bore nozzles, within the scope of this program, are RCS piping and nozzles less than 4-inch NPS that do not receive volumetric inspection in accordance with ASME Section XI. The aging effect managed by this program is cracking. A risk-informed ISI method has been implemented to select RCS piping welds for inspection. The risk-informed approach consists of two essential elements. A degradation mechanism evaluation is performed to assess the failure potential of the piping system under consideration, and a consequence evaluation is performed to assess the impact on plant safety in the event of a piping failure. The results from these two independent evaluations are coupled to determine the risk significance of piping segments within the system, and priority is then given to the most risk significant piping segments during the selection of RCS piping welds for inspection.

16.2.4 BOLTING AND TORQUING ACTIVITIES

Bolting and Torquing Activities prevent degradation of bolting or identify and correct degradation of bolting. The scope of Bolting and Torquing Activities applies to pressure boundary bolting applications associated with components within the scope of license renewal. Applications include bolted flange connects for vessels (i.e., manways and inspection ports), flanged joints in piping, body-to-body joints in valves, and pressure-retaining bolting associated with pumps or valves and miscellaneous process components. The aging effects addressed by Bolting and Torquing Activities are cracking, loss of material, and loss of mechanical closure integrity.

16.2.5 BORIC ACID CORROSION PREVENTION

The Boric Acid Corrosion Prevention Program prevents corrosion damage due to leakage from the borated water systems. The Boric Acid Corrosion Prevention Program is concerned with the RCS and other structures and components containing, or exposed to, borated water. This program is credited with monitoring the boric acid corrosion of carbon steel external surfaces of structures and components exposed to leakage from borated water. Carbon steel is utilized for bolting on many of the systems that contain borated water. This program manages the loss of material of bolts that could eventually result in a loss of pressure integrity for bolted connections.

16.2.6 CHEMISTRY CONTROL

The following subsections address the individual ANO-specific chemistry control programs in more detail:

- Primary Chemistry Monitoring
- Secondary Chemistry Monitoring
- Auxiliary Systems Chemistry Monitoring
- Diesel Fuel Monitoring
- Service Water Chemical Control

16.2.6.1 Primary Chemistry Monitoring

The Primary Chemistry Monitoring Program maximizes long-term availability of primary systems by minimizing system corrosion, fuel corrosion, and radiation field build-up. The scope of the Primary Chemistry Monitoring Program includes sampling activities and analysis on the following systems: RCS, borated water storage tanks, spent fuel pool system, letdown purification demineralizers, and reactor makeup water. The Primary Chemistry Monitoring Program provides assurance that an elevated level of contaminants and oxygen does not exist in the systems covered by the program. This prevents or minimizes the occurrence of cracking and other aging effects.

16.2.6.2 Secondary Chemistry Monitoring

The Secondary Chemistry Monitoring Program maximizes the availability and operating life of major components. The scope of the Secondary Chemistry Monitoring Program includes sampling activities and analysis on the main feedwater system, condensate storage system, and steam generators. The aging reviews for many of the safety-related, non-Class 1 systems also indirectly credit the Secondary Chemistry Monitoring Program since the condensate storage tanks are used as a source of makeup water to these systems. The Secondary Water Chemistry Monitoring Program ensures the levels of contaminants and oxygen are maintained within a range that prevents or minimizes the occurrence of loss of material and other aging effects.

16.2.6.3 Auxiliary Systems Chemistry Monitoring

The Auxiliary Systems Chemistry Monitoring Program maximizes the availability and operating life of the components used for the closed cooling loops. The scope of the Auxiliary Systems Chemistry Monitoring Program is limited to sampling activities and analysis on the ICW system, chilled water systems, emergency diesel generators, and the AAC diesel generator. The Auxiliary Systems Chemistry Monitoring Program is credited with minimizing the loss of material due to corrosion, cracking, fouling, and loss of mechanical closure integrity.

16.2.6.4 Diesel Fuel Monitoring

The Diesel Fuel Monitoring Program ensures that adequate diesel fuel quality is maintained to prevent plugging of filters, fouling of injectors, and corrosion of the fuel systems. The scope of the Diesel Fuel Monitoring Program is limited to sampling activities and analysis on the following tanks: bulk fuel oil storage tank, EDG fuel tanks, EDG day tanks, fire pump diesel day tank, and the AAC diesel generator day tank. The aging management reviews credit the sampling and monitoring as providing an adequate control of the fuel oil to ensure water and contamination (including microbiological) are not present in the system.

16.2.6.5 Service Water Chemical Control

The Service Water Chemical Control Program maximizes the availability and operating life of the components in the service water system. The scope of the Service Water Chemical Control Program includes sampling activities and analysis on the service water system. The scope also includes chemical injection into the service water bays. The fire protection system takes suction from the service water bays. The Service Water Chemical Control Program has been credited for aging management in the service water system, auxiliary cooling water system, and the fire protection system since these systems draw suction from the intake structure. The chemical additions are only credited with reducing corrosion, not eliminating this mechanism.

16.2.7 CONTROL ROD DRIVE MECHANISM NOZZLE AND OTHER VESSEL CLOSURE PENETRATIONS INSPECTION

The CRDM Nozzle and Other Vessel Closure Penetrations Inspection Program verifies the assumptions made in the BWOG safety evaluation of the susceptibility and consequence of PWSCC in B&W-designed CRDM nozzles. The scope of the program includes the B&W-designed reactor vessel closure head CRDM nozzles and other closure head penetrations. The aging effect is PWSCC of Alloy-600 nozzles with partial penetration welds that cause high circumferential residual stresses on the inner diameter of the nozzles opposite the welds.

16.2.8 FIRE PROTECTION

Fire Protection Program activities, with respect to aging management, include: fire barrier inspections, fire hose station inspections, fire suppression water supply system surveillance, fire suppression sprinkler system surveillance, fire water piping thickness evaluation, control room halon fire system inspection, NFPA 25 testing of sprinkler head components that are 50 years old, and RCP oil collection system visual inspection.

16.2.8.1 Fire Barrier Inspections

Fire Barrier Inspections provide for periodic surveillance of fire barriers separating redundant safe shutdown systems to assure that they perform their separation functions. The scope includes 10CFR50.48-required fire walls and fire floors as indicated on the fire protection drawings. Fire doors/hatches, fire damper mountings, fire wraps, and penetration fire stops associated with 10CFR50.48-required fire walls and fire floors are within the scope. The aging effects for fire barriers are cracking, loss of material, and change in material properties.

16.2.8.2 Fire Hose Station Inspections

The Fire Hose Station Inspections assure that manual fire suppression is available to safety-related equipment. Fire hose reels associated with 10CFR50.48-required fire hose stations are within the scope of license renewal. The aging effect for fire hose reels is a loss of material.

16.2.8.3 Fire Suppression Water Supply System Surveillance

This surveillance verifies operability of fire suppression water supply system components. The Fire Suppression Water Supply System Surveillance applies to fire water system supply piping and valves. The surveillance applies to several diesel fire pump subsystems including the intake air, exhaust, lube oil, and cooling water. Fire protection system heat exchangers are also within the scope of this surveillance. This program verifies that loss of material due to internal surface corrosion and fouling of carbon steel, stainless steel, brass, or bronze components is managed. Cracking of stainless steel, brass or bronze components is also an aging effect being managed.

16.2.8.4 Fire Suppression Sprinkler System Surveillance

This surveillance verifies operability of fire suppression sprinkler system components. Within the scope of license renewal, the Fire Suppression Sprinkler System Surveillance applies to fire suppression sprinkler system piping, valves, and nozzles. This surveillance verifies that loss of material due to internal surface corrosion and fouling of carbon steel, stainless steel, brass or bronze components is managed. Cracking of stainless steel, brass, or bronze components is also an aging effect being managed.

16.2.8.5 Fire Water Piping Thickness Evaluation

The Fire Water Piping Thickness Evaluation provides a method for the examination and evaluation of pipe wall thickness changes in the fire water system. Within the scope of license renewal, the Fire Water Piping Thickness Evaluation applies to fire water system piping. A loss of material by internal surface corrosion of cast iron, carbon steel, or stainless steel fire water system components is the aging effect managed by the Fire Water Piping Thickness Evaluation.

16.2.8.6 <u>Control Room Halon Fire System Inspection</u>

The Control Room Halon Fire System Inspection assures that frequently manipulated components are free of aging effects. The components within the scope of the Control Room Halon Fire System Inspection are the halon discharge tube assembly and the halon pilot header flexible tubing and fittings and discharge tube assembly fittings. The aging effects addressed by the Control Room Halon Fire System Inspection are loss of material due to wear from frequent manipulations and cracking.

16.2.8.7 <u>Reactor Coolant Pump Oil Collection System Visual Inspection</u>

The Reactor Coolant Pump Oil Collection System Inspection ensures integrity of the reactor coolant pump oil leakage collection system. The Reactor Coolant Pump Oil Collection System Inspection applies to the shrouds, drip pans, dammed areas, accessible piping, collection tanks, and spray protection. The aging effects addressed by the Reactor Coolant Pump Oil Collection System Inspection are a loss of material and a loss of mechanical closure integrity. These aging effects would be caused by general corrosion of the carbon steel internal surfaces or external surfaces due to the potential for water leakage into the system.

16.2.9 FLOW ACCELERATED CORROSION PREVENTION

The Flow Accelerated Corrosion Prevention Program provides a programmatic approach for identifying, inspecting, and managing loss of material for components that are adversely affected by flow accelerated corrosion (also known as erosion/corrosion). Within the scope of license renewal, only the main feedwater and main steam systems are identified as susceptible to flow-accelerated corrosion. The aging effect is a phenomenon that results in metal loss from components made of carbon steel, which occurs only under certain conditions of flow, chemistry, geometry, and material. The aging management reviews credit this program with determining which systems are susceptible to flow-accelerated corrosion and monitoring the loss of material for those systems.

16.2.10 INSPECTION AND PREVENTIVE MAINTENANCE OF THE POLAR CRANE

The program provides for the inspection and preventive maintenance of the polar crane. The scope of this program includes the structural steel associated with the polar crane. The aging effect managed by the Inspection and Preventive Maintenance of the Polar Crane is a loss of material.

16.2.11 INSTRUMENT AIR QUALITY

With respect to license renewal, the Instrument Air Quality Program ensures that the instrument air supplied to components is maintained free of water and significant contaminants. The Instrument Air Quality Program applies to those components, within the scope of license renewal, supplied with instrument air where pressure boundary integrity is required for the component to perform its intended function. The aging effects requiring management addressed by the Instrument Air Quality Program are loss of material and cracking.

16.2.12 LEAKAGE DETECTION IN REACTOR BUILDING

Leakage detection in the reactor building monitors leakage to manage the consequences of cracking, loss of material, or loss of mechanical closure integrity. Leakage detection in the reactor building is focused on RCS leakage, but also includes other systems that have the potential to leak in the reactor building.

16.2.13 MAINTENANCE RULE

Maintenance Rule system and structural walk downs are conducted to detect and manage aging effects of structures and components within the scope of the license renewal. This includes coatings inspections of coated surfaces on structures and components. The Maintenance Rule is utilized to manage cracking, loss of material, and change in material properties of structures and components within the scope of license renewal.

16.2.14 OIL ANALYSIS

The Oil Analysis Program ensures the oil environment in the mechanical systems is maintained to the quality required. Oil analysis program controls are credited as a program for maintaining oil systems free of contaminants (primarily water and particulates) thereby preserving an environment that is not conducive to corrosion. The scope of the Oil Analysis Program, with respect to license renewal, is limited to sampling activities and analysis on the auxiliary building electrical room chillers, emergency diesel generators, decay heat pumps, reactor building spray pumps, primary makeup pumps, diesel driven fire pump and engine, EFW pumps and turbine, the AAC diesel generator, and the control room chiller compressor. The Oil Analysis Program has been credited for ensuring the oil is free of water or contaminants. This manages the aging effects of cracking and loss of material.

16.2.15 PREVENTIVE MAINTENANCE

The purpose of the Preventive Maintenance Program is to perform preplanned, repetitive maintenance tasks on plant components and systems to extend equipment operating-life and to minimize the possibility of in-service component failures. The scope of the Preventive Maintenance Program, with regard to license renewal, is the preventive maintenance tasks credited with managing the aging effects listed below:

Preventive Maintenance Activity	Aging Effect
BWST internal inspection	Loss of material
BWST external inspection	Loss of material and loss of mechanical closure integrity
Reactor building ventilation cooling coil cleaning and inspection	Fouling and loss of material
Hydrogen sampling system cabinet/heat exchanger cleaning, inspection, and lubrication	Fouling
Emergency fire diesel cooling water quarterly sampling for corrosion inhibitor	Loss of material
Penetration room floor drain check valves inspection	Loss of material and cracking
Decay heat room drain valves inspection	Loss of material, cracking, and loss of mechanical closure integrity
EDG fuel oil tank inspection	Loss of material
EDG HVAC components inspection	Loss of material and loss of mechanical closure integrity
Control room ventilation inspections	Fouling and loss of material
Battery Charger and Penetration Room Cooler Cleaning and Inspection	Loss of material and loss of mechanical closure integrity
Auxiliary Building Switchgear Room Coolers Cleaning and Inspection	Loss of material, loss of mechanical closure integrity, and fouling

Preventive Maintenance Activity	Aging Effect
Auxiliary Building Decay and Heat Room Coolers Cleaning and Inspection	Loss of material, loss of mechanical closure integrity, and fouling
HPI Room Coolers Cleaning and Inspection	Loss of material, loss of mechanical closure integrity, and fouling

16.2.16 REACTOR BUILDING LEAK RATE TESTING

The Reactor Building Leak Rate Testing Program provides assurance that leakage from the reactor building does not exceed required maximum values for reactor building leakage. The Reactor Building Leak Rate Testing Program is comprised of Integrated Leak Rate Testing and Local Leak Rate Testing. The integrated leak rate test measures the primary reactor building overall integrated leakage rate. The scope of the integrated leak rate test is the entire reactor building. Integrated leak rate testing identifies loss of material or cracking. The local leak rate test measures the leakage across individual penetration components and determines the leakage of each penetration. Local leak rate testing identifies changes in material properties, loss of material, and cracking.

16.2.17 REACTOR BUILDING SUMP CLOSEOUT INSPECTION

The Reactor Building Sump Closeout Inspection detects significant degradation of the sump components and removes foreign objects that could impede suction from the sump. The scope of the Reactor Building Sump Closeout Inspection applies to reactor building sump, the area immediately surrounding the sump, the screening materials, and the equipment inside the sump. The aging effects addressed by the Reactor Building Sump Closeout Inspection are loss of material for the carbon steel components and cracking for stainless steel components due to the presence of borated water.

16.2.18 REACTOR VESSEL INTEGRITY

The ANO-1 Reactor Vessel Integrity Program consists of the following five interrelated subprograms:

- Master Integrated Reactor Vessel Surveillance Program (MIRVP)
- Cavity Dosimetry Program
- Fluence and Uncertainty Calculations
- Pressure/Temperature Limits
- Monitoring Effective Full Power Years

The purpose of the MIRVP is to monitor reactor pressure vessel materials containing Linde 80 high copper beltline welds to determine the reduction of material toughness by neutron irradiation embrittlement. The purpose of the Cavity Dosimetry Program is to verify the accuracy of fluence calculations and to determine fluence uncertainty values. The purpose of the reactor vessel fluence and uncertainty calculations is to provide an accurate prediction of the actual reactor vessel accumulated neutron fast fluence value, for use in development of the pressure/temperature limit curves and pressurized thermal shock calculations. The purpose of the pressure/temperature limit curves is to establish the normal operating limits for the RCS. The pressure/temperature limit curves apply to the reactor vessel. The purpose of determining the EFPY is to accurately monitor and tabulate the accumulated operating time experienced by the reactor vessel. The reduction of material toughness by neutron irradiation embrittlement is the aging effect addressed by these five subprograms.

16.2.19 SERVICE WATER INTEGRITY

The service water integrity program ensures the service water system components continue to operate and perform their safety-related functions for the remaining life of ANO-1. The scope of the Service Water Integrity Program, with respect to license renewal, is limited to activities on service water system components and structures, including the emergency cooling pond. The Service Water Integrity Program has been credited with managing the following aging effects:

- The flow rate testing ensures the effects of fouling do not reduce flow rates below required values.
- The heat exchanger testing manages the aging effect of fouling by ensuring the heat exchangers can remove the necessary heat load.
- The thickness mapping and visual inspections manage the aging effects of loss of material from the service water components.
- Visual inspections of a sample of safety-related valves and heat exchangers manage the effect of cracking of the components.
- The service water bay inspection is credited for managing loss of material of the mechanical components in the service water bay.

16.2.20 STEAM GENERATOR INTEGRITY

The Steam Generator Integrity Program ensures the steam generator integrity is maintained under normal operating, transient, and postulated accident conditions. The Steam Generator Integrity Program applies to the steam generator internals, tubing, and associated repair techniques and components, such as plugs and sleeves. The aging effects addressed by the Steam Generator Integrity Program are loss of material, cracking, and fouling.

16.2.21 SYSTEM AND COMPONENT MONITORING, INSPECTIONS, AND TESTING

16.2.21.1 Annual Emergency Cooling Pond Sounding

This program verifies the availability of a sufficient supply of cooling water in the emergency cooling pond to handle design basis accidents, with a concurrent loss of Lake Dardanelle. The scope includes the emergency cooling pond and surrounding structural components. The aging effect managed by this program is a loss of form of the emergency cooling pond due to sedimentation.

16.2.21.2 Battery Quarterly Surveillance

The battery rack inspections ensure their structural integrity. Seismically-qualified battery racks are within the scope. Battery racks and associated threaded fasteners are inspected for physical damage or abnormal deterioration including a loss of material.

16.2.21.3 Control Room Ventilation Testing

With respect to license renewal, the Control Room Ventilation Testing is credited as one of the programs to manage the aging effects. The control room ventilation testing applies to the control room emergency cooling coils. Fouling on the external surfaces of the cooling coil tubes is the aging effect managed by this program.

16.2.21.4 Core Flood Tank Monitoring

With respect to license renewal, the core flood tank monitoring provides a method to manage the aging effect of loss of material due to boric acid corrosion. The core flood tank monitoring applies to both core flood tanks. The loss of material due to boric acid corrosion on parts wetted by leaks from the core flood tanks may be detected through core flood tank monitoring.

16.2.21.5 Emergency Diesel Generator Testing and Inspections

With respect to license renewal, EDG Testing and Inspections provide a means of detecting aging effects associated with the various emergency diesel generator subsystems. The scope for these activities includes the emergency diesel generator assembly and associated support components. Loss of material is an aging effect for the carbon steel components in the EDG starting air system. Loss of material is identified as an aging effect for the unpainted carbon steel internal surfaces and the outer portion of the intake that could be wetted by rain. Loss of material and fouling are considered aging effects for the EDG intake air after coolers. Loss of material from the piping and muffler internal surfaces and from external surfaces exposed to the weather is an aging effect for the EDG exhaust components. Loss of material and fouling are aging effects for the lube oil coolers. The cooling water carbon steel components are susceptible to a limited loss of material from corrosion and the stainless steel components have the aging effect of cracking. Loss of material and fouling are aging effects for the cooling water heat exchangers. Since the portions of the subsystems on the engine are exposed to high vibration, loss of bolted closure integrity was identified as an aging effect for the skid mounted and connected components.

16.2.21.6 Emergency Feedwater Pump Testing

With respect to license renewal, the Emergency Feedwater Pump Testing is credited as one of the programs for managing the effects of aging. The scope includes the turbine and electric motor driven EFW pumps and associated components. Fouling in the system heat exchangers is the primary aging effect that this testing will identify. This testing also is credited with identifying the aging effects of loss of material and loss of mechanical closure integrity for system components.

16.2.21.7 NaOH Tank Level Monitoring

The NaOH tank level monitoring provides a method of detecting changes in the tank level that might indicate leakage from the NaOH tank or system. The NaOH tank level monitoring applies to the NaOH system components. This inspection is credited with managing the aging effects of loss of material, loss of mechanical closure integrity, and cracking.

16.2.21.8 Spent Fuel Pool Level Monitoring

The spent fuel pool level monitoring provides a method of detecting changes in the spent fuel pool level that might indicate cracks in the spent fuel pool liner. The spent fuel pool level monitoring applies to the detection of leakage through the spent fuel pool liner. Cracking of the spent fuel pool liner is the aging effect addressed by spent fuel pool level monitoring.

16.3 <u>TIME-LIMITED AGING ANALYSIS ACTIVITIES</u>

16.3.1 REACTOR VESSEL NEUTRON EMBRITTLEMENT

The reactor vessel is described in Sections 4.3.3. Time-limited aging analyses applicable to the reactor vessel are:

- neutron embrittlement of the beltline region, including pressurized thermal shock and Charpy upper-shelf energy reduction; and
- intergranular separation in the heat affected zone of low alloy steel under austenitic stainless steel weld cladding is addressed in Section 16.3.7.

The Reactor Vessel Integrity Program as described in Section 16.2.18 is being utilized to ensure that the time dependent parameters used in the time-limited aging analysis evaluations for pressurized thermal shock and Charpy upper-shelf energy reduction are tracked such that the time-limited aging analysis remains valid through the period of extended operation.

16.3.2 METAL FATIGUE

Cyclic loads are described in Section 4.1.2.4. For the extension of plant service-life from 40 years to 60 years, metal fatigue resulting from thermal transient cyclic loads is considered a time-limited aging analysis as defined by 10CFR54.21(c). The time-limited aging analysis requires metal fatigue evaluations to remain valid for the period of extended operation. This is achieved by maintaining adequate documentation of fatigue stress analyses to show the allowable design cycles of the RCS components for the applicable transient events and monitoring or tracking the actual operating cycles to ensure the allowable cycles are not exceeded. Fatigue evaluations are performed based on the design-allowable cycles specified in Table 4-8 and the NSSS vendor functional specification. As such, the fatigue evaluations and the fatigue life of the RCS components are dependent on the actual operating cycles and independent of the service-life of the plant. As long as the number of applicable transient cycles are below the design-allowable cycles, the fatigue evaluations originally performed for a 40-year plant service-life are applicable for the 60-year plant service-life. Transient cycle logging will be maintained to ensure the fatigue analysis assumptions remain valid during the period of extended operation.

In addition to the fatigue analysis, credit is taken for risk-informed inservice inspection of the pressurizer surge line, HPI nozzles, and decay heat removal suction line to manage the potential effects of environmentally assisted fatigue. The risk-informed inservice inspections also address commitments for augmented inspections related to NRC Bulletins 88-08 and 88-11. Additional augmented inspections related to the HPI/MU nozzles in B&W plants will be continued during the period of extended operation.

16.3.3 ENVIRONMENTAL QUALIFICATION

Environmental qualification evaluations of electrical equipment are identified as time-limited aging analyses. Equipment covered by the EQ Program has been evaluated to determine if existing EQ aging analyses can be projected to the end of the period of extended operation by re-analysis or additional analysis. Qualification into the license renewal period will be treated the same as equipment initially qualified for 40 years or less. When aging analysis cannot justify a qualified life into the license renewal period, then the component or parts will be replaced prior to exceeding the qualified life in accordance with the EQ Program.

16.3.4 CONCRETE REACTOR BUILDING TENDON PRESTRESS

The analysis of tendon prestress over the original license term is a time-limited aging analysis that was performed at the time of initial licensing. The containment tendon elements are acceptable for the period of extended operation based on projected tendon relaxation. In addition, the ASME Section XI, Inservice Inspection Program IWL inspections will be adequate to manage the effects of aging on the intended function for the period of extended operation.

16.3.5 REACTOR BUILDING LINER PLATE FATIGUE ANALYSIS

Several thermal cycling conditions, which include annual outdoor temperature variations, changes in interior temperature during start-up and shutdown of the reactor coolant system, and DBA conditions were considered in the fatigue analysis of the liner plate. This analysis is a time-limited aging analysis. The projected number of cycles for these loadings for 60 years is bounded by the existing fatigue analysis. Therefore, the original design assumptions for addressing thermal fatigue of the liner plate and piping penetrations remain valid for the period of extended operation.

16.3.6 AGING OF BORAFLEX IN SPENT FUEL POOL RACKS

The Boraflex in the spent fuel pool racks is discussed in Section 9.6.2.3. Potential stressors for the Boraflex in the spent fuel pool racks include gamma flux, which changes the material characteristics of the base polymer, and chemical environment, from the exposure to borated water. Continued monitoring and analyses of the Boraflex degradation was committed to in response to Generic Letter 96-04. In order to ensure that the 5 percent subcriticality margin can be maintained for the life of the spent fuel storage racks, the existing coupon monitoring program will be continued. Spent fuel pool silica levels will continue to be monitored and silica evaluations will continue to be performed in order to confirm that the 5 percent subcriticality margin will be maintained through the next evaluation period. These reanalysis and sampling actions provide reasonable assurance that the effects of aging on the Boraflex in the spent fuel pool racks will be adequately managed for the period of extended operation.

16.3.7 REACTOR VESSEL UNDERCLAD CRACKING

Intergranular separations (underclad cracking) in low alloy steel heat-affected zones under austenitic stainless steel weld claddings were detected in SA-508, Class-2 reactor vessel forgings manufactured to a coarse grain practice and clad by high-heat-input submerged arc processes. BAW-10013 contains a fracture mechanics analysis that demonstrates the critical crack size required to initiate fast fracture is several orders of magnitude greater than the assumed maximum flaw size, plus predicted flaw growth due to design fatigue cycles. The flaw growth analysis was performed for a 40-year cyclic loading, and an end-of-life assessment of radiation embrittlement (i.e., fluence at 32 EFPY) was used to determine fracture toughness properties. The report concluded that the intergranular separation found in B&W vessels would not lead to vessel failure. This conclusion was accepted by the AEC. To cover the period of extended operation, an analysis was performed using current ASME Code requirements. This analysis is fully described in BAW-2251A. The maximum applied stress intensity factor for the emergency and faulted conditions occurs in the closure head to head flange region and the fracture toughness margin was determined to be 2.24, which is greater than the IWB-3612 acceptance criterion of 1.41. It is therefore concluded that the postulated intergranular separations in the ANO-1 reactor vessel 508 Class-2 forgings are acceptable for continued safe operation through the period of extended operation.

16.3.8 REACTOR VESSEL NOZZLES – FLOW-INDUCED VIBRATION ENDURANCE LIMIT

Report BAW-10051, "Flow Induced Vibration Endurance Limit Assumptions," calculated stress values for the reactor vessel incore nozzles and compared them to endurance limit (stress) values. These endurance limit values were based on an assumed value of 10^{12} cycles for 40 years of operation. The number of fatigue cycles was extended for 60 years, and the component item stress values were compared to the recalculated endurance limit values and shown to be acceptable.

16.3.9 LEAK-BEFORE-BREAK

The successful application of leak-before-break to the main RCS piping is described in report BAW-1847 "The B&W Owners Group Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSSS." This report provides the technical basis for evaluating postulated flaw growth in the main RCS piping under normal plus faulted loading conditions. LBB LOCA loadings are considered for the faulted analyses of fuel assemblies as described in Section 3.3.3.3.2.1, and LBB is credited in the Section 4.2.6.6 to justify pipe whip restraints being no longer required on the main RCS piping and may be removed as needed to facilitate maintenance or other activities. In addition, the analysis of reactor building internal pressure differentials following a LOCA applies LBB in the selection of breaks to be analyzed (see Section

14.2.2.5.5.2). In summary, analyses have been performed to demonstrate that LBB remains valid for the period of extended operation.

16.3.10 RCP MOTOR FLYWHEEL

Flaw growth analysis associated with the RCP motor flywheel is a time-limited aging analysis. The analysis for fatigue crack growth addresses the growth of pre-existing cracks. The crack growth analysis was performed for 4,000 startup/shutdown cycles for the RCP motors, which exceeds the number of design cycles by a factor of eight. Therefore, the existing crack growth analysis remains valid for the period of extended operation.

Appendix B

Aging Management Programs and Activities

1.0 INTRODUCTION

The ANO-1 Integrated Plant Assessment comprises four major activities. The first two activities, “Identification of Structures and Components that are Subject to Aging Management Review,” and “Identification of Aging Effects Requiring Management,” have been described previously in the body of the LRA. The third major activity of the ANO-1 IPA is the identification of plant-specific programs and activities that will manage the aging effects identified as requiring management. These programs and activities are described in this appendix, “Aging Management Programs and Activities.” The fourth major activity of the ANO-1 IPA, the aging management demonstration for programs and activities, is also presented in this appendix.

The ANO-1 programs and activities that are credited for managing aging may be divided into new actions and existing actions. The new programs and activities are described in Section 3.0 of this appendix. The existing programs and activities are described in Section 4.0 of this appendix. The descriptions have used a series of specific attributes to facilitate the description of the programs and activities. These attributes are defined in Section 2.0. Summary descriptions of new and existing programs and activities are contained in the ANO-1 SAR Supplement, which is provided in Appendix A of the LRA.

ANO-1 programs and activities that are credited during the aging management review are described in this appendix. These programs and activities provide reasonable assurance that the effects of aging will be adequately managed so that the structures and 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. The demonstrations, along with the program and activity descriptions, meet the requirements specified in 10CFR54.21(a)(3). Along with the technical information contained in the body of the LRA, this appendix is intended to allow the NRC to make the finding contained in 10CFR54.29(a)(1).

2.0 PROGRAM AND ACTIVITY ATTRIBUTES

The attributes that are used to describe aging management programs and activities are described in this section. NEI 95-10, Revision 0, Sections 4.2 and 4.3 [Reference B-1] and the Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants [Reference B-2] served as primary inputs to the attribute definitions used in this appendix.

Two attributes common to all programs and activities described in this appendix are corrective actions and administrative controls. The Entergy Quality Assurance Program applies to ANO-1 safety related structures and components. Corrective actions and administrative (document) control for both safety related and non-safety related structures and components are accomplished per the existing ANO corrective action program and the ANO document control program. These two programs apply to the corrective actions and administrative controls for the programs and activities within the scope of license renewal. Accordingly, discussion of corrective actions and administrative control is not included in the summary descriptions of the individual programs and activities in this appendix.

The remaining attribute definitions used to describe new and existing programs and activities are provided below.

Purpose: A clear statement of the reason why the program or activity exists for ANO-1 license renewal.

Scope: A description of the ANO-1 structures and components within the scope of license renewal and subject to an aging management review that are encompassed by the program or activity.

Aging Effects: A description of the aging effects requiring management or the relevant physical conditions to be monitored for the identified scope of structures and components.

Method: A description of the type of action or technique used to identify or manage the aging effects or relevant conditions (e.g., visual examination of the component).

Sample Size: For new programs or activities, a sample can be identified from the total population of affected structures and components for inspection or monitoring. If a sample is chosen for inspection or monitoring, a description of the sample is provided.

Industry Codes or Standards: A description of an industry code (e.g., ASME Section XI, IEEE) or an industry standard (e.g., ASTM or NRC-approved BWOG report) that guides or governs the program or activity. This attribute may not be applicable to some programs and activities.

Frequency: A description of the frequency of action that is established for detection of aging effects or relevant physical conditions.

Acceptance Criteria or Standard: Acceptance criteria or standards are described for the relevant conditions to be monitored or the chosen examination methods.

Timing of New Program or Activity: For new or modified programs or activities, an identification of the specific timing for the implementation or modification of the program or activity.

Regulatory Basis: For existing programs and activities, an identification of any existing regulatory basis for these actions, such as the technical specifications. This attribute may not be applicable to some programs and activities.

Operating Experience and Demonstration: (For existing programs and activities) A demonstration that the program or activity can adequately manage the aging effects. In addition, operating experience relevant to the demonstration is provided.

Demonstration: (For new programs and activities) A demonstration that the program or activity can adequately manage the aging effects.

3.0 NEW ACTIVITIES

3.1 BURIED PIPE INSPECTION

Purpose: Buried Pipe Inspections will be performed to ensure that a loss of material due to external surface corrosion of buried piping is adequately managed.

Scope: The safety-related portions of underground carbon steel piping on the ANO-1 service water and fuel oil systems are within the scope of this inspection.

Aging Effects: The aging effect addressed by the Buried Pipe Inspection is a loss of material due to corrosion of the external surfaces of pipe caused by loss of the protective coating.

Method: Visual inspection of the protective coating will be performed.

Sample Size: When underground piping is uncovered during plant maintenance or modification activities, inspection of the protective coating will be performed. Sampling of underground pipe would become warranted if observations of defective protective coatings or losses of material on external surfaces of piping were seen during inspections.

Industry Code or Standards: Not applicable.

Frequency: Inspections will be performed when plant maintenance or modifications uncover buried piping. If defective protective coatings or loss of material is observed, then the frequency will be evaluated.

Acceptance Criteria or Standard: Acceptance criteria will be defined in plant procedures. The extent of the defective coatings or corrosion on the external surfaces of piping will determine the nature of the corrective action.

Timing of New Program or Activity: This inspection will be initiated prior to the end of the initial 40-year license term.

Demonstration: The Buried Pipe Inspection will be effective in the future for managing aging effects since it incorporates proven monitoring techniques, acceptance criteria, corrective actions, and administrative controls from existing programs and procedures. The implementation of the new Buried Pipe Inspection provides reasonable assurance that the effects of aging will be adequately managed so that the components within the scope of this program will perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.2 ELECTRICAL COMPONENT INSPECTION

Purpose: The Electrical Component Inspection Program will inspect splices, connectors, and cables located in areas that may be conducive to accelerated aging.

Scope: The scope of this inspection includes cables exposed to elevated temperatures, wet environments, or corrosive chemicals. The scope also includes cables that can experience elevated temperatures due to the current they are carrying and connectors used in impedance-sensitive circuits and cable splices subject to aging-related stressors.

Aging Effect: The aging effect for cables and cable splices is change of material properties as evidenced by cracking or discoloration of the insulation. The aging effect for connectors in impedance-sensitive circuits is corrosion of connector pins.

Method: Visual inspection will be used to detect aging effects in the cables, splices, and connectors in the scope of this program.

Sample Size: Samples may be used for this program. If used, an appropriate sample size will be determined prior to the inspection or test.

Industry Codes or Standards: Not applicable.

Frequency: Cables, splices, and connectors in the selected sample will be inspected at least once every 10 years.

Acceptance Criteria or Standard: No unacceptable visual indications of age related degradation of the cables, splices, and connectors in scope.

Timing of New Program or Activity: The Electrical Component Inspection Program will be formally implemented and the first inspection of in-scope cables, splices, and connectors completed prior to the expiration of the initial 40-year licensing term.

Demonstration: The Electrical Component Inspection will be effective in the future for managing aging effects since it incorporates proven monitoring techniques, acceptance criteria, corrective actions, and administrative controls from existing programs and procedures. The implementation of the new Electrical Component Inspection provides reasonable assurance that the effects of aging will be adequately managed so that the components within the scope of this program will perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.3 HEAT EXCHANGER MONITORING

Purpose: The Heat Exchanger Monitoring Program will inspect heat exchangers to the extent required to ensure seismic qualification is maintained.

Scope: The Heat Exchanger Monitoring Program manages aging effects on the following ANO-1 safety-related systems and components:

<u>Service Water System</u>

Reactor building coolers

Emergency diesel generator jacket cooling water heat exchangers

Make-up pump lube oil coolers

Make-up pump room coolers

Decay heat room coolers

Decay heat system heat exchangers

Electrical room chillers and coolers

<u>Control Room Ventilation System</u>

Control room chillers and coolers

<u>Emergency Feedwater System</u>

Emergency feedwater system lube oil coolers

Aging Effects: The aging effects addressed by the Heat Exchanger Monitoring Program are cracking and loss of material that could result in degradation in the seismic qualification of the heat exchangers.

Method: Non-destructive examinations, such as eddy-current inspections or visual inspections, will be performed.

Sample Size: An appropriate sample population of heat exchangers will be determined based on operating experience prior to the inspections.

Industry Codes or Standards: Not applicable.

Frequency: Inspections will be performed periodically at a frequency to be determined prior to implementation.

Acceptance Criteria or Standard: No unacceptable indication of loss of material or cracking as determined by engineering analysis.

Timing of New Program or Activity: Inspection will be initiated prior to the end of the initial 40-year license term for ANO-1.

Demonstration: The Heat Exchanger Monitoring activity will be effective in the future for managing aging effects since it incorporates proven monitoring techniques, acceptance criteria, corrective actions, and administrative controls from existing programs and procedures. The implementation of the new Heat Exchanger Monitoring activity provides reasonable assurance that the effects of aging will be adequately managed so that the systems within the scope of this program will perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.4 PRESSURIZER EXAMINATIONS

Section 2.3.1 of the ANO-1 LRA identifies the pressurizer as subject to aging management review. Section 3.2 of the ANO-1 LRA and BAW-2244A [Reference B-3] identify the aging effects that will require new or additional inspections for license renewal. These aging effects are cracking of pressurizer cladding, including items attached to the cladding (e.g., tripod legs), which may result in cracking or loss of underlying ferritic steel; cracking of the structural welds that connect the heater sheaths to the diaphragm plates; and cracking of small bore nozzles and safe-ends.

Management of aging effects for the pressurizer Alloy-600 small bore nozzles is addressed in the Alloy-600 Aging Management Program. Small bore safe ends are addressed in the Small Bore Piping and Small Bore Nozzle Inspections. The Pressurizer Examinations include the pressurizer cladding examination, and the pressurizer heater penetration weld examination, which are described in the following sections.

3.4.1 Pressurizer Cladding Examination

Purpose: The pressurizer cladding examination will assess the condition of the pressurizer cladding.

Scope: The scope of this activity will include the cladding and attachment welds to the cladding of the ANO-1 pressurizer.

Aging Effects: The aging effect is cracking of cladding by thermal fatigue, which may propagate to the underlying ferritic steel.

Method: Volumetric examination of the pressurizer items that are most susceptible to thermal fatigue will provide assurance that cracking of cladding has not extended into the base metal of the pressurizer. Pressurizer items with the highest cumulative usage factor include the circumferential weld that connects the shell to the lower head and the full penetration weld that connects the pressurizer surge nozzle to the lower head.

In accordance with ASME Section XI, Examination Category B-B, volumetric examination of the circumferential shell-to-head weld is performed each inspection interval. In addition, 1 ft of the longitudinal weld adjacent to the heater belt forging is volumetrically examined. The weld that connects the surge nozzle to the lower head receives volumetric examination each inspection interval in accordance with Examination Category B-D. Continuation of these inspections during the period of extended operation will manage any cracking of cladding that may extend into the base metal at the locations that are most susceptible to thermal fatigue.

Sample Size: The pressurizer design report was reviewed and the stainless steel clad carbon steel items with the highest cumulative usage factors include the circumferential weld that connects the shell to the lower head and the weld that connects the surge nozzle to the lower head. Inspection of these items will bound the remaining stainless steel clad carbon steel items in the pressurizer.

Industry Code or Standards: ASME Section XI, 1992 Edition with portions of the 1993 Addenda, including mandatory Appendices VII and VIII (Appendix VIII in accordance with 10CFR50.55a).

Frequency: The examination frequency is defined by ASME Section XI, Table IWB-2500-1, Examination Categories B-B and B-D.

Acceptance Criteria or Standard: Acceptance standards for volumetric examination in accordance with ASME Section XI, IWB-3510 and IWB-3512.

Timing of New Program or Activity: The inspections discussed above are included in the current ANO-1 ISI program and will be carried forward to the period of extended operation.

Demonstration: As a result of pressurizer cladding cracking that occurred at Haddam Neck, cracking of cladding in the pressurizer was evaluated as a potential aging effect requiring aging management. The concern is that cracks in the cladding may extend into the underlying ferritic steel and subsequent growth of the crack may propagate and remain undetected. Based on differences in design, fabrication, and operation, cladding cracking and propagating into the ferritic base material in the ANO-1 pressurizer is not expected.

The Pressurizer Cladding Examination includes multiple volumetric examinations of pressurizer items having the highest fatigue usage factors. Any cracking of the cladding that extends into the base metal would be detected by ASME Section XI volumetric examinations at these locations. The multiple volumetric inspections being performed in accordance with ASME Section XI requirements conservatively envelope the one time inspection recommended in the SER in BAW-2244A. The Pressurizer Cladding Examination will provide reasonable assurance that the aging effects associated with the cladding and ferritic base material will be managed such that the applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.4.2 Pressurizer Heater Bundle Penetration Welds Examination

Purpose: The pressurizer heater bundle penetration welds examination will assess the condition of the pressurizer heater penetration welds.

Scope: This examination will be applicable to the heater sheath-to-diaphragm plate penetration welds inside the pressurizer. The pressurizer contains three heater bundles.

Aging Effects: The aging effect is cracking at the heater bundle penetration welds, which may lead to reactor coolant leakage.

Method: For the first heater bundle that is removed for replacement, a surface examination of sixteen peripheral welds will be performed. A visual examination (VT-3 or equivalent) of the remaining welds of the heater bundle will be performed.

In addition, ANO-1 inspects the exterior portions of the heater bundle each outage in accordance with Examination Category B-P of ASME Section XI. In accordance with IWA-5242 as modified by Code Case N-533 for bolted connections, ANO-1 will remove the insulation surrounding the penetrations and perform a VT-1 inspection. This addresses Open Item 2 in the NRC SER of BAW-2244A.

Sample Size: The examination will include sixteen peripheral heater penetration welds on one heater bundle. However, if the surface examination of the Oconee heater bundle penetration welds is performed before the ANO-1 bundle removal and indicates that cracking of the welds is not an aging effect, the heater bundle penetration welds at ANO-1 will not be inspected. The Oconee (Units 2 and 3) and ANO-1 heater bundle designs are identical, and inspection of the Oconee Unit 1 welds would bound ANO-1 since the Oconee Unit 1 welds are fabricated from Alloy 82/182 [BAW-2244A].

Industry Code or Standards: ASME Section XI, 1992 Edition, including mandatory Appendices VII and VIII (Appendix VIII in accordance with 10CFR50.55a).

Frequency: Pressurizer heater bundle penetration welds examination is a one-time inspection. If the results of the inspection are not acceptable, then the results may be used as a baseline inspection for establishing further programmatic actions covering the other two ANO-1 heater bundles.

Acceptance Criteria or Standard: Acceptance standards for surface examinations and visual examination (VT-3) will be in accordance with ASME Section XI.

Timing of New Program or Activity: The heater bundle examination may occur prior to entering the period of extended operation or during the period of extended operation. If the Oconee inspection occurs prior to the ANO-1 bundle removal and indicates that cracking of the heater penetration welds is an aging effect, the examination will be performed upon removal of an ANO-1 pressurizer heater bundle.

Operating Experience and Demonstration: No stainless steel heater sheath-to-diaphragm plate penetration welds have cracked to date on a B&W plant. Failures have occurred at other non-B&W plants on similar heater penetrations. However, these failures occurred on more susceptible Alloy-600 penetrations through hemispherical heads. The ANO-1 heaters are welded to a flat plate. Cracking is not expected on the ANO-1 pressurizer heaters during the period of extended operation. If through-wall cracking occurs, the resulting leakage will be detected by Leakage Detection in Reactor Building or ASME Section XI-IWB Inspection Programs. Because of the design of the heater sheath assembly, if leakage occurs, it will not be a safety concern. The mechanical design of the heater will limit leakage and provide adequate structural support even with a failure of the sheath-to-diaphragm welds. If a heater bundle is removed for replacement, surface examinations of the 16 peripheral heater penetrations and VT-3 or equivalent examinations of the remaining welds of the heater bundle will determine if cracking of these welds is an applicable aging effect. This will provide reasonable assurance that the aging effects associated with stainless steel heater penetration welds will be managed such that the applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.5 REACTOR VESSEL INTERNALS AGING MANAGEMENT

The establishment of the ANO-1 Reactor Vessel Internals Aging Management Program involves the combination of several activities culminating in the inspection of the ANO-1 reactor vessel internals once during the 20-year period of extended operation. Ongoing industry efforts are aimed at characterizing the aging effects associated with the reactor vessel internals. As described in BAW-2248A, [Reference B-4] aging effects were qualitatively assessed based on operating conditions and operating experience. Further understanding of these aging effects will be developed by the industry over time and will provide additional bases for the inspections under this program. The purpose of the ANO-1 Reactor Vessel Internals Inspection is to provide visual inspections and non-destructive examinations of the ANO-1 reactor vessel internals during the period of extended operation. The major activities associated with this program include participation in industry activities, reporting results to the NRC, and performance of inspections.

Entergy Operations will participate in the BWOG Reactor Vessel Internals Aging Management Program and other industry programs, as appropriate, to continue investigation of aging effects for the reactor vessel internals. These activities will assist in establishing the appropriate monitoring and inspection programs for the reactor vessel internals.

Entergy Operations will provide updates after the completion of significant milestones in the preparation of the ANO-1 Reactor Vessel Internals Inspection, commencing within one year of the issuance of the renewed license. Entergy Operations will submit a report to the NRC at, or about, the end of the initial 40-year operating license term. This report will summarize the understanding of aging effects applicable to the reactor vessel internals and will contain the Entergy Operations inspection plan including methods for each inspection.

Entergy Operations will perform the ANO-1 Reactor Vessel Internals Inspection as provided in the following summary description. Should data or evaluations indicate that this inspection can be modified or eliminated, Entergy Operations will provide plant-specific justification to demonstrate the basis for the modification or elimination.

Purpose: The purpose of the Reactor Vessel Internals Inspection is to inspect and examine the reactor vessel internals to assure that the aging effects will not result in loss of the intended functions of the internals during the period of extended operation.

Scope: The inspection applies to the reactor vessel internals stainless steel items for ANO-1. These items can be separated into three groups- (1) items comprised of plates, forgings, welds, core barrel bolts, and thermal shield bolts; (2) baffle bolts and; (3) items fabricated from CASS and martensitic steel. More specifically, the items fabricated from CASS and martensitic steel include control rod guide tube spacers, vent valve bodies, and incore guide tube assembly spiders. The vent valve retaining rings, fabricated from martensitic stainless steel, are also included in this inspection.

Aging Effects: The aging effects for plates, forgings, welds, core barrel bolts, and thermal shield are cracking due to stress corrosion and irradiation assisted stress corrosion, reduction of fracture toughness due to irradiation embrittlement, dimensional changes due to void swelling, and loss of bolted closure integrity due to stress relaxation.

The aging effects for baffle bolts are cracking due to irradiation assisted stress corrosion, reduction of fracture toughness due to irradiation embrittlement, and dimensional changes due to void swelling.

The aging effects for items fabricated from CASS and martensitic steel are reduction of fracture toughness by thermal embrittlement and irradiation embrittlement.

Method: Current plans are to perform a visual inspection of the plates, forgings, welds, core barrel bolts, and thermal shield bolts. Activities are in progress to develop and qualify the inspection method.

Current plans are to perform a volumetric inspection of the baffle bolts. Activities are in progress to develop and qualify the inspection method.

For items fabricated from CASS and martensitic steel, reduction of fracture toughness cannot be measured through traditional in-situ examination techniques, thus necessitating an analytical approach to assess the effect of reduction of fracture toughness on the applicable reactor vessel internals items. The specific inspection method will depend on the results of these analyses.

Sample Size: The sample size for the inspection of ANO-1 will be determined as part of the development of the inspection method.

Industry Codes or Standards: Not applicable.

Frequency: The Reactor Vessel Internals Inspection will be performed once during the twenty-year period of extended operation.

Acceptance Criteria or Standard: For plates, forgings, welds, core barrel bolts, and thermal shield bolts that will be visually inspected, critical crack size will be determined by analysis. Acceptance criteria will be developed prior to the inspection.

For baffle bolts, any detectable crack indication is unacceptable for a particular baffle bolt. The number of baffle bolts needed to be intact and their locations will be determined by analysis. Acceptance criteria for dimensional changes due to void swelling will be developed prior to the inspection.

For items fabricated from CASS and martensitic steel, critical crack size will be determined by analysis. Acceptance criteria for all aging effects will be determined prior to the inspection.

Timing of New Program or Activity: This inspection will begin during the period of extended operation.

Demonstration: Current plans are to perform a visual inspection of the plates, forgings, welds, core barrel bolts, and thermal shield bolts and to perform a volumetric inspection of the baffle bolts. ANO will participate in the BWOG Reactor Vessel Internals Aging Management Program and other industry programs as necessary. ANO will also participate in industry studies establishing appropriate monitoring and inspection programs for reactor vessel internals items. The final inspection program will provide reasonable assurance that the aging effects associated with the reactor vessel internals will be managed such that the applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.6 SPENT FUEL POOL MONITORING

Purpose: The purpose of the Spent Fuel Pool Monitoring Program will be to manage the aging effects requiring management of the ANO-1 spent fuel pool liner.

Scope: The scope of the Spent Fuel Pool Monitoring Program is concerned with the ANO-1 spent fuel pool liner.

Aging Effects: Stress corrosion cracking is possible from the external surface of the liner in weld heat-affected zones since this was not verified to be chloride free during construction.

Method: The spent fuel pool monitoring trench drains will be monitored to detect liner leakage. This will identify through-wall cracks existing in the spent fuel pool liner. During this test, trench drain valves will be opened, drained, and monitored.

Sample Size: Not applicable

Industry Code or Standards: Not applicable.

Frequency: Spent fuel pool leakage to the monitoring trench drains will be monitored quarterly.

Acceptance Criteria or Standard: No unacceptable leakage due to age-related degradation of the spent fuel pool liner.

Timing of New Program or Activity: This program will be initiated before the end of the initial 40-year license term.

Demonstration: The ANO-1 history of successful operation demonstrates that leakage monitoring and inspections have been effective in managing the effects of aging on components. This is indicative that the Spent Fuel Pool Monitoring activity will be effective in the future for managing aging effects since it incorporates proven monitoring techniques, acceptance criteria, corrective actions, and administrative controls from existing programs and procedures. Based on this experience, the implementation of the new Spent Fuel Pool Monitoring activity provides reasonable assurance that the effects of aging will be adequately managed so that the components within the scope of this program will perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.7 WALL THINNING INSPECTION

Purpose: Wall Thinning Inspections will be performed to ensure wall thickness is above the minimum required in order to avoid leaks or failures under normal conditions and postulated transient and accident conditions, including seismic events.

Scope: Wall Thinning Inspections cover the following safety-related systems and components:

<u>Emergency Feedwater System</u>

- Emergency feedwater pump casing and carbon steel discharge piping and valves
- Emergency feedwater steam supply components downstream of steam admission valves
- Emergency feedwater steam exhaust piping and valves
- Carbon steel cooling water, seal water, and instrument piping and valves
- Turbine lube oil cooler
- Carbon steel emergency feedwater supply header piping and valves (condensate supply)

<u>Chemical Addition System</u>

- NaOH Tank

<u>Main Steam System</u>

- Carbon steel piping and components

<u>Reactor Building Isolation</u>

- Carbon steel components of penetrations 11, 42, 43, 48, 49, 52, 54, 58, 60, 62, and 64
- Carbon steel components of penetrations 51 and 59 (chilled water system)

Aging Effects: The aging effect to be addressed by Wall Thinning Inspection is a loss of material due to corrosion of the internal surfaces of carbon steel piping and components.

Method: Non-destructive examinations, will be performed on susceptible component locations.

Sample Size: An appropriate sample size will be determined based on operating experience prior to these inspection activities.

Industry Codes or Standards: Components will be inspected in accordance with the applicable code for each respective system.

Frequency: Inspections will be performed periodically at a frequency to be determined prior to implementation. The frequency of inspections will depend upon results of previous inspections, calculated rate of material loss, and industry and plant operating experience.

Acceptance Criteria or Standard: Wall thickness measurements greater than minimum wall thickness values for the components design code of record will be acceptable.

Timing of New Program or Activity: This inspection will be initiated prior to the end of the initial 40-year license term for ANO-1.

Demonstration: The ANO-1 history of successful operation demonstrates that visual inspections have been effective in managing the effects of aging on components. This is indicative that the Wall Thinning Inspection activity will be effective in the future for managing aging effects since it incorporates proven monitoring techniques, acceptance criteria, corrective actions, and administrative controls from existing programs and procedures. Based on this experience, the implementation of the new Wall Thinning Inspection activity provides reasonable assurance that the effects of aging will be adequately managed so that the components within the scope of this program will perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.0 EXISTING ACTIVITIES

4.1 ALLOY-600 AGING MANAGEMENT

Purpose: The ANO-1 Alloy-600 Aging Management Program will manage cracking by PWSCC of Alloy-600 and Alloy 82/182 locations for the period of extended operation.

Scope: The Alloy-600 Aging Management Program will be applicable to the Alloy-600 items and Alloy 82/182 weld material in the RCS, including the hot leg flow meter element. The scope does not include the Alloy-600 OTSG tubes, which are addressed separately in the Steam Generator Integrity Program.

Aging Effects: The aging effect managed by the Alloy-600 Aging Management Program is cracking due to primary water stress corrosion of Alloy-600 items and Alloy 82/182 weld metal in the ANO-1 RCS.

Method: ANO-1 has implemented an augmented inspection program for Alloy-600 nozzles attached to the pressurizer. The augmented inspection program consists of visual examination (i.e., VT-2) of each nozzle from the exterior of the vessel each refueling outage. In addition, on the repaired Alloy-600 level sensing nozzle, the ferritic steel in the nozzle-bore is periodically examined using ultrasonic testing. ANO-1 will continue to monitor the pressurizer Alloy-600 partial penetration welded nozzles by performing VT-2 examinations of the nozzles from the exterior of the vessel each refueling outage, during the period of extended operation

The augmented visual examination from the exterior of the vessel is sufficient to monitor the most susceptible Alloy-600 items since the industry has proven through safety evaluations that PWSCC will form axial cracks that result in RCS leakage but do not compromise the structural integrity of the pressurizer. Surface or volumetric examination of the partial penetration welded nozzles is not justified due to the inaccessibility of the welded joint (1-inch Schedule 160 pipe would have to be cut) and the demonstrated low risk associated with longitudinal cracking of these penetrations.

The pressurizer spray nozzle Alloy-600 safe end to clad carbon steel spray nozzle dissimilar metal welded joint is volumetrically inspected each interval. In addition, the welded joint that connects the spray nozzle safe end to the stainless steel spray line is volumetrically inspected each interval. Since the most probable location for cracking is at, or near, the heat-affected zone, these volumetric inspections will include the heat-affected zones. The volumetric inspections of the dissimilar metal welds associated with the spray nozzle safe end will be carried forward to the period of extended operation.

Sample Selection: From the total population of Alloy-600 items and Alloy 82/182 weld locations at ANO-1, the top three location groupings with respect to susceptibility to PWSCC will be selected for inspection. Monitoring the most susceptible locations will bound the Alloy-600 items and Alloy 82/182 weld locations that are not inspected.

The method used to perform the susceptibility ranking of the Alloy-600 items and Alloy 82/182 welds includes the following steps.

1. Identify all Alloy-600 items and Alloy 82/182 weld metal used at ANO-1

2. Select a PWSCC susceptibility model

   The model used to rank the susceptibility of Alloy-600 items and Alloy 82/182 welds to PWSCC is similar to the CRDM Nozzle PWSCC Inspection and Repair Strategic Evaluation (CIRSE) model that was applied to the CRDM penetrations [See section 2.3.3 and Appendix B of Reference B-9]. Use of a concordant method ensures consistency between the Alloy-600 Aging Management Program and the CRDM nozzle and Other Vessel Head Penetration Inspection Program, which are two separate but related programs at ANO-1.

3. Select a reference Alloy-600 item for calculation of relative time to crack initiation.

   The reference item chosen for calculation of relative time to crack initiation is the ANO-1 pressurizer instrumentation nozzle that was identified as leaking in 1990.

4. Evaluate the differences in material and operating parameters between the reference Alloy-600 part and the remaining Alloy-600 items and Alloy 82/182 welds.

   The specific material and operational parameters that were compared include maximum inside surface stress, operating temperature, microstructure, surface condition, and water chemistry.

5. Calculate the relative susceptibility factor for the Alloy-600 items and Alloy 82/182 welds relative to the time of crack initiation for the reference Alloy-600 item.

The differences in material and operational parameters were used to calculate a relative susceptibility factor, which is described in Appendix B to BAW-2301 [Reference B-9], for each Alloy-600 item and Alloy 82/182 weld. The relative susceptibility factors were used to calculate an estimated time to crack initiation for a specific Alloy-600 part or Alloy 82/182 weld relative to the time of crack initiation for the reference part. The susceptibility study was completed as a BWOG project and is described in site documentation.

Three most susceptible location groupings include (1) pressurizer sample nozzles, level tap nozzles, and thermowell nozzles; (2) pressurizer vent nozzle; and (3) the 4-inch NPS Alloy-600 safe end that connects the stainless steel spray line to the stainless steel clad carbon steel spray nozzle. Groups 1 and 2 may be combined since all of these nozzles are fabricated from Alloy-600 and are attached to the pressurizer using partial penetration welds.

The augmented inspections on the three most susceptible groupings will be carried forward to the period of extended operation. Additional inspection locations may be added based on the results of visual examinations and may be based on a qualitative assessment of risk.

The CRDM penetrations at ANO-1 are not among the top three Alloy-600 groupings with respect to susceptibility to PWSCC. For a description of the program to manage PWSCC of CRDM penetrations, see the Control Rod Drive Mechanism Nozzle and Other Vessel Closure Penetration Inspection Program, which is a separate program to address ANO-1 commitments to NRC Generic Letter 97-01, “Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations.”

Validation of Sample

At present, the B&W operating plants have experienced only two instances (other than OTSG tubes) of cracking by PWSCC: the ANO-1 pressurizer level nozzle and the CRDM nozzle at Oconee Unit 2. The model described above was used to benchmark the events at the two B&W operating plants and the predicted time to crack initiation was within a factor of 2 of the actual time to crack discovery. While the uncertainty may be high, ranking of items using the relative susceptibility factors described above provides a quantitative means of selecting candidate items for inspection.

Industry Code or Standards: ASME Section XI, 1992 Edition, including mandatory Appendices VII and VIII (Appendix VIII in accordance with 1989 Addenda).

Frequency: Visual inspection (VT-2) of each Alloy-600 penetration (i.e., all Group 1 and 2 locations) attached to the pressurizer will be conducted each refueling outage. Volumetric inspection of the welded joints that connect the pressurizer spray nozzle safe end to the spray nozzle and stainless steel spray line will be conducted each inspection interval in accordance with ASME Section XI. In addition, on the repaired Alloy-600 level sensing nozzle, the ferritic steel in the nozzle-bore will be examined every other refueling outage using ultrasonic testing. The purpose of the inspection is to ensure that erosion/corrosion of the exposed ferritic steel in the nozzle-bore annulus is not an active damage mechanism. ANO-1 may eliminate or change the frequency of the ultrasonic testing examination should future examinations indicate that erosion/corrosion is inactive.

Acceptance Criteria or Standard: Acceptance criteria for identified flaws are in accordance with ASME Section XI.

Regulatory Basis: BAW-2243A [Reference B-5] and BAW-2244A [Reference B-3], Action Item 5.

Operating Experience and Demonstration: In 1990, a pressurizer Alloy 600 level sensing nozzle at ANO-1 developed a through-wall axial crack in the nozzle near the heat-affected zone of the J weld on the inside diameter of the vessel. The failure was due to primary water stress corrosion cracking. The crack was identified by audible indications and a dye penetrant inspection. Evaluations concluded that the crack did not compromise the structural integrity of the pressurizer. Several through-wall axial cracks have occurred in Alloy 600 nozzles throughout the industry. These cracks are small and axial due to the self-limiting high hoop stress caused by weld shrinkage during the nozzle installation process. Nozzle ejection will not occur on these type nozzle failures since the cracks are not circumferential. ANO and industry experience has proven that the visual inspection method used in this program is able to identify small leaks resulting from aging effects. In almost all leaks discovered to date, no corrosion or erosion was observed on the vessels or piping base metal. Only limited corrosion/erosion has ever been observed on the base material of a few penetrations. Conservative corrosion testing and analysis have further shown that only minimal corrosion/erosion could occur on the carbon steel base metal of the vessel or piping during one fuel cycle of operation in the event of a through wall crack. Such leaks from Alloy 600 items or Alloy 82/182 weld material will not cause sufficient loss of material to threaten the structural integrity of the vessel or piping. Based on operating experience in conjunction with analyses and testing, the Alloy 600 Aging Management Program provides reasonable assurance that the aging effects associated with Alloy 600 items and Alloy 82/182 weld material will be managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.2 ALTERNATE AC DIESEL GENERATOR TESTING AND INSPECTIONS

Purpose: The purpose of the Alternate AC Diesel Generator Testing and Inspections is to ensure that the effects of aging are managed before the loss of the intended functions of the system.

Scope: The Alternate AC Diesel Generator Testing and Inspections applies to the Alternate AC diesel generator and its components, as well as the fuel oil heat exchanger.

Aging Effects: The aging effects addressed by the Alternate AC Diesel Generator Testing and Inspections include the following.

- Loss of material or loss of mechanical closure integrity for the starting air subsystem components
- Loss of material or loss of mechanical closure integrity for the intake combustion air subsystem components
- Loss of material, fouling, and loss of mechanical closure integrity for the intake air aftercooler
- Loss of material for carbon steel components, cracking of the stainless steel components or loss of mechanical closure integrity for the exhaust subsystem components
- Loss of mechanical closure integrity for the lube oil subsystem components; fouling, loss of material from wear, and loss of mechanical closure integrity for the lube oil cooler
- Loss of material and loss of mechanical closure integrity for the cooling water subsystem components; fouling and a loss of material for the AAC radiator
- Loss of material from wetted portions of the exhaust fan housings
- Fouling of the fuel oil heat exchanger

Method: The Alternate AC Diesel Generator Testing and Inspections are a series of proceduralized surveillance activities. The AAC generator is started for quarterly operability testing and once every 18 months to verify the ability to satisfy station blackout requirements. The quarterly testing verifies operability by starting and slowly increasing the load on the AAC diesel.

The entire AAC diesel generator assembly is verified by this testing to be able to perform its intended function of starting and supplying power. This testing is performed in accordance with ANO commitments to the station blackout rule. This testing provides one method for managing the aging effects listed above since both pressure boundary integrity and heat transfer functions are verified by the operation of the diesel.

Inspections are periodically performed in accordance with the manufacturer's recommendations for this class of standby service of the engine and its supporting subsystems. This provides a second method for managing the aging effects listed above since the inspections and preventive maintenance could detect loss of material, leakage and fouling.

Industry Codes or Standards: Not applicable

Frequency: The AAC diesel generator is tested quarterly and once every 18 months in accordance with ANO commitments to station blackout requirements. Inspections are periodically performed in accordance with the manufacturer's recommendations for this class of standby service.

Acceptance Criteria or Standard: Acceptance criteria are based on guidance provided in the site procedures and vendor manuals.

Regulatory Basis: Responses [Reference B-10 and B-11] to Regulatory Guide 1.155, *Station Blackout*, provide a regulatory basis for these activities.

Operating Experience and Demonstration: A review of ANO-1 condition report summaries identified a loss of mechanical closure integrity on the lube oil supply header, an air leak from the starting air system, and a leak from the fuel oil day tank. These conditions identified through system testing and inspections are typical of those expected to result from the effects of aging. The continued implementation of the Alternate AC Diesel Generator Testing and Inspections provides reasonable assurance that the aging effects will be managed on these various system components so that they will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.3 ASME SECTION XI INSERVICE INSPECTION

4.3.1 IWB Inspections

Section 2.3.1 of the ANO-1 LRA identifies the Reactor Coolant System components subject to aging management review. Section 3.2 identifies cracking, loss of material and loss of closure integrity as aging effects requiring management for these components. ASME Section XI, 1992 Edition with portions of the 1993 Addenda for pressure testing, Subsection IWB Inspections, under the ANO-1 Inservice Inspection Plan, will manage these aging effects for the period of extended operation. The specific reactor coolant system component or component feature, aging effect requiring management, and credited ASME Section XI examination category are identified in Table 3.2-1. The ASME Section XI Inservice Inspection Program, IWB Inspections, has the following attributes.

Purpose: The purpose of the ASME Section XI, Subsection IWB Inspections under the scope of the ANO-1 Inservice Inspection Plan is to identify and correct degradation of ASME Class 1 pressure retaining components and their integral attachments in accordance with 10CFR50.55a and ANO-1 Technical Specification 4.0.5.

Scope: The scope of the ASME Section XI, Subsection IWB Inspections, credited for license renewal, is identified specifically for each component and for applicable component features in Table 3.2-1. Items listed in Table 3.2-1 selected for inservice inspection at ANO-1 are consistent with the items contained in ASME Section XI, Table IWB-2500-1, with the exception of RCS piping and approved alternatives.

ANO-1 has adopted Code Case N-481 from Regulatory Guide 1.147. In addition to N-481, which requires a VT-3 visual examination of the internal surfaces whenever a pump is disassembled for maintenance, if an RCP has not been disassembled for maintenance, ANO-1 will disassemble an RCP and do a VT-3 visual inspection of the internal surface of one pump casing before entering the period of extended operations.

ANO-1 has implemented a risk-informed methodology to select RCS piping welds for inspection in lieu of the requirements specified in the 1992 Edition of ASME Section XI, Table IWB-2500-1, Examination Category B-J. The risk-informed approach is based on Code Case N-560 and consists of the following two essential elements: (1) a degradation mechanism evaluation is performed to assess the failure potential of the piping system under consideration, and (2) a consequence evaluation is performed to assess the impact on plant safety in the event of a piping failure.

The results from these two independent evaluations are coupled to determine the risk-significance of piping segments within the system and are used to prioritize the selection of welds for inspection. The results of the ANO-1 implementation of risk-informed inspection based on Code Case N-560 are reported in Reference B-7 [Correspondence 1CAN069804, June 3, 1998, Arkansas Nuclear One-Unit 1, Docket No. 50-313, License No. DPR-51, Risk Informed Inservice Inspection Pilot Plant Submittal]. In the NRC SER of the ANO-1 submittal [Letter from the NRC to Mr. C. Randy

Hutchinson entitled “Risk-Informed Alternative to Certain Requirements of ASME Code Section XI, Table IWB-2500-1 at Arkansas Nuclear One, Unit 1 (TACMA2023),” August 25, 1999, Docket No. 50-313], the staff states that the alternative method described by ANO-1 provides equivalent, or better, examination criteria for Class 1 Category B-J welds than that provided by the current Section XI requirements (i.e., 1992 Edition which is equivalent to the 1989 Edition for B-J welds).

The risk-informed process used to select piping elements for inspection is consistent with the method used to identify applicable aging effects in BAW-2243A for stainless steel piping, Alloy-600 branch connections and piping, and clad carbon steel piping. Therefore, the risk-informed method that supercedes the current Examination Category B-J requirements is appropriate for the period of extended operation and adequately addresses the applicable aging effect of cracking at welded joints for clad carbon steel and stainless steel piping. See the ANO-1 program entitled "Small Bore Piping and Small Bore Nozzles Inspections," for a discussion of the application of risk-based inspection to small bore piping (i.e., NPS less than 4 inches) and small bore nozzles. Aging management for Alloy-600 items is addressed by the ANO-1 program entitled "Alloy-600 Aging Management Program."

Aging Effects: The aging effects managed as part of the ASME Section XI, Subsection IWB Inspections include cracking, loss of mechanical closure integrity at bolted connections, and loss of material.

Method: Detection of flaws is performed using nondestructive examination techniques. Three different types of examinations are performed: volumetric, surface, and visual examinations. Volumetric examinations are the most extensive, using methods such as radiographic, ultrasonic or eddy current examinations to locate surface and subsurface flaws. Surface examinations use methodologies such as magnetic particle or dye penetrant testing to locate surface flaws.

Three levels of visual examinations are specified. The VT-1 visual examination is conducted to assess the condition of the surface of the part being examined, looking for cracks, symptoms of wear, corrosion, erosion or physical damage. It can be done with either direct visual observation or with remote examination using various optical/video devices. The VT-2 examination is conducted specifically to locate evidence of leakage from pressure retaining components (period pressure tests). While the system is under pressure for a leakage test, visual examinations are conducted to detect direct or indirect indication of leakage. The VT-3 examination is conducted to determine the general mechanical and structural condition of components and supports and to detect discontinuities and imperfections such as loss of integrity at bolted connections.

Industry Code or Standards: The ASME B&PV Code Section XI, 1992 Edition, 1993 Addenda for Pressure Testing was used to develop this program. For examination category B-J, the risk-informed approach is based on code case N-560.

Frequency: The frequency of inspections is specified in ASME Section XI Tables IWB-2500-1 for applicable examination categories identified in Table 3.2-1. The inspection intervals are not restricted by the Code to the current term of operation and are valid for any period of extended operation.

Acceptance Criteria or Standard: Flaws detected during examination are evaluated by comparing the examination results to the acceptance standards established in ASME Section XI, Tables IWB-2500-1 for all applicable examination categories identified in Table 3.2-1.

Timing of New Program or Activity: An RCP will be disassembled and inspected prior to the end of the initial 40-year license term for ANO-1.

Regulatory Basis: The regulatory basis for the inservice inspection program is 10CFR50.55a(g), which specifically requires ISI be performed per ASME Section XI Code. ANO-1 Technical Specification 4.0.5 specifically requires ISI per the ASME Section XI Code. NRC Reg. Guide 1.147, *Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1,* also provides a regulatory basis for this program.

Operating Experience and Demonstration: The IWB inspections are implemented in accordance with NRC approved versions of ASME Section XI using proven techniques and methods to detect and evaluate flaws. Repairs or replacement are accomplished in accordance with ASME Section XI standards. The continued implementation of the IWB Inspections provides reasonable assurance that the aging effects will be managed so that the applicable structures and components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.3.2 IWC Inspections

Purpose: The purpose of the ASME Section XI, Subsection IWC inspections is to identify and correct degradation of ASME Class 2 pressure retaining components and their integral attachments in accordance with 10CFR50.55a and ANO-1 Technical Specification 4.0.5.

Scope: The scope of the ASME Section XI, Subsection IWC Inspections, credited for license renewal, includes inspection of selected components in the following systems.

- Core Flood
- Reactor Building Spray
- Main Feedwater
- Spent Fuel
- Service Water
- High Pressure Injection/Makeup and Purification
- Low Pressure Injection/Decay Heat
- Emergency Feedwater
- Main Steam
- Reactor Building Isolation
- Chilled Water System

Aging Effects: The aging effects that are managed as part of the ASME Section XI, Subsection IWC Inspections include cracking, loss of mechanical closure integrity, and loss of material.

Method: Detection of flaws is performed using nondestructive examination techniques. Three different types of examinations performed are volumetric, surface, and visual examinations. Volumetric examinations consist of radiographic, ultrasonic or eddy current examinations performed to locate surface and subsurface flaws. Surface examinations use methodologies such as magnetic particle or dye penetrant testing to locate surface flaws.

Three levels of visual examinations are specified. The VT-1 visual examination is conducted to assess the condition of the surface of the part being examined, looking for cracks, symptoms of wear, corrosion, erosion or physical damage. It can be done with either direct visual observation or with remote examination using various optical/video devices. The VT-2 examination is conducted specifically to locate evidence of leakage from pressure retaining components (period pressure tests). While the system is under pressure for a leakage test, visual examinations are conducted to detect direct or indirect indication of leakage. The VT-3 examination is conducted to determine the general mechanical and structural condition of components and supports and to detect discontinuities and imperfections.

Industry Code or Standards: The ASME Code Section XI, 1992 Edition, 1993 Addenda for Pressure Testing was used to develop this program.

Frequency: The frequency of inspections is specified in ASME Section XI Table IWC-2500-1. The inspection intervals are not restricted by the Code to the current term of operation and are valid for any period of extended operation.

Acceptance Criteria or Standard: Flaws detected during examination are evaluated by comparing the examination results to the acceptance standards established in ASME Section XI, Subsection IWC.

Regulatory Basis: The regulatory basis for the inservice inspection program is 10CFR50.55a(g), which specifically requires ISI be performed per ASME Code Section XI. ANO-1 Technical Specification 4.0.5 specifically requires ISI per the ASME B&PV Code, Section XI. NRC Reg. Guide 1.147, *Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1*, also provides a regulatory basis for this program.

Operating Experience and Demonstration: The IWC inspections are implemented in accordance with NRC approved versions of ASME Section XI using proven techniques and methods to detect and evaluate flaws. Repair and replacement are accomplished in accordance with ASME Section XI standards. The continued implementation of the IWC Inspections provides reasonable assurance that the aging effects will be managed so that the applicable structures and components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.3.3 IWD Inspections

Purpose: The purpose of the ASME Section XI, Subsection IWD Inspections is to identify and correct degradation of ASME Class 3 pressure-retaining components and their integral attachments in accordance with 10CFR50.55a and ANO-1 Technical Specification 4.0.5.

Scope: The scope of the ASME Section XI, Subsection IWD Inspections, credited for license renewal includes inspections of selected components in the following systems.

- Service Water
- Spent Fuel
- Main Steam
- Emergency Feedwater
- Sodium Hydroxide
- Condensate Storage

Aging Effects: The aging effects that are managed as part of the ASME Section XI, Subsection IWD Inspections include cracking, loss of mechanical closure integrity, and loss of material.

Method: Detection of flaws is performed using visual inspection techniques. Two levels of visual examinations are specified. The VT-2 examination is conducted specifically to locate evidence of leakage from pressure retaining components (period pressure tests). While the system is under pressure for a leakage test, visual examinations are conducted to detect direct or indirect indication of leakage. The VT-3 examination is conducted to determine the general mechanical and structural condition of components and supports and to detect discontinuities and imperfections.

Industry Code or Standards: The ASME Code Section XI, 1992 Edition, 1993 Addenda for Pressure Testing was used to develop this program.

Frequency: The frequency of inspections is specified in ASME Section XI Table IWD-2500-1. The inspection intervals are not restricted by the Code to the current term of operation and are valid for any period of extended operation.

Acceptance Criteria or Standard: Flaws detected during examination are evaluated by comparing the examination results to the acceptance standards established in ASME Section XI, Subsection IWD.

Regulatory Basis: The regulatory basis for the inservice inspection program is 10CFR50.55a(g), which specifically requires ISI be performed per ASME B&PV Code Section XI. ANO-1 Technical Specification 4.0.5 specifically requires ISI per the ASME

B&PV Code, Section XI. NRC Reg. Guide 1.147, *Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1*, also provides a regulatory basis for this program.

Operating Experience and Demonstration: The IWD inspections are implemented in accordance with NRC approved versions of ASME Section XI using proven techniques and methods to detect and evaluate flaws. Repair and replacement are accomplished in accordance with ASME Section XI standards. The continued implementation of the IWD Inspections provides reasonable assurance that the aging effects will be managed so that the applicable structures and components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation

4.3.4 IWE Inspections

Purpose: The purpose of the ASME Section XI, Subsection IWE Inspections is to identify and correct degradation of Class MC pressure retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure retaining components and their integral attachments in accordance with 10CFR50.55a.

Scope: The scope of the ASME Section XI, Subsection IWE Inspections, credited for license renewal includes inspections of the reactor building liner plate.

Aging Effects: The aging effect managed as part of the ASME Section XI, Subsection IWE Inspections is a loss of material of the steel surfaces.

Method: Detection of flaws is performed using nondestructive examination techniques. Three different types of examinations performed are volumetric, surface, and visual examinations. Volumetric examinations consist of radiographic and ultrasonic examinations. Surface examinations use methodologies such as magnetic particle or dye penetrant testing to locate surface flaws.

Two levels of visual examinations are specified. The VT-1 visual examination is conducted to assess the condition of the surface of the part being examined, looking for cracks, symptoms of wear, corrosion, erosion or physical damage. It can be done with either direct visual observation or with remote examination using various optical/video devices. The VT-3 examination is conducted to determine the general mechanical and structural condition of components and supports and to detect discontinuities and imperfections.

Industry Code or Standards: The ASME B&PV Code Section XI, 1992 Edition, 1993 Addenda for Pressure Testing was used to develop this program.

Frequency: The frequency of inspections is specified in ASME Section XI Table IWE-2500-1. The inspection intervals are not restricted by the Code to the current term of operation and are valid for any period of extended operation.

Acceptance Criteria or Standard: Flaws detected during examination are evaluated by comparing the examination results to the acceptance standards established in ASME Section XI, Subsection IWE.

Regulatory Basis: The regulatory basis for the inservice inspection program is 10CFR50.55a(g), which specifically requires ISI be performed per ASME B&PV Code Section XI. NRC Reg. Guide 1.147, *Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1*, also provides a regulatory basis for this program.

Operating Experience and Demonstration: The IWE inspections are implemented in accordance with NRC approved versions of ASME Section XI using proven techniques and methods to detect and evaluate flaws. Repair and replacement are accomplished in accordance with ASME Section XI standards. The continued implementation of the IWE

Inspections provides reasonable assurance that the aging effects will be managed so that the applicable structures and components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.3.5 IWF Inspections

Purpose: The purpose of the ASME Section XI Inservice Inspection Program, IWF Inspections is to identify and correct degradation of ASME Class 1, 2, 3, or MC component supports in accordance with 10CFR50.55a and ANO-1 Technical Specification 4.0.5.

Scope: The scope of the ASME Section XI, Subsection IWF Inspections, credited for license renewal includes component supports for ASME Class 1, 2, 3, or MC components.

Aging Effects: The aging effects managed as part of ASME Section XI, Subsection IWF include cracking, loss of material, and change in material properties.

Method: Visual examinations (i.e., VT-3) are conducted to determine the general mechanical and structural condition of component supports within the scope as defined for the applicable component support type in ASME Section XI Table IWF-2500-1.

Industry Code or Standards: The ASME Section XI, 1992 Edition, 1993 Addenda for Pressure Testing was used to develop this program.

Frequency: The frequency of inspections is specified in ASME Section XI Tables IWF-2500-1. The inspection intervals are not restricted by the Code to the current term of operation and are valid for any period of extended operation.

Acceptance Criteria or Standard: Flaws detected during examination are evaluated by comparing the examination results to the acceptance standards established in ASME Section XI, Subsection IWF-3400.

Regulatory Basis: The regulatory basis for the inservice inspection program is 10CFR50.55a(g), which specifically requires ISI be performed per ASME B&PV Code Section XI. ANO-1 Technical Specification 4.0.5 specifically requires ISI per the ASME B&PV Code, Section XI. NRC Reg. Guide 1.147, *Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1*, also provides a regulatory basis for this program.

Operating Experience and Demonstration: The IWF inspections are implemented in accordance with NRC approved versions of ASME Section XI using proven techniques and methods to detect and evaluate flaws. Repair and replacement are accomplished in accordance with ASME Section XI standards. The continued implementation of the IWF Inspections provides reasonable assurance that the aging effects will be managed so that the applicable structures and components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.3.6 IWL Inspections

Purpose: To provide instructions and documentation requirements for assessing the quality and structural performance of the reactor building's post-tensioning systems and concrete surfaces.

Scope: IWL inspections are performed on the reactor building's post-tensioning systems and concrete components that are subject to an aging management review as identified in Sections 2.4 and 3.6 of the ANO-1 LRA. Items exempt from the examination requirements include inaccessible tendon end anchors and concrete surfaces.

Aging Effects: The aging effects requiring management are loss of material for tendon anchorage and cracking and change in material properties for concrete.

Method: ASME Code Section XI, Subsection IWL provides the rules and requirements for inservice examination, inservice inspection and repair of the reinforced concrete and the post-tensioning systems of Class CC components. Such inspections are performed since degradation could lead to a crack or break in tendon wires or anchorage, thereby rendering the tendon unable to maintain compressive force on the reactor building structure during an accident.

Industry Codes or Standards: ASME Code Section XI, Subsection IWL provides requirements for inservice inspection and repair or replacement activities of the post-tensioning systems of concrete reactor building.

Frequency: Tendon surveillance is currently performed at 5-year intervals. Concrete surface examinations are conducted within a year of tendon surveillance.

Acceptance Criteria of Standard: Acceptance standards are specified in IWL-3000.

Regulatory Basis: 10CFR50.55a and technical specifications.

Operating Experience and Demonstration: During the twentieth year in-service inspection performed in the latter part of 1993, signs of degradation included an observable quantity of water in one of the tendons, corrosion on a shim at the end of one tendon, and one tendon found to have slightly low ultimate strength. The corroded shim was replaced. Metallurgical analysis found that the slightly low tensile strength was an original condition from the wire mill. The surveillance findings indicated that the tendons are experiencing normal relaxation. This experience demonstrates that the IWL inspections are effective in identifying indications of potential aging effects. In addition, the tendon surveillance and concrete inspections are performed in accordance with Subsection IWL of the ASME Code. Continued implementation of this program provides reasonable assurance that aging effects will be managed so that the reactor building post-tensioning system will continue to perform its intended function in accordance with the current licensing basis during the period of extended operation.

4.3.7 Augmented Inspections

Purpose: The purpose of the ASME Section XI, Augmented Inspections is to identify and correct degradation of components outside of the jurisdiction of ASME Section XI.

Scope: As required by the ANO-1 Technical Specification 4.15, augmented periodic inspections are completed for several main feedwater and main steam system welds, not in the Class 2 piping, to support the high energy line break analysis. Augmented inspections are completed for the BWST header including the lines from the reactor building sump. Augmented inspections that will be added to the program because of ANO-1 license renewal include the following.

- A special augmented inspection on the welds of the piping wetted by the reactor building sump water

- Some supplemental inspections of the "Q" stainless piping of the main steam system

- At least a one-time inspection of the penetration 68 piping and components and the decay heat pump room drain valves to ensure the seismic qualification is maintained to manage the aging effects of loss of material and cracking

- Special inspections of penetrations 10, 47, 58 and 64 to verify that there is no cracking in these penetrations

Aging Effects: The aging effects managed by these inspections are cracking and a loss of material.

Method: The methods utilized for augmented inspections have been discussed in the previous ASME Section XI sections.

Industry Code or Standards: The ASME Code Section XI, 1992 Edition, 1993 Addenda for Pressure Testing was used to develop this program.

Frequency: The frequency of inspections is or will be specified in the Inservice Inspection Plan. The inspection intervals are not restricted by the Code to the current term of operation and are valid for any period of extended operation.

Acceptance Criteria or Standard: Flaws detected during examination are evaluated by comparing the examination results to the acceptance standards established in ASME Section XI.

Timing of New Program or Activity: The new inspections will be initiated prior to the end of the initial 40-year license term for ANO-1.

Regulatory Basis: The regulatory basis for the inservice inspection program is 10CFR50.55a(g), which specifically requires ISI be performed per ASME B&PV Code Section XI. ANO-1 Technical Specification 4.15 specifically requires ISI per the ASME

B&PV Code, Section XI. NRC Reg. Guide 1.147, *Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1*, also provides a regulatory basis for this program.

Operating Experience and Demonstration: Augmented Inspections use the same non-destructive examination methods that are used for Section XI inspections on Class 1, 2, and 3 structures and components. These methods have proven effective in the industry for identifying cracking and loss of material. The continued implementation of the Augmented Inspections provides reasonable assurance that the aging effects will be managed so that the applicable structures and components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.3.8 Small Bore Piping and Small Bore Nozzles Inspections

Purpose: The Small Bore Piping and Small Bore Nozzles Inspections identify aging effects on small bore piping and nozzles.

Scope: The small bore piping and small bore nozzles, within the scope of this program, are defined as reactor coolant system piping and nozzles less than 4-inch NPS that do not receive volumetric inspection in accordance with ASME Section XI. Alloy-600 small bore branch connections, small bore safe ends, and small bore nozzles are addressed by the Alloy-600 Aging Management Program.

Aging Effect: BAW-2243A and BAW-2244A identify cracking as an aging effect for small bore piping and small bore nozzles.

Method: Section 4.4.2 of BAW-2243A states that additional inspections of small bore piping may be appropriate to assure the management of potential weld cracking for the period of extended operation. Selection of additional inspection locations should be based on detailed evaluations of material susceptibility, operating environment, stress, and risk.

ANO-1 has implemented a risk-informed method to select RCS piping welds for inspection in lieu of the requirements specified in the 1992 Edition of ASME Section XI, Table IWB-2500-1, Examination Category B-J. The risk-informed approach is based on Code Case N-560 and consists of two essential elements: (1) a degradation mechanism evaluation to assess the failure potential of the piping system under consideration, and (2) a consequence evaluation to assess the impact on plant safety in the event of a piping failure.

The results from these two independent evaluations are coupled to determine the risk significance of piping segments within the system. Priority is then given to the most risk significant piping segments during the selection of RCS piping welds for inspection.

As part of the risk-informed ISI program, ANO-1 has selected for volumetric examinations, a sample population of welds in the following Class 1 small bore piping: 1½-inch pressurizer spray line, 2½-inch makeup and purification lines, 2½-inch letdown line, and 1½-inch cold leg suction drain line.

Industry Codes or Standards: ASME Code Case N-560 is the industry code used to develop this program.

Frequency: The inspection frequencies are defined in Table 1 of ASME Code Case N-560.

Acceptance Criteria or Standard: Acceptance criteria are provided in ASME Section XI IWB-3400 and IWB-3132 as provided in ASME Code Case N-560.

Regulatory Basis: The regulatory basis for the inservice inspection program is 10CFR50.55a(g), which specifically requires ISI be performed per ASME B&PV Code Section XI. ANO-1 Technical Specification 4.0.5 also provides a regulatory basis for this program.

Operating Experience and Demonstration: Following the discovery of a cracked weld in an RCS drain line in 1989, ANO-1 implemented a program to investigate the potential for cracking of other similar lines. The root cause of the cracking was determined to be a weld defect that propagated by vibrational fatigue. A document search for records of small bore pipe failures for the past 10 years at ANO-1 revealed no piping failures caused by thermal fatigue.

Vibration induced socket weld failures at ANO have occurred, almost exclusively, on small bore (2-inch NPS and under) vents and drains. Engineering personnel performed a comprehensive root cause analysis and developed a corrective action plan for the prevention of ANO-1 piping vibration failures. Several socket welds at locations of high vibration loads were reinforced. Plant changes that may introduce new vibration sources or new vents or drains are thoroughly evaluated before implementation. At ANO-1, failures of ASME Class 1 small bore pipe have been rare.

The ANO-1 risk-informed method for selecting welds for inspection incorporates the elements necessary to manage cracking of small bore piping and small bore nozzles during the period of extended operation. The inspections are implemented in accordance with NRC approved versions of ASME section XI using proven techniques and methods to detect and evaluate flaws. Repair or replacement is accomplished in accordance with ASME Section XI standards. The continued implementation of this program provides reasonable assurance that the aging effects will be managed so that the applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.4 BOLTING AND TORQUING ACTIVITIES

Purpose: Bolting and torquing activities performed at ANO-1 prevent degradation of bolting or identify and correct degradation of bolting.

Scope: The scope of bolting and torquing activities is pressure boundary bolting applications associated with components within the scope of license renewal and subject to aging management review. Applications include bolted flange connections for vessels (i.e., manways and inspection ports), flanged joints in piping, body-to-body joints in valves, and pressure-retaining bolting associated with pumps or valves and miscellaneous process components.

Aging Effects: The aging effects addressed by Bolting and Torquing Activities are cracking, loss of material, and loss of mechanical closure integrity.

Method: An ANO site procedure provides guidance regarding inspections and preparation of mating surfaces, threaded fasteners, and bolted joints. Instructions are provided for proper tightening of fasteners and use of wrenching devices.

Industry Code or Standards: Not applicable.

Frequency: Bolting and Torquing activities apply when performing maintenance activities that involve threaded fasteners.

Acceptance Criteria or Standard: Acceptance criteria are provided in the ANO site procedures. Typical criteria are that mating surfaces are smooth and free of major defects. Male and female threads are inspected for major defects (nicks, burrs, evidence of galling, etc.). Other criteria include proper and adequate thread engagement, no loose fasteners, and use of appropriate torque values.

Regulatory Basis: 0CAN088201, AP&L Response to IEB 82-02 – Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants [Reference B-12].

Operating Experience and Demonstration: Procedures for bolting and torquing activities at ANO-1 are based on generic industry guidance. This guidance was based on industry experience regarding bolted closures and has been proven effective in maintaining the integrity of bolted closures. Based on this information, the continued implementation of the Bolting and Torquing Activities provides reasonable assurance that the aging effects associated with bolted closures will be managed such that applicable structures and components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.5 BORIC ACID CORROSION PREVENTION

Purpose: The purpose of the Boric Acid Corrosion Prevention Program is to prevent corrosion damage due to leakage from the borated water systems at ANO.

Scope: The Boric Acid Corrosion Prevention Program is concerned with the RCS and other structures or components containing, or exposed to, borated water.

Aging Effects: This program is credited with monitoring the boric acid corrosion of carbon steel external surfaces exposed to leakage from borated water. Carbon steel is utilized for bolting on many of the systems that contain borated water. This program has been identified as managing the loss of material of bolting that could eventually result in a loss of mechanical closure integrity for bolted connections.

Method: In addition to RCS leakage monitoring during normal operation, ANO-1 completes visual inspections to identify pressure boundary leakage. A partial inspection of the RCS is performed during plant cooldowns (as long as the cooldown was not an emergency) to identify locations needing repair. Detailed post outage pressure testing and visual inspections of RCS components are completed to demonstrate the reactor coolant system integrity prior to the return to criticality. This inspection is utilized to identify leakage for evaluation in accordance with the Boric Acid Corrosion Prevention Program. Prior to plant startup, active leaks that contact carbon steel are repaired, redirected, or evaluated as acceptable for continued service. The implementing procedures provide guidance on the system walkdown requirements and leakage evaluation criteria.

Industry Code or Standards: Not applicable.

Frequency: Visual inspections are performed during plant cooldowns and heatups. Additional inspections may be performed based on the nature of RCS leakage detected during normal operations.

Acceptance Criteria or Standard: The acceptance criteria for the Boric Acid Corrosion Prevention Program are contained in site procedures. Evaluations are accomplished for each identified leak. Consideration is given to the possibility of flow paths from the leak to carbon steel components, or the accumulation of boric acid in insulation. Any staining or buildup of boric acid crystals is evaluated to ensure no components are damaged to the extent that they would be unable to fulfil their intended safety function.

Regulatory Basis: 0CAN058813, Entergy's response to Generic Letter 88-05, *Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants*. [Reference B-13]

Operating Experience and Demonstration: The Boric Acid Corrosion Prevention Program has been successful in ensuring the proper identification, evaluation, and repair of boric acid leakage. Leakage is being reported not only on the reactor coolant system components, but also on other systems that contain borated water. This program has helped in the reduction of unidentified reactor coolant system leakage. This experience demonstrates the success of the program in detecting and initiating corrective action for boric acid leakage. In conjunction with other programs, the continued implementation of the Boric Acid Corrosion Prevention Program provides reasonable assurance that the aging effects associated with boric acid corrosion will be managed such that the applicable structures and components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.6 CHEMISTRY CONTROL

Procedures for chemistry inspections of ANO-1 systems and heat exchangers outline inspections checking for corrosion, deposits, structural damage, general cleanliness, appearance and biological growth. Chemistry inspections are frequently performed on components and systems that are available due to routine or corrective maintenance. These inspections help to verify the adequacy of the existing chemistry controls and ensure unanalyzed degradation is not occurring.

The following subsections address the individual ANO-specific chemistry control programs in more detail**:**

- Primary Chemistry Monitoring Program
- Secondary Chemistry Monitoring Program
- Auxiliary Systems Chemistry Monitoring
- Diesel Fuel Monitoring Program
- Service Water Chemical Control Program

4.6.1 Primary Chemistry Monitoring

Purpose: The purpose of the Primary Chemistry Monitoring Program is to maximize long-term availability of primary systems by minimizing system corrosion, fuel corrosion, and radiation field build-up.

Scope: The scope of the Primary Chemistry Monitoring Program, with respect to license renewal, includes sampling activities and analysis on the following systems:

- Reactor Coolant System
- Borated Water Storage Tanks
- Spent Fuel Pool System
- Letdown Purification Demineralizers
- Reactor Makeup Water

Aging Effects: The Primary Chemistry Monitoring Program provides assurance that elevated levels of contaminates and oxygen do not exist in the systems covered by the program. This prevents or minimizes the occurrence of cracking and other aging effects.

Method: The ANO-1 Primary Chemistry Monitoring Program consists of sampling criteria, frequencies, locations, and allowable values with specific guidance for parameters exceeding allowable values.

- **Industry Code or Standards:** Not applicable.

Frequency: The frequency of sampling is daily, weekly, monthly, quarterly or as required, based on plant operating conditions. This frequency has been established based on technical specification requirements, EPRI guidelines, and ANO-specific experience.

Acceptance Criteria or Standard: The acceptance criteria for the Primary Chemistry Monitoring Program are contained in site procedures and are based on the sampling parameter, the sampling location, and plant operating conditions. These criteria have been established based on technical specification requirements, EPRI guidelines, and ANO-specific experience.

Regulatory Basis: ANO-1 Technical Specification 3.1.5, Chemistry.

Operating Experience and Demonstration: Operating experience on the primary systems demonstrates the effectiveness of the Primary Water Chemistry Monitoring Program. No significant chemistry related degradation of primary components has been experienced. Experience has shown that implementation of a primary chemistry program in accordance with accepted industry standards is effective in managing the effects of aging. Based on this experience, the continued implementation of the Primary Chemistry Monitoring Program provides reasonable assurance that aging effects will be managed so that primary system components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.6.2 Secondary Chemistry Monitoring

Purpose: The purpose of the Secondary Chemistry Monitoring Program is to maximize the availability and operating life of major components at ANO-1.

Scope: The scope of the Secondary Chemistry Monitoring Program includes sampling activities and analysis on the main feedwater system, condensate storage system, and steam generators. The aging reviews for many of the safety-related, non-Class 1 systems also indirectly credit the Secondary Chemistry Monitoring Program since the condensate storage tanks are used as a source of makeup water to these systems.

Aging Effects: Since the Secondary Water Chemistry Monitoring Program has adequate processes to ensure the levels of contaminants and oxygen are well below the assumptions of the aging management reviews, this prevents or minimizes the occurrence of loss of material and other aging effects.

Method: The ANO-1 Secondary Chemistry Monitoring Program consists of sampling parameters, frequencies, locations, and allowable values with specific guidance given when parameters exceed specified allowable ranges.

Industry Code or Standards: Not applicable.

Frequency: The frequency of sampling is daily, weekly, monthly, quarterly or as required, based on plant operating conditions. This frequency has been established based on technical specification requirements, EPRI guidelines, or ANO-specific experience.

Acceptance Criteria or Standard: The acceptance criteria for the Secondary Chemistry Monitoring Program are located in site procedures and are based on the sampling parameter, the sampling location, and the plant operating conditions. These criteria have been established based on EPRI guidelines and ANO-specific experience.

Regulatory Basis: The regulatory basis for this program includes Facility Operating License DPR-51 Section 2.c.(7).

Operating Experience and Demonstration: Operating experience with the secondary systems demonstrates the effectiveness of the Secondary Chemistry Monitoring Program. Chemistry parameters are controlled well within EPRI guidelines and industry standards. These standards have been proven, through industry experience, to be appropriate for minimizing the effects of aging on secondary system components. Based on this experience, the continued implementation of the Secondary Chemistry Monitoring Program provides reasonable assurance that aging effects will be managed so that secondary components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.6.3 Auxiliary Systems Chemistry Monitoring

Purpose: The purpose of the Auxiliary Systems Chemistry Monitoring Program is to maximize the availability and operating life of the components used for the closed cooling loops at ANO-1.

Scope: The scope of the Auxiliary Systems Chemistry Monitoring Program, with respect to license renewal, is limited to sampling activities and analysis on the following systems.

- Intermediate Cooling Water System
- Chilled Water Systems
- Emergency Diesel Generators
- Alternate AC Diesel Generator

Aging Effects: The Auxiliary Systems Chemistry Monitoring Program is credited with minimizing the loss of material due to corrosion, cracking, fouling, and loss of mechanical closure integrity.

Method: The water in the applicable system is sampled. Control parameters are monitored and corrective actions are taken if the parameters are outside the acceptable range. Corrosion inhibitors may be utilized in these systems.

- **Industry Code or Standards:** Not applicable.

Frequency: The frequency of sampling is established based on EPRI guidelines and ANO-specific experience.

Acceptance Criteria or Standard: The acceptance criteria for the Auxiliary Systems Chemistry Monitoring Program are located in site procedures and are based on the sampling parameter, the sampling location, and the plant operating conditions. These criteria have been established based on equipment specification requirements, EPRI guidelines, or ANO-specific experience.

Regulatory Basis: None.

Operating Experience and Demonstration: As part of the Auxiliary Systems Chemistry Monitoring Program, corrosion inhibitor levels, biological activities and corrosion rates are monitored and trended. Based on these trends, there appear to be no major corrosion mechanisms that could prevent the closed loop systems of ANO-1 from performing their intended function for many years. Corrosion inhibitor concentrations are maintained well within their effective concentrations. Biocides are added as necessary to keep biological activity within industry recommended specifications. Corrosion rates in the intermediate cooling water system are monitored with coupons and are well within EPRI guidelines for closed loop systems. During system visual inspections conducted in conjunction with maintenance activities, no deposits or visible pitting or corrosion have been observed on the ANO-1 closed-loop cooling systems. Based on this experience, continuation of the Auxiliary Systems Chemistry Monitoring Program will provide reasonable assurance that the aging effects will be managed such that the applicable structures and components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.6.4 Diesel Fuel Monitoring

Purpose: The purpose of the Diesel Fuel Monitoring Program is to ensure that adequate diesel fuel quality is maintained to prevent plugging of filters, fouling of injectors, and corrosion of the fuel systems.

Scope: The scope of the Diesel Fuel Monitoring Program is limited to sampling activities and analysis on the following tanks.

- Bulk fuel oil storage tank
- Emergency diesel fuel tanks
- Emergency diesel day tanks
- Fire pump diesel day tank
- AAC diesel generator day tank

Aging Effects: The aging management reviews credit the sampling and monitoring as providing an adequate control of the fuel oil to ensure water and contamination (including microbiological) are not present in the system.

Method: The ANO-1 Diesel Fuel Monitoring Program consists of sampling parameters, frequencies, locations, and allowable values with specific guidance given when parameters exceed specified allowable ranges.

Industry Code or Standards: References used to develop this program include:

- ASTM D975-1981, Standard Specification for Diesel Fuel Oils
- VV-F-800D, Military Specifications
- Other ASTM Standards

Frequency: Monthly and quarterly samples are taken from the tanks within the scope of the program. In addition, each new shipment of diesel fuel is sampled prior to unloading into the bulk fuel oil storage tank.

Acceptance Criteria or Standard: The acceptance criteria for the Diesel Fuel Monitoring Program are contained in site procedures. These criteria have been established based on technical specification requirements or industry codes or standards.

Regulatory Basis: ANO-1 Technical Specification 4.6.1, Diesel Generators.

Operating Experience and Demonstration: Past operating experience involving diesel fuel at ANO has included problems with water in the fuel, particulate contamination, and biological fouling. Based on these experiences, a comprehensive diesel fuel monitoring program was developed at ANO. The ANO Diesel Fuel Monitoring Program was implemented to prevent the problems experienced in the past. The program provides for addition of biocides and stabilizers, as necessary, to prevent biological breakdown of diesel fuel. Procedures provide instructions for routine tank bottom draining to remove any water accumulation. Diesel tanks at ANO are cleaned at least once every 10 years to remove sediment. Diesel tank filtration provides for removal of particulate buildup. Tanks that are drained for cleaning are inspected for corrosion and other physical degradation. Tank inspections over the past years have not shown tank degradation or corrosion. The continuation of the Diesel Fuel Monitoring Program will provide reasonable assurance that the aging effects will be managed such that the diesel fuel oil system structures and components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.6.5 Service Water Chemical Control

Purpose: The purpose of the Service Water Chemical Control Program is to maximize the availability and operating life of the components in the ANO-1 service water system.

Scope: The scope of the Service Water Chemical Control Program includes sampling activities and analysis on the service water system. The scope also includes chemical injection into the service water bays. The fire protection system also takes suction from the service water bays.

Aging Effects: The Service Water Chemical Control Program has been credited in the aging management reviews for the service water system, the auxiliary cooling water system, and the fire protection system since these systems draw suction from the intake structure. The chemical additions are only credited with reducing corrosion, and are not credited with elimination of this mechanism.

Method: The Service Water Chemical Control Program consists of sampling parameters, frequencies, locations, and allowable values with specific guidance given for parameters exceeding allowable values.

Industry Code or Standards: Not applicable.

Frequency: The frequency of sampling is daily, twice per week, weekly, or as required, based on plant conditions. This frequency has been established based on ANO-specific experience.

Acceptance Criteria or Standard: The acceptance criteria for the Service Water Chemical Control Program are contained in site procedures and are based on the sampling parameter, the sampling location, and the plant operating conditions. These criteria have been established based on EPRI guidelines and ANO- specific experience.

Regulatory Basis: ANO response to NRC Generic Letter 89-13 [Reference B-31, B-32, B-33, B-34, and B-35].

Operating Experience and Demonstration: The Service Water Chemical Control Program supplements the Service Water Integrity Program in managing the effects of aging on the service water and auxiliary cooling water systems. Corrosion inhibitor concentrations are monitored and maintained well within their effective concentrations. Biocide is continuously added to control biological activity. Corrosion rates of the service water system are monitored with coupons. System visual inspections are conducted when maintenance activities allow. Based on the results of these inspections, in addition to testing and inspections performed as part of the Service Water Integrity Program, system components or piping may be scheduled for replacement. Based on this experience, the continued implementation of the Service Water Chemical Control Program, in conjunction with the Service Water Integrity Program, provides reasonable assurance that the effects of aging will be adequately managed so that the service water and auxiliary cooling water components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.7 CONTROL ROD DRIVE MECHANISM NOZZLE AND OTHER VESSEL CLOSURE PENETRATION INSPECTION PROGRAM

Section 2.3.1.5 of the ANO-1 LRA and BAW-2251A identify the control rod drive mechanism nozzles and other vessel closure penetrations as subject to aging management review. Section 3.2.4 of the ANO-1 LRA and BAW-2251A [Reference B-16] identify primary water stress corrosion cracking as an aging effect of concern that must be managed for the period of extended operation. The CRDM Nozzle and Other Vessel Closure Penetrations Inspection Program in conjunction with the Chemistry Control Program, Inservice Inspection Program, Leakage Detection in Reactor Building, and Boric Acid Corrosion Prevention Program will manage PWSCC for the period of extended operation.

Purpose: The purpose of the CRDM Nozzle and Other Vessel Closure Penetrations Inspection Program is to verify the assumptions made in BAW-2301 [Reference B-9], BWOG Integrated Response to Generic Letter 97-01, of the susceptibility and consequence of PWSCC in B&W-designed CRDM nozzles.

Scope: The scope of the program includes the B&W-designed reactor vessel closure head CRDM nozzles and other closure head penetrations.

Aging Effects: The aging effect is PWSCC of Alloy-600 nozzles with partial penetration welds that cause high circumferential residual stresses on the inner diameter of the nozzles opposite the welds.

Method: The current BWOG program requires the re-inspection of from two to twelve Oconee Unit 2 CRDM nozzles from the top of the head and an inspection of all CRDM penetrations at Crystal River Unit-3. The Oconee Unit 2 re-inspection was completed in 1999 and no change was reported relative to previous inspections. At present, the Crystal River Unit-3 inspection is scheduled for 2001. The method used to select the most susceptible B&W-designed nozzles for inspection is described in BAW-2301, Section 2.3. The ANO-1 CRDM nozzles are among the lowest in susceptibility of the B&W operating plants with a predicted relative time to failure well in excess of 48 EFPY. In addition, the ANO-1 CRDM nozzles are ranked in the "beyond 15 year" category in the NEI integrated "Industry Histogram for Reactor Vessel Head Penetrations." ANO-1 will continue to monitor the inspection results from the B&W operating plants and other plants during the current term of operation and during the period of extended operation. The need for inspections during the period of extended operation at ANO-1 will be determined based on the results of inspections at the other B&W operating plants.

At Oconee and Crystal River Unit-3, eddy current inspection will be utilized for detection and eddy current, ultrasonic, and liquid penetrant will be used for sizing.

Industry Code or Standard: Not applicable.

Frequency: The inspection frequency is dependent on plant-specific, BWOG, and industry-wide inspection results. The inspections at Crystal River Unit-3 are planned for 2001. Plans for future inspections at specific B&W operating plants will be adjusted based upon review of ongoing inspections.

Acceptance Criteria or Standard: Should ANO-1 inspect, axial flaws detected during inspection will be analyzed and evaluated using the NUMARC acceptance criteria, which were approved by the NRC in their Safety Evaluation, dated November 19, 1993. Circumferential flaws will be analyzed and addressed with the NRC on a case-by-case basis.

Regulatory Basis: 0CAN079703 and 0CAN029908, ANO-1 responses to NRC Generic Letter 97-01. [References B-17 and B-18].

Operating Experience and Demonstration: A full inspection of Oconee Unit 2 from beneath the reactor vessel head was performed in 1994. In addition, a re-inspection was completed on two Oconee Unit 2 CRDM nozzles in 1996 from above the reactor vessel head. The results of these inspections were submitted to the NRC. A subsequent re-inspection at Oconee Unit 2 was completed in 1999.

The Oconee Unit 2 inspections in 1994 identified a small number of nozzles with crack-like indications that were insignificant in depth. Re-inspection showed no growth after one cycle of operation. Re-inspection in 1999 showed no change relative to the 1996 and 1994 inspections. Future inspections will be performed in a manner consistent with these previous inspections. Based on the above review, the continued implementation of the CRDM Nozzle and Other Vessel Closure Penetration Inspection Program provides reasonable assurance that the aging effects will be managed such that the CRDM nozzle penetrations will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.8 FIRE PROTECTION

The primary objectives of ANO-1's Fire Protection Program are:

- The prevention of fire
- The prompt detection and suppression of fires , and
- The protection of systems, structures, and components essential to plant safety so that they are able to withstand a fire without a loss function.

The activities performed to achieve the Fire Protection Program objectives that are credited for the management of aging effects are:

- Fire Barrier Inspections,
- Fire Hose Station Inspections,
- Fire Suppression Water Supply System Surveillance,
- Fire Suppression Sprinkler System Surveillance,
- Fire Water Piping Thickness Evaluation,
- Control Room Halon Fire System Inspection,
- NFPA 25 Testing of Sprinkler Head Components that are 50 years old, and
- Reactor Coolant Pump Oil Collection System Visual Inspection.

4.8.1 Fire Barrier Inspections

Fire barriers subject to aging management review include fire walls/floors, fire doors/hatches, fire damper mountings, fire wraps, and fire stops. The aging effects for fire barriers are discussed in Section 3.6 of the ANO-1 LRA.

Purpose: Fire Barrier Inspections provide for periodic surveillance of fire barriers separating redundant safe shutdown systems to assure that they perform their separation functions.

Scope: The scope includes 10CFR50.48-required fire walls and fire floors as indicated on the fire protection drawings. Fire doors/hatches, fire damper mountings, fire wraps, and penetration fire stops associated with 10CFR50.48-required fire walls and/or fire floors are within the scope.

Aging Effects: The aging effects requiring management for fire barriers are as follows.

Fire walls (masonry blockwalls)	Cracking
Fire doors/hatches (including threaded fasteners)	Loss of material
Fire wraps (and associated banding)	Loss of material, cracking/ delamination, and/or change in material properties
Penetration fire stops	Loss of material, cracking/ delamination/separation, and change in material properties

There are no aging effects that require management associated with fire walls comprised of concrete (including fire floors) or with fire dampers.

Method: Fire barriers are visually inspected.

Industry Code or Standard: Not applicable.

Frequency: Fire barriers are periodically inspected as follows:

Each fire barrier: At least once per 18 months (excluding penetration seals).

Fire doors and hardware: At least once per 18 months.

Sealed penetrations: At least 10 percent of each type are inspected at least once per 18 months. If penetration is found inoperable, an additional 10 percent of the degraded type are inspected until a 10 percent sample is found with no visual degradation. Samples are selected so that each penetration seal is inspected at least once every 15 years.

Repaired fire barriers: Inspected prior to returning fire barrier to operable status.

Acceptance Criteria or Standard: Fire barriers are considered operable when the visually observed condition is the same as the as-designed condition. Acceptance criteria are indicated in the Fire Barrier Inspection procedures.

Regulatory Basis: The regulatory basis for this program is 10CFR50.48, 10CFR Part 50 Appendix R, and Operating License DPR-51 Paragraph 2.c.(8).

Operating Experience and Demonstration: The condition reporting system was utilized to review the ANO-1 operating experience pertaining to fire barriers. The review indicated that voids and gaps have been discovered at fire barrier penetration seals. Other damage to fire barriers has included torn boot seals, loose damming material from building vibration or contraction from temperature changes, and holes associated with blockwall cracks in the vicinity of penetrations. Damaged penetration seals and fire barriers have been repaired. This experience demonstrates that Fire Barrier Inspections are effective in being able to identify and initiate corrective action for fire barrier deficiencies. Based on this experience, continuation of Fire Barrier Inspections will provide reasonable assurance that the aging effects will be managed such that the fire barriers will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.8.2 Fire Hose Station Inspections

Purpose: The purpose of Fire Hose Station Inspections is to assure that manual fire suppression is available to safety-related equipment.

Scope: Fire hose reels associated with 10CFR50.48-required fire hose stations are within the scope of license renewal.

Aging Effects: The aging effect for fire hose reels (including threaded fasteners) is a loss of material.

Method: Fire hose reels are visually inspected.

Industry Codes or Standards: Fire hose station inspections were developed using guidance of the National Fire Protection Association.

Frequency: Fire hose reel inspections are performed once every 31 days.

Acceptance Criteria or Standard: Fire hose stations, protecting areas containing safety related equipment, are to be operable whenever the safety related equipment is required to be operable. Acceptance criteria are indicated in the fire hose station testing procedure.

Regulatory Basis: The regulatory basis for this program is 10CFR50.48, 10CFR Part 50 Appendix R, and Operating License DPR-51 Paragraph 2.c.(8).

Operating Experience and Demonstration: A review of ANO-1 condition report summaries did not identify indications of a loss of material associated with fire hose reels. Hose reels not operating properly were typically either misaligned or there were problems with their swivel connections. While these inspections have identified no applicable aging effect, visual inspections have proven effective in identifying indications typical of the applicable aging effect for fire hose reels. Continuation of Fire Hose Station Inspections will provide reasonable assurance that the aging effect will be managed such that fire hose reels will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.8.3 Fire Suppression Water Supply System Surveillance

Purpose: The purpose of this surveillance is to verify operability of fire suppression water supply system components.

Scope: The Fire Suppression Water Supply System Surveillance applies to ANO-1 fire water system supply piping and valves. The surveillance applies to several diesel fire pump subsystems including the intake air, exhaust, lube oil, and cooling water. Fire protection system heat exchangers are also within the scope of this surveillance.

Aging Effects: This activity verifies that loss of material due to internal surface corrosion and fouling of carbon steel, stainless steel, brass or bronze components is managed. This activity also manages cracking of stainless steel, brass, or bronze components.

Method: The following surveillance activities are performed on the fire suppression water supply system

- At least once per 31 days, on a staggered test basis, each fire water pump is started by automatic actuation and operated for 15 minutes with flow through a relief line.
- A flush of the system main is performed at least once every six months.
- A system functional test is performed at least once per 18 months, which includes simulated automatic actuation of the system throughout its operating sequence, and includes verification of pump flow, discharge pressure, and fire suppression water system pressure requirements.
- At least once per three years a flow test of the system is completed in accordance with Chapter 5, Section 11, of the Fire Protection Handbook 14th edition.

These tests ensure the pumps are capable of starting and supplying the required flow rate and ensure the heat exchangers for the fire pumps operate as required. The flushing and flow testing helps to ensure flow blockage is not present from a buildup of corrosion products or fouling.

Industry Codes or Standards: Standards and recommended practices of the National Fire Protection Association were used as guidance in the development of fire suppression water system surveillances.

Frequency: The frequency of surveillance activities is listed under the Method discussion of this program.

Acceptance Criteria or Standard: Acceptability of surveillance results is determined in accordance with site procedures.

Regulatory Basis: The regulatory basis for this program is 10CFR50.48, 10CFR Part 50 Appendix R, and operating license DPR-51 paragraph 2.c.(8).

Operating Experience and Demonstration: Inspections of the underground cement lined cast iron piping have shown negligible corrosion degradation. The above ground carbon steel pipe is inspected during repair or replacement of fire water components. There have been replacements of components and piping in the small-bore carbon steel piping due to internal corrosion. Based on this experience, these surveillance activities will continue to provide assurance that aging effects will be adequately managed so that fire protection system components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.8.4 Fire Suppression Sprinkler System Surveillance

Purpose: The purpose of this surveillance is to provide a method for verifying operability of fire suppression sprinkler system components.

Scope: Within the scope of license renewal, the Fire Suppression Sprinkler System Surveillance applies to ANO-1 fire suppression sprinkler system piping, valves, and nozzles.

Aging Effects: This activity verifies that loss of material due to internal surface corrosion and fouling of carbon steel, stainless steel, brass or bronze components is managed. This activity also manages cracking of stainless steel, brass, or bronze components.

Method: The following surveillance activities are performed on the fire suppression sprinkler system.

- Each testable valve in the flow path is cycled at least once per 12 months. An exception for this surveillance applies to the valves in the reactor building, which shall be inspected when in cold shutdown.
- An inspection to verify the integrity of the spray nozzles and headers is performed at least once every 18 months.
- Deluge spray system flush is performed quarterly.

Industry Codes or Standards: The standards and recommended practices of the National Fire Protection Association were used as guidance in the development of the fire suppression sprinkler system surveillance.

Frequency: The frequency of surveillance activities is listed under the Method section of this program.

Acceptance Criteria or Standard: Acceptability of surveillance results is determined in accordance with site procedural instructions.

Regulatory Basis: The regulatory basis for this program is 10CFR50.48, 10CFR Part 50 Appendix R, and Operating License DPR-51 Paragraph 2.c.(8).

Operating Experience and Demonstration: Internal visual inspections of sprinkler system components are performed during maintenance activities that require the system to be opened. Inspection reports have noted general corrosion, but no severe cases. In addition to visual inspections, ultrasonic testing is performed on sprinkler piping on a regular basis. These examinations have not identified any piping that failed the initial screening process and required further evaluation. This method has proven effective on other systems in identifying wall thinning as a result of corrosion mechanisms. This experience indicates that continuation of the Fire Suppression Sprinkler System

Surveillance inspections will provide assurance that aging effects will be adequately managed so that fire protection sprinkler systems will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.8.5 Fire Water Piping Thickness Evaluation

Purpose: The purpose of Fire Water Piping Thickness Evaluation is to provide a method for the examination and evaluation of pipe wall thickness changes in the fire water system.

Scope: Within the scope of license renewal, the Fire Water Piping Thickness Evaluation applies to ANO-1 fire water system piping.

Aging Effects: A loss of material by internal surface corrosion of cast iron, carbon steel, or stainless steel fire water system components is the aging effect managed by the Fire Water Piping Thickness Evaluation.

Method: Minimum and nominal design wall thickness specifications are obtained prior to performing the examination. The wall thickness at the deepest pit and the average wall thickness are obtained, using non-destructive examination methods, as outlined in the fire water piping thickness evaluation procedure.

Industry Codes or Standards: Not applicable.

Frequency: Fire water piping is examined at a frequency determined by the system engineer. The frequency and locations for the inspections are based on results of previous inspections, the time since previous inspections, inspections of nearby or representative piping, and the need for additional inspection locations to characterize the condition of a pipe section. The consequences of failure of the subject piping are also considered when determining examination frequency and location.

Acceptance Criteria or Standard: Acceptability of examination results for each inspection location is determined in accordance with site procedural instructions.

Regulatory Basis: Not applicable.

Operating Experience and Demonstration: Ultrasonic thickness examinations have been performed on the fire water system. UT examinations have determined that there is loss of material due to pitting in the fire water system piping. There have been repairs to the fire water system piping due to excessive pipe wall thinning. This experience demonstrates that the Fire Water Piping Thickness Evaluation Program is able to identify loss of material and initiate corrective action. Based on this experience, these examinations will continue to provide assurance that aging effects will be adequately managed so that fire protection system components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.8.6 Control Room Halon Fire System Inspection

Purpose: The purpose of the Control Room Halon Fire System Inspection, with respect to license renewal, is to assure that frequently manipulated components are free of aging effects.

Scope: The components within the scope of the Control Room Halon Fire System Inspection are listed in Table 3.4-6.

Aging Effects: The aging effects addressed by the Control Room Halon Fire System Inspection are loss of material due to wear from frequent manipulations and cracking.

Method: The Control Room Halon Fire System Inspection provides for periodic inspections to ensure halon system operability. Leakage in the pressurized portion of the halon system would be detected by this inspection. During these inspections, the cylinders are disconnected from the headers and are weighed.

The procedure verifies the nitrogen bottle pressure is adequate and that cracking or loss of material has not caused a leak to occur. Steps in the procedure verify the correct reinstallation. The components are visually inspected during this activity.

Industry Code or Standards: The NFPA Standard 12A was utilized in the development of this inspection program.

Frequency: Inspections are performed at least once every 6 months.

Acceptance Criteria or Standard: Acceptance standards for minimum halon cylinder weights, minimum halon and nitrogen pressures and maximum nitrogen cylinder pressures are listed in the inspection procedure.

Regulatory Basis: The regulatory basis for this program is 10CFR50.48, 10CFR Part 50 Appendix R, and Operating License DPR-51 Paragraph 2.c.(8).

Operating Experience and Demonstration: Semi-annual testing has identified several types of system degradation. Included are the loss of nitrogen or halon pressure or weight, and the perturbation of the protected enclosure halon vapor barrier. Past system halon loss has been, in part, due to removal of factory gauges to install calibrated test gauges during cylinder testing. The cylinders have been fitted with permanently installed calibrated gauges to lesson the losses of nitrogen pressure or halon during testing. Spare cylinders have been removed entirely from the testing program. In addition, nitrogen pressure or halon has been lost because of bad seals in the cylinder control heads.

The ceramic board and ceiling tile fire barriers exist to contain the halon in the control room ceiling and to provide a fire barrier between the occupied control room workspace and the protected overhead space. Semiannual inspection will ensure the identification and repair of any degradation of the barriers in the overhead. Based on this operating experience, the continued implementation of the Control Room Halon Fire System

Inspection provides reasonable assurance that the effects of aging will be adequately managed so that the system components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.8.7 Reactor Coolant Pump Oil Collection System Inspection

Purpose: The purpose of the Reactor Coolant Pump Oil Collection System Inspection, with respect to license renewal, is to ensure integrity of the reactor coolant pump oil leakage collection system.

Scope: The scope of the Reactor Coolant Pump Oil Collection System Inspection applies to the shrouds, drip pans, dammed areas, accessible piping, collection tanks, and spray protection.

Aging Effects: The aging effects addressed by the Reactor Coolant Pump Oil Collection System Inspection are a loss of material and a loss of mechanical closure integrity. These aging effects would be caused by general corrosion of the carbon steel internal surfaces or external surfaces due to the potential for water leakage into the system.

Method: The Reactor Coolant Pump Oil Collection System Inspection is a visual inspection. Guidance for this inspection is contained in a site procedure.

Industry Code or Standards: Not applicable.

Frequency: The inspection is performed during shutdown and prior to startup for each refueling outage.

Acceptance Criteria or Standard: Acceptance criteria are based on guidance provided in site procedures. There should be no accumulation of oil outside the collection system**.**

Regulatory Basis: 10CFR50 Appendix R Section III.O and 0CAN049705, response to IR 96-027 [Reference B-19].

Operating Experience and Demonstration: The Reactor Coolant Pump Oil Collection System Inspection is normally performed during shutdown and prior to startup for each refueling outage. If an abnormal accumulation of oil is found or the integrity of the collection system is found deficient, corrective action will be initiated and documented per the inspection procedure. Additionally, if oil is found that is not being collected, corrective action will be initiated in accordance with the inspection procedure. Visual inspections have been effective in identifying these types of deficiencies. If any of these events occur, corrective actions will be taken to reestablish design requirements and prevent reoccurrence. Based on this review, continuation of the Reactor Coolant Pump Oil Collection System Inspection will provide reasonable assurance that the aging effects associated with the oil collection system will be managed such that the applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.9 FLOW ACCELERATED CORROSION PREVENTION

Purpose: The purpose of the Flow Accelerated Corrosion Prevention Program is to provide a programmatic approach for identifying, inspecting, and managing loss of material for components that are adversely affected by flow accelerated corrosion (also known as erosion/corrosion).

Scope: For the systems within the scope of license renewal, only the main feedwater and main steam systems are identified as susceptible to flow-accelerated corrosion.

Aging Effects: The aging effect is a phenomenon that results in metal loss from components made of carbon steel, which occurs only under certain conditions of flow, chemistry, geometry, and material. The aging management reviews credit this program with determining which systems are susceptible to flow-accelerated corrosion and monitoring the loss of material for those systems.

Method: The Flow Accelerated Corrosion Prevention Program utilizes a combination of computer codes, previous examination results, industry experience, and engineering judgment to determine specific locations to be inspected. Ultrasonic inspections and visual inspections (where applicable) are utilized to quantify the amount of wall thinning on a component. The inspection data is documented in engineering reports that are developed for each refueling outage. Inspection information is input into the CHECWORKS Program to refine the PASS 2 analysis to predict wall thinning rates more accurately.

Per procedural requirements for developing modification packages, program-screening checklists are completed to ensure impact evaluations are accomplished for modifications affecting systems included in the flow-accelerated corrosion program. The current CHECWORKS database tracks safety and non-safety related large bore components in the main steam, main feedwater, condensate, reheat steam, extraction steam, and heater vents and drains systems.

Industry Code or Standards: Not applicable.

Frequency: Inspection frequency for each location is based on consideration of previous inspection results, CHECWORKS predictions resulting from PASS-2 analysis, changes in plant operating or chemistry conditions, and pertinent industry events.

Acceptance Criteria or Standard: The acceptance criteria for the Flow Accelerated Corrosion Prevention Program are located in site procedures. Any measured wall thickness below, or projected to be below, 70% of nominal wall at the next refueling outage is evaluated to determine if additional areas need to be examined. Any component with a measured wall thickness below, or projected to be below, the ASME B31.1 minimum wall will be replaced, unless a local wall thinning evaluation can show acceptability for continued service.

Regulatory Basis: The documents that provide the regulatory basis for the ANO-1 Flow Accelerated Corrosion Prevention Program include:

- NRC Bulletin 87-01, *Thinning of Pipe Walls in Nuclear Power Plants*
- NRC Generic Letter 89-08, *Erosion/Corrosion-Induced Pipe Wall Thinning*

Operating Experience and Demonstration: From the start of the eighth refueling outage to the end of the fifteenth refueling outage, approximately nine hundred inspections have been accomplished. Resulting from these inspections, approximately one hundred twenty-five components were replaced. When replacements are required, the materials used are resistant to flow accelerated corrosion damage. This program has proven effective in managing the loss of material caused by flow accelerated corrosion. Based on operating experience, the continued implementation of the Flow Accelerated Corrosion Program provides reasonable assurance that the effects of aging will be adequately managed so that the main steam and main feedwater systems will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.10 INSPECTION AND PREVENTIVE MAINTENANCE OF THE ANO-1 POLAR CRANE

Purpose: This program provides for the inspection and preventive maintenance of the ANO-1 polar crane.

Scope: Structural steel associated with the ANO-1 polar crane.

Aging effects: The aging effect managed by the Inspection and Preventive Maintenance of the ANO-1 Polar Crane is a loss of material.

Method: The polar crane steel components are visually inspected in accordance with the governing procedure. In addition to a visual inspection, the cranes bridge system bolting tightness is tested by hand.

Industry Codes of Standards: The polar crane is inspected, tested, and maintained in compliance with ANSI B30.2.

Frequency: The polar crane steel components are inspected annually, in conjunction with other periodic crane inspection activities.

Acceptance Criteria of Standard: The acceptance criteria for this inspection are no visual indications of deformation, cracking, loose or corroded (i.e., loss of material) members, and no loose or missing bolts.

Regulatory Basis: ANO commitment to NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants*.

Operating Experience and Demonstration: Inspection findings related to the polar crane structural steel components have not identified indications of a loss of material (i.e., corrosion). However, visual inspections have proven effective in identifying indications of the applicable aging effects for the polar crane structures and components. The continued implementation of the Inspection and Preventive Maintenance of the ANO-1 Polar Crane provides reasonable assurance that the aging effect are managed such that the polar crane will continue to perform its intended function consistent with the current licensing basis for the period of extended operation.

4.11 INSTRUMENT AIR QUALITY

Purpose: The purpose of the Instrument Air Quality Program, with respect to license renewal, is to ensure that the instrument air supplied to components is maintained free of water and significant contaminants.

Scope: The Instrument Air Quality Program applies to those components within the scope of license renewal, supplied with instrument air where pressure boundary integrity is required for the component to perform its intended function.

Aging Effects: The aging effects addressed by the Instrument Air Quality Program are loss of material and cracking.

Method: Sampling is performed to verify the instrument air is dry. Testing also checks for contaminants or foreign material in the air supply.

Industry Code or Standards: ISA Quality Standard for Instrument Air, S7.3-1975 [Reference B-20] is an applicable industry standard used to develop the Instrument Air Quality Program.

Frequency: The frequency is in accordance with site procedures.

Acceptance Criteria or Standard: Acceptance criteria are based on guidance provided in ISA Quality Standard for Instrument Air, S7.3-1975.

Regulatory Basis: ANO commitment to NRC Generic Letter 88-14, *Instrument Air Supply System Problems Affecting Safety-Related Equipment*.

Operating Experience and Demonstration: The results of the periodic testing have verified the instrument air quality is being maintained. Based on this information, the continued implementation of the Instrument Air Quality Program provides reasonable assurance that the aging effects will be managed such that components supplied with instrument air will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.12 LEAKAGE DETECTION IN REACTOR BUILDING

Purpose: The purpose of leakage detection in the reactor building, with respect to license renewal, is to monitor for leakage to manage the consequences of cracking, loss of material, and loss of mechanical closure integrity.

Scope: Leakage detection in the reactor building is focused on RCS leakage, but also includes other systems that have the potential to leak in the reactor building.

Aging Effects: Monitoring for leakage in the reactor building is credited as one of the methods of managing the aging effects of cracking, loss of material, and loss of mechanical closure integrity.

Method: Leakage detection in the reactor building is accomplished by three different means. These are the inventory balance, the reactor building sump monitoring, and the reactor building atmosphere radioactivity monitoring.

Reactor Coolant Inventory Balance

ANO-1 Technical Specification Table 4.1-2 requires the RCS leak rate to be determined periodically. RCS allowable leakage is limited, during power operation, as specified in ANO-1 Technical Specification 3.1.6. The envelope for leakage monitoring is the RCS and makeup system with all leakage assumed to be from the RCS unless proven otherwise.

Reactor Building Sump Monitoring

The reactor building sump fill rate is trended, and in conjunction with the RCS leak rate, is used for an indication of the source of leakage into the sump (i.e. leakage inside/outside the reactor building, RCS leakage or non-RCS leakage such as from main feedwater or main steam systems, etc.).

Reactor Building Radioactivity Monitoring

The reactor building leak detector consists of the reactor building atmosphere particulate detector and the reactor building atmosphere gaseous detector. These monitor readings are recorded on operator logs.

Industry Code or Standards: Not applicable.

Frequency: Potential indicators of RCS leakage are monitored continuously throughout each shift. The RCS leak rate determination is performed as required by ANO-1 Technical Specification 3.1.6 and Table 4.1-2 and ANO-1 site procedures.

Acceptance Criteria or Standard: The RCS allowable leakage is limited as specified in ANO-1 Technical Specification 3.1.6. Identified non-RCS leakage is evaluated on a case-by-case basis.

Regulatory Basis: ANO-1 Technical Specification 3.1.6 and Table 4.1-2 provide the regulatory basis for this program.

Operating Experience and Demonstration: A review of ANO-1 operating experience (i.e. ANO-1 specific licensee event reports dating back to 1984) confirms that these activities are effective in detecting leakage in the reactor building. In 1989, a non-isolable leak on a reactor coolant system drain line was detected when RCS leakage increased from the previous day's leak rate by approximately 0.2 gpm (to a total of 0.74 gpm). Although the cause of this leak was attributed to a weld defect and not an aging effect, this example demonstrates the program is effective in detecting small amounts of RCS leakage. Leaks caused by cracking, loss of material, or loss of mechanical closure integrity would be detected by this program. Based on operating experience, the continued implementation of Leakage Detection in the Reactor Building provides reasonable assurance that the effects of aging will be adequately detected and managed so that the systems within scope of this program will perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.13 MAINTENANCE RULE

Purpose: Maintenance Rule system and structural walkdowns are conducted to detect and manage aging effects of structures and components within the scope of the license renewal.

Scope: Structural components and commodities within the scope of license renewal and managed under the Maintenance Rule are listed in Tables 3.6-1 through 3.6-8. Coatings inspections apply to coated surfaces of ANO-1 structures and components within the scope of license renewal.

Aging Effects: The Maintenance Rule is utilized to manage cracking, loss of material, and change in material properties of structures and components within the scope of license renewal.

Method: Visual inspections of structures and components are performed.

Industry Codes or Standards: Not applicable.

Frequency: Structural and component walkdowns are performed periodically, and the frequency varies depending on the structure or component being inspected.

Acceptance Criteria or Standards: No unacceptable visual indications of cracking, loss of material, or change of material properties of structures or components.

Timing of New Program or Activity: Incorporate additional guidance for coatings inspections as part of existing system and structural walkdowns during next revision of System Engineering Desk Guide, but no later than the end of the initial 40-year license term for ANO-1.

Regulatory Basis: The Maintenance Rule is consistent with the requirements of 10CFR50.65.

Operating Experience and Demonstration: The ANO-1 history of successful operation demonstrates that visual inspections have been effective in managing the effects of aging on structures and components. This is indicative that the Maintenance Rule will be effective in the future for managing aging effects, since it incorporates proven monitoring techniques, acceptance criteria, corrective actions, and administrative controls from existing programs and procedures. Based on this experience, the continued implementation of the Maintenance Rule provides reasonable assurance that the effects of aging will be adequately managed so that the systems within the scope of this program will perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.14 OIL ANALYSIS

Purpose: The purpose of the Oil Analysis Program is to ensure the oil environment in the mechanical systems is maintained to the quality required. Oil analysis program controls are credited as a program for maintaining oil systems free of contaminants (primarily water and particulates) thereby preserving an environment that is not conducive to corrosion.

Scope: The scope of the Oil Analysis Program, with respect to license renewal, is limited to sampling and analysis of lubricants in the following components.

- Auxiliary building electrical room chillers
- Emergency diesel generators
- Decay heat pumps
- Reactor building spray pumps
- Primary makeup pumps
- Diesel driven fire pump and engine
- EFW pumps and EFW turbine
- Alternate AC diesel generator
- Control room ventilation compressors

Aging Effects: The Oil Analysis Program has been credited for ensuring the oil is free of water or contaminants. This manages the aging effects of cracking and loss of material.

Method: For components that have oil changes at intervals that satisfy the recommendations of the equipment manufacturer, a particle count and check for water are performed to detect evidence of abnormal wear rates, contamination by moisture, or excessive corrosion. For components that do not have regular oil changes, viscosity and neutralization number are determined to verify the oil is suitable for continued use. Specialized sampling and testing is performed for some components based on their specific use and vendor recommendations.

Industry Code or Standards: References used to develop this program include:

- ASTM D95, *Standard Test Method for Water in Petroleum Products and Bituminous Materials by Distillation* [Reference B-21]
- ASTM D664, *Standard Test Method for Neutralization Number by Potentiometric Titration* [Reference B-22]

Frequency: The frequency of sampling is based upon equipment manufacturer recommendations and standard industry practices for each component.

Acceptance Criteria or Standard: The acceptance criteria for the Oil Analysis Program are contained in site procedures.

Regulatory Basis: Not applicable.

Operating Experience and Demonstration: Review of historical oil sampling plots for components in the scope of license renewal verifies that the lubricating oil is maintained free of excess water. For all non-engine oils, the total acid number is measured to verify contamination is not occurring that has impacted oil acidity. For engine oils, the total base number is measured and trended to verify that the oil is being maintained with enough additives to neutralize any acid that forms due to engine operation. Based on this information, the continued implementation of the Oil Analysis Program provides reasonable assurance that the aging effects will be managed such that the applicable components will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

4.15 PREVENTIVE MAINTENANCE

Purpose: The purpose of the Preventive Maintenance Program is to perform preplanned, repetitive maintenance tasks on plant components and systems with the intent to extend equipment operating life and to minimize the possibility of in-service component failures.

Scope: The scope of the Preventive Maintenance Program, with regard to license renewal, is those preventive maintenance tasks credited with managing the aging effects identified in Section 3.0 of the LRA. Below is a list of preventive maintenance activities and the aging effects that they address.

Preventive Maintenance Activity	Aging Effect
Borated water storage tank internal inspection	Loss of material
Borated water storage tank external inspection	Loss of material and loss of mechanical closure integrity
Reactor building ventilation cooling coil cleaning and inspection	Fouling and loss of material
Hydrogen sampling system cabinet / heat exchanger cleaning, inspection, and lubrication	Fouling
Emergency fire diesel cooling water quarterly sampling for corrosion inhibitor	Loss of material
Penetration room floor drain check valves inspection	Loss of material and cracking
Decay heat room drain valves inspection	Loss of material, cracking, and loss of mechanical closure integrity
Emergency diesel fuel oil tank inspection	Loss of material
Emergency diesel generator HVAC components inspection	Loss of material and loss of mechanical closure integrity
Control room ventilation inspections	Fouling and loss of material
Battery Charger and Penetration Room Cooler Cleaning & Inspection	Loss of material and loss of mechanical closure integrity

Preventive Maintenance Activity	Aging Effect
Auxiliary Building Switchgear Room Coolers Cleaning and Inspection	Loss of material, loss of mechanical closure integrity, and fouling
Auxiliary Building Decay and Heat Room Coolers Cleaning and Inspection	Loss of material, loss of mechanical closure integrity, and fouling
HPI Room Coolers Cleaning and Inspection	Loss of material, loss of mechanical closure integrity, and fouling

Aging Effects: The aging effects addressed by the Preventive Maintenance Program are identified in the above table.

Method: The Preventive Maintenance Program consists of preplanned repetitive tasks for maintenance activities on plant components. These activities include periodic inspections, tests, calibrations, measurements and adjustments, cleaning, sampling or analysis, lubrication, and the replacement of limited life parts or components. The required maintenance activities are defined by preventive maintenance engineering evaluations. Vendor recommendations, ANO commitments, equipment history, industry experience and owners group recommendations are considered in these evaluations.

Industry Code or Standards: Not applicable.

Frequency: The frequency of the performance of preventive maintenance tasks is based on recommendations in the preventive maintenance engineering evaluations. The scheduling of preventive maintenance task is established to coincide with any previously schedule-related tasks, system outages, or component outages as applicable.

Acceptance Criteria or Standard: Applicable acceptance criteria are provided for each repetitive maintenance task based on the preventive maintenance engineering evaluations. Existing preventive maintenance procedures that do not adequately address inspection criteria for aging effects will be updated to provide appropriate inspection criteria.

Timing of New Program or Activity: New preventive maintenance activities that address inspection criteria for aging effects will be incorporated into existing preventive maintenance procedures prior to the end of the initial 40-year license term for ANO-1.

Regulatory Basis: Not applicable.

Operating Experience and Demonstration: The ANO-1 history of successful operation demonstrates that typical preventive maintenance activities, such as visual inspections, cleaning, and sampling, have been effective in managing the effects of aging on components. This is indicative that the Preventive Maintenance activity will be effective in the future for managing aging effects since it consists of proven monitoring techniques, acceptance criteria, corrective actions, and administrative controls in existing programs and procedures. Based on this experience, the continuation of the Preventive

Maintenance activity provides reasonable assurance that the effects of aging will be adequately managed so that the components within the scope of this program will perform their intended functions consistent with the current licensing basis for the period of extended operation

4.16 REACTOR BUILDING LEAK RATE TESTING

The Reactor Building Leak Rate Testing Program provides assurance that leakage from the reactor building will not exceed required maximum values for reactor building leakage. The Reactor Building Leak Rate Testing Program consists of Type A, Type B, and Type C testing.

Type A testing measures the primary reactor building overall integrated leakage rate. This is also known as Integrated Leak Rate Testing.

Type B testing measures the leakage across air locks, door seals, equalizing valves, and test ports whose designs incorporate resilient seals, gaskets, and flexible metal seals.

Type C testing measures primary reactor building isolation valve leakage rates, whether the valves are manual or automatic. Generically, Type B and Type C tests are also known as local leak rate testing.

For the purposes of aging management programs for license renewal, only Type A and Type C tests are considered.

4.16.1 Integrated Leak Rate Testing

Purpose: The purpose of the integrated leak rate test is to measure the primary reactor building overall integrated leakage rate.

Scope: The scope of the integrated leak rate test is the reactor building.

Aging Effects: Type A integrated leak rate testing identifies loss of material or cracking.

Method: The integrated leak rate test pressurizes the reactor building to the peak calculated reactor building internal pressure, for the design basis loss of coolant accident, and monitors the rate of pressure drop. A leak rate is calculated based on the rate of the pressure drop observed.

Industry Code or Standards: Guidelines for the testing is provided by:

- ANSI/ANS 56.8, 1994, *Containment System Leakage Testing Requirements* [Reference B-14]
- ANSI N45.4, 1972, *Leakage Rate Testing of Containment Structures for Nuclear Reactors* [Reference B-15]

Frequency: Integrated leak rate tests are required once per ten years as long as calculated leakage remains less than the maximum allowable leakage rate.

Acceptance Criteria or Standard: The maximum allowable reactor building leakage rate is defined in the technical specifications.

Regulatory Basis: The regulatory basis for this program includes 10CFR50.54; 10CFR50 Appendix J option B; Regulatory Guide 1.163, *Performance-Based Containment Leak-Test Program* and ANO-1 Technical Specification 6.8.4.

Operating Experience and Demonstration: Historically, integrated leakage rates have been well within the maximum allowable leakage rates specified in the technical specifications. The continued implementation of the integrated leak rate testing provides reasonable assurance that the aging effects will be managed such the reactor building will continue to perform its intended functions consistent with the current licensing basis for the period of extended operation.

4.16.2 Local Leak Rate Testing

Purpose: The purpose of the local leak rate test is to measure the leakage across individual penetration components and determine the leakage of each penetration.

Scope: Local Leak Rate Testing addresses leakage across reactor building penetrations.

Aging Effects: Local leak rate testing identifies changes in material properties, loss of material, and cracking.

Method: The local leak rate testing measures leakage on individual components by pressurizing the penetration to the specified test pressure.

Industry Code or Standards: Guidance for the local leak rate testing is provided by:

- ANSI/ANS 56.8, 1994, *Containment System Leakage Testing Requirements [Reference B-14]*
- ANSI N45.4, 1972, *Leakage Rate Testing of Containment Structures for Nuclear Reactors [Reference B-15]*

Frequency: Local leak rate tests are based on past local leak rate test results, service conditions, design, safety impact, previous failure causes, and common mode failure detection. Local leak rate tests are required when any adjustment or maintenance on an isolation barrier is performed that can affect sealing characteristics. Examples are replacement of packing, adjustments to valve stroke or closure switch settings, or lapping of valve seats.

Acceptance Criteria or Standard: The maximum allowable reactor building leakage rate is defined in ANO-1 Technical Specification 6.8.4.

Regulatory Basis: The regulatory basis for this program includes 10CFR50.54; 10CFR50 Appendix J; Regulatory Guide 1.163, *Performance-Based Containment Leak-Test Program* and ANO-1 Technical Specification 6.8.4.

Operating Experience and Demonstration: The sum of the leakage rates, at accident pressure, of Type B tests and pathway leakage rates from Type C tests, have been maintained less than the maximum allowable leakage rate, with margin, as specified in the technical specifications. The continued implementation of the local leak rate testing provides reasonable assurance that the aging effects will be managed such the reactor building will continue to perform its intended functions consistent with the current licensing basis for the period of extended operation.

4.17 REACTOR BUILDING SUMP CLOSEOUT INSPECTION

Purpose: The purpose of the Reactor Building Sump Closeout Inspection is to detect significant degradation of the sump components and remove any foreign objects that could impede suction from the sump.

Scope: The Reactor Building Sump Closeout Inspection applies to ANO-1 reactor building sump, the area immediately surrounding the sump, the screening materials, and the equipment and structural components inside the sump.

Aging Effects: The aging effects addressed by the Reactor Building Sump Closeout Inspection are loss of material for the carbon steel components and cracking for stainless steel components due to the presence of borated water.

Method: The Reactor Building Sump Closeout Inspection is a visual inspection of the exterior and interior surfaces of the sump. If the inspection is associated with a limited scope outage and the screens are not unbolted, or controls for a foreign material exclusion area are in place, then only exterior surfaces of the sump are inspected.

Industry Code or Standards: Not applicable.

Frequency: As a minimum, this inspection is performed at the end of each refueling outage.

Acceptance Criteria or Standard: Acceptance criteria are based on guidance provided in the closeout inspection procedure. Surfaces are inspected for evidence of the following: significant structural distress, corrosion, excessive rust, significant physical degradation, obvious loose or missing bolts, excessive hatch gap, or tears in sump screens. The reactor building sump screen is inspected for excessive openings or gaps in the screen. The sump internal inspection also verifies that there is no obvious loose bolting in the internal area of the sump and that no excessive corrosion or loss of material exists on the bolting and yokes of valves or on the divider plate. The inspection verifies that there is no excessive pitting or corrosion on piping external surfaces or flued heads.

Regulatory Basis: Not applicable.

Operating Experience and Demonstration: As detailed during the 1998 inspection, wetted portions of the sump were inspected for indications of aging effects. Structural members showed very little evidence of corrosion. Some light boron was removed which had formed due to a leaking valve above the sump. No service induced, or environmentally induced deficiencies were found with respect to carbon or stainless steel valve parts. The carbon steel portions of the divider plate were beginning to show light corrosion and rust along the bottom edge and up both sides. Flued heads showed no signs of service induced or environmentally induced pitting, cracking, or corrosion.

This experience demonstrates that continuation of the Reactor Building Sump Closeout Inspection provides reasonable assurance that aging affects will be managed such that the reactor building sump will continue to perform its intended functions consistent with the current licensing basis during the period of extended operation.

4.18 REACTOR VESSEL INTEGRITY

Section 2.3.1 of the ANO-1 LRA identifies the reactor vessel as a component that is subject to aging management review for license renewal. For the reactor vessel, Section 3.2.5 and Table 3.2-1 identify reduction in fracture toughness as the aging effect requiring management for the period of extended operation. The ANO-1 Reactor Vessel Integrity Program will manage the aging effect of reduction in fracture toughness of the reactor vessel. The ANO-1 Reactor Vessel Integrity Program consists of the following five interrelated subprograms.

- Master Integrated Reactor Vessel Surveillance Program
- Cavity Dosimetry Program
- Fluence and Uncertainty Calculations
- Pressure/Temperature Limits
- Monitoring Effective Full Power Years

Entergy Operations complies with the requirements of 10CFR50.60, Appendices G and H, and 10CFR50.61, through the ANO-1 Reactor Vessel Integrity Program. In accordance with 10CFR50.60 and 10CFR50.61, periodic updates for the five subprograms are provided to the NRC for review.

Continuation of the Reactor Vessel Integrity Program provides reasonable assurance that applicable aging effects will be managed such that the reactor vessel will continue to perform its intended functions consistent with the current licensing basis during the period of extended operation.

4.18.1 Master Integrated Reactor Vessel Surveillance

Entergy is a participant in the BWOG Master Integrated Reactor Vessel Surveillance Program. The Master Integrated Reactor Vessel Program meets the requirements of Appendix H of 10CFR Part 50, with regard to integrated surveillance programs (paragraph III.C). In addition, the Master Integrated Reactor Vessel Program addresses reference temperature shift concerns and pressurized thermal shock in accordance with 10CFR50.61. A description of the Master Integrated Reactor Vessel Program is provided in BAW-1543A, Revision 2 [Reference B-23] and in BAW-2251A [Reference B-24].

Purpose: The purpose of the Master Integrated Reactor Vessel Program is to provide a method to monitor reactor pressure vessel materials containing Linde 80 high copper beltline welds for determining the reduction of material toughness by neutron irradiation embrittlement.

Scope: The scope of the Master Integrated Reactor Vessel Program includes beltline plate and weld material for the beltline region of the ANO-1 reactor vessel.

Aging Effects: The aging effect requiring management is the reduction of material toughness by neutron irradiation embrittlement.

Method: Fracture toughness specimens are irradiated within two operating B&W reactor vessels (i.e., Davis-Besse and Crystal River–3) and the participating Westinghouse reactor vessels. The specimens are irradiated in capsules located near the reactor vessel inside wall, thus enabling reactor vessel materials to become irradiated to beyond anticipated license renewal fluence levels. The fracture toughness specimens are tested in accordance with the applicable ASTM standards identified in Section 5.0 of BAW-1543A, Revision 2.

Industry Code or Standard: ASTM E185 [Reference B-25]; ASTM E900 [Reference B-30]; ASTM standards as identified in Section 5.0 of BAW-1543.

Frequency: The capsule withdrawal schedules are presented in BAW-1543, Revision 4, Supplement 3. The Master Integrated Reactor Vessel Program schedule may be altered due to unscheduled downtimes or extended outages at the host plants. In addition, certain surveillance capsules may receive additional irradiation to fully satisfy license renewal fluence requirements.

Acceptance Criteria or Standard: Fracture toughness specimens removed from the surveillance capsules will be laboratory tested to ensure reactor vessel fracture toughness properties exhibit upper shelf energy greater than 50 ft-lbs. If the Charpy upper shelf energy drops below 50 ft-lbs, then it must be demonstrated that margins of safety against fracture are equivalent to those of Appendix G of ASME Section XI. In addition, calculations of reference temperature for pressurized thermal shock (RT_{PTS}) must be below the screening criteria of 270ºF for plates, forgings, and longitudinal welds and 300ºF for circumferential welds. If the projected reference temperature exceeds the screening criteria, licensees are required to submit an analysis and schedule for such flux reduction programs as are reasonably practicable, to avoid exceeding the screening criteria. If no reasonably practicable flux reduction program will avoid exceeding the screening criteria, licensees shall submit a safety analysis to determine what actions are necessary to prevent potential failure of the reactor vessel if continued operation beyond the screening criteria is allowed.

Regulatory Basis: 10CFR50.60, *Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation*; 10CFR50.61, *Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock*; Appendix G to Part 50, *Fracture Toughness Requirements*; Appendix H to Part 50, *Reactor Vessel Material Surveillance Program Requirements*; and ANO-1 Technical Specification 3.1.2, *Pressurization, Heatup, and Cooldown Limitations.*

4.18.2 Cavity Dosimetry

Purpose: The purpose of the Cavity Dosimetry Program is to verify the accuracy of fluence calculations and to determine fluence uncertainty values.

Scope: The ANO-1 reactor vessel has installed cavity dosimetry.

Aging Effects: The reduction of material toughness by irradiation embrittlement.

Method: Dosimeters (i.e., U_{238}, Np_{237}, Ni, Cu, etc.) are irradiated in the cavity region outside of the ANO-1 reactor vessel. Cavity dosimetry was irradiated at ANO-1 for Cycles 10, 11, and 12, and combined Cycles 13 and 14. At present, cavity dosimetry is being irradiated at ANO-1 for combined cycles 15 and 16.

The cavity dosimeters are measured to determine the activity resulting from the fast fluence irradiation. In addition, calculations of the dosimetry activities are performed using operational data.

Industry Code or Standard: ASTM E185 [Reference B-25] and ASTM E900 [Reference B-30].

Frequency: At present, cavity dosimetry is changed out on an every-other-cycle basis. Projections indicate extending the frequency to an every-third-cycle exchange period or longer may be acceptable. The cavity dosimetry exchange schedule may be altered due to changes in fuel type, fuel loading pattern, or power rating of ANO-1.

Acceptance Criteria or Standard: Not applicable. This information is used in conjunction with the fluence and uncertainty calculations.

Regulatory Basis: 10CFR50.60, *Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation*; Appendix H to Part 50, *Reactor Vessel Material Surveillance Program Requirements*; BAW-2241A-P, and 10CFR50.61.

4.18.3 Fluence and Uncertainty Calculations

Purpose: The purpose of the reactor vessel fluence and uncertainty calculations is to provide an accurate prediction of the actual reactor vessel accumulated neutron fast fluence value for use in development of the pressure/temperature limit curves and pressurized thermal shock calculations.

Scope: The fluence and uncertainty calculations apply to the ANO-1 reactor vessel.

Aging Effect: The reduction of material toughness by neutron irradiation embrittlement.

Method: The cavity dosimetry program yields irradiated dosimeters that are analyzed based on ANO-1 specific geometry models (i.e., fuel, reactor vessel, capsule holders, concrete structures), macroscopic cross sections, cycle-specific sources using the DORT and GIP computer codes, and reference microscopic cross section (BUGLE 93). Specific attention is given to target fluence values for limiting reactor vessel beltline weld locations. Recently updated fluence and uncertainty calculations were based on cavity dosimetry irradiated at ANO-1 for cycles 10 through 14. Future calculation revisions will be based on cavity dosimetry being irradiated at ANO-1 for combined cycles 15 and 16.

Industry Code or Standard: ASTM E900 [Reference B-30] and ASTM E185 [Reference B-25].

Frequency: Fluence and uncertainty calculations are expected to follow each cavity dosimetry analysis for the next few years. The frequency of updating fluence and uncertainty calculations may change as additional data are obtained. Future decisions concerning the frequency of withdrawal of dosimetry will be based on changes in fuel type or fuel loading pattern.

Acceptance Criteria or Standard: The fluence uncertainty values are to be within the NRC-suggested limit of ±20%. Calculated fluence values for fluence levels above 1.0 MeV are compared with the measurement values to determine if calculations contain any errors. The method represents a continuous validation process to ensure that no biases have been introduced and that the uncertainties remain comparable to the reference benchmarks.

Regulatory Basis: Appendix H of 10CFR Part 50, *Reactor Vessel Material Surveillance Program Requirements*; BAW-2241A-P; and BAW-2251A.

4.18.4 Pressure/Temperature Limit Curves

Purpose: The purpose of the pressure/temperature limit curves is to establish the normal operating, inservice leak test, and hydrostatic test transient limits for the RCS.

Scope: The pressure/temperature limit curves apply to the ANO-1 reactor vessel.

Aging Effects: The change of material properties by neutron irradiation embrittlement.

Method: Pressure/temperature curves are generated assuming a postulated 1/4T surface flaw in accordance with ASME Section XI, Appendix G. Bounding input heatup and cooldown transients are used to develop the pressure/temperature curves.

Industry Code or Standard: ASME Section XI, Appendix G, 1989 Edition; ASME Code Case N-514; ASTM E900 [Reference B-30].

Frequency: Pressure/temperature limit curves are valid for a period expressed in effective full power years. The curves are required to be updated prior to exceeding this time period.

Acceptance Criteria or Standard: Pressure/temperature limit curves must be in place for continued plant operation.

Regulatory Basis: ANO-1 Technical Specification 3.1.2 and BAW-10046 [Reference B-27]

4.18.5 Effective Full Power Years

Purpose: The purpose of determining the EFPY is to accurately monitor and tabulate the accumulated operating time experienced by the reactor vessel.

Scope: The EFPY activity applies to the ANO-1 reactor vessel.

Aging Effect: The reduction of material toughness by neutron irradiation embrittlement.

Method: The effective full power days of plant operation are based on reactor incore power readings. The Nuclear Applications Software, which runs on the plant computer, collects incore instrument data. Site reactor engineers determine effective full power days values by comparing the burnup to the thermal power calculated burnup. The data is collected continuously.

Industry Code or Standard: Not applicable.

Frequency: ANO-1 is continuously computer monitored and updated weekly by site reactor engineers to determine the effective full power days of reactor coolant system operation during the previous seven day period.

Acceptance Criteria or Standard: For a given fuel cycle, the updated effective full power days calculation based on the power history must be within ±0.3 EFPD of the plant computer generated value.

Regulatory Basis: ANO-1 Technical Specification 3.1.2.

4.19 SERVICE WATER INTEGRITY

Purpose: The purpose of the Service Water Integrity Program is to ensure the ANO service water system components continue to operate and perform their safety-related functions for the remaining life of ANO-1. The Service Water Integrity Program activities are integrated into the normal system engineering responsibilities that include such duties as performance monitoring, Maintenance Rule administration, and Generic Letter 89-13 compliance.

Scope: The scope of the Service Water Integrity Program, with respect to license renewal is limited to activities on ANO-1 service water system components and structures, including the emergency cooling pond.

Aging Effects: The Service Water Integrity Program is credited with managing the following aging effects.

- The flow rate testing ensures the effects of fouling do not reduce flow rates below required values. System cleaning is performed when necessary to maintain cleanliness in accordance with GL 89-13 commitments.

- The heat exchanger testing manages the aging effect of fouling by ensuring the heat exchangers can remove the necessary heat load.

- The thickness mapping and visual inspections manage the effects of loss of material from the service water components.

- Visual inspections of a sample of safety-related valves and heat exchangers manage the effect of cracking of the components.

- The service water bay inspection manages loss of material for the mechanical components in the service water bay.

Method: The ANO-1 Service Water Integrity Program consists of the following testing activities to achieve the program objectives committed to in the Generic Letter 89-13 responses.

- Flow testing of safety related heat exchangers every refueling outage

- Heat transfer testing or inspections of heat exchangers

- Pump performance testing in accordance with ASME Section XI

- Flushing and minimum flow testing of the reactor building coolers in accordance with the technical specifications

- Testing of the sluice gates and system boundary valves for leakage

Non-destructive examinations commitments in Generic Letter 89-13 responses include the following.

- Cleaning and inspecting the intake structure service water bay and sluice gates on a periodic basis
- Mapping pipe thickness at selected locations
- Visual inspections of a sample of the safety related heat exchangers and valves
- Inspections of the epoxy coating internal to the ECP return line

Chemical controls commitments in Generic Letter 89-13 responses include the following.

- Treatment with biocides to minimize macrobiological fouling and MIC
- A side stream corrosion rack or other acceptable methods for detailed monitoring of the corrosion and chemical effects
- Addition of a corrosion inhibitor, to the extent practical to reduce the corrosion rate of the system carbon steel piping

Other activities also include:

- The periodic flushing of pump bearing coolers and stagnant portions of the service water piping
- Mechanical or chemical cleaning, when necessary, to remove fouling

Industry Code or Standards: Not applicable.

Frequency: The frequency of activities is specified in site procedures. The frequency may be adjusted based on the results of testing and inspection**.**

Acceptance Criteria or Standard: The acceptance criteria for the Service Water Integrity Program are located in site procedures.

Regulatory Basis: The regulatory basis for the Service Water Integrity Program is the ANO commitment to Generic Letter 89-13 [Reference B-31, B-32, B-33, B-34, and B-35].

Operating Experience and Demonstration: The Service Water Integrity Program, initially established for ANO-1 in 1980, is a comprehensive program to ensure operability of the service water system. The program involves chemistry controls, system and component testing, non-destructive examinations, and miscellaneous preventive maintenance activities that have proven effective in managing the effects of aging on plant components. These program activities consist of proven monitoring techniques, acceptance criteria, corrective actions, and administrative controls in existing site procedures. The continuation of the Service Water Integrity Program provides reasonable assurance that the aging effects will be managed such that the components of the service water system will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.20 STEAM GENERATOR INTEGRITY

Purpose: The purpose of the Steam Generator Integrity Program is to ensure the steam generator integrity is maintained under normal operating, transient, and postulated accident conditions. The Steam Generator Integrity Program is structured to meet the Nuclear Energy Institute *Steam Generator Program Guidelines* (NEI 97-06) [Reference B-28], which includes the following essential elements.

- Assessment of potential degradation mechanisms
- Tube inspection
- Tube structural and leakage assessment
- Maintenance and repairs
- Primary-to-secondary leakage monitoring
- Secondary side water chemistry
- Primary side water chemistry
- Foreign material exclusion
- Secondary side integrity
- Self-assessments
- NRC reporting

Scope: The scope of the Steam Generator Integrity Program applies to the ANO-1 steam generator internals, tubing, and associated repair techniques and components, such as plugs and sleeves.

Aging Effects: The aging effects addressed by the Steam Generator Integrity Program are loss of material, cracking, and fouling.

Method: Eddy current inspections are completed as required by technical specifications and the Steam Generator Integrity Program. The technical specifications require eddy-current testing, or other equivalent technique, that is capable of detecting defects with a penetration of 20% or more of the minimum allowable, as-manufactured tube wall thickness. To ensure adequate detection sensitivity, the techniques used during the eddy current inspections are qualified or demonstrated equivalent to the EPRI PWR Steam Generator Examination Guidelines. In many cases, this requires testing of the steam generator tubes with multiple techniques.

The secondary side internals of the steam generators are assessed each outage and inspections are periodically performed in accordance with procedural requirements. Leak testing may be performed to locate a primary to secondary leak, and tube pulls can be performed, when necessary, to further evaluate flaw morphology and assess structural and leakage integrity.

Industry Code or Standards: Industry codes and standards utilized for this program include the ASME Boiler and Pressure Vessel Code, Section XI.

Frequency: The frequency of inspections is based on the technical specifications. Additionally, the post outage tube integrity evaluation is performed to ensure the operating interval is justified.

Acceptance Criteria or Standard: Site procedures, technical specifications, and engineering evaluations provide the acceptance criteria. The detailed tube repair criterion is documented in an engineering evaluation prior to each eddy current inspection.

Regulatory Basis: ANO-1 Technical Specification 4.18.

Operating Experience and Demonstration: Detailed OTSG operating experience is routinely compiled by the BWOG and included in the plant-to-plant trending report. Additionally, the evaluations performed for the tube integrity include ANO-1 specific operating experience. Steam generator inspection and testing activities have proven effective in identifying indications of aging effects such as cracking and loss of material. Corrective actions within this program have been successful in correcting the identified deficiencies. Based on this information, the Steam Generator Integrity Program provides reasonable assurance that the aging effects will be managed such that the steam generators will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.21 SYSTEM AND COMPONENT MONITORING, INSPECTIONS, AND TESTING

A number of miscellaneous system and component monitoring, inspection, and testing activities are credited for managing the effects of aging. These existing activities are typically surveillance activities required by the technical specifications that are not considered part of a larger program or activity. In general, these activities are conducted on a periodic basis to verify the continuing capability of safety-related structures, systems, and components to meet established performance requirements. These credited activities are described in the following sections.

4.21.1 Annual Emergency Cooling Pond Sounding

Purpose: The annual emergency cooling pond sounding verifies the availability of a sufficient supply of cooling water to handle design basis accidents, with a concurrent loss of the Dardanelle Reservoir.

Scope: The emergency cooling pond and surrounding structural components.

Aging Effects: Loss of form of the emergency cooling pond due to sedimentation.

Method: Accessible and exposed surfaces are visually inspected along with sounding for pond level. Areas of the cooling pond are inspected for excessive erosion, degradation of rip rap, or silt build-up.

Industry Codes or Standards: Not applicable.

Frequency: The emergency cooling pond and its structural components are inspected annually.

Acceptance Criteria: Acceptance criteria established in applicable site procedures are based on technical specifications to ensure that sufficient inventory will be maintained in the emergency cooling pond.

Regulatory Basis: Emergency cooling pond inspections meet the surveillance requirements of ANO-1 Technical Specification 4.13.

Operating Experience and Demonstration: A review of in-house documentation showed that emergency cooling pond deficiencies have been identified during ECP inspections. The identified deficiencies include torn sandbags, out-of-place riprap, eroded banks and a broken drain. Applicable station procedures ensure that adequate corrective measures are taken when such deficiencies are identified during inspection.

Based on the above review, continued implementation of the Annual Emergency Cooling Pond Sounding provides reasonable assurance that the effects of aging will be managed such that the emergency cooling pond will continue to perform its intended functions consistent with the current licensing basis for the period of extended operation.

4.21.2 Battery Quarterly Surveillance

Purpose: The purpose of the battery rack inspections is to ensure their structural integrity.

Scope: Seismically-qualified battery racks are within the scope.

Aging Effects: Battery racks and associated threaded fasteners are inspected for physical damage or abnormal deterioration, including a loss of material.

Method: The battery racks are visually inspected as part of the battery quarterly surveillance.

Industry Code or Standard: ANSI/IEEE Standard 450-1980: *IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations* [Reference B-29].

Frequency: In association with other battery maintenance activities, battery rack inspections are performed on a quarterly basis.

Acceptance Criteria: The battery racks are considered acceptable if there are no visual indications of degradation. Deficiencies, when noted, are evaluated to determine if the damage or deterioration affects the ability of the battery banks to perform their function.

Regulatory Basis: Not applicable.

Operating Experience and Demonstration: A review of ANO-1 condition report summaries did not identify indications of damage to, or deterioration of, the battery racks. While battery rack inspections have identified no evidence of loss of material, visual inspections in general have proven effective in identifying degradation that would be indicative of this aging effect. The continuation of this activity will provide reasonable assurance that the effects of aging will be adequately managed so that the battery racks will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

4.21.3 Control Room Ventilation Testing

Purpose: With respect to license renewal, Control Room Ventilation Testing is credited as one of the programs to manage aging effects.

Scope: The control room ventilation testing applies to the control room emergency cooling coils.

Aging Effects: Fouling on the external surfaces of the cooling coil tubes is the aging effect managed by this program.

Method: Per ANO-1 Technical Specification 4.10, each train of the control room emergency air conditioning system is demonstrated operable at least once per 31 days. This testing provides evidence that excessive fouling is not present**.**

Industry Standard or Codes: Not applicable.

Frequency: Each train is tested once per 31 days. At least once per 18 months, the system flow rate is verified.

Acceptance Criteria: The acceptance criteria for testing are provided in ANO-1 Technical Specification 4.10.

Regulatory Basis: ANO-1 Technical Specification 4.10

Operating Experience and Demonstration: A review of ANO-1 condition report summaries did not identify fouling of the cooling coils. In conjunction with control room ventilation system inspection under the PM Program and the Heat Exchanger Monitoring Program, the continued implementation of Control Room Ventilation Testing will provide reasonable assurance that fouling of the cooling coils will be adequately managed so that the control room ventilation system will continue to perform its intended function consistent with the current licensing basis for the period of extended operation.

4.21.4 Core Flood Tank Monitoring

Purpose: With respect to license renewal, the core flood tank monitoring provides a method to manage the aging effect of loss of material due to boric acid corrosion.

Scope: The core flood tank monitoring applies to both core flood tanks at ANO-1.

Aging Effects: The loss of material due to boric acid corrosion on parts wetted by leaks from the core flood tanks may be detected through core flood tank monitoring.

Method: Core flood tank level and pressure are monitored per ANO-1 operating procedures using installed control room instrumentation. Alarms activate if pressure or level moves outside the acceptable range**.**

Industry Standard or Codes: Not applicable.

Frequency: Core flood tank level and pressure are monitored once per shift during plant operation. Alarms activate if pressure or level moves outside the acceptable range.

Acceptance Criteria: The acceptance criteria for level and pressure are provided in site procedures.

Regulatory Basis: ANO-1 Technical Specification 4.1.a.

Operating Experience and Demonstration: Core flood tank level monitoring has proven effective in identifying level changes resulting from small amounts of leakage. This same monitoring activity is expected to identify aging effects resulting in similar small amounts of leakage. Based on the general effectiveness of core flood tank level monitoring in identifying leakage, the continuation of this activity will provide reasonable assurance that the effects of aging will be adequately managed so that the core flood system will continue to perform its intended functions consistent with the current licensing basis for the period of extended operation.

4.21.5 Emergency Diesel Generator Testing and Inspections

Purpose: With respect to license renewal, Emergency Diesel Generator Testing and Inspections provide a means of detecting aging effects associated with the various emergency diesel generator subsystems.

Scope: The scope for testing and inspections includes the emergency diesel generator assembly and associated support components.

Aging Effects: Loss of material is an aging effect requiring management for the carbon steel components in the EDG starting air system. Loss of material is identified as an aging effect for the unpainted carbon steel internal surfaces and the outer portion of the intake that could be wetted by rain. Loss of material and fouling are considered aging effects for the EDG intake air aftercoolers. Loss of material from the piping and muffler internal surfaces and from external surfaces exposed to the weather is an aging effect for the EDG exhaust components. Loss of material and fouling are aging effects for the lube oil coolers. The cooling water carbon steel components are susceptible to a limited loss of material from corrosion and the stainless steel components have the aging effect of cracking. Loss of material and fouling are aging effects for the cooling water heat exchangers. Since the portions of the subsystems on the engine are exposed to high vibration, loss of bolted closure integrity was identified as an aging effect for the skid mounted and connected components.

Method: Per ANO-1 Technical Specification 4.6.1, each diesel generator is started each month and operated until the temperatures stabilize. Once every 18 months, the diesel generator is automatically started and operated for greater than one hour after the temperatures have stabilized. Also once every 18 months each diesel generator is given an inspection following the manufacturer's recommendations. The following are examples of the maintenance actions that support the management of aging effects.

- A pressure drop test is performed on the aftercoolers and the aftercoolers are cleaned if a high differential pressure is indicated. This would detect fouling of these heat exchangers.

- A check is made for exhaust leaks, cooling water leaks, or lube oil leaks while the engine is running. This would detect loss of bolted closure integrity, cracking or loss of material in these subsystems that had progressed to the point of allowing leakage.

- The air start components on the skid are disassembled and inspected. A leak check is performed on the tubing. This would detect significant loss of material or loss of integrity of the air start components.

- The exhaust manifold screen assembly is removed and inspected, which helps to detect loss of material at this location in the exhaust subsystem.

- The lube oil cooler is disassembled, inspected, and cleaned (interior and exterior of tubes) when necessary. This would detect loss of material or fouling.

- Throughout the inspection, the bolt torque is checked on a large number of components, thereby managing a loss of bolted closure integrity.

Industry Standard or Codes: Not applicable.

Frequency: Each emergency diesel generator is manually started monthly and automatically started every 18 months. Inspections are performed on an 18-month interval.

Acceptance Criteria: The acceptance criteria for testing and inspections are documented in emergency diesel generator operation and inspection procedures.

Regulatory Basis: ANO-1 Technical Specification 4.6.1 and NRC Regulatory Guide 1.108, *Periodic Testing of Emergency Diesel Generator Units Used As On-site Power Systems At Nuclear Power Plants.*

Operating Experience and Demonstration: Operational testing of the emergency diesel generators has proven effective in identifying loss of mechanical closure integrity of air, lube oil, and fuel oil systems components. Testing includes a 24-hour endurance run once per 18 months. Based on a review of industry experience and ANO condition reports regarding the emergency diesel generators, these tests and inspections provide reasonable assurance that the aging effects will be managed such that the emergency diesel generators will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.21.6 Emergency Feedwater Pump Testing

Purpose: With respect to license renewal, Emergency Feedwater Pump Testing is credited as one of the programs for managing the effects of aging.

Scope: The scope of Emergency Feedwater Pump Testing includes the turbine and electric motor driven emergency feedwater pumps and associated components.

Aging Effects: Fouling in the system heat exchangers is the primary aging effect that this testing will identify. This testing also is credited with identifying the aging effects of loss of material and loss of mechanical closure integrity for system components.

Method: Per ANO-1 Technical Specification 4.8, each train is demonstrated operable by verifying that each pump starts, and operates, through the test loop flow path.

Industry Standard or Codes: Not applicable.

Frequency: Each train is tested once per 31 days.

Acceptance Criteria: The acceptance criteria for testing are documented in ANO-1 Technical Specification 4.8.

Regulatory Basis: ANO-1 Technical Specification 4.8

Operating Experience and Demonstration: Emergency feedwater system testing challenges the pressure boundary and heat transfer functions for system components. A review of ANO-1 condition report summaries did not identify any occurrence of fouling of the system heat exchangers. The continued implementation of emergency feedwater pump testing provides reasonable assurance that fouling of the system heat exchangers will be detected and corrected so that the emergency feedwater system will continue to perform its intended function consistent with the current licensing basis for the period of extended operation.

4.21.7 NaOH Tank Level Monitoring

Purpose: The purpose of the NaOH tank level monitoring, with respect to license renewal, is to provide a method of detecting changes in the tank level that might indicate leakage from the NaOH tank or system.

Scope: The NaOH tank level monitoring applies to the NaOH system components.

Aging Effect: This inspection is credited with managing the aging effects of loss of material, loss of mechanical closure integrity, and cracking.

Method: The NaOH tank level is monitored through the use of a level alarm that actuates prior to exceeding the ANO-1 Technical Specification 3.3.4.B limits.

Industry Standard or Codes: Not applicable.

Frequency: The NaOH tank level is continuously monitored by the low level alarm feature.

Acceptance Criteria or Standards: Corrective action is initiated upon receipt of the NaOH tank low level alarm.

Regulatory Basis: ANO-1 Technical Specification 3.3.4.B

Operating Experience and Demonstration: A review of ANO-1 condition report summaries did not identify documentation of leakage from the chemical addition system boundary that could lead to the above identified aging effects. However, level monitoring in general has proven effective in leading to the identification and correction of small leaks. Continuation of NaOH tank level monitoring will provide reasonable assurance that the effects of aging will be adequately managed so that the chemical addition system will continue to perform its intended functions consistent with the current licensing basis for the period of extended operation.

4.21.8 Spent Fuel Pool Level Monitoring

Purpose: The purpose of spent fuel pool level monitoring, with respect to license renewal, is to provide a method of detecting changes in the spent fuel pool level that might indicate cracks in the spent fuel pool liner.

Scope: The spent fuel pool level monitoring applies to the detection of leakage through the spent fuel pool liner.

Aging Effect: Cracking of the spent fuel pool liner is the aging effect addressed by spent fuel pool level monitoring.

Method: Operators record the spent fuel pool level during their rounds. Alarms are provided to indicate decreasing spent fuel pool level. If the spent fuel pool level dropped without explanation, operations will determine the cause and initiate corrective action**.** Leakage due to a through wall crack of the spent fuel liner would be identified by a decrease in spent fuel pool level.

Industry Standard or Codes: Not applicable.

Frequency: Operations records the spent fuel pool level once per shift. Spent fuel pool level is continuously monitored by the low level alarm feature.

Acceptance Criteria: The acceptance criterion for the spent fuel pool level is provided in site procedures.

Regulatory Basis: Not applicable.

Operating Experience and Demonstration: As documented in NRC Inspection Report 50-313/95-11; 50-368/95-11, 0CNA039524 [Reference B-8], the refueling cavity liner plate was found to have cracking during refueling outage 11. Welding of the new permanent seal plate is believed to have caused the previously existing cracks to propagate. Based on this condition, cracking is considered an applicable aging effect for the spent fuel pool liner. The cracking of the refueling canal liner plate occurred in the heat affected zone of the liner plate welds. Spent Fuel Pool Level Monitoring is one of the means of managing cracking of the spent fuel pool liner plate. In conjunction with the Spent Fuel Pool Monitoring Program described in Section 3.7 of this appendix, Spent Fuel Pool Level Monitoring will provide reasonable assurance that cracking of the liner plate will be managed such that the spent fuel pool will continue to perform its intended functions consistent with the current licensing basis for the period of extended operation.

4.22 REFERENCES FOR APPENDIX B

B-1 Industry Guideline for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule, NEI 95-10, Revision 0, Nuclear Energy Institute, March 1996.

B-2 Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants, Working Draft, NRC, September 1997.

B-3 BAW-2244A, *Demonstration of the Management of Aging Effects for the Pressurizer*, The B&W Owners Group Generic License Renewal Program, December 1997.

B-4 BAW-2248A, Demonstration of the Management of Aging Effects for the Reactor Vessel Internals, The B&W Owners Group Generic License Renewal Program, July 1997.

B-5 BAW-2243A, Demonstration of the Management of Aging Effects for the Reactor Coolant system Piping. The B&W Owners Group Generic License Renewal Program, June 1996.

B-6 1CAN059902, ANO letter dated May 6, 1999 to NRC, Annual Radiological Environmental Operating Report – 1998.

B-7 1CAN069804, ANO letter dated June 3, 1998 to NRC, Risk Informed Inservice Inspection Pilot Plant Submittal

B-8 0CNA039524, NRC letter dated March 31, 1995 to ANO, *NRC Inspection Report 50-313/95-11; 50-368/95-11*

B-9 BAW-2301, Degradation of Control Rod Drive Mechanism and Other Vessel Closure Head Penetrations, The B&W Owners Group Response to Generic Letter 97-01, July 1997.

B-10 1CAN048908, ANO letter dated April 13, 1989 to NRC, Information Submittal for Station Blackout Rule, 10CFR50.63

B-11 0CAN049107, ANO letter dated April 15, 1991 to NRC, Response to Station Blackout Safety Evaluation Report (TAC Nos.68508 and 68509).

B-12 0CAN088201, John R. Marshall (AP&L) letter dated August 2, 1982 to John T. Collins (NRC), Initial AP&L Response to IEB 82-02 – Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants.

B-13 0CAN058813, Dan R. Howard (AP&L) letter dated May 27, 1988 to Document Control Desk (NRC), Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants (Generic Letter 88-05).

B-14 ANS 56.8, Containment System Leakage Testing Requirements, ANS, 1994.

B-15 ANSI N45.4, Leakage Rate Testing of Containment Structures for Nuclear Reactors, ANSI, 1972.

B-16 BAW-2251A, Demonstration of the Management of Aging Effects for the Reactor Vessel, The B&W Owners Group Generic License Renewal Program, June 1996.

B-17 0CAN079703, ANO letter dated July 29, 1997 to NRC, *120 Day Response to Generic Letter 97-01*.

B-18 0CAN029908, ANO letter dated February 24, 1999 to NRC, Generic Letter 97-01 Request for Additional Information.

B-19 0CAN049705, ANO letter dated April 15, 1997 to NRC, *Monthly Operating Report*.

B-20 ISA S7.3, Quality Standard for Instrument Air, ISA, 1975.

B-21 ASTM D95, Standard Test Method for Water in Petroleum Products and Bituminous Materials by Distillation, ASTM, June 1999.

B-22 ASTM D664, Standard Test Method for Neutralization Number by Potentiometric Titration, ASTM, October 1995.

B-23 BAW-1543A, *Integrated Reactor Vessel Surveillance Program,* B&W Owners Group Materials Committee, May 1985.

B-24 BAW-2251A, *Demonstration of the Management of Aging Effects for the Reactor Vessel,* The B&W Owners Group Generic License Renewal Program, June 1996.

B-25 ASTM E185, Standard Practice for Conducting Surveillance Test for Light-Water Cooled Nuclear Power Reactor Vessels.

B-26 BAW-2241A-P, Fluence and Uncertainty Methodologies, April 1997.

B-27 BAW-10046, Revision 1, *Methods of Compliance with 10CFR50, Appendix G*, The B&W Owners Group.

B-28 NEI 97-06, Steam Generator Program Guidelines.

B-29 IEEE 450, IEEE Recommended Practice for Maintenance, Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Substations, IEEE, 1980.

B-30 ASTM E900, Standard Guide for Predicting Neutron Radiation Damage to Reactor Vessel Materials, ASTM, 1987.

B-31 0CAN019012, ANO letter dated January 26, 1990 to NRC, *Response to Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment"*.

B-32 0CAN119010, ANO letter dated November 30, 1990 to NRC, *Revision to Response to Generic Letter 89-13.*

B-33 0CAN099110, ANO letter dated October 3, 1991 to NRC, Implementation Status of Four Issues at ANO.

B-34 1CAN039209, ANO letter dated March 20, 1992 to NRC, *Modified Component Testing List for Generic Letter 89-13.*

B-35 0CAN109205, ANO letter dated October 30, 1992 to NRC, *Revised Approach for Compliance to NRC Generic Letter 89-13; Service Water*.

Appendix C

Process for Identifying Aging Effects Requiring Aging Management for Non-Class 1 Mechanical Components

1.0 INTRODUCTION

Entergy Operations utilized generic guidance developed as part of the BWOG Generic License Renewal Program as the primary basis for determining applicable aging effects for the ANO-1 non-Class 1 mechanical components. This appendix does not contain the detailed derivation of the aging effects that are applicable for the materials and environment combinations, but summarizes the process and the results to assist in the understanding of the aging effects identified in the LRA.

The potential aging effects for all mechanical components include the following.

- Loss of material
- Cracking
- Change in material properties
- Distortion
- Loss of mechanical closure integrity

An additional potential aging effect for heat exchangers and similar components, whose intended function is heat transfer, is fouling (loss of heat transfer capability).

Operating environments for mechanical systems within the scope of license renewal are discussed in the following sections of this appendix.

- Borated water (Section 2.0)
- Treated water (Section 3.0)
- Raw water (Section 4.0)
- Sodium hydroxide (Section 5.0)
- Oil and fuel oil (Section 6.0)
- Gas (Section 7.0)
- External environments (Section 8.0)

The following two special topics are also within the scope of license renewal.

- Bolted closures (Section 9.0)
- Heat exchangers (Section 10.0)

1.1 MECHANICAL COMPONENTS SUBJECT TO AGING MANAGEMENT REVIEW

In accordance with Appendix B of NEI 95-10 and the guidance provided in the 10CFR Part 54, only passive mechanical components are in the scope of review. Within the systems that are within the scope of license renewal, the following are typical components subject to aging management review.

- Heat exchangers
- Tanks/vessels
- Pump casings
- Valve bodies and bonnets
- Pipe, tubing, fittings, and branch connections
- Bolting
- Miscellaneous process components
- Filter housings
- Flex hose
- Expansion joints
- Traps
- Flow orifices
- In-line flowmeters
- Cyclone separators

Many of the mechanical components within the scope of license renewal contain gaskets, packing, and seals. However, these items are not subject to aging management review since they are not long lived and are defined as consumables. In addition, the NRC, in the SER on BAW-2244A [Reference 1-1], agreed that an aging management review was not required since a gasket, as a part of the bolted connection, exists to minimize leakage and is not solely responsible for providing the pressure boundary or supporting a structural load.

1.2 Materials Used in Non-Class 1 Components

The following materials are present in the non-Class 1 systems within the scope of license renewal.

- Stainless steels (wrought and cast)
- Nickel-base alloys (inconel)
- Carbon and low alloy (chrome-moly) steels
- Cast iron
- Copper alloys (bronze, brass, Admiralty, and copper-nickel)
- Aluminum
- Copper
- Glass
- Incoloy-800

1.3 Environments and Special Topics

1.3.1 Borated Water

Borated water is demineralized water containing boric acid. Aging effects for materials typically found in borated water environments are summarized in Section 2.0 of this appendix.

1.3.2 Treated Water

Treated water is demineralized water and is the base water for all clean systems. Depending on the system, treated water may require additional processing. Treated water could be deaerated and include corrosion inhibitors, biocides, or some combination of these treatments. Aging effects for materials typically found in treated water environments are summarized in Section 3.0 of this appendix.

1.3.3 Raw Water

At ANO-1, raw water systems use water from Lake Dardanelle that has been filtered by traveling water screens. In addition, the floor drains and reactor building and auxiliary building sumps may be exposed to a variety of untreated water that is classified as raw water for the determination of aging effects. Aging effects for materials typically found in raw water environments are summarized in Section 4.0 of this appendix.

1.3.4 Sodium Hydroxide

The sodium hydroxide (chemical addition) system contains sodium hydroxide in demineralized water. Aging effects for materials typically found in a sodium hydroxide environment are summarized in Section 5.0 of this appendix.

1.3.5 Lubricating Oil and Fuel Oil

The lubricating oil and fuel oil environment is applicable to components holding or using oil lubricants or fuel oil. Lubricating oil is low to medium viscosity hydrocarbons used for bearing, gear, and engine lubricating. Fuel oil is defined as diesel oil, No. 2 oil or other liquid hydrocarbons used to fuel diesel engines. Aging effects for materials found in an oil or fuel oil environment are summarized in Section 6.0 of this appendix.

1.3.6 Gas

Section 7.0 of this appendix discusses the aging effects applicable to the materials exposed to internal environments consisting of various gases. Non-Class 1 mechanical components within the scope of license renewal are exposed to the following internal gas environments.

- Air – both at atmospheric pressure in ventilation systems and compressed air used as a working fluid, e.g., instrument air
- Nitrogen
- Carbon dioxide
- Freon
- Halon

Ventilation, halon, compressed air, and refrigeration systems contain an internal environment of gas as the process fluid. Other systems only have an internal environment of gas above the liquid level in storage tanks or in portions of the system that are normally open to the atmosphere.

1.3.7 External Surface Environments

Sections 2.0 through 7.0 of this appendix discuss the aging effects of various fluids on the internal surfaces of components that ordinarily contain those fluids. Component external surfaces may experience aging due to exposure to the environment. Applicable external environments include the ambient atmosphere (including airborne contaminants and moisture), leakage, and the underground environment. Aging effects for materials exposed to these environments are discussed in Section 8.0 of this appendix. Degradation of external surfaces of bolted closures is addressed in Section 9.0 of this appendix.

1.3.8 Bolted Closures

Bolting applications within the scope of license renewal may be divided into pressure boundary bolting and structural or component support bolting. Pressure boundary bolting applications, which are addressed in Section 9.0 of this appendix, include bolted flange connections for vessels (i.e., manways and inspection ports), flanged joints in piping, body-to-bonnet joints in valves, and pressure retaining bolting associated with pumps and miscellaneous process components. These bolted joints are hereafter referred to as bolted closures. A bolted closure includes the entire bolted joint, seating surfaces (e.g., flange set surfaces), gasket, and pressure retaining bolting. Aging mechanisms affecting bolted closure integrity for the pressure boundary intended function are discussed in Section 9.0 of this appendix.

1.3.9 Heat Exchangers

Section 10.0 of this appendix discusses the aging effects applicable to heat exchangers.

1.4 Potential Aging Effects

Potential aging effects are considered applicable if the effects could cause a component to lose function during the period of extended operation. The potential aging effects for non-Class 1 mechanical system components are as follows.

Loss of Material

Loss of material may be due to general corrosion, pitting corrosion, boric acid wastage, galvanic corrosion, crevice corrosion, erosion (including erosion caused by abrasive wear, erosive wear, cavitation wear, and droplet impingement wear), erosion/corrosion, microbiologically influenced corrosion, or selective leaching.

General corrosion is the result of a chemical or electrochemical reaction between the material and the environment when both oxygen and moisture are present. General corrosion is characterized by uniform attack resulting in material dissolution and sometimes corrosion product buildup. General corrosion on components exposed to air tends to form a protective oxide film on the component that prevents further significant corrosion. This is typically true for components not exposed to other sources of moisture such as rain, condensation, or frequent leakage.

Pitting corrosion is a form of localized attack that results in depressions in the metal. Oxygen is required for initiation of pitting corrosion with contaminants such as halogens or sulfates required for continued material dissolution. Pitting corrosion is more common with passive materials such as austenitic stainless steels than with non-passive materials. Most materials of interest are susceptible to pitting corrosion under certain conditions. Most pitting is associated with the presence of halide ions, chlorides, bromides, and hypochlorites.

Loss of material due to boric acid wastage is an applicable aging effect for the external surfaces of carbon steel and low-alloy or chrome-molly component materials exposed to the leakage of borated water. Leaking fluid from a borated water system may expose the external surfaces of components made from these materials to a concentrated boric acid solution that can cause loss of material.

Loss of material due to galvanic corrosion can occur only when materials with different electrochemical potentials are in contact in the presence of oxygenated water or a corrosive environment. Generally, the effects of galvanic corrosion are precluded by design (e.g., isolation to prevent electrolytic connection or using similar materials). In galvanic couples involving admiralty, brass, carbon steel, cast iron, copper, low-alloy steel and stainless steel materials, the lower potential (more anodic) carbon steel, cast iron and low-alloy steel materials would be preferentially attacked.

Crevice corrosion occurs when a crevice exists in a component that allows a corrosive environment to develop within the crevice. It occurs most frequently in joints and connections, or points of contact between metals and nonmetals, such as gasket surfaces, lap joints, and under bolt heads. Crevice corrosion is strongly dependent on the presence of dissolved oxygen. Oxygen is required for crevice corrosion initiation; however, once initiated, the corrosion process does not require oxygen to continue. For environments with extremely low oxygen content (<0.1 ppm), crevice corrosion is considered insignificant.

Erosion-corrosion is a term used to describe the alternating pattern of oxide erosion due to fluid flow followed by corrosion of the newly exposed material surface that is again followed by oxide erosion as the pattern repeats. Physical parameters such as fluid temperature, fluid (steam) quality, fluid velocity, fluid pH, and mechanical component configuration affect the degree of erosion-corrosion.

Microbiologically influenced corrosion is a localized, corrosive attack accelerated by the influence of microbiological activity. Microbiologically influenced corrosion usually occurs at temperatures between 50°F and 120°F. Microbiological organisms can produce corrosive substances, as a byproduct of their biological processes, that disrupt the protective oxide layer on the component materials, leading to a material depression similar to pitting corrosion.

Selective leaching is the dissolution of one element from a solid alloy by corrosion processes.

<u>Cracking</u>

Cracking is service-induced cracking (initiation and growth) of base metal or weld metal due to hydrogen damage, stress corrosion, intergranular attack, or vibration. The analysis of the potential for cracking due to low cycle, thermal fatigue is a time-limited aging analysis and is addressed in the ANO-1 LRA.

Hydrogen damage to carbon steel results from the absorption of hydrogen into the metal. This effect is prevalent in carbon steel only for very high yield strengths that are not utilized in the non-Class 1 components at ANO-1.

Stress corrosion cracking and intergranular attack require a combination of a susceptible material, a corrosive environment, and tensile stress. Since the level of tensile stress required for stress corrosion cracking in a component is unknown, the stresses are conservatively assumed to be sufficient to initiate stress corrosion cracking and intergranular attack if the other conditions are met. Intergranular attack is similar to stress corrosion cracking, except that stress is not necessary for it to proceed. In the case of stress corrosion cracking of carbon and low-alloy steels, the literature shows the mechanism is possible citing stress corrosion cracking in aqueous chlorides as the most common form. However, in the discussion of prevention and control, one of the most reliable methods of preventing stress corrosion cracking of carbon and low-alloy steels is to select a material with a yield strength of less than 100 ksi. The yield strength of carbon steels typically used in non-Class 1 systems is on the order of 30 to 45 ksi. Industry data does not indicate a significant problem of stress corrosion cracking in low strength carbon steels. For these reasons, stress corrosion cracking of carbon and low-alloy steels is considered not applicable.

For stainless steels exposed to atmospheric conditions, stress corrosion cracking is considered plausible when exposed to high levels of contaminants (e.g., saltwater environment) and only if the material is in a sensitized condition. For ANO-1 non-Class 1 mechanical systems, this applies only to components whose exterior surfaces may be exposed to sodium hydroxide.

Stainless steel components at ANO-1 that are exposed to high levels of chlorides, fluorides, or sulfates have been reviewed for susceptibility to cracking from stress corrosion cracking. Cracking has been identified as an applicable aging effect at ANO-1 for stainless steel components that are exposed to high levels of chlorides, fluorides, or sulfates. ANO has not experienced cracking of the low carbon stainless steel components exposed to water from Lake Dardanelle. Low carbon stainless steel in an environment of low temperature water has a reduced susceptibility to cracking. This is consistent with NPRDS industry failure data that indicates stress corrosion cracking of stainless steels in a fresh water lake water environment is unlikely.

Change in Material Properties

Change in material properties is a reduction in fracture toughness due to hydrogen embrittlement, radiation embrittlement, or thermal aging. Change in material properties was considered in all mechanical system components falling within the scope of license renewal. For non-Class 1 mechanical system components, change in material properties is not an applicable aging effect.

Distortion

Distortion is a physical property change in a component caused by plastic deformation due to the temperature-related phenomenon of creep. Materials within non-Class 1 components are not exposed to the required high temperatures necessary for this mechanism to occur. Therefore, distortion is not an applicable aging effect for the non-Class 1 mechanical system components at ANO-1.

Loss of Mechanical Closure Integrity

The loss of mechanical closure integrity is an aging effect resulting in failure of a mechanical closure to provide a required pressure boundary. Loss of mechanical closure integrity may be attributed to one or more of the following conditions affecting bolting material.

- Loss of pre-load
- Cracking of bolting material
- Loss of bolting material
- Reduction of fracture toughness of bolting material (only applicable to reactor coolant system components)

Fouling

Fouling may be due to macro-organisms, precipitation or silting. Fouling is not a material degradation phenomenon, but is an aging effect that could cause loss of the heat transfer intended function, or a reduction in flow rate, for components in ANO-1 systems.

1.5 SECTION 1.0 REFERENCES

1-1 BAW-2244A, "*Demonstration of the Management of Aging Effects for the Pressurizer*," The B&W Owners Group Generic License Renewal Program, December 1997

1-2 "*Aging Management Guideline for Commercial Nuclear Power Plants-Heat Exchangers*," SAND93-7070, prepared by MDC-Ogden Environmental and Energy Services under contract to Sandia National Laboratories for the U.S. Department of Energy, June 1994.

2.0 EFFECTS REQUIRING AGING MANAGEMENT IN BORATED WATER ENVIRONMENTS

2.1 ATTRIBUTES OF BORATED WATER ENVIRONMENTS

Borated water is demineralized water with varying concentrations of boric acid.

2.2 MATERIALS USED IN BORATED WATER ENVIRONMENTS

The majority of the components within the scope of license renewal exposed to internal borated water are constructed of stainless steel. Other components are constructed of carbon steel but lined with stainless steel or Plastite to protect the carbon steel from direct contact with the borated water. Materials in direct contact with borated water include stainless steels and inconel.

2.3 AGING EFFECTS IN BORATED WATER ENVIRONMENTS

2.3.1 Loss of Material

Loss of material due to pitting corrosion is an applicable aging effect for inconel and stainless steel in borated water under certain conditions. For a borated water environment, two sets of conditions can lead to pitting corrosion. The first set of conditions needed for pitting corrosion to occur is the presence of halogens in excess of 150ppb, oxygen in excess of 100ppb and stagnant or low flow conditions. A second set of conditions leading to pitting corrosion is the presence of sulfates in excess of 100ppb, oxygen in excess of 100ppb and stagnant or low flow conditions. If either set of conditions is satisfied, loss of material due to pitting corrosion is an applicable aging effect for inconel and stainless steel materials in borated water.

2.3.2 Cracking

Cracking due to stress corrosion and intergranular attack of inconel and stainless steel materials in a borated water environment is an applicable aging effect under certain conditions. For inconel and stainless steel, the relevant conditions required for stress corrosion cracking are the presence of halogens in excess of 150ppb or sulfates in excess of 150ppb. In addition, stress corrosion cracking has been observed in high-purity water (i.e., sulfates and halogens less than 150ppb) at temperatures greater that 200°F with dissolved oxygen levels greater than 100ppb.

2.4 INDUSTRY EXPERIENCE IN BORATED WATER ENVIRONMENTS

In order to validate the applicable aging effects for components exposed to borated water, industry experience was reviewed. The review included an NPRDS search on relevant topics and NRC generic communications and NUREG documents. No unique aging effects were identified in these documents beyond those described in this section.

3.0 EFFECTS REQUIRING AGING MANAGEMENT IN TREATED WATER ENVIRONMENTS

3.1 ATTRIBUTES OF TREATED WATER ENVIRONMENTS

Treated water is demineralized water and is the base water for all clean systems. Depending on the system, treated water may require additional processing. Treated water could be deaerated, include corrosion inhibitors, biocides, or some combination of these treatments. In the determination of aging effects, steam is considered treated water.

Water chemistry of the main feedwater system is closely monitored to minimize the potential for degradation of the once-through steam generators. The pH of the feedwater is maintained by the addition of amines to reduce the potential for flow-assisted corrosion by reducing iron transport. Deaeration and the addition of hydrazine control dissolved oxygen. Impurities such as chlorides and sulfates are controlled to reduce the stress corrosion cracking of OTSG tubes.

Chemistry requirements for the demineralized or makeup water are stringent since makeup water is used for reactor coolant, secondary, and other auxiliary systems in which high quality water is required.

Emergency feedwater systems and condensate systems have strict limit on contaminants, but the safety related condensate storage tank at ANO-1 is vented to atmosphere so these systems contain water saturated with oxygen.

The chemistry in the auxiliary systems is treated water with corrosion inhibitors.

3.2 MATERIALS USED IN TREATED WATER ENVIRONMENTS

The following materials are exposed to an internal treated water environment.

- Stainless steels
- Carbon and low alloy (chrome-moly) steels
- Cast iron
- 90/10 Cu-Ni
- Copper
- Brass
- Bronze
- Admiralty
- Glass

3.3 AGING EFFECTS IN TREATED WATER ENVIRONMENTS

3.3.1 Loss of Material

Loss of material due to general corrosion is an applicable aging effect for admiralty, brass, carbon steel, cast iron, copper, and low-alloy steel in ANO-1 treated water environments due to the presence of oxygen. Stainless steel in treated water environments is resistant to general corrosion.

Loss of material due to pitting corrosion is an applicable aging effect for admiralty, brass, carbon steel, cast iron, copper, low-alloy steel, and stainless steel materials in a treated water environment under certain conditions. For a treated water environment, two sets of conditions can lead to pitting corrosion. The first set of conditions is the presence of halogens in excess of 150ppb, oxygen in excess of 100ppb and stagnant or low flow conditions. A second set of conditions is the presence of sulfates in excess of 150ppb, oxygen in excess of 100ppb and stagnant or low flow conditions.

Loss of material due to galvanic corrosion can occur only when materials with different electrochemical potentials are in contact in the presence of oxygenated water.

Loss of material due to erosion-corrosion is an applicable aging effect for carbon steel in treated water under certain conditions. Fluid conditions in the main steam and main feedwater systems can lead to erosion-corrosion.

3.3.2 Cracking

Cracking due to stress corrosion of stainless steel materials in treated water is an applicable aging effect under certain conditions. For stress corrosion cracking to occur in stainless steel, the concentration of halogens or sulfates must exceed 150ppb. In addition, stress corrosion cracking of stainless steel has been observed in high-purity water (i.e., sulfates and halogens less than 150ppb) at temperatures greater that 200°F with dissolved oxygen levels greater than 100ppb. If these relevant conditions are satisfied, stress corrosion cracking is an applicable aging effect for stainless steel in a treated water environment.

3.4 INDUSTRY EXPERIENCE WITH TREATED WATER ENVIRONMENTS

To validate the applicable aging effects for mechanical components exposed to a treated water internal operating environment, industry experience was reviewed. The survey of industry experience included an NPRDS search on relevant topics and a review of NRC generic communications and NUREG documents. No unique aging effects were identified in these documents beyond those discussed in this section.

4.0 EFFECTS REQUIRING AGING MANAGEMENT IN RAW WATER ENVIRONMENTS

4.1 ATTRIBUTES OF RAW WATER ENVIRONMENTS

The majority of raw water for ANO-1 is water from Lake Dardanelle. In general, the water has been rough-filtered to remove large particles and may contain a biocide additive for control of microorganisms, zebra mussels, and Asiatic clams. Lake Dardanelle is considered fresh water, i.e., it has a sodium chloride content below 1000 mg/l. In addition, the floor drains and reactor building and auxiliary building sumps may be exposed to a variety of untreated water that is classified as raw water for the determination of aging effects.

4.2 MATERIALS USED IN RAW WATER ENVIRONMENTS

Materials within the scope of license renewal at ANO-1 that are exposed to raw water include the following.

- Stainless steel
- Carbon and low alloy steel
- Cast iron
- Brass
- Admiralty
- Copper
- 90-10 copper nickel
- Bronze

4.3 AGING EFFECTS IN RAW WATER ENVIRONMENTS

4.3.1 Loss of Material

Loss of material due to general corrosion is an applicable aging effect for admiralty, brass, bronze, carbon steel, low alloy steel, cast iron, copper, and 90-10 copper-nickel component materials in a raw water environment. The stainless steel materials in the plant raw water environments are resistant to general corrosion.

Loss of material due to pitting corrosion is an applicable aging effect for admiralty, brass, bronze, carbon steel, low alloy steel, cast iron, copper, 90-10 copper-nickel, cast iron, and stainless steel materials in a raw water environment. Maintaining an adequate flow rate, which prevents impurities from adhering to the material surface, can inhibit pitting corrosion. The more susceptible locations for pitting corrosion in materials in a raw water environment are locations of low or stagnant flow.

Loss of material due to galvanic corrosion in a raw water environment can occur when materials with different electrochemical potentials are in contact.

Microbiological organisms present in raw water can produce corrosive substances, as a byproduct of their biological processes, that disrupt the protective oxide layer on the component materials, leading to a material depression similar to pitting corrosion. Loss of material due to microbiologically influenced corrosion is an applicable aging effect for admiralty, brass, bronze, carbon steel, cast iron, copper, 90-10 copper-nickel, and stainless steel materials exposed to raw water.

Loss of material due to selective leaching is an applicable aging effect for cast iron component materials in a raw water environment.

4.3.2 Cracking

Cracking due to stress corrosion cracking or intergranular attack is a potential concern for stainless steels and copper-based alloys in a raw water environment due to possible chemical concentrations. ANO-1 has not experienced cracking of stainless steel components exposed to service water because of the low temperature of the water and the use of low carbon stainless steel. This is consistent with NPRDS industry failure data that indicates stress corrosion cracking of stainless steels in a fresh water lake environment is not likely. Cracking has been conservatively identified as an applicable aging effect for components exposed to fluid in floor drains or sumps due to the potential for very high levels of contaminants in this environment.

4.3.3 Fouling

For a raw water system, fouling is an applicable aging effect. Fouling can be categorized by particulate fouling (sediment, silt, dust, corrosion products, etc.), marine biofouling (clamshells, mussels, etc.) or macro fouling (peeled coatings, debris, etc.). Fouling in a raw water system can occur on the piping, valves, and heat exchangers. Fouling can result in a reduction of the function of heat transfer and a reduction in the system flow rate.

4.4 Industry Experience with Raw Water Environments

To validate the applicable aging effects for mechanical components exposed to a raw water internal operating environment, industry experience was reviewed. The survey of industry experience included an NPRDS search on relevant topics and a review of NRC generic communications and NUREG documents. No unique aging effects were identified in these documents beyond those discussed in this section.

5.0 EFFECTS REQUIRING AGING MANAGEMENT IN SODIUM HYDROXIDE ENVIRONMENTS

5.1 Attributes of Sodium Hydroxide Environments

Sodium hydroxide solutions in demineralized water are used in the sodium hydroxide (chemical addition) system to mix with the reactor building spray during loss of coolant accidents. The sodium hydroxide system is supplied only from treated water that is free from microorganisms. Microbiologically influenced corrosion has not occurred in this system. The sodium hydroxide in the system is a caustic soda that is maintained at a concentration of 18 +2.8/-3.0 wt.% as required by the ANO-1 Technical Specifications. High levels of chlorides exist in sodium hydroxide solutions due to the production methods for the sodium hydroxide. Since the tank is vented to the atmosphere, oxygen levels of the tank contents are expected to be near saturation.

5.2 Materials used in Sodium Hydroxide Environments

The materials exposed to sodium hydroxide solution within the scope of license renewal include stainless steels and carbon and low alloy steels.

5.3 Aging Effects in Sodium Hydroxide Environments

5.3.1 Loss of Material

Carbon and low alloy steel are susceptible to general corrosion in a sodium hydroxide solution. Carbon and low alloy steels exposed to sodium hydroxide will form a protective film and further corrosion of the carbon steel is greatly affected by the liquid agitation and the temperature. Pitting and crevice corrosion are applicable aging mechanisms for carbon and low alloy steel due to the low flow and high chlorides. Galvanic corrosion is an applicable mechanism for carbon and low alloy steel at the interfaces with the stainless steel piping.

The corrosion rate of stainless steel in sodium hydroxide solutions at low temperatures is expected to be less than 0.1 mil per year, which will not cause significant loss of material even in the period of extended operation. Pitting and crevice corrosion are applicable aging mechanisms due to the stagnant conditions and high levels of chlorides.

5.3.2 Cracking

Stress corrosion cracking and intergranular separation are applicable aging mechanisms for the stainless steel materials since the water contains high levels of chlorides.

5.4 Industry Experience with Sodium Hydroxide Environments

To validate the applicable aging effects for mechanical components exposed to sodium hydroxide, industry experience was reviewed. The review included an NPRDS search on relevant topics and review of NRC generic communications and NUREG documents. No unique aging effects were identified in these documents beyond those discussed in this section.

6.0 EFFECTS REQUIRING AGING MANAGEMENT IN LUBRICATING OIL AND FUEL OIL ENVIRONMENTS

6.1 ATTRIBUTES OF LUBRICATING OIL AND FUEL OIL ENVIRONMENTS

Separate evaluations are completed for lubricating oil and fuel oil.

6.1.1 Fuel Oils

At ANO-1, the fuel oil within the scope of licensing renewal is primarily No. 2 diesel oil. Diesel fuel oil is delivered to ANO-1 in tanker trucks and is stored in large tanks to provide an on-site supply of diesel fuel for a specified period of diesel generator operating time. This fuel oil is supplied to the diesel engines through pumps, valves, and piping. Strainers, filters and other equipment assure that the diesel fuel supplied to the engines is clean and free of contaminants.

6.1.2 Lubricating Oils

Lubricating oils within the scope of license renewal are low to medium viscosity hydrocarbons used for bearing, gear, and engine lubricating.

6.2 MATERIALS USED IN LUBRICATING OIL AND FUEL OIL ENVIRONMENTS

The non-Class 1 mechanical components within the scope of license renewal exposed to lubricating oil and fuel oils contain the following materials.

- Stainless steels
- Carbon and low alloy steels
- Cast iron
- Brass
- Bronze
- Aluminum
- Admiralty
- Glass
- Copper

6.3 AGING EFFECTS IN THE LUBRICATING OIL AND FUEL OIL ENVIRONMENTS

6.3.1 Fuel Oil Environment

Loss of material due to general corrosion is an applicable aging effect for carbon and low alloy steel in a fuel oil environment at locations containing water. The stainless steel, brass, admiralty, cast iron, glass, aluminum, bronze, and copper are inherently resistant to general corrosion in the plant fuel oil environments.

Loss of material due to pitting corrosion is an applicable aging effect for brass, bronze, carbon steel, low alloy steel, copper, and stainless steel materials in a fuel oil environment at location containing oxygenated water and contaminants such as halide ions, particularly chloride ions.

Loss of material due to crevice corrosion is an applicable aging effect for brass, bronze, carbon steel, copper, and stainless steel materials in a fuel oil environment at locations containing oxygenated water. Oxygen is required for the initiation of crevice corrosion. Fuel oil that is not contaminated does not contain oxygen in sufficient quantities for crevice corrosion to occur. Water contamination of the fuel oil is required for the introduction of oxygen.

Loss of material due to galvanic corrosion in a fuel oil environment can occur only when materials with different electrochemical potentials are in contact in the presence of water.

Loss of material due to microbiologically influenced corrosion is an applicable aging effect for brass, carbon steel, copper, and stainless steel materials exposed to fuel oil if microorganisms are present.

Cracking due to stress corrosion of the stainless steel material in a fuel oil environment is an applicable aging effect at locations containing oxygenated water.

Due to the high quality of fuel oil received at ANO-1, the system configuration, and sampling performed, little water is expected in the fuel oil systems. Any significant amount of water contamination would accumulate at the bottoms of the tanks, due to the higher density of water relative to fuel oil and the relatively low flow velocities in large tanks. Due to the addition of biocides, microbiologically influenced corrosion is not a concern for the ANO-1 fuel oil.

6.3.2 Lubricating Oil Environment

Loss of material due to general corrosion is an applicable aging effect for carbon and low alloy steel in a lubricating oil environment at locations containing water. The stainless steel, brass, admiralty, cast iron, glass, aluminum, bronze, and copper in the plant lubricating oil environments are inherently resistant to general corrosion.

Loss of material due to pitting corrosion is an applicable aging effect for brass, bronze, carbon steel, low alloy steel, copper, and stainless steel materials in a lubricating oil environment at location containing oxygenated water with contaminants such as halide ions, particularly chloride ions.

Loss of material due to crevice corrosion is an applicable aging effect for brass, bronze, carbon steel, copper, and stainless steel materials in an oil environment at locations containing oxygenated water. Oxygen is required for the initiation of crevice corrosion. Lube oil does not contain oxygen in sufficient quantities for crevice corrosion to occur. Water contamination of the lubricating oil is required for the introduction of oxygen.

Loss of material due to galvanic corrosion in a lubricating oil environment can occur only when materials with different electrochemical potentials are in contact in the presence of water.

Loss of material due to microbiologically influenced corrosion is an applicable aging effect for brass, carbon steel, copper, and stainless steel materials exposed to lubricating oil.

Cracking due to stress corrosion of the stainless steel material in a lubricating oil environment is an applicable aging effect at locations containing oxygenated water.

Due to the high quality of lubricating oil received at ANO-1 and the periodic sampling performed, water or contaminants are not expected in the lubricating oil systems. Microbiologically influenced corrosion has not been a concern for the ANO-1 lubricating oil for the systems in the scope of license renewal.

6.4 Industry Experience with Lubricating Oil and Fuel Oil Environments

To validate the applicable aging effects for mechanical components exposed to lubricating oil or fuel oil internal operating environment, industry experience was reviewed. The survey of industry experience included an NPRDS search on relevant topics and a review of NRC generic communications and NUREG documents. No unique aging effects were identified in these documents beyond those discussed in this section.

7.0 EFFECTS REQUIRING AGING MANAGEMENT IN GAS ENVIRONMENTS

The gas environments within the scope of license renewal at ANO-1 include atmospheric air (filtered and unfiltered), instrument air (clean and dry), and compressed gases (nitrogen, carbon dioxide, freon, and halon). A steam environment is considered a treated water environment as discussed in Section 3.0 of this appendix.

7.1 ATTRIBUTES OF GAS ENVIRONMENTS

This discussion includes a majority of the gaseous internal environments to which components within the scope of license renewal may be subjected. Numerous components may be subjected to different gaseous environments depending on plant and system operating conditions. The various gaseous environments covered by this discussion are described below.

7.1.1 Air

Air is composed of mostly nitrogen and oxygen with smaller fractions of various other constituents. The internal surfaces of a majority of components are at some time exposed to air. External surface contact with air is described in Section 8.0 of this appendix. Where air is the intended internal fluid (e.g., compressed air and instrument air systems), it is supplied in either its natural state or in a "dry" condition. The ANO instrument air system supplies air free of water or contaminants.

7.1.2 Nitrogen

Nitrogen is an inert gas used in many nuclear plant applications to place components in a dry lay-up condition or to provide a cover gas to prevent exposure to oxygen. The commercial grade nitrogen provided to ANO-1 is a high quality product with little, if any, contaminants.

7.1.3 Carbon Dioxide

Carbon dioxide is a colorless, odorless incombustible gas. The carbon dioxide systems of interest at ANO-1 contain dry carbon dioxide in gaseous form. Without the presence of moisture, this gaseous carbon dioxide is not a significant contributor to corrosion or other aging effects.

7.1.4 Freon

Fluorocarbons constitute a large family of fluorinated hydrocarbon compounds that exhibit similar chemical properties and a wide range of physical characteristics. The fluorocarbons used at ANO-1 are inert, nonflammable, colorless and relatively nontoxic. Their inert character and the range of their vapor pressures, boiling points and other physical properties makes them especially well suited for use as the working fluid in refrigeration and air conditioning systems.

7.1.5 Halon

Halon 1301 (bromotrifluoromethane-CF_3Br) is a halogenated extinguishing agent used in the ANO-1 main control room fire system for its ability to chemically react with fire and smother flames. The high purity Halon supplied to ANO-1 is essentially a non-corrosive gas. In use, it is combined with nitrogen gas (used as a propellant) in the fire suppression system.

7.2 Materials Used in Gaseous Environments

The materials exposed to gases within the scope of license renewal include the following.

- Stainless steels
- Carbon and low alloy steels
- Cast iron
- Brass
- Bronze
- Aluminum
- 90-10 copper nickel
- Copper
- Admiralty

7.3 Aging Effects in Gaseous Environments

For the most part, gases provide an environment for aging effects only in the presence of moisture or other contaminants.

7.3.1 Aging Effects in Air Environments

General Corrosion

At ordinary temperatures, oxygen and moisture are the basic factors for the corrosion of iron. Both oxygen and moisture must be present because oxygen alone or water free of dissolved oxygen does not corrode iron to any significant extent. Carbon and low-alloy steels, as well as cast iron are susceptible to general corrosion. Stainless steels, nickel-based alloys, aluminum, copper alloys and galvanized steel are inherently resistant to general corrosion.

General corrosion is an electrolytic reaction and, regardless of the particular gas environment, depends on the presence of oxygen and moisture. Corrosion in a nonaqueous environment only occurs by direct chemical reaction and only at high temperatures well above those encountered at ANO-1. Nitrogen and halon environments should have negligible amounts of free oxygen. Therefore, corrosion of carbon steel and cast iron components in these environments should not be a concern. The air environments within plant systems and components can vary from clean, dry air to moist, contaminated air whose purity is dictated by the source of the air. Portions of compressed air systems contain air that has been processed through dryers, which provide dry, oil free air to the downstream portions of the system. Moisture should not be a concern for these portions of systems and general corrosion is not expected.

Galvanic Corrosion

The severity of galvanic corrosion depends largely on the type and amount of moisture present. Galvanic corrosion does not occur when the metals are completely dry since there is no electrolyte to carry the current between the two electrode areas. Any gas and moisture interface that contains dissimilar materials with significant potential differences may be susceptible to galvanic corrosion. Air systems can be susceptible to galvanic corrosion due to the different materials used and the potential for moisture in crevices and other low points of systems. Aluminum to brass connections, as well as steel to copper connections, are susceptible to galvanic corrosion.

Crevice Corrosion

With any gas environment other than air, the oxygen content may be low enough to preclude crevice corrosion concerns. Crevice corrosion is a concern where moisture may pool in the presence of contaminants such as halides or sulfates.

Pitting Corrosion

All ANO-1 materials of interest are susceptible to pitting corrosion under certain conditions. Most pitting is associated with the presence of halide ions, chlorides, bromides, or hypochlorites.

Microbiologically Influenced Corrosion

Microbiological organisms that could induce corrosion are generally not found in a gaseous environment. Microbiologically influenced corrosion, therefore, is only a potential problem where contamination from untreated water or soil may have introduced bacteria. Air and gas systems are only affected where stagnant conditions and the pooling of an untreated aqueous solution provide an environment suitable for propagation of the mechanism.

7.3.2 Aging Effects in Nitrogen Environments

Carbon steel, cast iron and stainless steel in a nitrogen environment have no aging effects since the nitrogen has negligible amounts of free oxygen.

7.3.3 Aging Effects in Carbon Dioxide Environments

Carbon dioxide environments should have negligible amounts of free oxygen; therefore, corrosion of carbon steel and cast iron components in these environments should not be a concern.

7.3.4 Aging Effects in Freon Environments

Fluorocarbons show no appreciable decomposition at temperatures up to 400°F and oxidize only at very high temperatures. Fluorocarbons are non-corrosive to common metals except at very high temperatures.

The ANO-1 non-Class 1 refrigerant systems are typically pressurized closed loop systems containing freon mixed with an oil lubricant. These systems contain in line refrigerant dryers for enhancement of both performance and corrosion prevention. Unless contamination of the closed system with moisture or sulfur occurs, the conditions necessary for internal pressure boundary degradation due to corrosion do not exist.

7.3.5 Aging Effects in Halon Environments

The Halon environment is non-corrosive for the materials in the Halon system.

7.4 Industry Experience with Gaseous Environments

To validate the applicable aging effects for mechanical components exposed to an internal gaseous operating environment, industry experience was reviewed. The survey of industry experience included an NPRDS search on relevant topics and a review of NRC generic communications and NUREG documents. No unique aging effects were identified from these documents beyond those discussed in this section.

8.0 EFFECTS REQUIRING AGING MANAGEMENT IN EXTERNAL SURFACE ENVIRONMENTS

The purpose of this section is to identify aging effects applicable for external surfaces of mechanical components at ANO-1. Sections 2.0 through 7.0 of this appendix focused on specific material and internal environment combinations and the associated aging effects. Degradation of external surfaces of bolted closures is addressed in Section 9.0 of this appendix. External environments include the ambient atmosphere, leakage, and the underground environment.

8.1 ATTRIBUTES OF EXTERNAL SURFACE ENVIRONMENTS

8.1.1 External Ambient Environment

The external ambient environment consists of atmospheric conditions, which may include humidity, condensation, and airborne contaminants such as sulfur dioxide, chlorine gas, sulfur gas, and ozone.

8.1.2 Leakage Environment

The leakage environment is created when fluids escape from their system boundaries (usually from bolted closures) and contact the external surfaces of components. The fluid in leakage environments of concern is typically borated, treated, or raw water.

8.1.3 Underground Environment

The underground environment applies to components buried in the soil and exposed to the soil and groundwater. The soil and groundwater are untreated and could be corrosive to materials. The factors affecting corrosiveness of soils are moisture, pH (alkalinity or acidity), permeability of water and air (compactness or texture), oxygen, salts, stray currents, and biological organisms. Most of these affect electrical resistance of the soil, which is a good measure of corrosivity. High-resistance dry soils are generally not very corrosive. Carbon steel and cast iron with organic coatings are most common for underground components. Buried components are assumed susceptible to corrosion due to the potential for exposure to oxygen, moisture, biological organisms, and contaminants.

8.2 MATERIALS EXPOSED TO EXTERNAL SURFACE ENVIRONMENTS

The materials exposed to external environments include the following.

- Stainless steels
- Carbon and low alloy (chrome-moly)steel
- Cast iron
- Brass
- Bronze
- 90/10 Cu-Ni
- Incoloy-800
- Aluminum
- Copper
- Admiralty
- Glass

8.3 AGING EFFECTS OF THE EXTERNAL AMBIENT ENVIRONMENT

8.3.1 Loss of Material

Loss of material due to general corrosion is an applicable aging effect for carbon steel and low-alloy steel, admiralty, brass, cast iron, and copper materials in ambient air environments if the material is in contact with moist air. Paint or protective coatings applied to external surfaces will prevent this aging effect.

Loss of material due to galvanic corrosion can occur when materials with different electrochemical potentials are in contact in the presence of water, which is needed to establish the galvanic couple. Systems continually operating at a temperature at which surface condensation occurs in the ambient air environment will have water present on their external surfaces.

8.3.2 Cracking

<u>Stress Corrosion Cracking/Intergranular Attack</u>

One of the most reliable methods of preventing stress corrosion cracking of carbon and low-alloy steels is to select a material with yield strength of less than 100 ksi. The yield strength of carbon steels typically used in non-Class 1 systems is on the order of 30 to 45 ksi. Industry data does not indicate a problem of stress corrosion cracking in low strength carbon steels. For these reasons, stress corrosion cracking of carbon and low-alloy steels is considered not applicable.

For stainless steels exposed to atmospheric conditions, stress corrosion cracking is only plausible when exposed to high levels of contaminants (e.g., saltwater environment) and then only if the material is in a sensitized condition.

8.4 AGING EFFECTS IN LEAKAGE ENVIRONMENTS

8.4.1 Loss of Material

Borated Water Leakage

Loss of material due to boric acid wastage is an applicable aging effect for the external surfaces of carbon steel and low-alloy or chrome-moly materials exposed to leakage of borated water. Leaking fluid from a borated water system may expose the external surfaces of components made from these materials to a concentrated boric acid solution that can cause loss of material.

Treated Water Leakage

Loss of material due to general corrosion is an applicable aging effect for admiralty brass, brass, carbon steel, cast iron, copper, and low-alloy steel exposed to treated water leakage due to the presence of oxygen. Stainless steel materials exposed to treated water leakage are resistant to general corrosion. Paint or protective coatings applied to external surfaces will prevent this aging effect.

Loss of material due to pitting corrosion is an applicable aging effect for admiralty brass, brass, carbon steel, cast iron, copper, low-alloy steel, and stainless steel materials exposed to treated water leakage. Paint or protective coatings applied to external surfaces will prevent this aging effect.

Raw Water Leakage

Loss of material due to general corrosion is an applicable aging effect for admiralty brass, brass, bronze, carbon steel, cast iron, copper, and 90-10 copper-nickel component materials exposed to raw water leakage. Stainless steel materials are resistant to general corrosion due to raw water leakage.

Loss of material due to pitting corrosion is an applicable aging effect for admiralty, brass, bronze carbon steel, cast iron, copper, 90-10 copper-nickel, and stainless steel materials exposed to raw water leakage.

Loss of material due to galvanic corrosion can occur in materials exposed to raw water leakage.

Loss of material due to microbiologically influenced corrosion is an applicable aging effect for admiralty, brass, bronze, carbon steel, cast iron, copper, 90-10 copper-nickel and stainless steel materials exposed to raw water leakage.

Paint or protective coatings applied to external surfaces will prevent exposure to the raw water environment eliminating the aging effect of loss of material.

8.5 Aging Effects of Underground Environments

8.5.1 Loss of Material

Carbon steel materials in the underground environment in contact with soil and untreated groundwater can experience loss of material due to various corrosion mechanisms.

Microbiological organisms present in the soil or groundwater can produce corrosive substances, as a byproduct of their biological processes, that disrupt the protective oxide layer on the component materials, leading to a material depression similar to pitting corrosion.

Galvanic corrosion in an underground environment can occur when materials with different electrochemical potentials are in contact in the presence of water. In the underground environment, galvanic corrosion can occur between the material and the surrounding soil and groundwater.

8.6 Industry Experience with External Aging Effects

To validate the applicable aging effects for mechanical components exposed to various external environments, industry experience was reviewed. The survey of industry experience included an NPRDS search on relevant topics and a review of NRC generic communications and NUREG documents. No unique aging effects were identified from these documents beyond those discussed in this section.

9.0 EFFECTS REQUIRING AGING MANAGEMENT FOR BOLTED CLOSURES

This section discusses aging affects applicable to bolted closures of non-Class 1 mechanical components whose intended function is maintenance of pressure boundaries including the following.

- Bolted flange connections for vessels (i.e., manways and inspection ports)
- Flanged joints in piping
- Body-to-bonnet joints in valves
- Pressure-retaining bolting on pumps and miscellaneous process components

A bolted closure includes the seating surfaces (e.g., flange set surfaces), gasket, and pressure retaining bolting. Pressure boundary bolting, typically referred to as threaded fasteners, includes nuts, bolts, studs, and capscrews. Gaskets do not require an aging management review because the gaskets are only a part of the bolted connection and exist to minimize leakage rather than to directly support the pressure boundary.

9.1 ATTRIBUTES OF BOLTED CLOSURE ENVIRONMENTS

Bolted closures within the scope of license renewal are found in all non-Class 1 mechanical systems. The non-Class 1 bolting and threaded connections are subject to external ambient environments and exposure to leakage of process fluids from within components.

9.2 MATERIALS EVALUATED FOR BOLTING/THREADED CONNECTIONS

The materials used in bolting and threaded connections within the scope of license renewal are primarily carbon and low-alloy steels and stainless steel.

9.3 AGING EFFECTS ON BOLTED CLOSURES

The governing aging effect to consider for bolted closures is loss of mechanical closure integrity. Loss of mechanical closure integrity may be attributed to one or more of the following conditions.

9.3.1 Loss of Pre-Load

The loss of bolted closure may be attributed to embedment, cyclic load embedment, gasket creep, thermal effects (e.g., yield stress effect, modulus of elasticity effect, and stress relaxation), and self loosening.

9.3.2 Loss of Material of Bolting Materials

<u>Corrosion of Bolting Materials</u>

Loss of material due to boric acid wastage is the most common aging effect observed for ferritic fasteners. Stainless steel fasteners are immune to loss of material due to general corrosion. Most bolting is normally in a dry environment and is coated with a lubricant and general corrosion is not expected. General corrosion of ferritic fasteners has only been observed due to leaking joints.

Wear of Bolting Materials

Wear could lead to the loss of bolting material in connections subject to frequent operational use. Proper maintenance and tool handling practices and infrequent opening and closing preclude significant wear on most bolted connections.

9.3.3 Cracking of Bolting Materials

Cracking of bolting materials may be caused by stress corrosion cracking or fatigue. Stress corrosion cracking may occur in bolting materials subjected to water or steam (e.g., from leakage) that contains various contaminants. When leakage is combined with contaminant species, such as sulfides or chlorides, an aggressive environment that can promote stress corrosion cracking may result.

9.3.4 Vibration-Induced Loosening

Threaded connections are assumed subject to loss of mechanical closure integrity in high vibration applications, such as on diesel generators. This mechanism is independent of material and environment.

9.4 INDUSTRY EXPERIENCE WITH BOLTED CLOSURES

To validate the applicable aging effects for non-Class 1 bolted closures, industry experience was reviewed. The survey of industry experience included an NPRDS search on relevant topics and a review of NRC generic communications and NUREG documents. No unique aging effects were identified from these documents beyond those discussed in this section.

10.0 AGING EFFECTS REQUIRING MANAGEMENT IN HEAT EXCHANGERS

Although most heat exchanger aging effects are material and environment driven, heat exchangers are also evaluated for the component specific mechanisms of erosion and wear. In addition, heat exchangers are typically exposed to several different environment and material combinations (i.e., tube material and fluid are different from shell material and fluid) [Reference 10-1].

10.1 ATTRIBUTES OF HEAT EXCHANGER ENVIRONMENTS

The internal environments of heat exchangers (both the internal surfaces of the tubes and the internal surfaces of the shells) are the same as those described for the contained fluids (i.e., borated, raw and treated water, oil or fuel oil, and various gases). External heat exchanger surfaces are exposed to the external ambient atmosphere or to external leakage as discussed in Section 8.0 of this appendix.

10.2 HEAT EXCHANGER AGING EFFECTS

In general, heat exchanger components are subject to the same aging mechanisms as other non-Class 1 mechanical components of similar materials exposed to similar environments as described in Sections 2.0 through 8.0 of this appendix. This section will focus on the aging mechanisms that affect the heat exchangers in addition to the standard material and environment mechanisms.

10.2.1 Loss of Material

Loss of material due to vibration and rubbing of the heat exchanger internal components is an effect for heat exchangers. The applicability of this mechanism is dependent on the heat exchanger configuration.

Depending on specific component geometry and contents of the entrained fluids, erosion or erosion-corrosion can also produce a loss of material on internal heat exchanger surfaces with high flow velocities (typically, where the velocity is high in a localized area).

10.2.2 Fouling

Fouling is any process that changes heat transfer surfaces such that it impairs heat transfer. The impairment is primarily from a decrease in the heat transfer coefficient of the heat transfer surface.

Entrained materials in the fluids precipitating out and adhering to the heat exchanger surfaces, influenced by local flow patterns and temperature conditions, can cause fouling. Many corrosion mechanisms also generate corrosion byproducts that adhere to the surface of the corroding metal. Since even a thin layer of fouling can impact the heat transfer capability of heat exchangers, fouling is considered an applicable aging effect for non-Class 1 heat exchangers subject to an aging management review.

10.2.3 Heat Exchanger Operating History

To validate the applicable aging effects for non-Class 1 heat exchangers, industry experience was reviewed. The survey of industry experience included an NPRDS search on relevant topics and NRC generic communications and NUREG documents. No unique aging effects were identified in these documents beyond those discussed in this section.

10.3 SECTION 10.0 REFERENCES

10-1 Sandia National Laboratories, "Aging Management Guideline for Commercial Nuclear Power Plants-Heat Exchangers," Contractor Report No. SAND93-7070, June 1994.

Appendix D

Technical Specification Changes

The Code of Federal Regulations, 10CFR54.22, requires applicants to include any technical specification changes, or additions, necessary to manage the effects of aging during the period of extended operation as a part of the renewal application. Based on a review of the information provided in the ANO-1 LRA and Technical Specifications, no changes to the ANO-1 Technical Specifications have been identified.