CP&L
A Progress Energy Company

Robinson Nuclear Plant

License Renewal Application

PREFACE

The following describes the content of the RNP License Renewal Application.

Chapter 1 provides the administrative information required by 10 CFR54.17 and 10 CFR 54.19.

Chapter 2 describes and justifies the methodology used to determine the systems, structures, and components within the scope of license renewal and the structures and components subject to an aging management review. The results of applying the methodology are provided in Tables 2.2-1, 2.2-2, and 2.2-3. These tables provide listings of the mechanical systems, structures, and electrical/instrumentation and control (I&C) systems within the scope of license renewal. Chapter 2 also provides a description of systems and structures and their intended functions and tables identifying components/commodities requiring aging management review and their intended functions. The tables provide a reference to the results of the aging management review for each component/commodity type. The descriptions of systems in Chapter 2 also identify the license renewal drawings that document the intended function boundaries for most of the mechanical systems. The drawings are provided in a separate submittal.

Chapter 3 describes the results of the aging management reviews of the components and structural components requiring aging management review. Chapter 3 is divided into six sections that address (1) the Reactor Vessel, Internals, and Reactor Coolant System, (2) Engineered Safety Features, (3) Auxiliary Systems, (4) Steam and Power Conversion Systems, (5) Containments, Structures, and Component Supports, and (6) Electrical and Instrumentation and Controls. The tables in Chapter 3 provide a summary of information concerning the aging effects requiring management and applicable aging management programs for component and commodity groupings in the scope of license renewal. The information presented in the tables is based on the format and content of NUREG-1800, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," U. S. Nuclear Regulatory Commission, April 2001, (the SRP-LR). The tables provide a discussion of the applicability of the component commodity group to RNP and details regarding the degree to which proposed aging management programs are consistent with those recommended in NUREG-1801, "Generic Aging Lessons Learned (GALL)," U.S. Nuclear Regulatory Commission, April 2001, (the GALL Report).

Chapter 4 addresses the time-limited aging analyses, as defined by 10 CFR 54.3. It includes the identification of the component or subject, and an explanation of the time-dependent aspects of the calculation or analysis. Chapter 4 demonstrates whether (1) the analyses remain valid for the period of extended operation, or (2) the analyses have been projected to the end of the period of extended operation, or (3) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. Chapter 4 also confirms that no 10 CFR 50.12 exemption involving a time-limited aging analysis as defined in 10 CFR 54.3 is required during the period of extended operation.

Appendix A, Updated Final 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, 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. RNP programs and activities that are credited for managing aging are divided into existing aging management programs, enhanced aging management programs, and new aging management programs.

Appendix C is not used.

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 Chapter 2, Chapter 3, and Appendix B fulfills the requirements in 10 CFR 54.21(a). Section 1.4 discusses how the requirements of 10 CFR 54.21(b) will be met. The information in Chapter 4 fulfills the requirements in 10 CFR 54.21(c). The information in Appendix A and Appendix D fulfills the requirements in 10 CFR 54.21(d) and 10 CFR 54.22, respectively. The supplement to the Environmental Report, as required by 10 CFR 54.23, is provided with the RNP License Renewal Application as a separate document.

ACRONYMS AND ABBREVIATIONS

AAC	Alternate AC
ACI	American Concrete Institute
AFW	Auxiliary Feedwater
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
AMR	Aging Management Review
ANSI	American National Standards Institute
API	American Petroleum Institute
ASA	American Standards Association
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transient Without Scram
AWS	American Welding Society
AWWA	American Water Works Association
BIT	Boron Injection Tank
CCW	Component Cooling Water
CLB	Current Licensing Basis
CMAA	Crane Manufacturers Association Of America, Inc.
CP&L	Carolina Power & Light Company, a Progress Energy Company
CRDM	Control Rod Drive Mechanism
CS	Carbon Steel
CSS	Containment Spray System
CST	Condensate Storage Tank
CV	Containment Vessel
CVCS	Chemical And Volume Control System
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DG	Diesel Generator
DS	Dedicated Shutdown
E&RC	Environmental and Radiation Control
ECCS	Emergency Core Cooling System
EDB	(PassPort) Equipment Database
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
ER	Environmental Report
ESF	Engineered Safety Features
FHB	Fuel Handling Building
FSAR	Final Safety Analysis Report
FW	Feedwater
GDC	General Design Criteria
GL	Generic Letter
GSI	Generic Safety Issue
HBR	H. B. Robinson
HELB	High Energy Line Break
HVAC	Heating, Ventilating, and Air Conditioning
I&C	Instrumentation and Control

ACRONYMS AND ABBREVIATIONS

IEEE	Institute Of Electrical And Electronic Engineers
ILRT	Integrated Leak Rate Test (Containment Type A Test)
IN	Information Notice
INPO	Institute Of Nuclear Power Operations
IPA	Integrated Plant Assessment
ISI	In-Service Inspection
IVSW	Isolation Valve Seal Water System
LBB	Leak-Before-Break
LOCA	Loss of Coolant Accident
LR	License Renewal
MSL	Mean Sea Level
NDE	Nondestructive Examination
NDTT	Nil-Ductility Transition Temperature
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PAP	Personnel Access Portal
pH	Concentration of Hydrogen Ions
PORV	Power-Operated Relief Valve
PPS	Penetration Pressurization System
PRT	Pressurizer Relief Tank
PTS	Pressurized Thermal Shock
PVC	Polyvinyl Chloride
PWR	Pressurized Water Reactor
PWST	Primary Water Storage Tank
QA	Quality Assurance
RAB	Reactor Auxiliary Building
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RMS	Radiation Monitoring System
RNP	Robinson Nuclear Plant
RPV	Reactor Pressure Vessel
RT_{NDT}	Reference Temperature, Nil-Ductility Transition
RWST	Refueling Water Storage Tank
SBO	Station Blackout
SCs	Structures and Components (per 10 CFR 54.21(a)(1)
SCC	Stress Corrosion Cracking
SER	Safety Evaluation Report
SFP	Spent Fuel Pit
SG	Steam Generator
SI	Safety Injection
SRP	Standard Review Plan
SS	Stainless Steel

ACRONYMS AND ABBREVIATIONS

SSCs	Systems, Structures, and Components
SSE	Safe Shutdown Earthquake
SWS	Service Water System
TLAA	Time-Limited Aging Analysis
UFSAR	Updated Final Safety Analysis Report
USAS	United States Of America Standards
USE	Upper Shelf Energy
UT	Ultrasonic Test
WCAP	Westinghouse Commercial Atomic Power
WOG	Westinghouse Owner’s Group

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1.0 ADMINISTRATIVE INFORMATION

1.1 PURPOSE AND GENERAL INFORMATION

In accordance with the requirements of Part 54 of Title 10 of the Code of Federal Regulations (10 CFR 54), Carolina Power & Light Company, a Progress Energy Company (hereinafter referred to as CP&L), has prepared this application to provide the technical and environmental information required for renewal of the operating license for the H. B. Robinson Steam Electric Plant, Unit No. 2, also referred to as Robinson Nuclear Plant (RNP). This application supports license renewal for an additional 20-year period beyond the end of the current license term of RNP Facility Operating License, DPR-23. The end of the current license term is July 31, 2010. The technical information consists of (1) an Integrated Plant Assessment, as defined in 10 CFR 54.21(a), (2) an evaluation of time-limited aging analyses, as defined in 10 CFR 54.21(c), (3) a supplement to the RNP Updated Final Safety Analysis Report (UFSAR), as required by 10 CFR 54.21(d), and (4) environmental information, as required by 10 CFR 54.23. The environmental information is provided as a separate report entitled "Applicant's Environmental Report – Operating License Renewal Stage." No changes to the RNP Technical Specifications are required to support this application.

This application also is applicable to renewal of the source, byproduct, and special nuclear material licenses that are part of current Facility Operating License, DPR-23.

The format and contents of this application are in accordance with Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," dated April 2001, and the guidance provided in Revision 3 of NEI 95-10, "Industry Guidelines for Implementing the Requirements of 10 CFR 54 – License Renewal Rule," dated March 2001.

This application and supporting environmental report are intended to provide sufficient information for the NRC to complete its technical and environmental reviews and allow the NRC to make the finding required by 10 CFR 54.29 in support of the issuance of a renewed operating license for RNP. The following is the application filing and content information required by 10 CFR 54.17 and 10 CFR 54.19.

1.1.1 NAME OF APPLICANT

Carolina Power & Light Company, a Progress Energy Company

1.1.2 ADDRESS OF APPLICANT

Carolina Power & Light Company
411 Fayetteville Street
Raleigh, NC 27601-1748

Address of Robinson Nuclear Plant:

Carolina Power & Light Company
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville, SC 29550

1.1.3 OCCUPATION OF APPLICANT

Carolina Power & Light Company (CP&L) is a subsidiary of Progress Energy, Inc. CP&L is a corporation primarily engaged in the generation, transmission, distribution, and sale of electricity in portions of North and South Carolina. CP&L serves more than 1.3 million customers. The company serves a territory encompassing over 34,000 square miles including the cities of Raleigh, Wilmington, Fayetteville, and Asheville in North Carolina and Florence and Sumter in South Carolina.

1.1.4 ORGANIZATION AND MANAGEMENT OF APPLICANT

CP&L is a corporation organized and existing under the laws of the State of North Carolina. CP&L is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. CP&L makes this application on its own behalf and is not acting as an agent or representative of any other person.

The names and addresses of Progress Energy directors and principal officers are listed below. All persons listed are U. S. citizens.

Director	Address
Edwin B. Borden	Goldsboro, NC
David L. Burner	Charlotte, NC
William Cavanaugh III	Raleigh, NC

Director	Address
Charles W. Coker	Hartsville, SC
Richard L. Daugherty	Raleigh, NC
W. D. (Bill) Frederick, Jr.	Orlando, FL
William O. McCoy	Chapel Hill, NC
E. Marie McKee	Corning, NY
John H. Mullin III	Brookneal, VA
Richard A. Nunis	Orlando, FL
Carlos A. Saladrigas	Miami, FL
J. Tylee Wilson	Ponte Vedra Beach, FL
Jean Giles Wittner	St. Petersburg, FL

Principal Officers	Address
William Cavanaugh III Chairman, President, and Chief Executive Officer	Progress Energy, Inc. 410 S. Wilmington Street Raleigh, NC 27601-1748
Robert B. McGehee Executive Vice President President and Chief Executive Officer, Progress Energy Service Company	Progress Energy Service Company, LLC 410 S. Wilmington Street Raleigh, NC 27601-1748
Donald K. Davis Executive Vice President - Energy Services	Progress Energy, Inc. 410 S. Wilmington Street Raleigh, NC 27601-1748
Fred N. Day IV Executive Vice President - Energy Delivery	Progress Energy, Inc. 410 S. Wilmington Street Raleigh, NC 27601-1748

Principal Officers	Address
H. William Habermeyer, Jr. President and Chief Executive Officer Florida Power Corporation	Progress Energy, Inc. 100 Central Avenue St Petersburg, Fl 33701-3324
William D. Johnson Executive Vice President General Counsel and Secretary	Progress Energy Service Company, LLC 410 S. Wilmington Street Raleigh, NC 27601-1748
Tom D. Kilgore President and Chief Executive Officer, Progress Ventures, Inc.	Progress Energy, Inc. 410 S. Wilmington Street Raleigh, NC 27601-1748
William S. Orser Group President - Energy Supply	Progress Energy, Inc. 410 S. Wilmington Street Raleigh, NC 27601-1748
Peter M. Scott III Executive Vice President and Chief Financial Officer	Progress Energy Service Company, LLC 410 S. Wilmington Street Raleigh, NC 27601-1748
Cecil L. Goodnight Senior Vice President - Administrative Services	Progress Energy Service Company, LLC 410 S. Wilmington Street Raleigh, NC 27601-1748
Bonnie V. Hancock Senior Vice President - Finance & Information Technology	Progress Energy Service Company, LLC 410 S. Wilmington Street Raleigh, NC 27601-1748
C. S. Hinnant Senior Vice President - Nuclear Generation	Progress Energy, Inc. 410 S. Wilmington Street Raleigh, NC 27601-1748
E. Michael Williams Senior Vice President - Power Operations	Progress Energy, Inc. 410 S. Wilmington Street Raleigh, NC 27601-1748
Joseph P. Hirl Senior Vice President - Trading and Marketing Progress Ventures, Inc.	Progress Energy, Inc. 410 S. Wilmington Street Raleigh, NC 27601-1748

1.1.5 CLASS AND PERIOD OF LICENSE SOUGHT

CP&L requests renewal of the Class 104b operating license for Robinson Nuclear Plant (Facility Operating License No. DPR-23) for a period of 20 years beyond the expiration of the current license. Approval of this license renewal request would extend the operating license for RNP from midnight July 31, 2010 until midnight July 31, 2030. The facility would continue to be known as the H. B. Robinson Steam Electric Plant Unit, No. 2, and will continue to generate electric power during the renewal period. CP&L also requests renewal of the source, byproduct, and special nuclear material licenses that are combined in the operating license.

1.1.6 ALTERATION SCHEDULE

CP&L does not propose to construct or alter any production or utilization facility in connection with this renewal application.

1.1.7 CHANGES TO THE STANDARD INDEMNITY AGREEMENT

10 CFR 54.19(b) requires that license renewal applications include, "…conforming changes to the standard indemnity agreement, 10 CFR 140.92, Appendix B, to account for the expiration term of the proposed renewed license." The current indemnity agreement for RNP 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 agreement. Item 3 of the Attachment to the indemnity agreement, as revised by Amendment No. 1, lists RNP Operating License DPR-23. CP&L requests that conforming changes be made to the indemnity agreement, and/or the Attachment to that agreement, specifying the extension of agreement until the expiration date of the renewed RNP operating license as sought in this application. In addition, should the license number be changed upon issuance of the renewed license, CP&L requests that conforming changes be made to the Attachment and any other sections of the indemnity agreement as appropriate.

1.1.8 RESTRICTED DATA AGREEMENT

This application does not contain any Restricted Data or other defense information, and CP&L does not expect that any activity under the renewed license for RNP will involve such information. However, if such information were to become involved, CP&L agrees that it will appropriately safeguard such information and not permit any individual to have access to, or any facility to possess, such information until the individual or facility has been approved for such access under the provisions of 10 CFR Part 25 and/or 10 CFR Part 95.

1.2 DESCRIPTION OF ROBINSON NUCLEAR PLANT

The Robinson Nuclear Plant (RNP), also known as Unit 2 of the H. B. Robinson Steam Electric Plant, is adjacent to Unit 1 of the H. B. Robinson Steam Electric Plant, a coal-fired steam power plant. The plant is located on Lake Robinson, a man-made lake, in South Carolina.

The RNP reactor is a pressurized light water moderated and cooled system. The nuclear power plant incorporates a three-loop, closed-cycle, pressurized water Nuclear Steam Supply System (NSSS) designed by Westinghouse Electric Corporation. The Turbine-Generator System utilizes saturated steam produced by the NSSS. Plant equipment includes systems for the processing of radioactive wastes, handling of fuel, electrical distribution, cooling, power generation structures, and all other onsite facilities required to provide a complete and operable nuclear power plant.

Additional descriptive information about RNP systems, structures, and components is provided in later chapters of this application.

CP&L operates an independent spent fuel storage installation at the RNP site in accordance with 10 CFR Part 72. The independent spent fuel storage installation is subject to licensing independent of the RNP operating license. The independent spent fuel storage installation is not within the scope of 10 CFR Part 54 or this application.

1.3 TECHNICAL INFORMATION REQUIRED FOR AN APPLICATION

In accordance with 10 CFR 54.21, four technical items are required to support an application for a renewed operating license. These are (1) an Integrated Plant Assessment (IPA), (2) an evaluation of time-limited aging analyses (TLAA), (3) a supplement to the RNP UFSAR that contains a summary description of the programs and activities for managing the effects of aging and the evaluation of the time-limited aging analyses, and (4) any changes to the current licensing basis (CLB) that occur during NRC review. In this application, the IPA information is provided in Chapter 2, Chapter 3 and Appendix B; the TLAA information, in Chapter 4; the UFSAR information, in Appendix A; and CLB changes, in Section 1.4.

In addition to the technical information, 10 CFR 54.22 requires applicants to submit any Technical Specification changes or additions necessary to manage the effects of aging during the period of extended operation. In this application, any information regarding Technical Specification changes is contained in Appendix D.

10 CFR 54.23 requires the application to include a supplement to the Environmental Report. A separate report entitled "Applicant's Environmental Report – Operating License Renewal Stage" has been provided as part of the application.

The IPA, as defined by 10 CFR 54.3, is a licensee assessment that demonstrates that a nuclear power plant's structures and components requiring aging management review in accordance with 10 CFR 54.21(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 RNP IPA includes:

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

The RNP IPA for license renewal, along with other information necessary to document compliance with 10 CFR 54, is maintained in an auditable and retrievable form, in accordance with 10 CFR 54.37(a). The RNP IPA is documented with site-specific reports and calculations that were generated in accordance with the RNP Quality Assurance Program.

1.4 CURRENT LICENSING BASIS CHANGES DURING NRC REVIEW

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

CP&L anticipates a revision to the CLB as a result of a planned Appendix K power uprate for RNP. Evaluations are underway to support an increase in licensed power, and related CLB changes that materially affect the contents of the License Renewal Application will be reported in amendments to the Application as required by 10 CFR 54.21(b).

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

This chapter describes the process for identifying structures and components subject to an aging management review in the RNP License Renewal review. The results of applying the process at RNP also are provided. 10 CFR 54.4 provides requirements for determining plant structures, systems, and components (SSCs) in scope for License Renewal. For those SSCs, 10 CFR 54.21(a)(1) requires a license renewal application to include an Integrated Plant Assessment (IPA) that identifies and lists the structures and components (SCs) subject to an aging management review. 10 CFR 54.21(a)(2) further requires that the methods used to identify and list these structures and components be described and justified. The technical information in this chapter is intended to satisfy these requirements.

The RNP License Renewal review methodology follows the approach recommended in NEI 95-10 [Reference 2.1-1]. The methodology consists of three processes: scoping, screening, and aging management reviews. These processes have been implemented in accordance with the RNP Quality Assurance Program.

Scoping and screening methodologies are described in Section 2.1. 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 structure components subject to an aging management review (screening) are contained in Section 2.3 for mechanical systems, Section 2.4 for structures, and in Section 2.5 for electrical/I&C systems.

The information provided in this Chapter provides the basis for the NRC to make the finding required by 10 CFR 54.29(a)(1) regarding identification of the SCs that require aging management review.

2.1 SCOPING AND SCREENING METHODOLOGY

Scoping is the initial step in the RNP evaluation methodology. Scoping is performed to identify SSCs that perform intended functions within the scope of license renewal as required by 10 CFR 54.4. The scoping methodology is described in Subsection 2.1.1.

Screening is the second step of the RNP methodology and addresses the requirements of an IPA defined in 10 CFR 54.21(a). The RNP screening process includes (1) a review of the systems and major structures identified as in scope for license renewal to identify the specific components of those structures and systems that support the functions of 10 CFR 54.4, and (2) a review of the components and structural components to identify those that satisfy the screening criteria of 10 CFR 54.21(a)(1). The screening process identifies those components and structural components that are subject to an aging management review. The screening process is described in Subsection 2.1.2.

2.1.1 SCOPING

SSCs that satisfy the criteria of 10 CFR 54.4(a)(1), (2), or (3) are within the scope of license renewal. Specifically, 10 CFR 54.4 states:

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

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

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

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

(3) *All systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrated compliance with the Commission's regulation for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR50.62), and station blackout (10 CFR 50.63).*

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

The RNP scoping process employed a multi-faceted approach to ensure that the systems and structures meeting the criteria of §54.4(a)(1) through (a)(3) were identified. The process was designed to make optimum use of existing plant documents and databases to populate the list of systems and structures in the scope of the Rule. The scoping methodology is consistent with guidance provided by the NRC in its letter from C. I. Grimes to D. J. Walters, of the Nuclear Energy Institute (NEI), dated August 5, 1999, entitled, “License Renewal Issue No. 98-0082, Scoping Guidance” [Ref. 2.1-2].

Details of the scoping process for safety related SSCs, in accordance with 10 CFR 54.4(a)(1), are provided in Subsection 2.1.1.1. Details of the scoping process for non-safety related SSCs whose failure could prevent satisfactory accomplishment of any of the functions identified for safety related SSCs, in accordance with 10 CFR 54.4(a)(2), are provided in Subsection 2.1.1.2. Details of the scoping process for SSCs relied on to demonstrate compliance with one of the regulated events, in accordance with 10 CFR 54.4(a)(3), are provided in Subsection 2.1.1.3.

2.1.1.1 Safety Related Criteria Pursuant to 10 CFR 54.4(a)(1)

The process of identification of safety related SSCs included use of the RNP PassPort Equipment Database (EDB) as the primary source to define a comprehensive list of the systems and structures that make up the Robinson Nuclear Plant and to identify those that are classified as safety related. The EDB was developed using the RNP Q-List and extends the classification of systems to the component level. For the purposes of license renewal, any system/structure, including support systems, that contains one or more safety related components was considered to be a safety related system/ structure.

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

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 exposure comparable to the guidelines in §50.34(a)(1), §50.67(b)(2), or §100.11 of this chapter as applicable.

These criteria are consistent with those used to develop the original Q-List at RNP, as documented in the RNP Continuing Quality Assurance Program Manual and RNP procedures that control the Q-List. Control of the RNP Q-List has evolved through several procedures. Currently, it is controlled by the “Q-List Control Procedure” containing the following criteria for inclusion of a structure, system, or component on the RNP Q-List:

“For qualifying as a Q-list plant item, the plant item in question must be necessary for any of the following:

a. Provide for structural integrity (or be a part) of the reactor coolant pressure boundary (RCPB).

b. Provide over pressure protection for the RCPB.
c. Make and hold the reactor subcritical during the occurrence of frequent, infrequent, and limiting plant process conditions.
d. Cool the core following normal reactor shutdown and during the occurrence of frequent, infrequent, and limiting plant process conditions.
e. Cool another safety system component during the occurrence of frequent, infrequent and limiting plant process conditions, such that its failure would inhibit the operation of the safety system.
f. Provide essential services to the reactor containment during the occurrence of infrequent and limiting plant process conditions.
g. Contain the radioactivity, control or reduce the radioactivity release to the environment during the occurrence of frequent, infrequent and limiting plant process conditions.
h. Meet other regulatory requirements involving specific seismic and/or environmental qualifications as determined by the Engineering Supervisor.
i. Regulatory Guide 1.97, Category 1 components required to follow the course of an accident."

A comparison finds that the criteria of the Q-List Control Procedure are broader in scope than those of §54.4(a)(1), such that the equipment needed to satisfy the latter is a subset of the former. The RNP scoping process for license renewal assumed a conservative interpretation that any system or structure that includes or contains a safety related component is a safety related system/structure. Therefore, it follows that the RNP Q-List can be used as a tool for system/structure scoping against the requirements of §54.4(a)(1).

Consistent with commitments in the RNP current licensing basis (CLB), the RNP Q-List criteria define the SSCs relied on to remain functional during and following design basis events described in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR) as well as in other sections of the UFSAR where the design bases for SSCs are defined by postulated events such as earthquakes and other external hazards.

Also, RNP design and CLB documentation was reviewed to compile a comprehensive list of functions that each system and structure at RNP is credited with performing. Primary sources of this information include Design Basis Documents, the PassPort EDB System, and the UFSAR. System functions that meet the criteria of §54.4(a)(1) were identified. These are the system/structure intended functions that are the basis for inclusion in license renewal scope.

Based on the above, the scoping process to identify safety related SSCs for RNP License Renewal is consistent with the guidance provided in Reference 2.1-2 regarding identification of safety related SSCs and satisfies the criteria in 10 CFR 54.4(a)(1).

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

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

The relationship by which the criterion of 10 CFR 54.4(a)(2) might be satisfied takes on one of two forms: (1) functional dependencies, wherein non-safety related equipment is required to perform a function in order to support the function of safety related equipment, and (2) physical interactions, wherein the failure of non-safety related equipment might inhibit the performance of nearby safety related equipment (seismic interaction, flooding effects, high energy line break effects, etc). In either case consistent with Reference 2.1-2, consideration of hypothetical failures that could result from system interdependencies that are not part of the plant's CLB, or that have not been previously experienced is not required.

At RNP, the procedural requirements for component classification state that components that do not perform a safety related function but whose failure could prevent the satisfactory accomplishment of a safety related function during or following design basis accidents and transients are to be classified as safety related. However, there are instances where the CLB permits use of non-safety related systems to support the function of safety related systems. In those cases, the systems are classified in accordance with CLB commitments. Therefore, an evaluation was performed to assure that all SSCs meeting the criterion of 10 CFR 54.4(a)(2) have been identified.

The following procedure steps describe the process that was employed to identify applicable non-safety related systems and structures.

First, certain non-safety related SSCs were not considered applicable to the review. These are discussed in the following paragraphs:

1. In some cases there will be overlap between §54.4(a)(2) and §54.4(a)(3). Evaluations for fires, ATWS, and Station Blackout rely on non-safety related SSCs in order to demonstrate acceptable results. While this equipment would otherwise fall under Criterion §54.4(a)(2), it was not specifically identified and addressed as such, because it is already within the scope of license renewal in accordance with §54.4(a)(3). The same logic also applies to any Environmental Qualification (EQ) Program components that are non-safety related.

2. The RNP design and licensing basis includes instances wherein non-safety related equipment, augmented with a suitable surveillance or monitoring program, is used to maintain safety related equipment or plant conditions within limits consistent with event assumptions. For example, station battery electrolyte temperature is assured to be within the required range by a combination of an area HVAC system, that is non-safety related and not in scope for license

renewal, augmented with regular monitoring of room and electrolyte temperature. For another example, plant chemistry is assumed to be within the specifications maintained by the Chemistry Program based upon regular monitoring and analysis. In these instances, it is the monitoring or surveillance program that is primarily credited with ensuring the appropriate initial conditions exist, rather than the reliability of any non-safety related equipment. Therefore, the function of non-safety related equipment to establish initial conditions for equipment operation or accident assumptions does not constitute the sole basis for inclusion in license renewal scope under §54.4(a)(2).

3. Malfunctions of non-safety related equipment which result in an actuation of safety related equipment do not constitute a basis for inclusion under §54.4(a)(2), since these malfunctions do not result in the loss of a safety related function. For example, loss of a condensate pump would result in a reactor trip and resultant challenge to plant safety systems. However, this would not result in the loss or degradation of any of the associated safety related equipment.

4. With regard to Regulatory Guide 1.97, those components used to monitor Category 1 variables are considered safety-related and seismic, and so within the scope of license renewal under §54.4(a)(1). Those non-safety related components (including sampling systems) that are used to monitor Category 2 and 3 variables are not included on the basis of their monitoring function, since those variables are not relied upon to perform a safety related function, nor would their failure to provide the monitoring function directly result in the failure of safety related SSCs.

Second, after eliminating the above categories of SSCs, the following steps were performed:

1. The RNP design and licensing basis information was reviewed to identify non-safety related SSCs that function to directly support a safety related system or structure and whose failure could prevent the performance of a required intended function. Sources of this information include Design Basis Documents, the UFSAR, EDB, Maintenance Rule Database, and docketed correspondence. Each instance was identified wherein non-safety related SSCs are credited in the performance of an intended function or whose failure could prevent the performance of an intended function. For each such instance, the specific function that is required of the non-safety related system/structure was identified. The SSCs meeting these criteria are designated as within the scope of license renewal per the §54.4(a)(2) criteria, and the associated function or interaction is considered to be a system/structure intended function.

2. The RNP design and licensing basis information was reviewed to identify non-safety related SSC interactions with safety related SSCs that could prevent the performance of a required intended function. Sources of this information include

Design Basis Documents, the UFSAR, plant drawings, and other CLB documentation, as well as the EDB and the Maintenance Rule Database. For each such instance, the specific interaction that may affect the function of safety related SSCs was identified. The SSCs meeting these criteria are designated as within the scope of license renewal per the §54.4(a)(2) criteria, and the associated interaction is considered to be a system/structure intended function.

3. Interactions of non-seismically-qualified SSC with seismically-qualified SSC (commonly referred to as Seismic II over I) is not part of the CLB for RNP. The RNP CLB, however, considers the effects of physical interactions on systems, structures and components necessary to achieve and maintain safe shutdown, consistent with the plant's responses pertaining to resolution of Unresolved Safety Issue (USI) A-46. The USI A-46 review imposed criteria for evaluating interactions between seismically-qualified SSC and non-seismically-qualified SSC associated with proximity, structural failure and falling, and flexibility of attached cables and piping. This type of interaction was considered in the license renewal process, and a spaces- or area-based approach was used to identify components in this category. As part of the screening process (refer to Subsection 2.1.2), a plant area-based approach was implemented to identify spatial interactions between non-safety related SSC and safety related SSC that could adversely affect the accomplishment of an intended function. Plant walkdowns were performed to identify potential seismic interactions; and non-safety related structural components (e.g., pipe supports, raceway supports, equipment supports, and miscellaneous structures) associated with seismic interactions were selected based on their location relative to safety related structures, systems, and components.

Based on the above, the scoping process for RNP License Renewal identified both non-safety related SSC that support the functioning of safety related SSC (functional interaction) and non-safety related SSC that may have adverse physical interactions with safety related SSC (spatial interaction). Therefore, the process is consistent with and satisfies the criteria in 10 CFR 54.4(a)(2).

2.1.1.3 Other Scoping Pursuant to 10 CFR 54.4(a)(3)

10 CFR 54.4(a)(3) states that SSCs relied upon in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63) are within the scope of license renewal. Current licensing basis evaluations have been performed and documented which facilitate the identification of those SSC credited in compliance of each of these regulations. For these SSCs, the system/structure level intended function is that it is relied upon in safety analyses or evaluations to demonstrate compliance with NRC requirements for the event in question. A system/structure function based approach is

not needed to identify intended functions, but can be used as necessary to identify the boundaries of credited equipment. Systems or structures that have one or more components credited for demonstrating compliance with one of the regulated events are within the scope of license renewal per the §54.4(a)(3) criteria.

Scoping based on each of the regulated events is described in the following paragraphs.

2.1.1.3.1 Fire Protection

Fire protection features and commitments are described in detail in the RNP UFSAR. UFSAR Section 9.5.1 describes the fire protection program and systems at RNP. UFSAR Appendix 9.5.1A provides the Fire Hazards Analysis that addresses fire protection features and related information on an area-by-area basis. A comparison of RNP fire protection administrative controls and features with NRC guidelines is provided in UFSAR Appendix 9.5.1B. And UFSAR Appendix 9.5.1C is an analysis to document separation between safe shutdown trains required to demonstrate compliance with the requirements of Section III.G of Appendix R of 10 CFR 50. The systems and structures at RNP that support either fire protection design or safe shutdown following a fire are considered within the scope of license renewal. Those items providing fire suppression, fire detection, and electrical power supply to that equipment are classified Quality class B-03 in the EDB. Any system with components classified as B-03 was considered in scope. Also, any systems, with components credited in plant design documents with supporting safe shutdown following a fire were considered in scope. Safe shutdown components credited in analyses to comply with 10 CFR 50, Appendix R, are included in the Appendix R Component Database. This database was used to identify systems in scope for license renewal. In addition, structures credited in the definition of fire areas and zones were considered in scope, as applicable.

The steps to identify SSCs relied on for fire protection to meet 10 CFR 54.4(a)(3) are:

1. The EDB, UFSAR Section 9.5.1, and UFSAR Appendices 9.5.1A, 9.5.1B, and 9.5.1C, design drawings, and component databases were reviewed, as applicable.

2. Based on the above, license renewal intended functions relative to the criteria or 10 CFR 54.4(a)(3) for fire protection were identified for each system and structure determined to meet this criteria.

The scoping process to identify systems and structures relied upon and/or specifically committed to for fire protection for RNP is consistent with and satisfies the criteria in 10 CFR 54.4(a)(3).

2.1.1.3.2 Environmental Qualification

10 CFR 50.49(b) defines electric equipment important to safety that is required to be environmentally qualified to mitigate certain accidents that result in harsh environmental conditions in the plant. The criteria for determining which equipment is subject to environmental qualification requirements are presented in Section 3.11 of the UFSAR and are identified on the RNP Environmental Qualification (EQ) List for 10 CFR 50.49.

The steps to identify SSCs relied on for environmental qualification to meet 10 CFR 54.4(a)(3) are:

1. The EDB identifies components that are on the RNP EQ List for 10 CFR 50.49. The EDB was used as a principal input document for scoping of SSCs. Any system that contained one or more components designated as "EQ" in EDB was considered in scope due to EQ.

2. The standard intended function for EQ was applied to the affected system: "Relied upon in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for Environmental Qualification (EQ)."

The scoping process to identify systems and structures relied upon and/or specifically committed to for environmental qualification for RNP is consistent with and satisfies the criteria in 10 CFR 54.4(a)(3). Note that EQ components may meet the requirements for Time-Limited Aging Analyses (TLAAs). EQ-related TLAAs are discussed in Section 4.4.

2.1.1.3.3 Pressurized Thermal Shock

10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, " requires that licensees evaluate the reactor vessel beltline materials against specific criteria to ensure protection against brittle fracture. CP&L has documented compliance with 10 CFR 50.61 with a number of submittals to the NRC dated September 15, 1993, documenting compliance with the Pressurized Thermal Shock (PTS) rule; and August 17, 1995, November 20, 1995 and July 23, 1998, responding to Generic Letter 92-01. The licensing correspondence related to PTS is listed as References 2.1-4 through 2.1-7.

Based upon the current analysis for PTS, RNP does not rely on a Regulatory Guide 1.154 analysis to satisfy the PTS Rule, and there are no structures, systems and components other than the reactor vessel that are within the scope of license renewal.

The scoping process to identify systems and structures relied upon and/or specifically committed to for PTS for RNP is consistent with and satisfies the criteria in 10 CFR 54.4(a)(3). Note that PTS is related to reactor pressure vessel embrittlement, which is a

Time Limited Aging Analysis (TLAA). The TLAA analysis associated with PTS is discussed in Section 4.2.

2.1.1.3.4 Anticipated Transients Without Scram

RNP design features related to Anticipated Transients Without Scram (ATWS) are within the scope of license renewal because they are relied on to meet the requirements of 10 CFR 50.62.

The steps to identify systems and structures at RNP relied upon for ATWS mitigation to meet the requirements of 10 CFR 54.4(a)(3) are outlined below:

1. The ATWS system description, EDB, and design drawings were reviewed.

2. Based on the above, the license renewal intended function relative to the criteria of 10 CFR 54.4(a)(3) for ATWS events was identified for each system and structure determined to meet this criteria.

The scoping process to identify systems and structures relied upon and/or specifically committed to for a postulated ATWS event for RNP is consistent with and satisfies the criteria in 10 CFR 54.4(a)(3).

2.1.1.3.5 Station Blackout

Design features to satisfy the Station Blackout (SBO) rule are described in the Station Blackout Coping Analysis [Reference 2.1-8].

The steps to identify systems and structures at RNP relied upon for SBO to meet the requirements of 10 CFR 54.4(a)(3) are outlined below:

1. The Station Blackout Coping Analysis, plant procedures, EDB, and design drawings were reviewed.

2. Based on the above, license renewal intended function relative to the criteria of 10 CFR 54.4(a)(3) for a postulated SBO was identified for each system and structure determined to meet this criteria.

The scoping process to identify systems and structures relied upon and/or specifically committed to for a postulated SBO for RNP is consistent with and satisfies the criteria in 10 CFR 54.4(a)(3).

2.1.2 STRUCTURE AND COMPONENT SCREENING

This subsection discusses the process used at RNP to (1) define particular structures and components (SCs) that perform an intended function within those SSC determined to be in-scope for license renewal, and (2) identify which of these SCs require an aging management review. In the RNP Integrated Plant Assessment (IPA), these two activities are considered part of the screening process.

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

Each application must contain the following information:

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

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

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

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

The screening process was performed by discipline: mechanical, civil/structural, and electrical/I&C. The screening process for mechanical systems is described in

Subsection 2.1.2.1; for civil structures, in Subsection 2.1.2.2; and for electrical and I&C systems, in Subsection 2.1.2.3.

During the screening process, some SCs were incorporated into commodity groups based on similarity of their design or materials of construction. Use of commodity groups made it possible to address an entire group of SCs with a single evaluation.

The RNP process for evaluating consumables is consistent with the NRC staff guidance on consumables provided in a letter from C. I. Grimes, NRC, to D. J. Walters, NEI, dated March 10, 2000 [Reference 2.1-3].

2.1.2.1 Mechanical Systems

For mechanical systems, the screening process is performed on each system identified to be within the scope of license renewal. This process evaluates the individual components included within in-scope mechanical systems to identify specific components or component groups that require an aging management review.

For the systems in-scope for license renewal, mechanical system evaluation boundaries were established. Generally, these boundaries were determined by mapping the pressure boundary associated with the license renewal system intended functions onto the system flow diagrams. License renewal system intended functions are the functions a system must perform relative to the scoping criteria of 10CFR 54.4(a)(1), 10CFR 54.4(a)(2), and 10CFR 54.4(a)(3). The flow diagram boundary drawings associated with each mechanical system within the scope of license renewal are identified in the mechanical system screening results described in Section 2.3.

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

1. The evaluation boundaries associated with license renewal system intended functions are mapped onto the system's flow diagram. The entire flow path is considered to include all components credited for the successful completion of each intended function.

2. Based on a review of flow diagrams, design drawings, plant documentation, and the system component list from the PassPort EDB, components that are included within the system intended function boundaries are identified. Although mechanical system intended function boundaries ordinarily occur at a valve location, the seismic boundary may extend to a support past the valve and may include a section of non-safety related piping. This piping segment and the associated support also are included in the scope of license renewal.

3. The components within the system intended function boundary that perform an intended function without moving parts or without a change in configuration or

properties, i.e., the screening criteria of 10 CFR 54.21(a)(1)(i), are identified. Active/passive screening determinations are based on the guidance in Appendix B to NEI 95-10 [Reference 2.1-1]. 10 CFR 54.21(a)(1)(i) provides a summary of specific component types that are excluded from the scope of license renewal. These specific component types are screened based on the provisions of the Rule. Some components were determined to be part of a complex assembly as discussed in NEI 95-10 and were screened accordingly.

4. The passive, in-scope components that are not subject to replacement based on a qualified life or specified time period, i.e., the screening criteria of 10 CFR 54.21(a)(1)(ii), are identified as requiring an aging management review. The determination of whether a passive, in-scope components has a qualified life or specified replacement time period was based on a review of plant specific information including the EDB, maintenance programs, and procedures.

5. The components that are within the scope of license renewal (i.e., required to perform a license renewal system intended function) are identified.

6. Component intended functions for in-scope components are identified. The component intended functions identified are based on the guidance of NEI 95-10.

2.1.2.2 Civil Structures

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

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

1. Using appropriate drawings and other documentation, the evaluation boundaries associated with each civil/structural intended function are identified and documented. Evaluation boundaries between mechanical components, electrical components, and structures and structural components are coordinated between the discipline reviewers. The civil/structural components include items such as walls, supports, and non-current carrying electrical and instrumentation and control components, i.e., conduit, cables trays, electrical enclosures, panels, and related supports. Civil structural intended functions were identified during performance of the scoping process discussed in Subsection 2.1.1.

2. Based on a review of the civil/structural evaluation boundaries, SCs and commodity types within the intended function boundaries for the given structure are identified and documented. A generic list of commodity types was developed using guidance from Table 4.1-1 of NEI 95-10 [Reference 2.1-1], and potential

intended functions for the commodity types are identified. Structural components are identified using the PassPort EDB as a starting point. In the screening process, no differentiation is made between individual component and commodity types; they have been grouped together under common types. For example, supports for fire protection piping or building drains are grouped under the same commodity type as hangers for safety related auxiliary feedwater piping. Implementation of this methodology conservatively includes many components and commodities within the scope of license renewal that otherwise would be screened out as not supporting any system intended function.

3. The in-scope SCs that perform an intended function without moving parts or without a change in configuration or properties, i.e., the screening criterion of 10 CFR 54.21(a)(1)(i), or that are not subject to replacement based on a qualified life or specified time period, i.e., the screening criterion of 10 CFR 54.21(a)(1)(ii), are identified. Active/passive screening determinations are based on the guidance in Appendix B to NEI 95-10.

4. Component intended functions for in-scope SCs are determined and documented. The component intended functions are based on the guidance of NEI 95-10. Those SCs that have a component or commodity group intended function that supports a structure intended function are subject to an aging management review.

2.1.2.3 Electrical and I&C Systems

The method used to determine which electrical and I&C components are subject to an aging management review is based on the component commodity group approach consistent with the guidance of NEI 95-10 [Reference 2.1-1]. The primary difference between this method and that used for mechanical system and structures is the order in which the component screening steps are performed. This method was selected for use with the electrical and I&C components since most electrical and I&C components are active. This method provides the most efficient means for determining electrical and I&C components that require an aging management review.

The sequence of steps for identification of electrical and I&C components that require an aging management review is as follows:

1. Using the EDB, appropriate plant design drawings, and other documentation, identify the different types of electrical components within the electrical and I&C systems determined to be in scope for license renewal.

2. Organize the component types associated with the electrical and I&C systems within the scope of license renewal into commodity groupings, i.e., circuit breakers, cables, sensors. In general, grouping of component types will use the

guidance in NEI 95-10 regarding grouping of components based on similar functions.

3. The electrical and I&C component commodity groups that perform an intended function without moving parts or without a change in configuration or properties, i.e., the screening criterion of 10 CFR 54.21(a)(1)(i), are identified. Active/passive screening determinations are based on the guidance in Appendix B to NEI 95-10. Commodity groups that have passive functions may be subject to an aging management review and are identified by this step.

4. For the passive electrical and I&C component commodity groups, component commodity groups that are not subject to replacement based on a qualified life or specified time period, i.e., the screening criterion of 10 CFR 54.21(a)(1)(ii), are identified as requiring an aging management review. Commodity group components that are replaced based on qualified life determined in accordance with the Environmental Qualification Program are not subject to aging management review.

2.1.3 GENERIC SAFETY ISSUES

In accordance with the guidance in NEI 95-10 [Reference 2.1-1] and Appendix A.3 (Branch Technical Position RLSB-2) of the "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants" [Reference 2.1-9], review of NRC Generic Safety Issues (GSIs) as part of the license renewal process is required to satisfy a finding per 10 CFR 54.29. GSIs that involve issues related to license renewal aging management reviews or time-limited aging analysis evaluations are to be addressed in the License Renewal application. Based on NEI and NRC guidance, NUREG-0933 [Reference 2.1-10], and previous license renewal applicants, RNP has identified the following GSIs to be addressed:

1. GSI-168, Environmental Qualification of Electrical Equipment – This GSI is related to aging concerns for equipment that is subject to the environmental qualification requirements of 10 CFR 50.49. Environmental qualification evaluations of electrical equipment are identified as time-limited aging analyses for RNP. Accordingly, this GSI is addressed in Subsection 4.4.2.

2. GSI-190, Fatigue Evaluation of Metal Components for 60-year Plant Life – This GSI addresses fatigue life of metal components and was closed by the NRC [Reference 2.1-11]. In the closure letter, however, the NRC concluded that licensees should address the effects of reactor coolant environment on component fatigue life as aging management programs are formulated in support of license renewal. Accordingly, the issue of environmental effects on component fatigue life is addressed in Subsection 4.3.3.

3. GSI-191, Assessment of Debris Accumulation on PWR Sump Performance - The issue is the potential impact on emergency core cooling system performance caused by blockage of containment sump screens by debris, especially failed coatings. However, degradation of coatings inside containment is an issue under the CLB and has been addressed. Refer to the CP&L response to NRC Generic Letter 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment" [Reference 2.1-12]. RNP does not credit coatings to assure that intended functions of coated SCs are maintained; thus, this is not specifically a license renewal concern. Nor is the issue related to the 40-year term of the current operating license. Thus it is not a TLAA.

Based on the above, the GSI review determined that the issues involving either aging effects for SCs subject to an aging management review or TLAAs are already being evaluated for license renewal.

2.1.4 CONCLUSIONS

The methods described in Subsections 2.1.1 through 2.1.3 were used to identify the systems, structures, and components that are within the scope of license renewal and require an aging management review. The methods are consistent with, and satisfy the requirements of, 10 CFR 54.4 and 10CFR 54.21(a)(1).

2.1.5 REFERENCES

2.1-1 NEI 95-10, “Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule,” Revision 3, Nuclear Energy Institute, March 2001.

2.1-2 Letter from C. I. Grimes, NRC, to D. J. Walters, NEI, dated August 5, 1999: “License Renewal Issue No. 98-0082, Scoping Guidance.”

2.1-3 Letter from C. I. Grimes, NRC, to D. J. Walters, NEI, dated March 10, 2000: “License Renewal Issue No. 98-12, Consumables.”

2.1-4 CP&L letter to NRC, dated September 15, 1993: “Request for License Amendment, Pressure-Temperature Curves.”

2.1-5 CP&L letter to NRC, dated August 17, 1995: “Response to NRC Generic Letter 92-01, Revision 1, Supplement 1: Reactor Vessel Structural Integrity.”

2.1-6 CP&L letter to NRC, dated November 20, 1995: “Response to NRC Generic Letter 92-01, Revision 1, Supplement 1: Reactor Vessel Structural Integrity.”

2.1-7 CP&L letter to NRC, dated July 23, 1998: “Response to Request for Additional Information Regarding Reactor Pressure Vessel Integrity.”

2.1-8 CP&L letter to NRC, dated March 3, 1989: “Response to Station Blackout Rule.”

2.1-9 NUREG-1800, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” U. S. Nuclear Regulatory Commission, April 2001.

2.1-10 NUREG-0933, “A Prioritization of Generic Safety Issues,” Supplement 25, June 2001.

2.1-11 NRC Memorandum, A. Thadani, Director, Office of Nuclear Regulatory Research, to W. Travers, Executive Director of Operations: Closeout of Generic Safety Issue 190, “Fatigue Evaluation of Metal Components for 60 Year Plant Life,” dated December 26, 1999.

2.1-12 CP&L letter to NRC, dated November 12, 1998: “Response to NRC Generic Letter 98-04, “Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment.”

2.2 PLANT LEVEL SCOPING RESULTS

The RNP Integrated Plant Assessment (IPA) methodology consists of three distinct processes: scoping, screening, and aging management reviews. This section provides the results of application of the scoping process described in Section 2.1.1.

Tables 2.2-1, 2.2-2, and 2.2-3 provide the results of applying the license renewal scoping criteria to mechanical systems, structures, and electrical/I&C systems. If a system or structure, in whole or in part, meets one or more of the license renewal scoping criteria, the system or structure is considered to be within the scope of license renewal. Also, included in the tables are references to the sections in the application that discuss screening results for in-scope systems and structures.

Figure 2.2-1 provides a layout view of RNP and identifies the major in-scope plant structures. This figure is based on UFSAR Figure 1.2.2-1.

TABLE 2.2-1 - LICENSE RENEWAL SCOPING RESULTS FOR MECHANICAL SYSTEMS

System Name	System in License Renewal Scope	Screening Results Application Subsection
Auxiliary Boiler/Steam System (AS)	No	
Auxiliary Feedwater (AFW)	Yes	2.3.4.9
Boric Acid Evaporator	No	
Chemical and Volume Control System	Yes	2.3.3.4
Circulating Water System	Yes	2.3.4.12
Circulating Water Treatment System	No	
Component/Closed Cooling Water System	Yes	2.3.3.3
Condensate Polishing Demineralizer System	No	
Condensate System (C)	Yes	2.3.4.10
Condenser Ball Cleaning System	No	
Condenser Vacuum System	No	
Containment Pressure Relief System (CVPRS)	Yes	2.3.2.5
Containment Purge System (CV PUR)	Yes	2.3.3.10
Containment Vacuum Breaker System (CV VBS)	Yes	2.3.2.5
Containment Vapor and Pressure Sampling System	Yes	2.3.3.1
Dedicated Shutdown Diesel Generator (DSD)	Yes	2.3.3.17
Diesel Generator System (DIESEL)	Yes	2.3.3.16
Electro-Hydraulic Control System	Yes	2.3.4.2
Emergency Diesel Generator Car Dox System (CARDOX)	Yes	2.3.3.15
EOF/TSC Security Emergency Diesel Gen. (EOF DG)	Yes	2.3.3.18
Exhaust Hood Spray System	No	
Extraction Steam System (ES)	Yes	2.3.4.4
Feedwater System (FW)	Yes	2.3.4.8
Fire Protection CO_2 System (CO2)	Yes	2.3.3.15
Fuel Oil System (FO)	Yes	2.3.3.19
Gaseous Waste Processing System	No	
Generator Gas System (GGS)	No	
Generator System	No	
Gland Seal System	No	
Halon Supply System (HALON)	Yes	2.3.3.15
Heater Vents, Drains, and Level Control	No	
HVAC Auxiliary Building (HVAC)	Yes	2.3.3.12
HVAC Containment Building System (HVAC)	Yes	2.3.2.4
HVAC Control Room Area (HVAC)	Yes	2.3.3.13
HVAC Fuel Handling Building (HVAC)	Yes	2.3.3.14
HVAC Radwaste Building (HVAC)	No	
HVAC Turbine Building (HVAC)	No	

TABLE 2.2-1 (continued) LICENSE RENEWAL SCOPING RESULTS FOR MECHANICAL SYSTEMS

System Name	System in License Renewal Scope	Screening Results Application Subsection
Hydrogen Seal Oil System (HSO)	No	
Hydrogen Supply Systems (H2 SUP)	No	
Instrument Air System (IA)	Yes	2.3.3.5
Isolation Valve Seal Water System (IVSW)	Yes	2.3.2.5
Liquid Waste Processing System (WDS)	Yes	2.3.2.5
Main Steam (MS)	Yes	2.3.4.5
Nitrogen Supply/Blanketing System (N2 SUP)	Yes	2.3.3.6
Oil Drains/Waste Oil Storage System (ODS/WOS)	No	
Penetration Pressurization Local Leak Rate Test	Yes	2.3.2.5
Post Accident Hydrogen System	Yes	2.3.2.5
Post Accident Sampling System	Yes	2.3.3.1
Potable Water System (PWS)	No	
Primary and Demineralized Water Makeup System (DW)	Yes	2.3.3.8
Primary Sampling System (PS)	Yes	2.3.3.1
Process/Area Radiation Monitoring System (RMS)	Yes	2.3.2.5
Radioactive Equipment Drains System (REDS)	Yes	2.3.3.7
Radwaste Solidification System (RWSS)	No	
Reactor Coolant System (RC)	Yes	2.3.1
Reactor Vessel and Internals System	Yes	2.3.1.4, 2.3.1.5
Residual Heat Removal System (RHR)	Yes	2.3.2.1
Rod Drive Cooling System (RDCS)	Yes	2.3.3.11
Rx Vessel Level Instrumentation / ICCM System	Yes	2.3.1.7
Safety Injection System	Yes	2.3.2.2, 2.3.2.3
Service Air System (SA)	Yes	2.3.2.5
Service Water System (SW)	Yes	2.3.3.2
Site Fire Protection System (SFPS)	Yes	2.3.3.15
Spent Fuel Cask Handling System (SFCASK)	No	
Spent Fuel Pool Cooling System (SFPCS)	Yes	2.3.3.9
Steam Cycle Sampling (Secondary)	Yes	2.3.4.7
Steam Generator	Yes	2.3.1.6
Steam Generator Blowdown System	Yes	2.3.4.6
Steam Generator Chemical Addition System	Yes	2.3.4.11
Turbine System	Yes	2.3.4.1
Turbine-Generator Lube Oil System	Yes	2.3.4.3

TABLE 2.2-2 - LICENSE RENEWAL SCOPING RESULTS FOR STRUCTURES

Structure Name	Structure in License Renewal Scope	Screening Results Application Subsection
Bldg 105: Unit 1 Plant Management	No	
Bldg 110: Administrative and Control Building	No	
Bldg 115: Unit 1 Tech Training	No	
Bldg 150: Gas Shed	No	
Bldg 155: Unit 1 Shop & Stock Room	No	
Bldg 160: Unit 1 Service Building	No	
Bldg 170: Security Building	No	
Bldg 175: Switchyard Relay Building	No	
Bldg 200: Reactor Containment Building	Yes	2.4.1
Bldg 205: Reactor Auxiliary Building	Yes	2.4.2.1
Bldg 210: Radwaste Building	Yes	2.4.2.5
Bldg 230: Contaminated Equipment Storage Building	No	
Bldg 250: New Outage Contaminated Storage Building	No	
Bldg 300: West Personnel Access Portal (PAP)	Yes	2.4.2.12
Bldg 310: EOF/TSC Security Emergency Diesel Generator Building	Yes	2.4.2.8
Bldg 315: Outage Management/Work Control Building	No	
Bldg 320: Modular Unit	No	
Bldg 325: Operations and Maintenance Facility	No	
Bldg 330: RCA Access Processing Building	No	
Bldg 335: Secondary Sampling Building	No	
Bldg 340: Dedicated Shutdown (DS) Diesel Generator Building	Yes	2.4.2.4
Bldg 345: Condensate Polishing Building	No	
Bldg 350: Turbine Building	Yes	2.4.2.3
Bldg 355: East Personnel Access Portal (PAP)	No	
Bldg 360: Auxiliary Boiler “C” Building	No	
Bldg 365: N2 and H2 Storage Shed	No	
Bldg 370: Oil Dispensing Building	No	
Bldg 375: Environmental and Radiation Control (E&RC) Building	No	
Bldg 385: Steam Generator Tomb	No	
Bldg 400: Chemical Storage building	No	
Bldg 405: Bulk Storage Warehouse	No	
Bldg 408: Administration Facility	No	
Bldg 410: Simulator Building	No	
Bldg 410: Tech Support Center/Emergency Operations Facility (TSC/EOF)	No	
Bldg 415: Weld Test Shop	No	

TABLE 2.2-2 (continued) LICENSE RENEWAL SCOPING RESULTS FOR STRUCTURES

Structure Name	Structure in License Renewal Scope	Screening Results Application Subsection
Bldg 420: EAPBX Building	No	
Bldg 425: Spare RCP Motor Storage	No	
Bldg 430: Contaminated Equipment Warehouse	No	
Bldg 435: General Employee Training (G.E.T.) Building	No	
Bldg 440: Channel Head Mockup	No	
Bldg 450: Visitor's Center	No	
Bldg N/A: Fuel Oil Unloading/Transfer Area	No	
Bldg N/A: Canal Outlet Structure	Yes	2.4.2.9
Bldg N/A: Construction Air Compressor	No	
Bldg N/A: Degasifier Foundation and Support Structure	No	
Bldg N/A: Discharge Block	No	
Bldg N/A: Discharge Canal	Yes	2.4.2.9
Bldg N/A: DSS-Main-XFMR	Yes	2.4.2.12
Bldg N/A: Intake Block	No	
Bldg N/A: Intake Structure	Yes	2.4.2.6
Bldg N/A: Lift Rig Storage Building	No	
Bldg N/A: Miscellaneous Yard Structures	Yes	2.4.2.12
Bldg N/A: North Service Water Header Enclosure	Yes	2.4.2.7
Bldg N/A: Picnic Area	No	
Bldg N/A: Plant Vent Stack	No	
Bldg N/A: Radioactive Materials Area Storage Slab	No	
Bldg N/A: Rest Rooms	No	
Bldg N/A: Seal Well #2	Yes	2.4.2.9
Bldg N/A: Security Lighting	Yes	2.4.2.12
Bldg N/A: Unit 1 Plant Stack	No	
Bldg N/A: Vehicle Barrier Structures	No	
Bldg N/A: Waste Condensate Pump room	No	
Bldg N/A: E&RC Lab/Waste Sump Tank	No	
Bldg N/A: HEPA/Vacuum Cleaner Storage Building	No	
Bldg N/A: Waste Evaporator Cooling Tower and Recirculation Pump Foundation	No	
Bldg T-331: Work Control Building	No	
Bldg: CV Access Extension	No	
Bldg N/A: Miscellaneous Concrete Slabs	No	
Bldg 215, 220, and 225: Fuel Handling Building	Yes	2.4.2.2
Bldg N/A: Caustic Tank Enclosure	No	
Concrete Trenches	Yes	2.4.2.12
Electrical Manholes and Duct Banks	Yes	2.4.2.12

TABLE 2.2-2 (continued) LICENSE RENEWAL SCOPING RESULTS FOR STRUCTURES

Structure Name	Structure in License Renewal Scope	Screening Results Application Subsection
Instrument Air Compressor D Receiver Air Tank	No	
Pipe Restraint Tower	Yes	2.4.2.11
Railroad Tracks, Switches and Foundations	No	
Refueling System (REFUEL)	Yes	2.4.2.13
Reservoir and Dam	Yes	2.4.2.10
Reservoir Blowdown System	No	
Roads	No	
Sewage Treatment System (SEWAGE)	No	
Spent Fuel Cask Decontamination Pump	No	
Storm Drain System	No	
Switchyard and Transformers	No	
Tank Foundation: Condensate Polish Neutralizer	No	
Tank Foundation: Condensate Storage	Yes	2.4.2.12
Tank Foundation: Contaminated Waste Oil #1	No	
Tank Foundation: Contaminated Waste Oil #2	No	
Tank Foundation: Contaminated Waste Oil #3	No	
Tank Foundation: Demineralized Water Acid	No	
Tank Foundation: Demineralized Water Caustic	No	
Tank Foundation: Demineralized Water Neutralizer	No	
Tank Foundation: Diesel Fire Pump Fuel Oil	Yes	2.4.2.12
Tank Foundation: Diesel Generator Fuel Oil Storage	Yes	2.4.2.12
Tank Foundation: DS Diesel Fuel Oil	Yes	2.4.2.12
Tank Foundation: Hypo Chlorate	No	
Tank Foundation: Monitor A	No	
Tank Foundation: Monitor B	No	
Tank Foundation: Primary Water Storage	Yes	2.4.2.12
Tank Foundation: Refueling Water Storage Tank	Yes	2.4.2.12
Tank Foundation: Spent Fuel Cask Decontamination Tank	No	
Tank Foundation: Steam Generator Blowdown Tank	Yes	2.4.2.12
Tank Foundation: Waste Condensate Tank C	No	
Tank Foundation: Waste Condensate Tank D	No	
Tank Foundation: Waste Condensate Tank E	No	
Tank Foundation: Water Conservation	No	
Tank Foundation: Unit 1 IC Fuel Oil Storage Facility	Yes	2.4.2.12

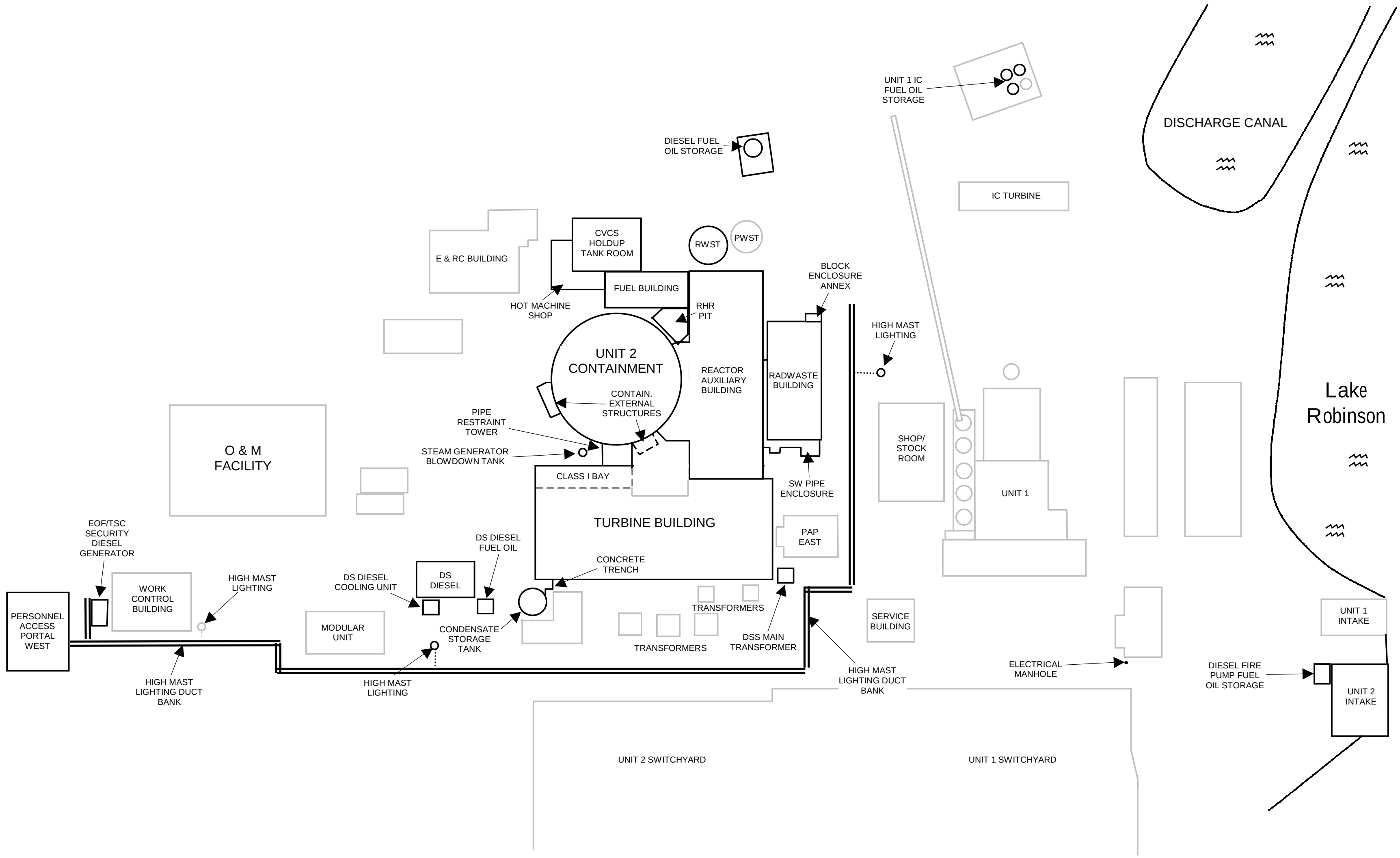
TABLE 2.2-3 LICENSE RENEWAL SCOPING RESULTS FOR ELECTRICAL/I&C SYSTEMS

System Name	System in License Renewal Scope	Screening Results Application Subsection
125 V DC Battery/Charger/Distribution System	Yes	2.5
208/120 VAC Distribution System	Yes	2.5
4 KV AC Distribution System (4KV)	Yes	2.5
480 V AC Distribution System (480 VAC)	Yes	2.5
Annunciator Systems (ANNUN)	Yes	2.5
Axial Power Distribution Monitoring System	No	
Card Reader/Access Control System (SEC)	No	
Closed Circuit T.V. System (SEC)	No	
Dedicated Shutdown System (DSS)	Yes	2.5
Emergency Communications System	No	
Emergency DC Lighting System (ELS)	Yes	2.5
ERFIS Computer System (ERFIS)	Yes	2.5
Excore Nuclear Instrument System	Yes	2.5
FCC Licensed Portable Radios (P RAD)	Yes	2.5
Fire Alarm Console Computer (FAC)	Yes	2.5
Freeze Protection System (FP)	No	
General Electrical Distribution & Hardware (GED)	No	
Generator Exciter System	No	
Generator Isolated Phase Bus System (GIPBS)	No	
Heat Tracing System (HT)	No	
Incore Nuclear Instrument System (INC)	Yes	2.5
Intrusion Devices (SEC)	Yes	2.5
Key Control and Hardware (KEY)	No	
Lightning Protection System (LPS)	No	
Load Frequency Control System (LFCS)	No	
Loose Parts Monitoring System (LPMS)	No	
Main Control Board (RTGB)	Yes	2.5
Met Tower System	No	
Meteorological and Environmental Systems	No	
Microwave System (MICRO)	No	
Normal AC Lighting System	Yes	2.5
Offsite/Non-Power Block Electrical Power System	No	
PA System (PA)	No	
PABX and Teleconferencing System (PABX)	No	
Physical Search System (SEC)	No	
Post Accident Monitoring System	Yes	2.5
Reactor Protection and Safeguards System (RPS)	Yes	2.5
Reactor Simulator	No	

TABLE 2.2-3 (continued) LICENSE RENEWAL SCOPING RESULTS FOR ELECTRICAL/I&C SYSTEMS

System Name	System in License Renewal Scope	Screening Results Application Subsection
RNP Artemis Mini-Computer (ARTEMIS)	No	
Rod Control System	Yes	2.5
Rod Position Indication System	No	
Security Communication System (SEC)	No	
Security Computer System (SEC)	No	
Security Fencing and Gates (SEC)	No	
Security Lighting System (SEC)	Yes	2.5
Site Fire Detection System (SFDS)	Yes	2.5
Site Grounding System (S GND)	No	
Sound Powered Telephone System (S TELE)	No	
Switchyard and Transformer System (SWTR)	No	

FIGURE 2.2-1 RNP PLANT STRUCTURES



2.3 SCOPING AND SCREENING RESULTS – MECHANICAL SYSTEMS

The determination of mechanical systems within the scope of license renewal is made by initially identifying RNP mechanical systems and then reviewing them to determine which ones satisfy one or more of the criteria contained in 10 CFR 54.4. This process is described in Section 2.1, and the results of the mechanical systems scoping review are contained in Section 2.2.

Section 2.1 also provides the methodology for determining the components within the scope of 10 CFR 54.4 that meet the requirements contained in 10 CFR 54.21(a)(1). The components that meet these screening requirements are identified in this section. These identified components consequently require an aging management review for license renewal.

The screening results are provided below in four sections:

- Reactor Vessel, Internals, and Reactor Coolant System,
- Engineered Safety Feature Systems,
- Auxiliary Systems, and
- Steam and Power Conversion Systems.

2.3.1 REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

The Reactor Vessel, Internals, and Reactor Coolant System consist of the systems and components designed to contain and support the nuclear fuel, contain the reactor coolant, and transfer the heat produced in the reactor to the steam and power conversion systems for the production of electricity. The Reactor Vessel, Internals, and Reactor Coolant System are in the scope of license renewal, because they contain SCs that are safety-related and are relied upon to remain functional during and following design basis events; SCs that are part of the Environmental Qualification Program; and SCs that are relied on during postulated fires, station blackout, and pressurized thermal shock events. The Reactor Vessel, Internals, and Reactor Coolant System for RNP is described in RNP UFSAR Sections 5.1, 5.3 and 5.4. The following components are included in this Subsection:

1. Reactor Coolant System Piping
2. Pressurizer
3. Reactor Coolant Pumps
4. Reactor Pressure Vessel
5. Reactor Vessel Internals
6. Steam Generators
7. Reactor Vessel Level Instrumentation/Inadequate Core Cooling Monitor System

Portions of the Incore Nuclear Instrumentation System (thimble tube pressure boundary), Chemical and Volume Control System (regenerative heat exchanger and Reactor Coolant Pump seal injection and return lines), Safety Injection System (piping and valves), Residual Heat Removal System (piping and valves), Reactor Vessel Level Instrumentation/Inadequate Core Cooling Monitor System, and the Primary Sampling System (piping and isolation valves) are included in this subsection, because they form part of the Class 1 Reactor Coolant System boundary.

Class 1 as used in this report means the Safety Class 1 definition in accordance with ANSI N18.2-1973 and ANSI N18.2a-1975. The Class 1 boundaries are shown on the system flow diagrams by means of flags.

The license renewal evaluation boundaries for the Reactor Vessel, Internals, and Reactor Coolant System are shown on the following flow diagrams. (Flow diagrams have been submitted separately for information only.)

Reactor Coolant	5379-1971LR Sheet 1
	5379-1971LR Sheet 2
Safety Injection[1]	5379-1082LR Sheet 4
Residual Heat Removal[1]	5379-1484LR Sheet 1
Chemical and Volume Control[1]	5379-685LR Sheet 1
Primary Sampling System[1]	5379-353LR Sheet 1
Reactor Vessel Level Instrumentation	HBR2-09067LR Sheet 1

Note 1: These systems are not part of the Reactor Coolant System; however, they have components that are part of the Class 1 boundary.

2.3.1.1 Reactor Coolant System Piping

Reactor Coolant System piping consists of piping (including fittings, branch connections, thermal sleeves, tubing, and thermowells), pressure retaining parts of valves, and bolted closures and connections. Reactor Coolant System piping is presented in two parts: (1) Class 1 piping, and (2) Non-Class 1 piping. The design code for the Reactor Coolant System piping is ASA B31.1-1955. The majority of Reactor Coolant System piping was designed to ASA B31.1; however, some small-bore piping was designed to ASME Boiler and Pressure Vessel Code, Section III.

2.3.1.1.1 Class 1 Piping

Class 1 piping includes the Reactor Coolant System main loop piping; pressurizer surge, spray, and safety and relief valve inlet lines; vents, drains, and instrument lines. Portions of ancillary systems attached to the Reactor Coolant System are also Class 1. Ancillary systems attached to the Reactor Coolant System include the Safety Injection System, Residual Heat Removal System, Chemical and Volume Control System, and Primary Sampling System. A description of the Reactor Coolant System piping is provided in UFSAR Sections 5.1 and 5.4.3.

The Regenerative Heat Exchanger is part of the Chemical and Volume Control System; however, it is addressed in this subsection because it is a part of the Reactor Coolant Class 1 boundary. The Regenerative Heat Exchanger is designed to recover the heat from the letdown stream from the Reactor Coolant System by reheating the charging stream from the charging pumps to the Reactor Coolant System during normal operation. The letdown stream flows thorough the shell of the Regenerative Heat Exchanger and the charging stream flows through the tubes. The Regenerative Heat Exchanger was designed and manufactured in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class C for both the tube side and shell side. The Regenerative Heat Exchanger is described in UFSAR Section 9.3.4.

Portions of the Primary Sampling System are designed to sample the Reactor Coolant System. Sample locations include the Reactor Coolant System loop 2 and loop 3 hot legs, Pressurizer liquid space, and Pressurizer steam space. Each of these connections to the Reactor Coolant System has a remotely operated isolation valve. The Class 1 portions of the Reactor Coolant System extend to these isolation valves. The Primary Sampling System is described in UFSAR Section 9.3.2.

During construction, the Class 1 piping was insulated in accordance with the applicable Westinghouse Equipment Specification CPL-RNP-MI-1. The specification included requirements regarding testing and compliance of insulation materials with regard to potential to induce cracking of austenitic stainless steel. External corrosion resistant surfaces in the reactor coolant system are insulated with low halide or halide free insulating material. Procurement of insulation is done in accordance with administrative controls which ensure that all the insulation used on the reactor coolant piping is low halide or halide free, such that the piping is not susceptible to stress corrosion cracking initiated by halides.

2.3.1.1.2 Non-Class 1 Piping

Several non-Class 1 piping components in the Reactor Coolant System are within the scope of license renewal for RNP:

1. The Pressurizer Relief Tank.

2. Pressurizer relief and safety valve discharge lines to the Pressurizer Relief Tank.

3. Auxiliary lines supporting RCS and PRT functions including containment isolation valves in those lines.

4. Reactor Vessel Level Instrumentation lines downstream of Class 1 boundary Bellows.

The Pressurizer Relief Tank is located inside containment. The Pressurizer Relief Tank normally contains water in a predominantly nitrogen atmosphere. Steam is discharged from relief and safety valves of the Reactor Coolant System into the Pressurizer Relief Tank where it is condensed and cooled by mixing with the water. The Pressurizer Relief Tank also collects leakage and liquid from various system pressure relief valves located inside the containment. The Pressurizer Relief Tank was designed to the ASME Boiler and Pressure Vessel Code, Section III, Class C.

2.3.1.2 Reactor Coolant Pumps

Each of the three reactor coolant loops contains a vertically mounted, single stage centrifugal reactor coolant pump that employs a controlled leakage seal assembly. The reactor coolant pumps provide the motive force for circulating the reactor coolant through the reactor core, piping, and steam generators. The RNP Reactor Coolant Pump casings were designed in accordance with ASME Boiler and Pressure Vessel

Code, Section III, Class A. A description of the reactor coolant pumps is provided in UFSAR Section 5.4.1.

The portion of the reactor coolant pump rotating element above the pump coupling, including the electric motor and the flywheel, is not subject to aging management review in accordance with 10 CFR 54.21(a)(1)(i). Reactor coolant pump seals are not subject to an aging management review for the following reasons:

1. Seal leakoff is closely monitored in the control room, and high leakoff flow rate is alarmed as an abnormal condition requiring corrective action,

2. The reactor coolant pump seal package and its constituent parts are periodically overhauled on a schedule established by the Preventive Maintenance Program. The seals are inspected and parts are replaced, as required.

Plant operating experience with pump seal performance has demonstrated the effectiveness of these activities.

2.3.1.3 Pressurizer

The pressurizer is a vertical cylindrical vessel containing electric heaters in its lower head and a water spray nozzle in its upper head. Sources of heat to the Reactor Coolant System are interconnected by piping to the pressurizer with no intervening isolation valves; the pressurizer lower head is connected to the Reactor Coolant System by the surge line. Pressure relief protection for the Reactor Coolant System is provided on the pressurizer. Overpressure protection consists of three code safety valves and two power operated relief valves. Piping attached to the pressurizer is Class 1 up to and including the safety and relief valves. The Pressurizer was designed and fabricated in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class A.

2.3.1.4 Reactor Pressure Vessel

The reactor vessel consists of the cylindrical vessel shell, lower vessel head, closure head, nozzles, interior attachments, and associated pressure retaining bolting. The vessel is fabricated of a low-carbon alloy steel with austenitic stainless steel cladding on all surfaces exposed to the reactor coolant fluid. Coolant flow enters the reactor vessel through three inlet nozzles in a plane just below the vessel flange and above the core. The coolant flows downward through the annular space between the vessel wall and the core barrel into a plenum at the bottom of the vessel where it reverses direction, passes up through the core into the upper plenum and then flows out of the vessel though three exit nozzles located on the same plane as the inlet nozzles. The Reactor Pressure Vessel was designed according to the 1965 Edition of the ASME Boiler and Pressure Vessel Code, Section III, Class A. A description of the reactor vessel is provided in UFSAR Section 5.3.

Control rod drive mechanism housings are attached to flanged nozzles, which penetrate the closure head. The active portions of the control rod drive mechanisms do not require an aging management review per 10 CFR 54.21(a)(1)(i). The control rod drive mechanisms are described in UFSAR Section 3.9.4.

Bottom-mounted instrumentation penetrates the reactor vessel lower head. The bottom head instrumentation support columns inside the reactor vessel, guide tubes attached to the bottom of the vessel, and the seal table inside containment are designed to allow instrumentation flux thimble tubes to be inserted into the reactor core. Neutron flux detectors that traverse the thimble tubes and thermocouples mounted in the double-wall thimble tubes provide the capability of monitoring core flux distribution and core exit temperature. The function of the thimble tubes is discussed in Subsection 2.3.1.5. The bottom-mounted instrumentation is described in UFSAR Section 7.7.1.5.

2.3.1.5 Reactor Vessel Internals

The reactor vessel internals are designed to support, align, and guide the core components, and to support and guide incore instrumentation. The reactor vessel internals consists of two basic assemblies: an upper internals assembly that is removed during each refueling operation to obtain access to the reactor core, and a lower internals assembly that can be removed, if desired, following a complete core unload. The reactor vessel internals are described in UFSAR Section 3.9.5.

The lower internals assembly is supported in the vessel by resting on a ledge in the vessel head-mating surface and closely guided at the bottom by radial support/clevis assemblies. The upper internals assembly is clamped at this same ledge by the reactor vessel head. The bottom of the upper internals assembly is closely guided by the core barrel alignment pins of the lower internals assembly.

The lower internals are comprised of the core barrel, thermal shield, core baffle assembly, lower core plate, intermediate diffuser plate, bottom support plate, and supporting structures. The upper internals package (upper core support structure) is a rigid member composed of the top support plate and deep beam sections, support columns, control rod guide tube assemblies, and the upper core plate. Upon upper internals assembly installation, the last three parts are physically located inside the core barrel.

The Reactor Pressure Vessel Internals for RNP were designed prior to the creation of ASME Boiler and Pressure Vessel Code, Section III, Subsection NG. The internals were designed using internal Westinghouse design criteria that effectively evolved to become the original NG criteria. The reactor vessel internals were designed using the allowable stress levels of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, as a guide. Most of the major internal structures are fabricated from A240 Type 304 stainless steel. However, the lower support forging is A182 Type 304.

Support column assemblies are stainless steel, A213, A276, or A182 Type 304. Control rod guide tube assemblies are A240, A249, and A479 Type 304. CASS (A351 Gr. CF8) components are the upper support tube base, lower support plate columns, and bottom mounted instrumentation column cruciforms. The holddown spring is Type 304 stainless steel. The secondary core support assembly is fabricated from A240 and A479 Type 304 stainless steel; the energy absorber is made from A182 Type 304 or A336 Type F8. Clevis inserts are Alloy 600 with Inconel X-750 insert bolts. Other bolts and alignment pins are of Type 316 or Type 347 stainless steel, except the head and vessel alignment pins and radial support keys which are Type 304.

The in-core instrumentation includes in-core flux guide thimbles to permit the insertion of movable detectors for measurement of the neutron flux distribution within the reactor core. Movable miniature neutron flux detectors are available to scan the active length of selected fuel assemblies to provide remote reading of the relative three-dimensional flux distribution. The thimbles are inserted into the reactor core through guide tubes, or conduits, extending from the bottom of the reactor vessel through the concrete shield area and then up to a thimble seal table. Since the movable detector thimbles are closed at the leading (reactor) end, they are dry inside. The thimbles thus serve as a pressure barrier between the reactor coolant pressure and the atmosphere. Mechanical seals between the retractable thimbles and the conduits are provided at the seal table. The bottom-mounted instrumentation is described in UFSAR Section 7.7.1.5. The guide tubes and seal table are shown on UFSAR Figure 1.2.2-4.

2.3.1.6 Steam Generators

The Steam Generators remove heat from the Reactor Coolant System by converting Feedwater into steam. The Steam Generators provide sufficient capacity to remove heat during normal operations and following postulated accidents and transients. An integral flow restrictor limits the flow rate of steam from a Steam Generator following a postulated steam line break accident. Steam Generator level instrumentation is provided to assure the heat removal capability is maintained following an accident.

There are three steam generators installed, one in each of the three RNP reactor coolant loops. Each steam generator is a vertical shell-and-tube heat exchanger that transfers heat from a single-phase fluid at high temperature and pressure (the reactor coolant) in the tube side, to a two-phase (steam-water) mixture at lower temperature and pressure in the shell side. The original Steam Generators (primary side) were designed to the 1965 Edition of the ASME Boiler and Pressure Vessel Code, Class A, through Summer 1965 Addenda. The shell side was designed according to the ASME Boiler and Pressure Vessel Code, Section III, Class C. The replacement Steam Generator Lower Assemblies were designed in accordance with the 1980 Edition of the ASME Boiler and Pressure Vessel Code, Section III, Class A, through Winter 1980 Addenda. UFSAR Section 5.4.2 and UFSAR Section 10.3 provide a description of the RNP steam generators.

Reactor coolant enters and exits the tube side of each steam generator through nozzles located in the lower hemispherical head. The Reactor Coolant System fluid flows through inverted U-tubes connected to the tube sheet. The lower head is divided into inlet and outlet chambers by a vertical partition plate extending from the lower head to the tube sheet. The steam-water mixture is generated on the secondary, or shell side, and flows upward through moisture separators and dryers to the outlet nozzle at the top of the vessel providing essentially dry, saturated steam. Manways and inspection ports are provided to permit access to both sides of the lower head and to the U-tubes and moisture separating equipment on the shell side of the steam generators.

The steam generator support system includes hydraulic snubbers. The snubbers are considered to be structural components; however, portions of the hydraulic equipment for each steam generator (manifold, hydraulic control unit, flex hoses, piping, reservoir) are subject to an aging management review to assure their pressure boundary integrity is maintained.

Lower assemblies of the steam generators, including the lower shell, tubes, and tube sheet, were replaced in 1984.

2.3.1.7 Reactor Vessel Level Instrumentation

An Inadequate Core Cooling Instrumentation System is provided to detect the approach to inadequate reactor core cooling and assess the adequacy of responses taken to restore core cooling. The system consists of three subsystems: Reactor Vessel Level Instrumentation System (RVLIS), Core Exit Thermocouple System (CETS), and the Core Cooling Monitor System (CCMS). Portions of the RVLIS consist of mechanical components that are part of the Reactor Coolant System pressure boundary or part of the containment pressure boundary.

2.3.1.8 Summary

The Reactor Vessel, Internals, and Reactor Coolant System are in the scope of license renewal because they contain:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs that are part of the Environmental Qualification Program

3. SCs that are relied on during postulated fires, station blackout, and pressurized thermal shock events

Table 2.3-1 below identifies the Reactor Vessel, Internals, and Reactor Coolant System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity group.

TABLE 2.3-1 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: REACTOR VESSEL, INTERNALS AND REACTOR COOLANT SYSTEM

Component/Commodity	Intended Function	AMR Results
Reactor Coolant System Piping		
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 22 Table 3.1-1, Item 26
Flow Orifices/Elements	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide flow restriction (throttle).	Table 3.1-1, Item 1 Table 3.1-1, Item 6 Table 3.1-2, Item 17
Pressurizer Relief Tank	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 26 Table 3.1-2, Item 13
RCP Thermal Barrier Heat Exchanger Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 24
Reactor Coolant Pumps	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 10 Table 3.1-1, Item 19 Table 3.1-1, Item 22
Valves, Piping, Tubing and Fittings (includes thermowells and thermal sleeves)	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components. Provide heat transfer. Provide insulation/thermal resistance.	Table 3.1-1, Item 1 Table 3.1-1, Item 6 Table 3.1-1, Item 10 Table 3.1-1, Item 19 Table 3.1-1, Item 20 Table 3.1-1, Item 22 Table 3.1-1, Item 24 Table 3.1-1, Item 26 Table 3.1-2, Item 2 Table 3.1-2, Item 8 Table 3.1-2, Item 17 Table 3.1-2, Item 18
Pressurizer		
Pressurizer Shell/Heads	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 24 Table 3.1-2, Item 2
Pressurizer Shell/Heads (outer surfaces)	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.1-1, Item 26

TABLE 2.3-1 (continued) COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: REACTOR VESSEL, INTERNALS AND REACTOR COOLANT SYSTEM

Component/Commodity	Intended Function	AMR Results
Pressurizer (continued)		
Pressurizer Spray Nozzle	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-2, Item 2
Pressurizer Surge Nozzle	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 24 Table 3.1-2, Item 2
PZR Thermal Sleeves, Instrument Nozzle, Safe End	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide insulation/thermal resistance.	Table 3.1-1, Item 1 Table 3.1-1, Item 24 Table 3.1-2, Item 2
PZR Spray and Surge Nozzles	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 24 Table 3.1-2, Item 2
Pressurizer Instrument Nozzles	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 24 Table 3.1-2, Item 2
Pressurizer Safe Ends	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 24 Table 3.1-2, Item 2
Pressurizer Manway Insert	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide corrosion protection to pressure boundary subcomponents.	Table 3.1-1, Item 1 Table 3.1-1, Item 24 Table 3.1-2, Item 2
Pressurizer Manway Cover and) Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 26
Pressurizer Manway Cover Bolts/Studs	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 22
Pressurizer Immersion Heater Sheaths/Sleeves	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 24 Table 3.1-2, Item 2
Pressurizer Support Skirt and Flange	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 26
Reactor Vessel And Internals System		
Dome	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 26
Dome Cladding	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 24 Table 3.1-2, Item 2
Head Flange	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 26
Head Flange Cladding	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 24 Table 3.1-2, Item 2

TABLE 2.3-1 (continued) COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: REACTOR VESSEL, INTERNALS AND REACTOR COOLANT SYSTEM

Component/Commodity	Intended Function	AMR Results
Reactor Vessel And Internals System (continued)		
Stud Assembly	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 26 Table 3.1-1, Item 34 Table 3.1-2, Item 1
CRD Head Penetration Nozzle	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 23 Table 3.1-2, Item 2
CRD Head Penetration Pressure Housing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 24 Table 3.1-2, Item 2
Nozzles - Inlet	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 3 Table 3.1-1, Item 4 Table 3.1-1, Item 26
Nozzles - Inlet Cladding	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item1 Table 3.1-1, Item 24 Table 3.1-2, Item 2
Nozzles - Outlet	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 3 Table 3.1-1, Item 4 Table 3.1-1, Item 26
Nozzles - Outlet Cladding	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 24 Table 3.1-2, Item 2
Nozzles - Safe End (Inlet)	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 24 Table 3.1-2, Item 2
Nozzles - Safe End (Outlet)	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 24 Table 3.1-2, Item 2
Vessel Shell - Upper Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 3 Table 3.1-1, Item 4 Table 3.1-1, Item 26
Vessel Shell - Upper Shell Cladding	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 24 Table 3.1-2, Item 2
Vessel Shell - Inter. And Lower Shell Cladding	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 24 Table 3.1-2, Item 2
Vessel Shell - Inter. Shell and Lower Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 3 Table 3.1-1, Item 4 Table 3.1-1, Item 26

TABLE 2.3-1 (continued) COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: REACTOR VESSEL, INTERNALS AND REACTOR COOLANT SYSTEM

Component/Commodity	Intended Function	AMR Results
Reactor Vessel And Internals System (continued)		
Vessel Shell - Vessel Flange	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 26 Table 3.1-1, Item 28
Vessel Shell - Vessel Flange Cladding	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 24 Table 3.1-2, Item 2
Vessel Shell - Bottom Head	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 26
Vessel Shell - Bottom Head Cladding	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 24 Table 3.1-2, Item 2
Core Support Pads (integral to pressure vessel wall)	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-2, Item 2 Table 3.1-2, Item 9
Penetrations - Instrumentation Tubes (Bottom Head)	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-2, Item 2 Table 3.1-2, Item 10
Penetrations - Head Vent Pipe	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 23 Table 3.1-2, Item 2
Penetrations – Instrumentation Tubes (Top Head)	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 23 Table 3.1-2, Item 2
Upper Support Plate	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 8 Table 3.1-1, Item 33
Upper Support Column	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 8 Table 3.1-1, Item 33
Upper Support Tube Base	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 8 Table 3.1-1, Item 33 Table 3.1-2, Item 14
Upper Support Column Bolts	Provide structural support to safety-related components.	Table 3.1-1, Item 8 Table 3.1-1, Item 33 Table 3.1-2, Item 15
Upper Core Plate	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 8 Table 3.1-2, Item 33
Upper Core Plate Alignment Pins	Provide structural support to safety-related components.	Table 3.1-1, Item 8 Table 3.1-1, Item 28 Table 3.1-2, Item 33

TABLE 2.3-1 (continued) COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: REACTOR VESSEL, INTERNALS AND REACTOR COOLANT SYSTEM

Component/Commodity	Intended Function	AMR Results
Reactor Vessel And Internals System (continued)		
Fuel Alignment Pins	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 8 Table 3.1-1, Item 33
Holddown Spring	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 8 Table 3.1-1, Item 33 Table 3.1-2, Item 15
RCCA Guide Tubes	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 8 Table 3.1-1, Item 33
RCCA Guide Tube Bolts	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 8 Table 3.1-1, Item 33
Core Barrel	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 8 Table 3.1-1, Item 31 Table 3.1-1, Item 33
Core Barrel Flange	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 8 Table 3.1-1, Item 33
Core Barrel Outlet Nozzle	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 8 Table 3.1-1, Item 33
Thermal Shield	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 8 Table 3.1-1, Item 33
Baffle and Former Plates	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 8 Table 3.1-1, Item 31 Table 3.1-1, Item 33
Baffle/Former Bolts	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 5 Table 3.1-1, Item 8 Table 3.1-1, Item 12 Table 3.1-1, Item 13
Lower Core Plate	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 8 Table 3.1-1, Item 31 Table 3.1-1, Item 33
Fuel Alignment Pins	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 8 Table 3.1-1, Item 31 Table 3.1-1, Item 33

TABLE 2.3-1 (continued) COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: REACTOR VESSEL, INTERNALS AND REACTOR COOLANT SYSTEM

Component/Commodity	Intended Function	AMR Results
Reactor Vessel And Internals System (continued)		
Lower Support Forging	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 8 Table 3.1-1, Item 31 Table 3.1-1, Item 33
Lower Support Plate Column Sleeves - non-CASS	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 8 Table 3.1-1, Item 33
Lower Support Plate Columns - CASS	Provide structural support to safety-related components.	Table 3.1-1, Item 8 Table 3.1-1, Item 33 Table 3.1-2, Item 14
Lower Support Plate Column Bolts	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 8 Table 3.1-1, Item 33 Table 3.1-2, Item 15
Clevis Inserts	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 8 Table 3.1-1, Item 28 Table 3.1-1, Item 33
Radial Support Keys	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 8 Table 3.1-1, Item 28 Table 3.1-1, Item 33
Clevis Insert Bolts	Provide structural support to safety-related components.	Table 3.1-1, Item 8 Table 3.1-1, Item 31 Table 3.1-1, Item 33 Table 3.1-2, Item 15
Flux Thimble Guide Tubes	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 33 Table 3.1-2, Item 2
Flux Thimbles	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 28 Table 3.1-2, Item 2 Table 3.1-2, Item 16
BMI Columns (non-CASS)	Provide structural support to safety-related components.	Table 3.1-1, Item 8 Table 3.1-1, Item 33
BMI Columns Cruciform (CASS)	Provide structural support to safety-related components.	Table 3.1-1, Item 8 Table 3.1-1, Item 33 Table 3.1-2, Item 14
Diffuser Plate	Provide structural support to safety-related components.	Table 3.1-1, Item 8 Table 3.1-1, Item 33
Head and Vessel Alignment Pins	Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 8 Table 3.1-1, Item 33

TABLE 2.3-1 (continued) COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: REACTOR VESSEL, INTERNALS AND REACTOR COOLANT SYSTEM

Component/Commodity	Intended Function	AMR Results
	Reactor Vessel And Internals System (continued)	
Head Cooling Spray Nozzle	Provide structural support to safety-related components.	Table 3.1-1, Item 8 Table 3.1-1, Item 33
Lifting Lugs (integral to pressure boundary)	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 26
Seal Table Valves and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.1-1, Item 6 Table 3.1-2, Item 2 Table 3.1-2, Item 17
Secondary Core Support	Provide structural support to safety-related components.	Table 3.1-1, Item 8 Table 3.1-1, Item 33
Upper Instrumentation Column, Conduit, and Supports	Provide structural support to safety-related components.	Table 3.1-1, Item 8 Table 3.1-1, Item 33
	Steam Generator	
Top Head	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 2
Steam Nozzle	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 2 Table 3.1-1, Item 21
Upper and Lower Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 2
Transition Cone including Girth Weld	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 2
Feedwater Nozzle	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 2 Table 3.1-1, Item 21
Secondary Manway and Handhole Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-2, Item 12
Lower Head	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 26
Lower Head Cladding	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide corrosion protection of pressure boundary components.	Table 3.1-1, Item 1 Table 3.1-1, Item 32 Table 3.1-2, Item 2
Primary Nozzles Cladding and Safe Ends	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide corrosion protection of pressure boundary components.	Table 3.1-1, Item 1 Table 3.1-1, Item 32 Table 3.1-2, Item 2

TABLE 2.3-1 (continued) COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: REACTOR VESSEL, INTERNALS AND REACTOR COOLANT SYSTEM

Component/Commodity	Intended Function	AMR Results
	Steam Generator (continued)	
Primary Nozzles	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 26
Primary Manway Cover	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 26
Primary Manway Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 26
Tube Bundle	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer.	Table 3.1-1, Item 1 Table 3.1-1, Item 15
Tube Plugs (Westinghouse))	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 15
Tube Support Plates	Provide structural support to pressure boundary components. Maintains structural integrity of pressure boundary components.	Table 3.1-1, Item 1 Table 3.1-1, Item 17
Snubber Reservoir Components	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-2, Item 7
Steam Generator Anti-vibration Bars	Provide structural support to pressure boundary components. Maintains structural integrity of pressure boundary components.	Table 3.1-1, Item 1 Table 3.1-2, Item 3
Steam Generator Feedwater Nozzle Thermal Sleeve	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide insulation/thermal resistance.	Table 3.1-1, Item 1 Table 3.1-1, Item 21 Table 3.1-2, Item 5
Steam Generator Feedwater Nozzle Thermal Sleeve Safe End	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-2, Item 4
Steam Generator Lower Head Divider Plate	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provides flow restriction or distribution.	Table 3.1-1, Item 1 Table 3.1-2, Item 2 Table 3.1-2, Item 11
Steam Generator Primary Manway Insert	Provide corrosion protection of pressure boundary components.	Table 3.1-1, Item 1 Table 3.1-1, Item 32 Table 3.1-2, Item 2
Steam Generator Secondary Side Manway, Inspection, and Handhole Covers	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-2, Item 5
Steam Generator Secondary Side Shell Penetrations	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-2, Item 5
Steam Generator Steam Flow Limiter	Provide flow restriction (throttle).	Table 3.1-1, Item 1 Table 3.1-2, Item 4 Table 3.1-2, Item 6

TABLE 2.3-1 (continued) COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: REACTOR VESSEL, INTERNALS AND REACTOR COOLANT SYSTEM

Component/Commodity	Intended Function	AMR Results
	Steam Generator (continued)	
Steam Generator Support Pad	Provide structural support to pressure boundary components. Maintains structural integrity of pressure boundary components.	Table 3.1-1, Item 1 Table 3.1-1, Item 26
Steam Generator Tube Bundle Wrapper System	Provide structural support to pressure boundary components. Maintains structural integrity of pressure boundary components. Provides flow restriction or distribution.	Table 3.1-1, Item 1 Table 3.1-2, Item 5 Table 3.1-2, Item 6
Steam Generator Tubeplate	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-2, Item 5 Table 3.1-2. Item 6
Steam Generator Tubeplate Cladding	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide corrosion protection of pressure boundary components.	Table 3.1-1, Item 1 Table 3.1-2, Item 2 Table 3.1-2, Item 11

2.3.2 ENGINEERED SAFETY FEATURES SYSTEMS

Engineered Safety Features Systems 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. Engineered Safety Features Systems may actuate automatically to mitigate a range of postulated accidents. The Engineered Safety Features Systems (1) provide emergency coolant to limit damage to the reactor core and limit energy release to containment in the event of a loss-of-coolant accident, (2) reduce pressure in the containment, (3) prevent leakage of radioactivity from containment, and (4) reduce the concentration of fission products in the containment atmosphere.

The following Engineered Safety Features Systems are included in this Subsection:

1. Residual Heat Removal System
2. Safety Injection System
3. Containment Spray System
4. Containment Air Recirculation Cooling System
5. Containment Isolation System

2.3.2.1 Residual Heat Removal System

The Residual Heat Removal System delivers borated water to the Reactor Coolant System during the injection phase of a design basis accident. Following a loss-of-coolant accident, the Residual Heat Removal System cools and recirculates water that is collected in the containment recirculation sump and returns it to the Reactor Coolant, Containment Spray, and Safety Injection Systems to maintain reactor core and containment cooling functions. In addition, during normal plant operations, the Residual Heat Removal System removes residual and sensible heat from the core during plant shutdown, cooldown, and refueling operations. The RHR system is used to achieve cold shutdown conditions following a postulated fire in accordance with 10 CFR 50, Appendix R, requirements. The Residual Heat Removal System is described in Sections 5.4.4 and 6.3 of the RNP UFSAR.

The license renewal evaluation boundaries for the Residual Heat Removal System are shown on the following flow diagram. (Flow diagrams have been submitted separately for information only.)

Residual Heat Removal System	5379-1484LR Sheet 1

The Residual Heat Removal System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events
2. SCs that are part of the Environmental Qualification Program
3. SCs that are relied on during postulated fires and station blackout events

Table 2.3-2 below identifies the Residual Heat Removal System components/ commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-2 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: RESIDUAL HEAT REMOVAL SYSTEM

Component/Commodity	Intended Function	AMR Results
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 26 Table 3.2-1, Item 11
Flow Orifices/Elements	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide flow restriction (throttle).	Table 3.2-1, Item 1 Table 3.2-1, Item 10 Table 3.2-2, Item 1 Table 3.2-2, Item 8
N2 Cylinder Tank(s)	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.2-1, Item 11
RHR Heat Exchanger Shell and Cover	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.2-1, Item 9 Table 3.2-1, Item 11 Table 3.2-2, Item 5
RHR Heat Exchanger Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer.	Table 3.2-1, Item 1 Table 3.2-1, Item 9 Table 3.2-1, Item 10 Table 3.2-2, Item 1 Table 3.2-2, Item 2 Table 3.2-2, Item 7
RHR Pump Seal Heat Exchanger Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.2-1, Item 9 Table 3.2-1, Item 11 Table 3.2-2, Item 5
RHR Pump(s)	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered	Table 3.2-1, Item 1 Table 3.2-1, Item 10 Table 3.2-2, Item 1 Table 3.2-2, Item 8

TABLE 2.3-2 (continued) COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: RESIDUAL HEAT REMOVAL SYSTEM

Component/Commodity	Intended Function	AMR Results
RHR SEAL WTR Heat Exchanger Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer.	Table 3.2-1, Item 1 Table 3.2-1, Item 9 Table 3.2-1, Item 10 Table 3.2-2, Item 1 Table 3.2-2, Item 2 Table 3.2-2, Item 7
Valves, Piping, Tubing and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 24 Table 3.1-2, Item 2 Table 3.1-2, Item 17 Table 3.2-1, Item 1 Table 3.2-1, Item 10 Table 3.2-2, Item 1 Table 3.2-2, Item 3 Table 3.2-2, Item 9 Table 3.2-2, Item 12

2.3.2.2 Safety Injection System

Following a postulated design basis accident, adequate emergency core cooling is provided by the Safety Injection System, whose components operate in three modes: passive accumulator injection, active safety injection, and residual heat removal recirculation. The primary purpose of the system is to deliver cooling water to the reactor core in the event of a loss-of-coolant accident. This limits the fuel cladding temperature and thereby ensures that the core will remain intact and in place, with its heat transfer geometry preserved. The system also provides a source of borated water for reactivity control. The Safety Injection System is described in Section 6.3 of the RNP UFSAR.

The license renewal evaluation boundaries for the Safety Injection System are shown on the following flow diagrams. (Flow diagrams have been submitted separately for information only.)

Safety Injection System	5379-1082LR Sheet 1 5379-1082LR Sheet 2 5379-1082LR Sheet 3 5379-1082LR Sheet 4 5379-1082LR Sheet 5

The Safety Injection System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs that are part of the Environmental Qualification Program

3. SCs that are relied on during postulated fires and station blackout events

Table 2.3-3 below identifies the Safety Injection System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-3 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: SAFETY INJECTION SYSTEM

Component/Commodity	Intended Function	AMR Results
Boron Injection Tank	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.2-2, Item 1 Table 3.2-2, Item 8
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 26 Table 3.2-1, Item 11
ECCS Sump Hood Filter	Provide filtration.	Table 3.2-2, Item 8
ECCS Screen Filter(s)	Provide filtration.	Table 3.2-2, Item 8
Equipment Frames and Housings	Provide filtration.	Table 3.2-1, Item 6 Table 3.2-1, Item 11
Flow Orifices/Elements	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide flow restriction (throttle).	Table 3.2-2, Item 1 Table 3.2-2, Item 8
Refueling Water Storage Tank	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.2-2, Item 1 Table 3.2-2, Item 8
Safety Injection Pump Outboard Bearing Heat Exchanger Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer.	Table 3.2-1, Item 8 Table 3.2-1, Item 11 Table 3.2-2, Item 4 Table 3.2-2, Item 11
Safety Injection Pump(s)	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.2-2, Item 1 Table 3.2-2, Item 8
SI Accumulator Tank(s)	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.2-1, Item 11 Table 3.2-2, Item 1 Table 3.2-2, Item 10
SI Pump Recirc Line Strainer Filter(s)	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.2-2, Item 1 Table 3.2-2, Item 8
SI Pump Seal Cooler Heat Exchanger Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer.	Table 3.2-1, Item 9 Table 3.2-2, Item 1 Table 3.2-2, Item 2
SI Pump Seal Heat Exchanger Shell and Cover	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.2-1, Item 9 Table 3.2-1, Item 11 Table 3.2-2, Item 5

TABLE 2.3-3 (continued) COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: SAFETY INJECTION SYSTEM

Component/Commodity	Intended Function	AMR Results
Valves, Piping, Tubing and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 19 Table 3.1-1, Item 22 Table 3.1-1, Item 24 Table 3.1-2, Item 2 Table 3.2-1, Item 6 Table 3.2-1, Item 11 Table 3.2-2, Item 1 Table 3.2-2, Item 9

2.3.2.3 Containment Spray System

The Containment Spray System and Containment Air Recirculation Cooling System limit the temperature and pressure that could be experienced in containment following a loss-of-coolant accident or a steam line break to less than the design values. These two systems employ different engineering principles for heat removal and serve as independent backups for each other. The Containment Air Recirculation Cooling System is discussed in Subsection 2.3.2.4.

Each train of the Containment Spray System includes a containment spray pump, valves, interconnecting piping, spray headers, and nozzles. Additionally, the Containment Spray System mixes the spray flow with sodium hydroxide from the spray additive tank to assist in removing iodine from the containment atmosphere. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. The system can utilize the flow from the residual heat removal system heat exchangers to provide a source and to cool the spray flow during the long-term recirculation phase. The Containment Spray System is described in Section 6.2.2 of the RNP UFSAR.

The license renewal evaluation boundaries for the Containment Spray System are shown on the following flow diagrams. (Flow diagrams have been submitted separately for information only.)

Containment Spray System	5379-1082LR Sheet 2 5379-1082LR Sheet 3 5379-1082LR Sheet 5

The Containment Spray System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs that are part of the Environmental Qualification Program

3. SCs that are relied on during postulated fires and station blackout events

Table 2.3-4 below identifies the Containment Spray System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-4 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: CONTAINMENT SPRAY SYSTEM

Component/Commodity	Intended Function	AMR Results
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.2-1, Item 6 Table 3.2-1, Item 11
CV Spray Pump Seal Cooler Heat Exchanger Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer.	Table 3.2-1, Item 9 Table 3.2-2, Item 1 Table 3.2-2, Item 2
CV Spray Pump Seal Heat Exchanger Shell and Cover	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.2-1, Item 9 Table 3.2-1, Item 11 Table 3.2-2, Item 5 Table 3.2-2, Item 6
CV Spray Pump(s)	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.2-2, Item 1 Table 3.2-2, Item 8
Eductors	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide flow restriction (throttle).	Table 3.2-2, Item 1 Table 3.2-2, Item 8
Flow Orifices/Elements	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide flow restriction (throttle).	Table 3.2-2, Item 1 Table 3.2-2, Item 8
Spray Additive Tank	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.2-1, Item 6 Table 3.2-1, Item 11 Table 3.2-2, Item 1
Valves, Piping, Tubing, and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.2-1, Item 6 Table 3.2-1, Item 11 Table 3.2-2, Item 1 Table 3.2-2, Item 9

2.3.2.4 Containment Air Recirculation Cooling System

The Containment Spray System and Containment Air Recirculation Cooling System limit the temperature and pressure that could be experienced in containment following a loss-of-coolant accident or a steam line break to less than the design values. These systems employ different engineering principles for heat removal and serve as independent backups for each other.

The Containment Air Recirculation Cooling System is part of the HVAC Containment Building System and was designed to recirculate and cool the containment atmosphere during and following an accident and to reduce containment pressure to atmospheric. Heat removed by the system is transferred to the Service Water System. The Containment Air Recirculation Cooling System consists of four air handling units, each including a fan, cooling coil, dampers and a duct distribution system, spaced around the operating floor adjacent to the containment wall. The Containment Air Recirculation Cooling System is described in Section 6.2.2 of the RNP UFSAR.

The license renewal evaluation boundaries for the Containment Air Recirculation Cooling System are shown on the following flow diagram. (Flow diagrams have been submitted separately for information only.)

Containment Air Recirculation Cooling System G-190304LR Sheet 1

The Containment Air Recirculation Cooling System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs that are part of the Environmental Qualification Program

Table 2.3-5 below identifies the Containment Air Recirculation System components/ commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-5 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: CONTAINMENT AIR RECIRCULATION SYSTEM

Component/Commodity	Intended Function	AMR Results
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13
Equipment Frames and Housings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer.	Table 3.3-1, Item 5 Table 3.3-1, Item 13 Table 3.3-2, Item 12 Table 3.3-2, Item 20 Table 3.3-2, Item 23
Flexible Collars	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 2
Heating/Cooling Coils	Provide pressure re-taining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer.	Table 3.3-1, Item 5 Table 3.3-1, Item 16 Table 3.3-2, Item 12
Valves	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13 Table 3.3-2, Item 19 Table 3.3-2, Item 23
Ductwork and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 20

TABLE 2.3-5 (continued) COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: CONTAINMENT AIR RECIRCULATION SYSTEM

Component/Commodity	Intended Function	AMR Results
Damper Mounting	Provide structural support to safety-related components.	Table 3.3-2, Item 20

2.3.2.5 Containment Isolation System

The Containment Isolation System is an engineered safety feature that provides for the closure or integrity of containment penetrations to prevent leakage of uncontrolled or unmonitored radioactive materials to the environment. The Containment Isolation System is described in Section 6.2.4 of the RNP UFSAR.

The pressure boundary portions of electrical penetrations and miscellaneous/spare mechanical penetrations that are not associated with a process system are included in the civil structural screening described in Section 2.4. The electrical portions of containment electrical penetrations are included in the electrical screening described in Section 2.5.

Process systems that have license renewal functions in addition to the containment isolation function are included in the system screening results described elsewhere in Section 2.3.

Process systems whose only license renewal intended function is the containment isolation function are:

1. Post Accident Hydrogen System
2. Service Air System
3. Process/Area Radiation Monitoring
4. Containment Pressure Relief System
5. Containment Vacuum Breaker System
6. Liquid Waste Processing System
7. Penetration Pressurization Local Leak Rate Test
8. Isolation Valve Seal Water System

All containment isolation valves at RNP have been screened and determined to be subject to aging management review (pressure boundary function only).

The license renewal evaluation boundaries for the process systems whose only license renewal intended function is the containment isolation function are shown on the following flow diagrams. (Flow diagrams have been submitted separately for information only.)

Containment Isolation Systems	HBR2-06933LR Sheet 1 G-190200LR Sheet 3 G-190304LR Sheet 1 5379-920LR Sheet 3 G-190261LR Sheet 1 G-190261LR Sheet 2 G-190261LR Sheet 3 G-190261LR Sheet 4 G-190262LR Sheet 1

The Post Accident Hydrogen System, Service Air System, Process/Area Radiation Monitoring, Containment Pressure Relief System, Containment Vacuum Breaker System, Liquid Waste Processing System, Penetration Pressurization Local Leak Rate Test, and Isolation Valve Seal Water System are in the scope of license renewal, because they contain:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs that are part of the Environmental Qualification Program

3. SCs that are relied on during station blackout events

Table 2.3-6 below Identifies the Containment Isolation System components/ commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/ commodity type.

TABLE 2.3-6 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: CONTAINMENT ISOLATION SYSTEM

Component/Commodity	Intended Function	AMR Results
Post Accident Hydrogen System		
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.2-1, Item 11
Valves, Piping and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.2-2, Item 3 Table 3.2-2, Item 9 Table 3.2-2, Item 13
Process/Area Radiation Monitoring System		
Valves, Piping and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.2-2, Item 9
Service Air System		
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.2-1, Item 11
Valves, Piping and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.2-1, Item 11 Table 3.2-2, Item 3 Table 3.2-2, Item 9 Table 3.2-2, Item 13 Table 3.2-2, Item 14
Containment Pressure Relief System		
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.2-1, Item 11
Valves, Piping and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.2-1, Item 11 Table 3.2-2, Item 14
Containment Vacuum Breaker System		
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.2-1, Item 11
Valves, Piping and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.2-1, Item 11 Table 3.2-2, Item 14
Liquid Waste Processing System		
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.2-1, Item 11
Valves, Piping and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.2-1, Item 3 Table 3.2-1, Item 4 Table 3.2-1, Item 11 Table 3.2-2, Item 1 Table 3.2-2, Item 9 Table 3.2-2, Item 14

TABLE 2.3-6 (continued) COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: CONTAINMENT ISOLATION SYSTEM

Component/Commodity	Intended Function	AMR Results
Penetration Pressurization Local Leak Rate Test		
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.2-1, Item 11
Valves, Piping and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.2-1, Item 11 Table 3.2-2, Item 9 Table 3.2-2, Item 12 Table 3.2-2, Item 14
Isolation Valve Seal Water System		
Valves, Piping and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.2-1, Item 3 Table 3.2-1, Item 4 Table 3.2-2, Item 1 Table 3.2-2, Item 9

2.3.3 AUXILIARY SYSTEMS

Auxiliary Systems are those systems used to support normal and emergency plant operations. The systems provide cooling, ventilation, sampling and other required functions. The following systems are included in this Subsection:

1. Sampling Systems
2. Service Water System
3. Component Cooling Water System
4. Chemical and Volume Control System
5. Instrument Air System
6. Nitrogen Supply/Blanketing System
7. Radioactive Equipment Drains
8. Primary and Demineralized Water System
9. Spent Fuel Pool Cooling System
10. Containment Purge System
11. Rod Drive Cooling System
12. HVAC Auxiliary Building
13. HVAC Control Room Area
14. HVAC Fuel Handling Building
15. Fire Protection System
16. Diesel Generator System
17. Dedicated Shutdown Diesel Generator
18. EOF/TSC Security Diesel Generator
19. Fuel Oil System

2.3.3.1 Sampling Systems

Sampling systems include the Primary Sampling System, the Steam Cycle Sampling System, the Containment Vapor and Pressure Sampling System, and the Post Accident Sampling System. (The Class 1 portions of the Primary Sampling System are addressed in Subsection 2.3.1.1, and Steam Cycle Sampling is addressed in Subsection 2.3.4.7.)

The Primary Sampling System provides representative samples for laboratory analysis to evaluate the chemistry of reactor coolant, residual heat removal system, safety injection system, steam system, and chemical and volume control system during normal operation. The system is operated manually on an intermittent basis. The Primary Sampling System is described in RNP UFSAR Section 9.3.2.1.

The Containment Vapor and Pressure Sampling System provides the means to monitor containment pressure.

The Post Accident Sampling System provides a means to remotely collect reactor coolant, containment atmosphere, and other samples following a postulated nuclear accident, to remotely indicate results, and to dilute the samples for subsequent radiological analysis. The Post Accident Sampling System is divided into two basic parts: reactor coolant sampling and containment air sampling. Reactor coolant samples are provided from the Primary Sampling System discussed above. Containment air samples are provided via the Penetration Pressurization System Local Leak Rate Test System from the Process/Area Radiation Monitoring System. The Post Accident Sampling System is described in RNP UFSAR Section 9.3.2.2.

The license renewal evaluation boundaries for the Sampling Systems are shown on the following flow diagrams. (Flow diagrams have been submitted separately for information only.)

Sampling Systems	5379-353LR Sheet 1 (Primary) HBR2-06490LR Sheet 1 (Containment) HBR2-08261LR Sheet 1 (Post Accident)

Based on the screening evaluation, the only component with a license renewal intended function in the Post Accident Sampling System is a sample heat exchanger, and its function is to maintain the pressure boundary of the Component Cooling Water System. The heat exchanger shell is cooled by the Component Cooling Water System; therefore, the shell and tubes form part of the pressure boundary of that system. Aging management review of the heat exchanger is addressed under the Component Cooling Water System in Subsection 2.3.3.3. (For the same reasons, sample heat exchangers in the Primary Sampling System have been evaluated under the Component Cooling Water System.) Therefore, the Post Accident Sampling System is not discussed further in this Subsection.

The Primary and Post Accident Sampling Systems are in the scope of license renewal, because they contain:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs that are part of the Environmental Qualification Program

The Containment Vapor and Pressure Sampling System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs that are relied on during station blackout events

Table 2.3-7 below identifies the Sampling System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-7 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: SAMPLING SYSTEMS

Component/Commodity	Intended Function	AMR Results
Primary Sampling System		
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 26
Valves, Piping and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 6 Table 3.1-2, Item 2 Table 3.3-1, Item 3 Table 3.3-2, Item 1 Table 3.3-2, Item 2 Table 3.3-2, Item 23
Containment Vapor And Pressure Sampling System		
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13
Valves, Piping, Tubing and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.3-1, Item 13 Table 3.3-2, Item 19 Table 3.3-2, Item 23

2.3.3.2 Service Water System

The Service Water System is an open loop system and provides makeup water to and removes heat from several plant systems. Redundant supply paths are provided to those systems required for safety either during normal operation or under postulated accident conditions. The system removes heat from the Component Cooling Water System; HVAC Systems in the Containment Building, Auxiliary Building, Control Room Area, Fuel Handling Building and safety related pump rooms; Emergency Diesel Generators; certain safety-related pumps; and various heat loads in the Turbine Building. The system contains four wet pit pumps and two full capacity booster pumps that supply water to the containment fan coolers. The system provides a backup, long-term water supply to the Auxiliary Feedwater System. The Service Water System is described in RNP UFSAR Section 9.2.1.

The license renewal evaluation boundaries for the Service Water System are shown on the following flow diagrams. (Flow diagrams have been submitted separately for information only.)

Service Water System	G-190199LR Sheet 1
	G-190199LR Sheet 2
	G-190199LR Sheet 3
	G-190199LR Sheet 4
	G-190199LR Sheet 5
	G-190199LR Sheet 6
	G-190199LR Sheet 7
	G-190199LR Sheet 8
	G-190199LR Sheet 9
	G-190199LR Sheet 10

The Service Water System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions

3. SCs that are part of the Environmental Qualification Program

4. SCs that are relied on during postulated fires and station blackout events

Table 2.3-8 below identifies the Service Water System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-8 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: SERVICE WATER SYSTEM

Component/Commodity	Intended Function	AMR Results
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13
Flow Orifices/Elements	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide flow restriction (throttle).	Table 3.3-1, Item 16 Table 3.3-2, Item 23
Service Water Booster Pumps	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 16 Table 3.3-2, Item 23
Service Water Pumps	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5 Table 3.3-1, Item 16 Table 3.3-1, Item 24
Service Water Supply Header Strainers	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide filtration.	Table 3.3-1, Item 5 Table 3.3-1, Item 16
Valves, Piping, Tubing and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components. Provide heat transfer.	Table 3.3-1, Item 5 Table 3.3-1, Item 13 Table 3.3-1, Item 16 Table 3.3-1, Item 17 Table 3.3-1, Item 24 Table 3.3-2, Item 11 Table 3.3-2, Item 14 Table 3.3-2, Item 23 Table 3.3-2, Item 29

2.3.3.3 Component Cooling Water System

The Component Cooling Water System provides a heat sink for the removal of process and operating heat from safety related components during postulated accidents or transients. During normal operation, the Component Cooling Water System also provides this function for various nonessential components, as well as the spent fuel storage pool. The Component Cooling Water System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Service Water System, and thus to the environment. The Component Cooling Water System consists of three pumps, two heat exchangers, a supply and return header, a surge tank, and associated piping, valves, and instrumentation. The Component Cooling Water System is described in RNP UFSAR Section 9.2.2.

The license renewal evaluation boundaries for the Component Cooling Water System are shown on the following flow diagrams. (Flow diagrams have been submitted separately for information only.)

Component Cooling Water System	5379-376LR Sheet 1 5379-376LR Sheet 2 5379-376LR Sheet 3 5379-376LR Sheet 4 HBR2-08261LR (Post Accident Sampling)

The Component Cooling Water System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs that are part of the Environmental Qualification Program

3. SCs that are relied on during postulated fires and station blackout events

Table 2.3-9 below identifies the Component Cooling Water System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-9 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: COMPONENT COOLING WATER SYSTEM

Component/Commodity	Intended Function	AMR Results
CCW Heat Exchanger Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13 Table 3.3-1, Item 14 Table 3.3-1, Item 16
CCW Heat Exchanger Tube Sheet	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 16 Table 3.3-2, Item 17
CCW Heat Exchanger Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer.	Table 3.3-1, Item 16 Table 3.3-2, Item 16 Table 3.3-2, Item 17
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13
Component Cooling Water Pumps	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13 Table 3.3-1, Item 14
Component Cooling Water Surge Tank	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13 Table 3.3-1, Item 14

TABLE 2.3-9 (continued) COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: COMPONENT COOLING WATER SYSTEM

Flow Orifices/Elements	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13 Table 3.3-1, Item 14
Hot Leg Sample Heat Exchanger Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13 Table 3.3-1, Item 14
Hot Leg Sample Heat Exchanger Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 14
Non Regenerative Heat Exchanger Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13 Table 3.3-1, Item 14
Non Regenerative Heat Exchanger Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 14
Pressurizer Liquid Sample Heat Exchanger Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 14
Pressurizer Steam Sample Heat Exchanger Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 14
PZR Liquid Sample Heat Exchanger Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13 Table 3.3-1, Item 14
PZR Steam Sample Heat Exchanger Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13 Table 3.3-1, Item 14
Rod Drive Cooling System Cooler Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer	Table 3.3-1, Item 5 Table 3.3-1, Item 24 Table 3.3-2, Item 16 Table 3.3-2, Item 25
Sample Vessel Heat Exchanger Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13 Table 3.3-1, Item 14
Sample Vessel Heat Exchanger Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 14
SFP Cooling Heat Exchanger Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13 Table 3.3-1, Item 14
SFP Cooling Heat Exchanger Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 14
SG Blowdown Heat Exchanger Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13 Table 3.4-1, Item 10
SG Blowdown Sample Heat Exchanger Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.4-1, Item 10
Valves, Piping, Tubing, and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.3-1, Item 13 Table 3.3-1, Item 14 Table 3.3-2, Item 23
Waste Gas Compressor Cooler Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 24 Table 3.3-2, Item 16
Waste Gas Compressor Coolers Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13 Table 3.3-1, Item 14

2.3.3.4 Chemical And Volume Control System

The Chemical and Volume Control System provides a continuous feed and bleed of reactor cooling water for the Reactor Coolant System to maintain proper water level and to adjust boron concentration. The Chemical and Volume Control System provides a means for injection of control poison in the form of boric acid solution, chemical additions for corrosion control, and reactor coolant cleanup and degasification. The system also adds makeup water to the RCS, reprocesses water letdown from the RCS and charging pump leakage, and provides seal water injection to the reactor coolant pump seals. Portions of the CVCS contain highly concentrated boric acid solution and may be used to maintain reactor Shutdown Margin in accordance with Technical Specification requirements. However, providing highly concentrated boric acid solution is not a safety related function; and the components that provide this solution are not in scope of license renewal unless they provide another function that is in scope. The Chemical and Volume Control System is described in RNP UFSAR Section 9.3.4.

The license renewal evaluation boundaries for the Chemical and Volume Control System are shown on the following flow diagrams. (Flow diagrams have been submitted separately for information only.)

Chemical and Volume Control System	5379-685LR Sheet 1 5379-685LR Sheet 2

The Chemical and Volume Control System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions

3. SCs that are part of the Environmental Qualification Program

4. SCs that are relied on during postulated fires and station blackout events

Table 2.3-10 below identifies the Chemical and Volume Control System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type. The tube side of the Regenerative Heat Exchanger is addressed as part of the Reactor Coolant System.

TABLE 2.3-10 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: CHEMICAL AND VOLUME CONTROL SYSTEM

Component/Commodity	Intended Function	AMR Results
Charging Pump Heat Exchanger Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 21 Table 3.3-2, Item 22
Regenerative Heat Exchanger Shell and Cover	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 23
Charging Pump Heat Exchanger Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer.	Table 3.3-2, Item 16 Table 3.3-2, Item 22
Charging Pump Heat Exchanger Waterbox	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13 Table 3.3-2, Item 22
Charging Pump LubeTank(s)	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 3 Table 3.3-2, Item 1 Table 3.3-2, Item 2 Table 3.3-2, Item 23
Charging Pump Suction Stabilizers and Pulsation Dampeners	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 1 Table 3.3-2, Item 23
Charging Pump(s)	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 1 Table 3.3-2, Item 23
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 26 Table 3.3-1, Item 13 Table 3.3-1, Item 23
Excess Letdown Heat Exchanger Shell and Cover	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 3 Table 3.3-1, Item 13 Table 3.3-1, Item 14 Table 3.3-2, Item 15
Excess Letdown Heat Exchanger Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 3 Table 3.3-1, Item 8 Table 3.3-1, Item 14 Table 3.3-2, Item 1 Table 3.3-2, Item 3
Flow Orifices/Elements	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 6 Table 3.1-2, Item 17 Table 3.3-1, Item 3 Table 3.3-2, Item 1 Table 3.3-2, Item 2 Table 3.3-2, Item 23
Regenerative Heat Exchanger Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-1, Item 6

TABLE 2.3-10 (continued) COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: CHEMICAL AND VOLUME CONTROL SYSTEM

Component/Commodity	Intended Function	AMR Results
Regenerative Heat Exchanger Tubing, Shell, and Cover	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 3 Table 3.3-1, Item 8 Table 3.3-2, Item 1 Table 3.3-2, Item 3 Table 3.3-2, Item 23
Seal Injection Filter	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 3 Table 3.3-2, Item 1 Table 3.3-2, Item 2 Table 3.3-2, Item 23
Seal Return Filter	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 3 Table 3.3-2, Item 1 Table 3.3-2, Item 2 Table 3.3-2, Item 23
Seal Wtr Heat Exchanger Shell and Cover	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13 Table 3.3-1, Item 14 Table 3.3-2, Item 15
Seal Wtr Heat Exchanger Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 8 Table 3.3-1, Item 14 Table 3.3-2, Item 1 Table 3.3-2, Item 3
Valves, Piping, Tubing and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.1-1, Item 1 Table 3.1-1, Item 6 Table 3.1-1, Item 19 Table 3.1-1, Item 22 Table 3.1-2, Item 2 Table 3.1-2, Item 17 Table 3.3-1, Item 3 Table 3.3-2, Item 1 Table 3.3-2, Item 2 Table 3.3-2, Item 9 Table 3.3-2, Item 23
Volume Control Tank	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 1 Table 3.3-2, Item 23

2.3.3.5 Instrument Air System

The Instrument Air System provides a reliable source of dry, oil free air for instrumentation and controls and pneumatic valves. The Instrument Air System provides motive power and control air to safety related and non-safety related components. The system contains air compressors, air dryers, air receivers and interconnecting piping and valves. The Instrument Air System is described in RNP UFSAR Section 9.3.1.

Safety related air operated valves that are required to operate following design basis events and are normally supplied by instrument air are provided with backup sources of either air (accumulators) or nitrogen. The Instrument Air System provides for isolation of safety related from non-safety related portions of the system.

The license renewal evaluation boundaries for the Instrument Air System are shown on the following flow diagrams. (Flow diagrams have been submitted separately for information only.)

Instrument Air System	G-190200LR Sheet 2 G-190200LR Sheet 3 G-190200LR Sheet 5 G-190200LR Sheet 9

The Instrument Air System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs that are part of the Environmental Qualification Program

3. SCs that are relied on during postulated fires and station blackout events

Table 2.3-11 below identifies the Instrument Air System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-11 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: INSTRUMENT AIR SYSTEM

Component/Commodity	Intended Function	AMR Results
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13
Instrument Air Filters	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 19 Table 3.3-2, Item 24
Instrument Air Regulator Body/Bonnet	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 19 Table 3.3-2, Item 21
Valves, Piping, Tubing and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.3-1, Item 13 Table 3.3-2, Item 5 Table 3.3-2, Item 14 Table 3.3-2, Item 19 Table 3.3-2, Item 21 Table 3.3-2, Item 23 Table 3.3-2, Item 24

2.3.3.6 Nitrogen Supply/Blanketing System

The Nitrogen Supply/Blanketing System provides gas for various plant functions. Nitrogen is used as the motive force for some gas-operated valves, to pressurize the Safety Injection System accumulators, and to provide inert cover gas for certain tanks. Portions of the system provide motive force for the pressurizer power operated relief valves.

The license renewal evaluation boundaries for the Nitrogen Supply/Blanketing System are shown on the following flow diagram. (Flow diagrams have been submitted separately for information only.)

Nitrogen Supply/Blanketing System	HBR2-08606LR Sheet 2

The Nitrogen Supply/Blanketing System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs that are relied on during postulated fires and station blackout events

Table 2.3-12 below identifies the Nitrogen Supply/Blanketing System components/ commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-12 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: NITROGEN SUPPLY/BLANKETING SYSTEM

Component/Commodity	Intended Function	AMR Results
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13
Flow Orifices/Elements	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide flow restriction (throttle).	Table 3.3-2, Item 19
PZR N_2 Accumulator Tank	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13 Table 3.3-2, Item 19
Steam Dump Accumulator Tank	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 19
Valves, Piping, Tubing and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.3-2, Item 19 Table 3.3-2, Item 21 Table 3.3-2, Item 23

2.3.3.7 Radioactive Equipment Drains

The Radioactive Equipment Drains route potentially radioactive floor drainage to the Liquid Waste Processing System. Portions of the system are relied on during postulated internal fires to drain fire protection water from rooms containing safety related equipment.

The evaluation boundaries for the portions of the Radioactive Equipment Drains that are within the scope of license renewal were determined on the basis of its function following actuation of fire suppression systems in the Reactor Auxiliary Building as described in UFSAR Appendix 9.5.1B. No flow diagrams were used to determine the evaluation boundaries. Portions of the Radioactive Equipment Drains piping is embedded in concrete and is considered to be part of the Reactor Auxiliary Building structure and will be screened as a civil commodity.

The Radioactive Equipment Drains are in the scope of license renewal, because they contain SCs that are relied on during postulated fires.

The following table identifies the Radioactive Equipment Drains System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-13 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: RADIOACTIVE EQUIPMENT DRAINS SYSTEM

Component/Commodity	Intended Function	AMR Results
Piping and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 8

2.3.3.8 Primary And Demineralized Water System

The Primary and Demineralized Water System supplies demineralized and deaerated water for process support functions and makeup supplies to various systems throughout the plant. UFSAR Section 9.2.3 provides a description of the Primary and Demineralized Water System.

The license renewal evaluation boundaries for the Primary and Demineralized Water System are shown on the following flow diagram. (Flow diagrams have been submitted separately for information only.)

Primary and Demineralized Water System G-190202LR Sheet 3

The Primary and Demineralized Water System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions

3. SCs that are relied on during postulated fires and station blackout events

Table 2.3-14 below identifies the Primary and Demineralized Water System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-14 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: PRIMARY AND DEMINERALIZED WATER MAKEUP SYSTEM

Component/Commodity	Intended Function	AMR Results
Valves, Piping and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.3-1, Item 5 Table 3.3-2, Item 6 Table 3.3-2, Item 7 Table 3.3-2, Item 23

2.3.3.9 Spent Fuel Pool Cooling System

The Spent Fuel Pool Cooling System removes decay heat generated by stored spent fuel elements from the spent fuel pool and provides filtering and demineralization of the water in the spent fuel pool. The Spent Fuel Pool Cooling System consists of three separate loops: cooling, purification, and skimmer loops.

The cooling loop removes heat from the spent fuel pool by circulating water through the spent fuel pool heat exchanger. Heat is removed from this heat exchanger by the Component Cooling Water System. The purification loop provides filtering and demineralization by circulating a portion of the cooling loop flow through a filter and demineralizer. The skimmer loop removes floating debris and surface contaminants that could affect water clarity by taking a suction on the skimmer and circulating the water through a strainer and filter. The functions involving heat removal, purification, and contaminant removal for the spent fuel pool are not intended functions for license renewal. Functions of the Spent Fuel Pool Cooling System within scope of license renewal involve maintaining a barrier to support the pressure boundaries of the Spent Fuel Pool and the Refueling Water Storage Tank.

The license renewal evaluation boundaries for the Spent Fuel Pool Cooling System are shown on the following flow diagram. (Flow diagrams have been submitted separately for information only.)

Spent Fuel Pool Cooling System 5379-1485LR Sheet 1

The Spent Fuel Pool Cooling System is in the scope of license renewal, because it contains SCs that are relied on during postulated fires and station blackout events. Valves in the Spent Fuel Pool Cooling System are used to isolate the refueling water storage tank to assure a cooling water supply.

Table 2.3-15 below identifies the Spent Fuel Pool Cooling System components/ commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-15 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: SPENT FUEL POOL COOLING SYSTEM

Component/Commodity	Intended Function	AMR Results
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13
Flow Orifices/Elements	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 1
Valves, Piping and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.3-2, Item 1

The spent fuel pool heat exchanger shell is cooled by the Component Cooling Water System; therefore, the shell and tubes form part of the pressure boundary of that system. Aging management review of the heat exchanger is addressed under the Component Cooling Water System in Subsection 2.3.3.3.

2.3.3.10 Containment Purge System

The Containment Purge System is designed to replenish the containment air at a rate to ensure that an effective purge can be accomplished within two hours. The system contains an outdoor air intake, supply and exhaust ducts that penetrate the containment and contain redundant isolation valves, and an exhaust filter bank. The Containment Purge System is described in RNP UFSAR Section 9.4.3.2.6.

The license renewal evaluation boundaries for the Containment Purge System are shown on the following flow diagram. (Flow diagrams have been submitted separately for information only.)

Containment Purge System G-190304LR Sheet 1

The Containment Purge System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events
2. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions
3. SCs that are part of the Environmental Qualification Program

Table 2.3-16 below identifies the Containment Purge System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-16 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: CONTAINMENT PURGE SYSTEM

Component/Commodity	Intended Function	AMR Results
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13
Ductwork and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.3-2, Item 20
Equipment Frames and Housings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13 Table 3.3-2, Item 19
Flexible Collars	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 2
Valves	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13 Table 3.3-2, Item 19

2.3.3.11 Rod Drive Cooling System

The Rod Drive Cooling System removes heat generated by the control rod drive mechanisms. The system consists of ventilation fans, ducts, and coolers. The Rod Drive Cooling System is described in RNP UFSAR Section 9.4.3.

The license renewal evaluation boundaries for the Rod Drive Cooling System are shown on the following flow diagram. (Flow diagrams have been submitted separately for information only.)

Rod Drive Cooling System G-190304LR Sheet 1

The Rod Drive Cooling System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs that are relied on during postulated fires

Table 2.3-17 below identifies the Rod Drive Cooling System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-17 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: ROD DRIVE COOLING SYSTEM

Component/Commodity	Intended Function	AMR Results
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13
Ductwork and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 20
Equipment Frames and Housings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13 Table 3.3-2, Item 19
Flexible Collars	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 2

The Rod Drive Cooling System cooling coil is cooled by the Component Cooling Water System and is part of the pressure boundary of that system. The cooling coil is included in the Component Cooling Water System in Subsection 2.3.3.3.

2.3.3.12 HVAC Auxiliary Building

The HVAC Auxiliary Building provides heat removal to ensure proper operation of safety related equipment in the auxiliary building. The system provides clean air to the operating areas of the Auxiliary Building and filters and exhausts air from the equipment rooms and open areas of the building. The HVAC Auxiliary Building includes a separate ventilation system for the waste evaporator enclosure on the roof of the building. Also, a separate ventilation supply and exhaust system is provided for each diesel generator room and operates when the diesel generator is operating. The system provides for local cooling of safety related pump rooms. The HVAC Auxiliary Building is described in RNP UFSAR Sections 9.4.4 and 9.4.8.

The license renewal evaluation boundaries for the HVAC Auxiliary Building are shown on the following flow diagrams. (Flow diagrams have been submitted separately for information only.)

HVAC Auxiliary Building	G-190304LR Sheet 2 G-190304LR Sheet 3

The HVAC Auxiliary Building is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs that are part of the Environmental Qualification Program

3. SCs that are relied on during postulated fires

Table 2.3-18 below identifies the HVAC Auxiliary Building System components/ commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-18 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: HVAC AUXILIARY BUILDING

Component/Commodity	Intended Function	AMR Results
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13
Ductwork and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.3-1, Item 5 Table 3.3-2, Item 20
Equipment Frames and Housings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5 Table 3.3-2, Item 20
Flexible Collars	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 2
Heating/Cooling Coils	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer.	Table 3.3-1, Item 5 Table 3.3-1, Item 16 Table 3.3-2, Item 17

2.3.3.13 HVAC Control Room Area

The HVAC Control Room Area provides heating, ventilation, cooling, filtration, air intake and exhaust isolation during normal operation and following a design basis accident. The system consists of two parts: an environmental control system and an air cleanup system. The environmental control system continually operates during normal and emergency conditions. The air cleanup system normally operates only during emergency conditions. The system has three operational modes: normal ventilation, emergency pressurization, and emergency recirculation. The HVAC Control Room Area is described in RNP UFSAR Section 9.4.2.

The license renewal evaluation boundaries for the HVAC Control Room Area are shown on the following flow diagram. (Flow diagrams have been submitted separately for information only.)

HVAC Control Room Area	G-190304LR Sheet 4

The HVAC Control Room Area is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs that are relied on during postulated fires

Table 2.3-19 below identifies the HVAC Control Room System components/ commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-19 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: HVAC CONTROL ROOM AREA

Component/Commodity	Intended Function	AMR Results
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 19
Equipment Frames and Housings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components. Provide heat transfer.	Table 3.3-2, Item 12 Table 3.3-2, Item 19 Table 3.3-2, Item 23
Flexible Collars	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 2
Flow Orifices/Elements	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide flow restriction (throttle).	Table 3.3-2, Item 23
Heating/Cooling Coils	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer.	Table 3.3-1, Item 16 Table 3.3-2, Item 21 Table 3.3-2, Item 23
Valves, Piping, Tubing and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.3-2, Item 19 Table 3.3-2, Item 21 Table 3.3-2, Item 23

2.3.3.14 HVAC Fuel Handling Building

The HVAC Fuel Handling Building provides ventilation and heat removal for the fuel handling building. The system provides clean air to the operating areas of the building and filters and exhausts air from the equipment rooms and open areas of the building. The exhaust air is routed through filters. The HVAC Fuel Handling Building is described in RNP UFSAR Section 9.4.5.

The license renewal evaluation boundaries for the HVAC Fuel Handling Building are shown on the following flow diagram. (Flow diagrams have been submitted separately for information only.)

HVAC Fuel Handling Building	G-190304LR Sheet 1

The HVAC Fuel Handling Building is in the scope of license renewal, because it contains SCs that are safety-related and are relied upon to remain functional during and following design basis events.

Table 2.3-20 below identifies the HVAC Fuel Handling Building components/ commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-20 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: HVAC FUEL HANDLING BUILDING

Component/Commodity	Intended Function	AMR Results
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13
Ductwork and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.3-2, Item 20
Equipment Frames and Housings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 20
Flexible Collars	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 2

2.3.3.15 Fire Protection System

The Fire Protection System protects plant equipment in the event of a fire to ensure safe plant shutdown and to minimize the risk of a radioactive release to the environment. The Fire Protection System consists of the fire suppression systems, fire detection and actuation system, and the fire barrier system. The fire suppression system employs diverse extinguishing agents consisting of water, carbon dioxide, and Halon. Other features required for fire protection are those used to assure safe shutdown of the plant following a fire. These include alternative and dedicated shutdown equipment and emergency lighting. The scoping methodology applicable to the Fire Protection System and the systems required for safe shutdown following a fire is described in Subsection 2.1.1.3.1. Passive fire barriers were screened with Structures (see Section 2.4). Fire detection and actuation systems and emergency lighting were screened with Electrical and Instrumentation and Controls (see Section 2.5). Components that support safe shutdown following a postulated fire in accordance with 10 CFR 50, Appendix R, have been screened with their respective systems.

The Fire Protection System is described in RNP UFSAR Section 9.5.1.4. Systems relied on for safe shutdown following a fire are discussed in UFSAR Appendices 9.5.1A and 9.5.1C and UFSAR Section 7.4.

Flow diagrams were not prepared to show the evaluation boundaries for the portions of the Fire Protection System that are within the scope of license renewal. The evaluation boundaries for the portions of the Fire Protection System that are within the scope of license renewal were defined on the basis of plant documents that identify fire protection features and on the basis of functional classifications assigned to fire protection components in the RNP equipment databases. The screening process for the Fire Protection System identified additional commodity groups to be included in the system boundary, such as, manifolds and nozzles. The Fuel Oil System flow diagrams listed in Subsection 2.3.3.19 show the evaluation boundaries for the portions of the Diesel Fire Pump fuel oil system that are within the scope of license renewal.

The Fire Protection System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions

3. SCs that are part of the Environmental Qualification Program

4. SCs that are relied on during postulated fires

The following table identifies the Fire Protection System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-21COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: FIRE PROTECTION SYSTEM

Component/Commodity	Intended Function	AMR Results
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 13
Diesel Driven Fire Pump	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 20 Table 3.3-2, Item 30
Ductwork and Fittings	Provide structural support to safety-related components.	Table 3.3-1, Item 19 Table 3.3-1, Item 5
Fire Hydrants	Provide pressure retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 20 Table 3.3-1, Item 24 Table 3.3-1, Item 5
Flow Orifices/Elements	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide flow restriction (throttle).	Table 3.3-1, Item 13 Table 3.3-1, Item 20 Table 3.3-1, Item 5
Jockey Fire Pump	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 20
Motor Driven Fire Pump	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 20 Table 3.3-2, Item 30
Sprinklers	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide flow restriction (throttle).	Table 3.3-1, Item 20 Table 3.3-1, Item 24 Table 3.3-2, Item 21
Valves, Piping, Tubing and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide filtration. Provide structural support to safety-related components. Provide flow restriction (throttle).	Table 3.3-1, Item 5 Table 3.3-1, Item 13 Table 3.3-1, Item 17 Table 3.3-1, Item 20 Table 3.3-1, Item 24 Table 3.3-2, Item 4 Table 3.3-2, Item 11 Table 3.3-2, Item 14 Table 3.3-2, Item 18 Table 3.3-2, Item 19 Table 3.3-2, Item 21 Table 3.3-2, Item 23 Table 3.3-2, Item 26 Table 3.3-2, Item 27 Table 3.3-2, Item 29

Portable fire protection equipment, such as self contained breathing apparatus and fire extinguishers, is considered short-lived, is replaced on condition, and is exempted from aging management review consistent with the treatment of consumables described in Section 4.1.2 of NEI 95-10 [Reference 2.3-2]. This equipment is monitored / replaced as needed in accordance with NFPA standards under the Site Fire Protection Program.

2.3.3.16 Diesel Generator System

The Diesel Generator System provides AC power to the onsite electrical distribution system to assure the capability for a safe plant shutdown. The Diesel Generator Support Systems necessary to ensure proper operation of the Diesel Generators are:

1. Starting Air Subsystem
2. Lube Oil Subsystem
3. Jacket Water Cooling Subsystem
4. Scavenging Air Subsystem
5. Scavenging Air Cooling Subsystem
6. Diesel Engine Fuel Oil Subsystem
7. Diesel Exhaust Subsystem

The Diesel Generator System is described in RNP UFSAR Section 8.3.1.1.5. Routing of the exhaust piping outside of the Diesel Generator Rooms in the Reactor Auxiliary Building is shown on UFSAR Figures 1.2.2-5 and 1.2.2-6. As shown on the figures, exhaust gases do not directly impinge on any structures that could fail and block Diesel Generator exhaust flow.

The license renewal evaluation boundaries for the Diesel Generator System are shown on the following flow diagrams. (Flow diagrams have been submitted separately for information only.)

Diesel Generator System	G-190204ALR Sheet 1 G-190204ALR Sheet 2 G-190204ALR Sheet 3

The Diesel Generator System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events
2. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions
3. SCs that are relied on during postulated fires

The diesel engine and generator form a complex assembly as described in Section 4.1.1 of NEI 95-10 [Reference 2.3-2]. Complex assemblies are discussed in Table 2.1-2 of the NRC Standard Review Plan [Reference 2.3-3].

Table 2.3-22 below identifies the Diesel Generator System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-22 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: DIESEL GENERATOR SYSTEM

Component/Commodity	Intended Function	AMR Results
D/G After Coolant Heat Exchangers Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 14 Table 3.3-2, Item 15
D/G After Coolant Heat Exchangers Shell & Waterbox Cover	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5
D/G After Coolant Heat Exchangers Tube Sheet	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 16 Table 3.3-1, Item 24 Table 3.3-2, Item 16
D/G After Coolant Heat Exchangers Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer.	Table 3.3-1, Item 16 Table 3.3-2, Item 16
D/G After Coolant Heat Exchangers Waterbox	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5 Table 3.3-1, Item 16 Table 3.3-1, Item 24
D/G After Coolant Heat Exchangers Waterbox Cover	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 16
D/G Jacket Water and After Coolant Regulators Body/Bonnet	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 14 Table 3.3-1, Item 24 Table 3.3-2, Item 19
D/G Jacket Water Heat Exchangers Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 14 Table 3.3-2, Item 15 Table 3.3-2, Item 19
D/G Jacket Water Heat Exchangers Shell & Waterbox Cover	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5 Table 3.3-2, Item 19
D/G Jacket Water Heat Exchangers Tube Sheet	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 16 Table 3.3-1, Item 24 Table 3.3-2, Item 16
D/G Jacket Water Heat Exchangers Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer.	Table 3.3-1, Item 16 Table 3.3-2, Item 16

TABLE 2.3-22 (continued) COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: DIESEL GENERATOR SYSTEM

Component/Commodity	Intended Function	AMR Results
D/G Jacket Water Heat Exchangers Waterbox	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered	Table 3.3-1, Item 5 Table 3.3-1, Item 16 Table 3.3-1, Item 24
D/G Jacket Water Heat Exchangers Waterbox Cover	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 16
D/G Jacket Water Stand-by Heater Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 14
D/G Lube Oil Heat Exchanger Tube Sheet	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 16 Table 3.3-1, Item 24 Table 3.3-2, Item 22
D/G Lube Oil Heat Exchanger Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer.	Table 3.3-1, Item 16 Table 3.3-2, Item 22
D/G Lube Oil Heat Exchanger Waterbox	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5 Table 3.3-1, Item 16 Table 3.3-1, Item 24
D/G Lube Oil Heat Exchanger Waterbox Cover	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 16
D/G Lube Oil Heat Exchangers Shell & Waterbox Cover	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5
D/G Standby Circulating Coolant Pumps	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 14 Table 3.3-1, Item 24 Table 3.3-2, Item 19
D/G Lube Oil Temperature Regulators Body/Bonnet	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 19 Table 3.3-2, Item 22
D/G Lube Oil Heat Exchanger Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 19 Table 3.3-2, Item 22
DG Main Bearing Oil Booster Regulators Body/Bonnet	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 19
DG Air Supply Regulators To Jacking Gear Body/Bonnet	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 19
DG Lube Oil Filters	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 19 Table 3.3-2, Item 22
DG Pre-Lube Oil Pumps	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 19 Table 3.3-2, Item 22
DG Lube Oil Heaters Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 19 Table 3.3-2, Item 22
DG Lube Oil Strainers	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide filtration	Table 3.3-2, Item 19 Table 3.3-2, Item 22

TABLE 2.3-22 (continued) COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: DIESEL GENERATOR SYSTEM

Component/Commodity	Intended Function	AMR Results
Diesel Air Exhaust Silencer	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.3-1, Item 5
Diesel Air Intake Silencer Filters	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.3-1, Item 5 Table 3.3-2, Item 19
Emergency Diesel Air Start Strainers	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide filtration.	Table 3.3-1, Item 18 Table 3.3-2, Item 19 Table 3.3-2, Item 21
Emergency Diesel Generator Air Receiver Tanks	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 18 Table 3.3-2, Item 19
Emergency Diesel Generator Jacket Water Expansion Tanks	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 14 Table 3.3-2, Item 19
Flow Orifices/Elements	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide flow restriction (throttle)	Table 3.3-1, Item 14 Table 3.3-2, Item 19
Lube Oil Recirc Standby Pumps	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 19 Table 3.3-2, Item 22
Starting Air Compressor Diesel Generator Unloaders Regulator Body/Bonnet	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.3-2, Item 19
Valves, Piping, Tubing and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components. Provide filtration.	Table 3.3-1, Item 5 Table 3.3-1, Item 14 Table 3.3-1, Item 16 Table 3.3-1, Item 18 Table 3.3-1, Item 24 Table 3.3-2, Item 5 Table 3.3-2, Item 16 Table 3.3-2, Item 19 Table 3.3-2, Item 21 Table 3.3-2, Item 22 Table 3.3-2, Item 23

2.3.3.17 Dedicated Shutdown Diesel Generator

The Dedicated Shutdown Diesel Generator provides an alternate AC power source to assure the capability for a safe plant shutdown following a fire or total loss of all AC power (Station Blackout). The skid-mounted, self-contained Dedicated Shutdown Diesel Generator is a component of the Dedicated Shutdown Electrical System. The Dedicated Shutdown Diesel Generator is described in RNP UFSAR Section 8.3.1.1.2 and Figure 8.3.1-4.

The license renewal evaluation boundaries for the Dedicated Shutdown Diesel Generator are shown on the following flow diagrams. (Flow diagrams have been submitted separately for information only.)

Dedicated Shutdown Diesel Generator	HBR2-08679LR Sheet 1 HBR2-08679LR Sheet 2

The Dedicated Shutdown Diesel Generator System is in the scope of license renewal, because it contains SCs that are relied on during postulated fires and station blackout events.

The diesel engine and generator form a complex assembly as described in Section 4.1.1 of NEI 95-10 [Reference 2.3-2]. Complex assemblies are discussed in Table 2.1-2 of the NRC Standard Review Plan for license renewal [Reference 2.3-3].

Table 2.3-23 below identifies the Dedicated Shutdown Diesel Generator System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-23 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: DEDICATED SHUTDOWN DIESEL GENERATOR

Component/Commodity	Intended Function	AMR Results
Diesel Air Exhaust Silencer	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5
DSD Air Vacuum Box Filter	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5
DSD Air Volume Tank	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5 Table 3.3-1, Item 18 Table 3.3-2, Item 10
DSD Expansion Tank	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5 Table 3.3-1, Item 14 Table 3.3-2, Item 15
DSD Immersion Heater	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 14 Table 3.3-2, Item 15 Table 3.3-2, Item 19
DSD Lube Oil Circulating Pump	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 19 Table 3.3-2, Item 22

TABLE 2.3-23 (continued) COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: DEDICATED SHUTDOWN DIESEL GENERATOR

Component/Commodity	Intended Function	AMR Results
DSD Lube Oil Cooler Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 14 Table 3.3-2, Item 15
DSD Lube Oil Cooler Tubing and Channels	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer.	Table 3.3-2, Item 22
DSD Lube Oil Cooler Channel and Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 19
DSD Lube Oil Cooler Tubing and Fins	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer.	Table 3.3-1, Item 24 Table 3.3-2, Item 16
DSD Lube Oil Filter	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 19 Table 3.3-2, Item 22
DSD Lube Oil Strainer	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide filtration.	Table 3.3-2, Item 19 Table 3.3-2, Item 22
DSD Radiator Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer.	Table 3.3-1, Item 5 Table 3.3-1, Item 14 Table 3.3-2, Item 15
DSD Radiator Waterbox	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5 Table 3.3-1, Item 14 Table 3.3-2, Item 15 Table 3.3-2, Item 16
DSD Soak Back Oil Filter	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5 Table 3.3-2, Item 22
DSD Turbocharger Oil Filter	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5 Table 3.3-2, Item 22
DSD Turbocharger Soak Back Pump	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5 Table 3.3-2, Item 22
DSDG Air Compressor Filter	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5
Ductwork and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5
Valves, Piping, Tubing and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered Provide flow restriction (throttle). Provide structural support to safety-related components.	Table 3.3-1, Item 5 Table 3.3-1, Item 14 Table 3.3-1, Item 17 Table 3.3-1, Item 18 Table 3.3-2, Item 5 Table 3.3-2, Item 10 Table 3.3-2, Item 12 Table 3.3-2, Item 13 Table 3.3-2, Item 16 Table 3.3-2, Item 19 Table 3.3-2, Item 21 Table 3.3-2, Item 22 Table 3.3-2, Item 23

2.3.3.18 EOF/TSC Security Diesel Generator

The EOF/TSC Security Diesel Generator provides backup electrical power to the Emergency Operations Facility/Technical Support Center Building and security systems upon loss of the normal power supplies. The backup electrical power supplied to security lighting in outside areas is relied on for performance of actions required for fire protection safe shutdown as identified in UFSAR Appendix 9.5.1B, Section D.5.

No flow diagram drawings are shown of the EOF/TSC Security Diesel Generator itself, because it is self-contained and skid-mounted. The Fuel Oil System flow diagrams listed in Subsection 2.3.3.19 show the evaluation boundaries for the portions of the fuel oil system supporting the EOF/TSC Security Diesel Generator that are within the scope of license renewal.

The EOF/TSC Security Diesel Generator System is in the scope of license renewal, because it contains SCs that are relied on during postulated fires.
The diesel engine and generator form a complex assembly as described in Section 4.1.1 of NEI 95-10 [Reference 2.3-2]. Complex assemblies are discussed in Table 2.1-2 of the NRC Standard Review Plan for license renewal [Reference 2.3-3].

Table 2.3-24 below identifies the EOF/TSC Security Diesel Generator System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-24 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: EOF/TSC SECURITY EMERGENCY DIESEL GENERATOR

Component/Commodity	Intended Function	AMR Results
Ductwork and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 20
EOF DG Intake Filters	Provide filtration.	Table 3.3-2, Item 23
EOF DG Exhaust Silencer	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5
EOF DG Jacket Water Immersion Heaters	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 14 Table 3.3-2, Item 19
EOF DG Radiator	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer.	Table 3.3-1, Item 5 Table 3.3-1, Item 14 Table 3.3-2, Item 15
Valves, Piping, Tubing and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 14 Table 3.3-1, Item 24 Table 3.3-2, Item 5 Table 3.3-2, Item 16 Table 3.3-2, Item 19 Table 3.3-2, Item 21 Table 3.3-2, Item 22

2.3.3.19 Fuel Oil System

The Fuel Oil System supplies fuel oil to the Emergency Diesel Engines, the Dedicated Shutdown Diesel Engine, and the Diesel Engine-Driven Fire Pump from fuel oil storage tanks on site. The Fuel Oil System also provides fuel oil to the EOF/TSC Security Diesel Generator. The Fuel Oil System provides a fuel oil storage capacity sufficient to operate an Emergency Diesel Generator at full load in accordance with Technical Specification requirements.

The license renewal evaluation boundaries for the Fuel Oil System are shown on the following flow diagrams. (Flow diagrams have been submitted separately for information only.)

Fuel Oil System	G-190204DLR Sheet 1 G-190204DLR Sheet 2 G-190204DLR Sheet 3

The Fuel Oil System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions

3. SCs that are relied on during postulated fires and station blackout events

Table 2.3-25 below identifies the Fuel Oil System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-25 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: FUEL OIL SYSTEM

Component/Commodity	Intended Function	AMR Results
Diesel Fire Pump Fuel Oil Tank	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 7 Table 3.3-1, Item 22
Diesel Oil Storage Tank Vent Filter	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide filtration.	Table 3.3-1, Item 22
DSD Fuel Oil Day Tank	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 7 Table 3.3-1, Item 22
DSD Fuel Oil Priming Pumps	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5 Table 3.3-1, Item 7

TABLE 2.3-25 (continued) COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: FUEL OIL SYSTEM

Component/Commodity	Intended Function	AMR Results
DSD Fuel Oil Pumps	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5 Table 3.3-1, Item 7
DSD Fuel Oil Tank	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 7 Table 3.3-1, Item 22
EDG Day Tank Vent Filters	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide filtration.	Table 3.3-1, Item 5 Table 3.3-1, Item 22
EDG Fuel Oil Day Tanks	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 7 Table 3.3-1, Item 22
EDG Fuel Oil Duplex Filters	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide filtration.	Table 3.3-1, Item 7 Table 3.3-2, Item 19
EDG Fuel Oil Hand Priming Pumps	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5 Table 3.3-1, Item 7
EDG Fuel Oil Storage Tank	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 7 Table 3.3-1, Item 22
EOF DG Fuel Oil Day Tank	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 7 Table 3.3-1, Item 22
EOF DG Fuel Oil Pump	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5 Table 3.3-1, Item 7
EOF/TSC Main Storage Tank	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 7 Table 3.3-2, Item 28
Flow Orifices/Elements	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5 Table 3.3-1, Item 7
Fuel Oil Transfer Pumps	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 5 Table 3.3-1, Item 7
Unit 1 IC Turbine Tanks	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 7 Table 3.3-1, Item 22
Valves, Piping, Tubing and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide filtration. Provide structural support to safety-related components.	Table 3.3-1, Item 5 Table 3.3-1, Item 7 Table 3.3-1, Item 17 Table 3.3-2, Item 5 Table 3.3-2, Item 12 Table 3.3-2, Item 13 Table 3.3-2, Item 19 Table 3.3-2, Item 21 Table 3.3-2, Item 23 Table 3.3-2, Item 28

2.3.4 STEAM AND POWER CONVERSION SYSTEMS

The Steam and Power Conversion Systems act as a heat sink to remove heat from the reactor and convert the heat generated in the reactor to the plant's electrical output. The following systems are included in this Subsection:

1. Turbine System
2. Electro-Hydraulic Control System
3. Turbine Generator Lube Oil System
4. Extraction Steam System
5. Main Steam System
6. Steam Generator Blowdown System
7. Steam Cycle Sampling
8. Feedwater System
9. Auxiliary Feedwater System
10. Condensate System
11. Steam Generator Chemical Addition
12. Circulating Water System

2.3.4.1 Turbine System

The Turbine System converts the thermal energy of the steam from the Main Steam System into mechanical energy used to drive the main generator and produce the plant's electrical output. Turbine System valves provide overspeed trip of the turbine to prevent generation of turbine blade missiles. The Turbine System is described in RNP UFSAR Section 10.2.2.

The evaluation boundaries for the applicable portions of the Turbine System were defined on the basis of plant documentation that presents a listing of components within the evaluation boundary of the system.

The Turbine System was conservatively included in the scope of license renewal, because it contains SCs that are non-safety related whose failure may prevent satisfactory accomplishment of safety related functions and SCs that are relied on during postulated anticipated transients without scram events. These functions are accomplished by providing protection from turbine overspeed or maintaining the integrity of the low-pressure turbine rotor. However, a review of the Turbine System design and component functions during the mechanical system screening process concluded that either (1) the system functions are performed by active components, or (2) any failure of component pressure boundary would not prevent the performance of the system intended functions. This conclusion is consistent with the information presented in the NRC Standard Review Plan for License Renewal, Table 2.1-5 for turbine controls that provide overspeed protection. The screening review concluded that the Turbine System components do not perform any intended functions for license renewal; therefore, none of the Turbine System components are subject to an aging management review.

2.3.4.2 Electro-Hydraulic Control System

The Electro-Hydraulic Control System controls the flow of steam to the Turbine System through all phases of turbine operation. The system also provides overspeed trip of the turbine to prevent generation of turbine blade missiles. The Electro-Hydraulic Control System is described in RNP UFSAR Section 10.2.2.

The evaluation boundaries for the applicable portions of the Electro-Hydraulic Control System were defined on the basis of plant documentation that presents a listing of components within the evaluation boundary of the system.

The Electro-Hydraulic Control System was conservatively included in the scope of license renewal, because it contains SCs which are non-safety related whose failure may prevent satisfactory accomplishment of safety related functions. However, a review of the Electro-Hydraulic Control System design and component functions during the mechanical system screening process concludes that (1) the system function is performed by active components, and (2) any failure of component pressure boundary would not prevent the performance of the system intended function. This conclusion is consistent with the information presented in the NRC Standard Review Plan for License Renewal, Table 2.1-5 for turbine overspeed trip components. The screening review concluded that the Electro-Hydraulic Control System components do not perform any intended functions for license renewal; therefore, none of the Electro-Hydraulic Control System components are subject to an aging management review.

2.3.4.3 Turbine Generator Lube Oil System

The Turbine Generator Lube Oil System provides oil for cooling and lubricating the turbine bearings and turning gear. The system also provides pressurized oil to the Turbine System overspeed and protective trip devices. The Turbine Generator Lube Oil System is described in RNP UFSAR Section 10.2.2.

The evaluation boundaries for the applicable portions of the Turbine Generator Lube Oil System were defined on the basis of plant documentation that presents a listing of components within the evaluation boundary of the system.

The Turbine Generator Lube Oil System was conservatively included in the scope of license renewal, because it contains SCs that are non-safety related whose failure may prevent satisfactory accomplishment of safety related functions. However, a review of the Turbine Generator Lube Oil System design and component functions during the mechanical system screening process concludes that (1) the system function is performed by active components, and (2) any failure of component pressure boundary would not prevent the performance of the system intended function. This conclusion is consistent with the information presented in the NRC Standard Review Plan for License Renewal, Table 2.1-5 for turbine controls. Therefore, none of the Turbine Generator Lube Oil System components are subject to aging management review.

2.3.4.4 Extraction Steam System

The Extraction Steam System provides reheating and moisture removal for the steam flow from the high pressure turbine before it is supplied to the low pressure turbines. The system also provides overspeed protection by providing valves to stop the flow of reheat steam to the low pressure turbine. The Extraction Steam System is described in RNP UFSAR Section 10.3.2.

The license renewal evaluation boundaries for the Extraction Steam System are shown on the following flow diagram. (Flow diagrams have been submitted separately for information only.)

Extraction Steam System	G-190196LR Sheet 1

The Extraction Steam System conservatively was included in the scope of license renewal, because it was identified as having SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions.

Following screening of the Extraction Steam System, it was concluded that none of the system components perform an intended function without moving parts or without a change in configuration. Therefore, none of the components in the Extraction Steam System boundaries is subject to an aging management review.

2.3.4.5 Main Steam System

The Main Steam System transports saturated steam from the steam generators to the main turbine and other secondary steam system components. The system is the principal heat sink for the Reactor Coolant System and protects the Reactor Coolant System and the steam generators from overpressurization. The Main Steam System provides isolation of the steam generators following a postulated accident, such as a steam line break, and provides steam supply to the Steam Driven Auxiliary Feedwater Pump. The Main Steam System is described in RNP UFSAR Section 10.3.

The license renewal evaluation boundaries for the Main Steam System are shown on the following flow diagram. (Flow diagrams have been submitted separately for information only.)

Main Steam System	G-190196LR Sheet 1

The Main Steam System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events
2. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions
3. SCs that are part of the Environmental Qualification Program
4. SCs that are relied on during postulated fires, anticipated transients without scram, and station blackout events

Table 2.3-26 below identifies the Main Steam System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-26 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: MAIN STEAM SYSTEM

Component/Commodity	Intended Function	AMR Results
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.4-1, Item 13
Flow Orifices/Elements	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide flow restriction (throttle).	Table 3.4-1, Item 1 Table 3.4-1, Item 7 Table 3.4-1, Item 13 Table 3.4-2, Item 2 Table 3.4-2, Item 7 Table 3.4-2, Item 8 Table 3.4-2, Item 13
MSIV Accumulator Tank(s)	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.4-1, Item 5 Table 3.4-2, Item 11

TABLE 2.3-26 (continued) COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: MAIN STEAM SYSTEM

Component/Commodity	Intended Function	AMR Results
Valves, Piping, Tubing, and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components. Provide filtration.	Table 3.4-1, Item 1 Table 3.4-1, Item 5 Table 3.4-1, Item 6 Table 3.4-1, Item 7 Table 3.4-1, Item 13 Table 3.4-2, Item 2 Table 3.4-2, Item 6 Table 3.4-2, Item 7 Table 3.4-2, Item 8 Table 3.4-2, Item 11 Table 3.4-2, Item 12 Table 3.4-2, Item 13

2.3.4.6 Steam Generator Blowdown System

The Steam Generator Blowdown System assists in maintaining required steam generator chemistry by providing a means for removal of foreign matter that concentrates in the steam generators. The system is fed by three independent blowdown lines (one per steam generator) that penetrate containment and tie to a common blowdown drain tank. The Steam Generator Blowdown System is described in RNP UFSAR Section 10.4.7.

The license renewal evaluation boundaries for the Steam Generator Blowdown System are shown on the following flow diagram. (Flow diagrams have been submitted separately for information only.)

Steam Generator Blowdown System	G-190234LR Sheet 1

The Steam Generator Blowdown System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events
2. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions
3. SCs that are part of the Environmental Qualification Program
4. SCs that are relied on during postulated station blackout events

Table 2.3-27 below identifies the Steam Generator Blowdown System components/ commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-27 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: STEAM GENERATOR BLOWDOWN SYSTEM

Component/Commodity	Intended Function	AMR Results
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.4-1, Item 13
Flow Orifices/Elements	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.4-1, Item 1 Table 3.4-1, Item 2 Table 3.4-2, Item 2 Table 3.4-2, Item 13
Valves, Piping, and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.4-1, Item 1 Table 3.4-1, Item 2 Table 3.4-1, Item 5 Table 3.4-1, Item 6 Table 3.4-1, Item 13 Table 3.4-2, Item 1 Table 3.4-2, Item 2 Table 3.4-2, Item 6 Table 3.4-2, Item 11 Table 3.4-2, Item 13

2.3.4.7 Steam Cycle Sampling

The Steam Cycle Sampling system provides for sampling and analysis of steam generator liquid via sample lines connected to the Steam Generator Blowdown System. A separate sample line is provided for each steam generator blowdown line.

The license renewal evaluation boundaries for the Steam Cycle Sampling system are shown on the following flow diagram. (Flow diagrams have been submitted separately for information only.)

Steam Cycle Sampling	HBR2-09006LR Sheet 2

The Steam Cycle Sampling system is in the scope of license renewal, because it contains SCs that are safety-related and are relied upon to remain functional during and following design basis events.

The only components with an intended function in the Steam Cycle Sampling system are sample heat exchangers. The heat exchanger shells are cooled by the Component Cooling Water System; therefore, the shell and tubes form part of the pressure boundary of that system. Aging management review of the heat exchanger is addressed under the Component Cooling Water System in Subsection 2.3.3.3.

2.3.4.8 Feedwater System

The Feedwater System provides pre-heated, high pressure feedwater to the steam generators under operating conditions. The system provides for feedwater and blowdown isolation following a postulated loss of coolant accident or steam line break event, and assists in maintaining steam generator water chemistry. SG level is controlled to ensure proper water inventory for various operational and accident conditions. The control is achieved by variations in the feedwater flowrate. The Feedwater System is described in RNP UFSAR Section 10.4.6.

The license renewal evaluation boundaries for the Feedwater System are shown on the following flow diagrams. (Flow diagrams have been submitted separately for information only.)

Feedwater System	G-190197LR Sheet 1
	G-190197LR Sheet 3
	G-190197LR Sheet 4

The Feedwater System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions

3. SCs that are relied on during postulated fires and station blackout events

Table 2.3-28 below identifies the Feedwater System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-28 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: FEEDWATER SYSTEM

Component/Commodity	Intended Function	AMR Results
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.4-1, Item 13
Feedwater Heater Heat Exchanger Cover / Tubesheet	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.4-1, Item 1 Table 3.4-1, Item 2 Table 3.4-2, Item 1 Table 3.4-2, Item 3
Feedwater Heater Heat Exchanger Cover	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.4-2, Item 11
Feedwater Heater Heat Exchanger Tube Sheet	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.4-1, Item 1 Table 3.4-1, Item 2 Table 3.4-2, Item 1 Table 3.4-2, Item 3
Feedwater Heater Heat Exchanger Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.4-1, Item 1 Table 3.4-1, Item 2 Table 3.4-2, Item 2
Flow Orifices/Elements	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide flow restriction (throttle).	Table 3.4-1, Item 1 Table 3.4-1, Item 2 Table 3.4-1, Item 5 Table 3.4-2, Item 1 Table 3.4-2, Item 2
Temperature Elements	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.4-1, Item 1 Table 3.4-1, Item 2 Table 3.4-1, Item 5 Table 3.4-1, Item 6 Table 3.4-2, Item 1 Table 3.4-2, Item 6
Valves, Piping, Tubing, and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.4-1, Item 1 Table 3.4-1, Item 2 Table 3.4-1, Item 5 Table 3.4-1, Item 6 Table 3.4-1, Item 13 Table 3.4-2, Item 1 Table 3.4-2, Item 2 Table 3.4-2, Item 6 Table 3.4-2, Item 11 Table 3.4-2, Item 12 Table 3.4-2, Item 13

2.3.4.9 Auxiliary Feedwater System

The Auxiliary Feedwater System supplies feedwater to the steam generators when normal feedwater sources are not available. The system provides for isolation of flow to a faulted steam generator following postulated accidents, such as a steam generator tube rupture or main steam line break.

The Auxiliary Feedwater System can provide feedwater to any combination of steam generators from any one or combination of three pumps; two are motor-driven, and the third is steam-driven. Steam can be supplied to the steam-driven pump from any of the steam generators. The pumps can take suction from the Condensate Storage Tank, which is the normal source, or from the Service Water System or the deepwell pumps if the Condensate Storage Tank is not available. The steam driven pump provides an independent and diversely powered means of providing feedwater to the steam generators. The steam driven system provides the required flow through injection lines that are separate from the motor driven subsystem. The Auxiliary Feedwater System is described in RNP UFSAR Section 10.4.8.

The license renewal evaluation boundaries for the Auxiliary Feedwater System are shown on the following flow diagram. (Flow diagrams have been submitted separately for information only.)

Auxiliary Feedwater System	G-190197LR Sheet 4

The Auxiliary Feedwater System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events
2. SCs that are relied on during postulated fires, anticipated transients without scram, and station blackout events

Table 2.3-29 below identifies the Auxiliary Feedwater System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-29 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: AUXILIARY FEEDWATER SYSTEM

Component/Commodity	Intended Function	AMR Results
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.4-1, Item 13
Flow Orifices/Elements	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide flow restriction (throttle).	Table 3.4-1, Item 2 Table 3.4-2, Item 1 Table 3.4-2, Item 2 Table 3.4-2, Item 11 Table 3.4-2, Item 13
SDAFW and MDAFW Pump Lube Oil Heat Exchanger Tubing	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer.	Table 3.4-2, Item 4 Table 3.4-2, Item 9
SDAFW and MDAFW Pump Lube Oil Heat Exchanger Waterbox	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.4-1, Item 9 Table 3.4-2, Item 4 Table 3.4-2, Item 10 Table 3.4-2, Item 11
SDAFW and MDAFW Pump Lube Oil Heat Exchanger Tubing and Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide heat transfer.	Table 3.4-2, Item 14
SDAFW and MDAFW Pump Lube Oil Heat Exchanger Shell	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.4-2, Item 12
SDAFW Pump Lube Oil Pump	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.4-2, Item 11 Table 3.4-2, Item 14
SDAFW and MDAFW Pump	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.4-1, Item 2 Table 3.4-2, Item 1 Table 3.4-2, Item 11
SDAFW Turbine	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.4-2, Item 7 Table 3.4-2, Item 11
Valves, Piping, Tubing, and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.4-1, Item 1 Table 3.4-1, Item 2 Table 3.4-1, Item 6 Table 3.4-1, Item 13 Table 3.4-2, Item 1 Table 3.4-2, Item 2 Table 3.4-2, Item 6 Table 3.4-2, Item 11 Table 3.4-2, Item 12 Table 3.4-2, Item 13 Table 3.4-2, Item 14 Table 3.4-2, Item 15

2.3.4.10 Condensate System

The Condensate System provides makeup grade water to the steam generators for removing decay and sensible heat from the Reactor Coolant System. The Condensate System provides a passive flow of water, by gravity, to the Auxiliary Feedwater System to support safe shutdown of the plant. The Condensate System consists of a condensate storage tank with piping to the suctions of all three Auxiliary Feedwater System pumps.

The Condensate System is described in RNP UFSAR Section 9.2.5.

The license renewal evaluation boundaries for the Condensate System are shown on the following flow diagram. (Flow diagrams have been submitted separately for information only.)

Condensate System	G-190197LR Sheet 1

The Condensate System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions

3. SCs that are relied on during postulated fires and station blackout events

Table 2.3-30 below identifies the Condensate System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-30 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: CONDENSATE SYSTEM

Component/Commodity	Intended Function	AMR Results
Condensate Storage Tank	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural and/or functional support to non-safety related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions.	Table 3.4-1, Item 2 Table 3.4-2, Item 5 Table 3.4-2, Item 13
Flow Orifices/Elements	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.4-1, Item 2 Table 3.4-2, Item 13
Valves, Piping, Tubing, and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.4-1, Item 2 Table 3.4-1, Item 5 Table 3.4-1, Item 6 Table 3.4-2, Item 1 Table 3.4-2, Item 6 Table 3.4-2, Item 13

2.3.4.11 Steam Generator Chemical Addition

The Steam Generator Chemical Addition system provides for chemical addition to the Feedwater System for proper steam generator chemistry control. Portions of the system provide pressure boundary integrity for the Feedwater and Auxiliary Feedwater Systems.

The license renewal evaluation boundaries for the Steam Generator Chemical Addition system are shown on the following flow diagram. (Flow diagrams have been submitted separately for information only.)

Steam Generator Chemical Addition G-190204CLR Sheet 1

The Steam Generator Chemical Addition system is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs that are relied on during postulated station blackout events

Table 2.3-31 below identifies the Steam Generator Chemical Addition system components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-31 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: STEAM GENERATOR CHEMICAL ADDITION SYSTEM

Steam Generator Chemical Addition System		
Component/Commodity	**Intended Function**	**AMR Results**
Valves, Piping and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.4-1, Item 1 Table 3.4-1, Item 2 Table 3.4-1, Item 5 Table 3.4-2, Item 1 Table 3.4-2, Item 11

2.3.4.12 Circulating Water System

The Circulating Water System provides cooling water from Lake Robinson to the main condensers to condense the steam discharged from the Turbine System. Portions of the system provide a flow path for the Service Water System flow. The Circulating Water System is described in RNP UFSAR Section 10.4.5.

The license renewal evaluation boundaries for the Circulating Water System are shown on the following flow diagram. (Flow diagrams have been submitted separately for information only.)

Circulating Water System	G-190199LR Sheet 1

The Circulating Water System is in the scope of license renewal, because it contains:

1. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions
2. SCs that are relied on during postulated fires and station blackout events

Table 2.3-32 below identifies the Circulating Water System components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.3-32 COMPONENT/COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: CIRCULATING WATER SYSTEM

Circulating Water System		
Component/Commodity	**Intended Function**	**AMR Results**
Piping and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-1, Item 20 Table 3.3-2, Item 18 Table 3.3-2, Item 31

2.3.5 REFERENCES

2.3-1 NUREG-1801, “Generic Aging Lessons Learned (GALL),” U. S. Nuclear Regulatory Commission, April 2001.

2.3-2 NEI 95-10, “Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule," Revision 3.

2.3-3 NUREG-1800, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” U. S. Nuclear Regulatory Commission, April 2001.

2.4 SCOPING AND SCREENING RESULTS - STRUCTURES

The determination of structures within the scope of license renewal is made by initially identifying RNP structures and then reviewing them to determine which ones satisfy one or more of the criteria contained in 10 CFR 54.4. A description of this process is provided in Section 2.1, and the results of the structures scoping review are contained in Section 2.2.

Section 2.1 also provides the methodology for determining the structures and structural components (SCs) within the scope of 10 CFR 54.4 that meet the requirements contained in 10 CFR 54.21(a)(1). The SCs that meet these screening requirements are identified in this section. These identified SCs subsequently require an aging management review for license renewal.

Note that the methodology for screening SCs, described in Subsection 2.1.2.2, includes screening of components and commodities that have been transferred to the Civil discipline from the Mechanical and Electrical disciplines. Evaluation boundaries between mechanical components, electrical components, and structures and structural components are coordinated between discipline reviewers. The types of components and commodities treated in this manner include pipe/component snubbers, fire damper penetration seals, electrical component supports, and electrical cabinets, consoles, cubicles, junction boxes, and panels.

The screening results are provided below in two Subsections: (1) Containment, and (2) Other structures.

2.4.1 CONTAINMENT

The Containment structure is a steel lined concrete shell in the form of a vertical right-circular cylinder with a hemispherical dome and a flat base. The Containment encloses the reactor and major components of the reactor coolant system and other important systems that interface with the Reactor Coolant System. Also, the Containment houses and supports components required for reactor refueling. This includes the polar crane, refueling cavity, and portions of the fuel handling system. The Containment is described in Section 3.8.1 of the RNP UFSAR.

Containment structural components requiring an aging management review are identified and discussed in the following three subsections: (1) Containment Structure, (2) Containment Internal Structural Components, and (3) Containment external structural components that surround and provide protection for the equipment and personnel hatches.

2.4.1.1 Containment Structure

2.4.1.1.1 Concrete

DOME AND CYLINDER WALLS

The Containment structure consists of a reinforced concrete cylinder, post-tensioned in the vertical direction, with a hemispherical dome. The dome is constructed of reinforced concrete. Cast-in-place concrete is used for the containment dome and shell (cylinder wall). The combined strength provided by the concrete, reinforcing steel, and the post tensioning system is used to satisfy the design loading conditions. Concrete specifications are identified in RNP UFSAR Section 3.8.1.6.

BASE SLAB

The dome and cylinder walls are supported by the base slab. The base slab is supported by steel pipe piles that restrain it both vertically and horizontally and are anchored to it where required to provide restraint for uplift. The reactor sump (also called the containment sump) is hung from the base slab.

Water stops are used at construction joints in the base slab and reactor sump and between the base slab and the RHR Pit. Also, waterproofing membrane was installed on the reactor sump structure to inhibit the intrusion of groundwater. Water stops and waterproofing membrane are considered to be subcomponents of the concrete walls and slabs.

FLOOR

A reinforced concrete floor is provided in containment, above the floor liner, to protect the liner plate from punctures and corrosion that could breach the essentially leak proof membrane. The containment floor slab is shown on UFSAR Figure 3.8.1-2.

2.4.1.1.2 Steel

LINER PLATE

The interior of the containment is lined with steel plates that are welded together. The liner plate covers the dome, cylinder walls, reactor sump, and the base slab and forms a leak proof membrane. The liner is not relied upon for the structural integrity of the containment except for resisting tangential shears in the dome. The Containment liner plate is discussed in UFSAR Section 3.8.1.

The inside surface of the liner plate was originally coated with a zinc-rich primer and an alkyd top coat from elevation 228 ft. to approximately elevation 352 ft. Above this, the inside surface of the liner was coated with a zinc based primer and phenolic epoxy topcoat. The liner plate is protected by insulation and sheathing up to elevation 367'10". The program for maintaining the coatings inside containment is provided in the RNP response to NRC Generic Letter 98-04 [Reference 2.4-1]. The containment floor liner is installed on top of the structural base slab and covered by the concrete floor. A moisture barrier seal is provided at the juncture of the floor and vertical liner plate to prevent moisture intrusion.

ANCHORS AND EMBEDMENTS

Structural steel commodities include anchors and embedments such as anchor studs that are welded to the liner and serve to anchor the liner to the containment shell. Penetration frames and reinforcing plates are embedded in concrete and provide continuity of the reinforcement of the containment in penetration areas and in the dome.

PENETRATIONS

In general, a penetration consists of a sleeve embedded in the concrete wall and welded to the containment liner. The process fluid or electrical penetration passes through the sleeve and the ends of the resulting annulus are sealed, either by welded plates, bolted flanges, bellows, or a combination of these.

Fuel Transfer Tube

The fuel transfer tube links the refueling canal inside the Containment to the spent fuel pool in the Fuel Handling Building. During normal operation, the inside and outside of the fuel transfer tube are dry; and a blind flange is installed and serves as part of the Containment essentially leak-tight barrier. Outside containment, the fuel transfer tube gate valve provides isolation, but the gate valve is not part of the essentially leak-tight barrier. The penetration consists of a 20 in. stainless steel pipe installed inside a 24 in. stainless steel sleeve. The Fuel Transfer Tube is described in UFSAR Sections 3.8.1 and 6.2.4 and on Figure 3.8.1-16.

Penetration Assemblies (Mechanical)

Mechanical penetrations provide the means for passage of process piping and ducts across the containment boundary. The process line may contain high or low temperature fluids. With some exceptions, double barrier piping penetrations are provided. This design consists of a sleeve welded to the liner and connected to the process line by bellows, end plates, or a combination of these. Connections are provided to pressurize the interior of double barrier penetrations to assure leak tight integrity. The piping in penetrations serving hot process fluids is insulated. Mechanical penetrations are described in UFSAR Section 3.8.1.1 and on Figure 3.8.1-15. The ECCS sump suction line guard pipes are described in UFSAR Section 6.3.2.5.

Penetration Assemblies (Electrical)

Electrical penetrations provide the means for electrical and instrumentation conductors to cross the Containment boundary while maintaining an essentially leak-tight barrier. Most electrical penetrations are the cartridge type consisting of a hollow cylinder sealed on both ends and welded to the penetration sleeve. The cartridge is provided with pressurization connections for leak detection. Certain electrical penetrations are the capsule type having a double pressure barrier design in their header plates; therefore, an endplate is required at one end only. Electrical penetrations design is discussed in UFSAR Section 3.8.1.1 and on Figures 3.8.1-14 and Figure 3.8.1-14A.

Equipment Hatch

The equipment hatch is a large flanged penetration that provides access to the containment interior for large equipment. The hatch consists of a bolted, dished door with a double-gasketed flange. The hatch barrel is embedded in the containment wall and is welded to the liner. Provision is made to pressurize the space between the double gaskets of the door flanges and the weld seam channels at the liner joint, hatch flanges, and dished door. The equipment hatch is described in UFSAR Section 3.8.1.1 and shown on UFSAR Figure 1.2.2-4.

Personnel Hatch

The containment personnel hatch (or airlock) consists of a cylindrical steel tube that passes through the concrete wall of containment and is welded to the liner. It has a bulkhead, with an air lock door, at each end. The doors are interlocked to prevent simultaneous opening. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity of the seals. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door). The personnel hatch is shown on UFSAR Figure 1.2.2-4.

REINFORCING STEEL (REBAR)

Rebar is used in the containment dome, cylinder, and base slab. Splices in rebar have been butt weld connected using Cadweld splices designed to develop the minimum ultimate tensile stress of the rebar. Embedding the reinforcing steel in structural concrete provides corrosion protection for the steel components. Cathodic protection for the reinforcing steel was not employed, because the steel embedded in properly compounded structural concrete is considered immune from the effects of galvanic corrosion. Rebar is discussed in UFSAR Sections 3.8.1.1.4 and 3.8.1.6.

PILINGS

The containment is supported on steel pipe pile foundations. Pilings restrain the containment base slab both vertically and horizontally and safely transmit the structural loads through the surface soils to the dense soils underlying the site. The piles consist of steel pipe driven closed-end and filled with concrete after driving.

The loss of steel from piles underneath the containment structure is expected to be very small in the absence of cathodic protection because of the high-resistivity soil environment. The only point where any additional protection was required was at the junction of bare pile steel and the concrete foundation. At this point, two coats of cold-applied “Bitumastic” sealer, or equal, has been applied so that the coating extends into the reinforced concrete and to a depth of at least 12 in. below the concrete.

The design and construction of the pile foundations for the containment are discussed in UFSAR Sections 3.8.1.1.2 and 3.8.5.1.1. The plan layout for the piles is shown on Figure 3.8.1-3 of the UFSAR.

2.4.1.1.3 Post Tensioning System

The post-tensioning system consists of vertical tendons located on the centerline of the wall spaced approximately every three feet around the periphery of the containment. Tendons made up of high-strength steel bars (six bars per tendon) are placed within 6-inch diameter, heavy wall galvanized steel pipe sheaths. After the tendons were tensioned, the sheaths were filled with Portland cement grout. Injection of Portland cement grouting material into the galvanized sheaths surrounding the tendons provides corrosion protection for the steel components of the Post Tensioning System. Cathodic protection for the post stressing tendons was not employed, because the steel embedded in sand-cement grout is considered immune from the effects of galvanic corrosion. The Containment post tensioning system is described in UFSAR Sections 3.8.1.1.8 and 3.8.1.6.

2.4.1.1.4 Containment Liner Insulation

The liner on the containment cylinder wall is insulated to limit stresses caused by the high containment temperature following a postulated loss-of-coolant accident. The containment liner insulation extends from the floor up to elevation 367'10" and consists of cross-linked PVC foam or polyimide foam panels with an outer sheathing of stainless steel. Various aspects of the containment liner insulation design are described in UFSAR Sections 3.8.1.1.3, 3.8.1.3.1, 3.8.1.4.5, and 3.8.1.6.1.7.

2.4.1.2 Containment Internal Structural Components

The Containment internal structural components consist mainly of the reactor primary shield wall, the secondary shield wall, and support structures for the reactor vessel, steam generators, reactor coolant pumps, and the pressurizer. The shield walls provide radiation shielding to permit access into the Reactor Containment during full power operation for inspection and maintenance. These walls also provide missile protection for the containment liner plate following postulated high energy line breaks. Also included in Containment Internal Structural Components are the equipment and piping supports and restraints. The Polar Crane and refueling cavity/refueling canal are included in this category of structures.

2.4.1.2.1 Concrete

The primary shield wall is a thick cylindrical wall that encloses the reactor vessel and provides biological shielding and structural support. The primary shield wall also acts as part of the missile barrier. The primary shield wall surrounds the reactor vessel and consists of an annular reinforced concrete structure extending from the base of the containment to elevation 275 ft. The lower portion of the shield is a minimum thickness of 6.5 ft and is an integral part of the main structural concrete support for the reactor vessel. Above the reactor vessel flange, it extends upward to the operating floor to form an integral portion of the refueling cavity. Concrete structures are shown on UFSAR Figures 1.2.2-2 through 1.2.2-4.

The secondary shield surrounds the reactor coolant loops and the primary shield. It consists of interior walls in the containment structure, the operating floor, and the reactor containment structure. The main portion of the secondary shield above grade internal to the Containment consists of a concrete crane wall, 3 ft thick, surrounding the reactor coolant loops. Interior walls divide the volume between the primary shield wall and the crane wall into three major compartments each housing one loop of the Reactor Coolant System. Each of these compartments contains a steam generator, reactor coolant pump, and Reactor Coolant System piping. The compartments are separated by the refueling canal, missile shield walls, and the in-core instrumentation room. The primary and secondary shield walls are described in UFSAR Section 12.3.1.

Other major concrete structures include the operating floor, which is supported by the crane wall and upper portion of the primary shield wall; the floor of the refueling canal; the floor supporting the in-core instrumentation seal table; and the control rod drive missile shield. The upper portion of the pressurizer is surrounded by a concrete structure that provides missile protection for the pressurizer and appurtenances and blocks any missiles generated by the pressurizer valves. Radiation shielding walls are constructed around the steam generators above elevation 275'.

The refueling canal is a stainless steel lined, reinforced concrete structure that forms a pool above the reactor when it is filled with borated water for refueling. It is irregularly shaped, formed by the upper portions of the primary shield concrete and other sidewalls. It contains space for storing the upper and lower reactor internals and refueling tools.

Barriers surround all high-pressure equipment, e.g., high-energy reactor coolant system piping and components, which could generate missiles as a result of a design basis accident. These barriers, principally the primary and secondary shield walls, prevent such missiles from damaging the containment liner, piping penetrations, and required engineered safeguards systems. The control rod drive missile shield is a removable structure located over the reactor vessel control rod drives during normal operation to provide missile protection for any missiles that could be generated from the control rod drives mechanisms. Curbs have been provided in Reactor Coolant Pump B and C pump bays to prevent an oil leak from spreading a fire.

Removable concrete hatch plugs are provided on the operating floor for access to "B" and "C" RCPs and the Seal Table Room. A removable concrete cover is provided over the pressurizer. Removable precast concrete pie blocks are provided on the operating floor over the Reactor Head Storage area and "A" Reactor Coolant pump for access and missile protection. Reinforced concrete masonry walls in accordance with IE Bulletin 80-11 are located in the Reactor Containment Building at three areas as shown on UFSAR Figure 3.8.4-2 and as described in UFSAR Section 3.8.4.1.5.

Concrete walls, floors, beams, equipment supports, and other miscellaneous concrete components are of reinforced concrete design.

2.4.1.2.2 Steel

Structural and miscellaneous steel platforms (grating and checkered plate), stairways, and ladders are provided inside Containment to allow access to the various elevations and areas for inspection and maintenance. Structural and miscellaneous steel also provides support for safety related and non-safety related systems and components, including: piping, ducts, miscellaneous equipment, electrical cable tray and conduit, instruments and tubing, electrical and instrumentation enclosures and racks, steel beams and columns. Steel commodities include connections and attachments to the concrete walls and liner, Reactor Sump blowout panels, the structural supports for the seal table, structural supports for the incore flux mapping drive equipment, and the Reactor Missile Shield. Containment internal structural steel is discussed in UFSAR Section 3.8.3.

Similar to the containment cylinder and dome liner plates discussed previously, the external surfaces of structural steel are coated. The application of protective coatings on steel surfaces ensures that the external metal surface is not in contact with a moist environment for extended periods of time.

Stainless steel liner plate is installed in the reactor cavity and the refueling canal.

REACTOR COOLANT SYSTEM SUPPORTS

Reactor Coolant System Supports include the support structures for the Major Class I Component such as the Reactor Vessel, Steam Generators, Reactor Coolant Pumps, Pressurizer, and Reactor Coolant System Piping Supports and Restraints. Major component supports are described in UFSAR Section 3.8.3.1.

Reactor Vessel Supports

The reactor vessel has three supports located at alternate nozzles. Each support bears on a support shoe, which is fastened to the support structure. The support shoe is a structural member that transmits the support loads to the supporting structure. Each support is designed to restrain vertical, lateral, and rotational movement of the reactor vessel, but allows for thermal growth by permitting radial sliding on bearing plates.

Reactor Coolant Pump Supports

Each reactor coolant pump is supported on a three-legged structural system consisting of three connected columns fabricated of carbon steel members, structural sections, and pipe. Provisions for limited movement of the structure in any horizontal direction to accommodate piping expansion is accomplished with a sliding "Lubrite" base plate arrangement and a system of tie rods and anchor bolts which restrain the structure from movement beyond the calculated limits. Sliding shoes at the top of the support structures permit radial thermal growth of the pumps during heat-up.

Steam Generator Supports

The steam generators are supported on a structural system consisting of four connected columns all welded together, fabricated of carbon steel members, with provisions for limited movement of the structure in a horizontal direction to accommodate piping expansion with a system of "Lubrite" plates, hydraulic snubbers, guides, and stops. The "Lubrite" plates, hydraulic snubbers, guides, and stops are designed as damped supports to resist the action of seismic and pipe break loads.

Sliding shoes at the top of the support structure permit radial thermal growth of the steam generators during heatup.

Steam generator lateral bracing is provided near the upper tube sheet elevation to resist lateral loads, including those resulting from seismic forces and pipe rupture forces. Additional bracing is provided at a lower elevation to resist pipe rupture loads.

Pressurizer Supports

The pressurizer is supported on a heavy concrete slab spread between the concrete shield walls of its compartment. The pressurizer is a bottom skirt support vessel, resting on a type of ring girder.

The supports for the reactor coolant system have been designed to withstand the blowdown forces associated with the sudden severance of the reactor coolant piping so that the coincidental rupture of the steam system is not considered credible. Major component supports are described in UFSAR Section 5.4.

POLAR CRANE

The reactor building polar crane is a cantilevered end gantry crane that operates on a circular track supported by the crane wall. The crane and associated rails are seismically qualified Class I structures. The polar crane has a main and an auxiliary hoist and provides a means of lifting and handling heavy loads inside Containment. The polar crane is shown on UFSAR Figure 1.2.2-4.

ECCS SUMP SCREENS

The ECCS sump is located outside the crane wall in the northeast quadrant of the Containment. Two suction lines lead from the sump, one line to each of the Residual Heat Removal pumps.

Filtration of water entering the ECCS sump begins with the emergency core cooling system coarse filter located inside the crane wall. The spilled mixture of reactor coolant and borated water from the Safety Injection System is filtered through the coarse screens as it flows through openings in the crane wall located in the reactor coolant pump bays. After entering the annular area between the crane wall and the Containment wall, the mixture level increases until it flows into the sump. Submerged and floating debris are prevented from entering the sump by baffles. Neutral and near-neutral buoyant material is stopped by the sump screens. A hood is located over the sump area to prevent debris from falling directly into the sump. The ECCS sump screen arrangement is described in UFSAR Section 6.3.2 and shown on Figure 6.3.2-3.

2.4.1.3 Containment External Structural Components

The Containment external structural components consist of the reinforced concrete structures that surround and provide protection for the equipment and personnel hatches. The structure associated with the equipment hatch also provides protection for the containment purge inlet valves that penetrate the containment wall.

The Containment external structural components are shown on UFSAR Figures 1.2.2-2 and 1.2.2-4.

2.4.1.3.1 Concrete

EQUIPMENT HATCH AREA STRUCTURE

The Equipment Hatch Area Structure consists of a reinforced concrete slab on grade and reinforced concrete walls that enclose the area around the equipment hatch and Containment purge inlet valve. The area over the purge inlet valve area is enclosed with a roof slab and has an exterior door. The wall directly in front of the equipment hatch is removable.

PERSONNEL LOCK SHIELD STRUCTURE

The Personnel Lock Shield Structure consists of a reinforced concrete slab on grade, reinforced concrete walls, and roof slab. The structure encloses the area around the personnel hatch. The wall directly in front of the personnel hatch (airlock) has an opening normally filled with radiation shield blocks (lead bricks) that are removed for direct access to the airlock during plant refueling outages. The Personnel Lock Shield Structure is located in the enclosed area between the Reactor Containment Building, the Reactor Auxiliary Building, and the Turbine Building.

2.4.1.3.2 Steel

EQUIPMENT HATCH AREA

The structure is separated from the Reactor Containment Building by a 2-inch space filled with expanded bead stock polystyrene and a structural angle used for flashing. There is a door for access and a ventilation damper in the exterior wall of the structure surrounding the inlet purge valve.

PERSONNEL LOCK SHIELD STRUCTURE

A stairway provides access from elevation 226 ft to 233 feet. The structure is separated from the Reactor Containment Building by a 2-inch space filled with expanded bead stock polystyrene and a structural angle used for flashing. A steel plate, mounted by expansion anchors, is installed over the lead brick radiation shielding after the lead bricks are installed. A security door is provided at elevation 233 ft.

2.4.1.4 Conclusion

The Containment is in scope of license renewal because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions

3. SCs that are relied on during postulated fires, anticipated transients without scram, and station blackout events.

The following table identifies the Containment Structure components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.4-1 COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: CONTAINMENT STRUCTURE

Component/Commodity	Intended Function	AMR Results
Anchorage/Embedments Exposed Surfaces	Provide pressure boundary and/or fission product barrier. Provide structural and/or functional support to safety-related equipment. Provide rated fire barrier to confine or retard a fired from spreading to or from adjacent areas of the plant. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-1, Item 27 Table 3.5-1, Item 28 Table 3.5-2, Item 2

TABLE 2.4-1 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: CONTAINMENT STRUCTURE

Component/Commodity	Intended Function	AMR Results
Anchorage/Embedments (Embedded/Encased in Concrete)	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 9
Bellows	Provide pressure boundary and/or fission product barrier.	Table 3.5-1, Item 2 Table 3.5-1, Item 3 Table 3.5-1, Item 19 Table 3.5-2, Item 11
Cable Tray and Conduit	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 2
Cavity Seal Ring Plate	Provide pressure boundary and/or fission product barrier.	Table 3.5-1, Item 19 Table 3.5-2, Item 12
Concrete Sump	Provide structural and/or functional support to safety-related equipment. Provide spray shield or curbs for directing flow (such as safety injection flow to containment sump).	Table 3.5-1, Item 7 Table 3.5-1, Item 15 Table 3.5-2, Item 10
Containment Liner Insulation and Penetration Insulation	Provide structural and/or functional support to safety-related equipment.	Table 3.5-2, Item 15
Containment Liner Plate (includes Liner Attachments and Liner Anchors).	Provide pressure boundary and/or fission product barrier. Provide structural and/or functional support to safety-related equipment. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 12 Table 3.5-2, Item 9 Table 3.5-2, Item 12

TABLE 2.4-1 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: CONTAINMENT STRUCTURE

Component/Commodity	Intended Function	AMR Results
Electrical & Instrument Panels and Enclosures	Provide structural and/or functional support to safety-related equipment. Provide shelter/protection to safety-related equipment (including radiation shielding). Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-2, Item 2 Table 3.5-2, Item 11 Table 3.5-2, Item 12
Electrical Component Supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-1, Item 27 Table 3.5-2, Item 2 Table 3.5-2, Item 11 Table 3.5-2, Item 12
Electrical Penetrations.	Provide pressure boundary and/or fission product barrier. Provide structural and/or functional support to safety-related equipment. Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 3 Table 3.5-2, Item 9
Equipment Hatch	Provide pressure boundary and/or fission product barrier. Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.	Table 3.5-1, Item 4 Table 3.5-1, Item 5 Table 3.5-2, Item 9

TABLE 2.4-1 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: CONTAINMENT STRUCTURE

Component/Commodity	Intended Function	AMR Results
Equipment Supports.	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-1, Item 27
Expansion Anchors	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-1, Item 27 Table 3.5-1, Item 28 Table 3.5-2, Item 11 Table 3.5-2, Item 12
External Reinforced Concrete Components (Missile Shield Slabs, Walls, Roof Slabs)	Provide structural and/or functional support to safety-related equipment. Provide shelter/protection to safety-related equipment (including radiation shielding). Serves as missile (internal or external) barrier.	Table 3.5-1, Item 16 Table 3.5-1, Item 17 Table 3.5-1, Item 21 Table 3.5-1, Item 22 Table 3.5-1, Item 23 Table 3.5-2, Item 10
Fire Hose Station	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.3-1, Item 13 Table 3.3-1, Item 20 Table 3.5-1, Item 16
Floor Drains	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-2, Item 9
Fuel Transfer Tube	Provide pressure boundary and/or fission product barrier.	Table 3.5-1, Item 3 Table 3.5-1, Item 19 Table 3.5-2, Item 11 Table 3.5-2, Item 12
Fuel Transfer Tube Blind Flange	Provide pressure boundary and/or fission product barrier.	Table 3.5-2, Item 12
Grouted Tendons	Provide structural and/or functional support to safety-related equipment.	Table 3.5-1, Item 14 Table 3.5-2, Item 9

TABLE 2.4-1 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: CONTAINMENT STRUCTURE

Component/Commodity	Intended Function	AMR Results
HVAC Duct Supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-1, Item 27
Instrument Line Supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-1, Item 27 Table 3.5-2, Item 11
Instrument Racks and Frames	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-1, Item 27

TABLE 2.4-1 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: CONTAINMENT STRUCTURE

Component/Commodity	Intended Function	AMR Results
Internal Reinforced Concrete Components (Beams, Walls, Floors, Columns, Radiation Shielding, Refueling Cavity, Equipment Pads, Missile Shields, Curbs, Hatches, Grout)	Provide structural and/or functional support to safety-related equipment. Provide shelter/protection to safety-related equipment (including radiation shielding). Serves as missile (internal or external) barrier. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO. Provide heat sink during SBO or design basis accidents. Provide spray shield or curbs for directing flow (such as safety injection flow to containment sump).	Table 3.5-1, Item 16 Table 3.5-1, Item 23 Table 3.5-2, Item 10
Masonry Walls	Provide shelter/protection to safety-related equipment (including radiation shielding).	Table 3.5-1, Item 20
Mechanical Penetrations	Provide pressure boundary and/or fission product barrier. Provide structural and/or functional support to safety-related equipment. Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 2 Table 3.5-1, Item 3 Table 3.5-2, Item 9 Table 3.5-2, Item 11 Table 3.5-2, Item 12
Miscellaneous Steel (Stairs & Ladders, Platforms & Connectors, Grating & Checker Plate)	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-2, Item 2

TABLE 2.4-1 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: CONTAINMENT STRUCTURE

Component/Commodity	Intended Function	AMR Results
Moisture Barrier	Provide shelter/protection to safety-related equipment (including radiation shielding).	Table 3.5-1, Item 6
NIS Detector Cover	Provide pressure boundary and/or fission product barrier. Provide structural and/or functional support to safety-related equipment.	Table 3.5-1, Item 19 Table 3.5-2, Item 12
Personnel Airlock	Provide pressure boundary and/or fission product barrier. Provide structural and/or functional support to safety-related equipment. Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO. Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.5-1, Item 4 Table 3.5-1, Item 5 Table 3.5-2, Item 9
Pilings	Provide structural and/or functional support to safety-related equipment.	Table 3.5-2, Item 6
Pipe Supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-1, Item 27 Table 3.5-1, Item 28
Pipe Whip Restraints	Provide pipe whip restraint and/or jet impingement protection.	Table 3.5-1, Item 25 Table 3.5-1, Item 27
Polar Crane	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions.	Table 3.3-1, Item 15

TABLE 2.4-1 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: CONTAINMENT STRUCTURE

Component/Commodity	Intended Function	AMR Results
Pressurizer and Pressurizer Surge Line Supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 27 Table 3.5-1, Item 28
Protective Enclosure (Structures sheltering or enclosing plant equipment)	Provide shelter/protection to safety-related equipment (including radiation shielding).	Table 3.5-1, Item 16 Table 3.5-2, Item 11 Table 3.5-2, Item 12
Reactor Cavity (Refueling Canal) Liner Plate	Provide pressure boundary and/or fission product barrier. Provide structural and/or functional support to safety-related equipment. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 19 Table 3.5-2, Item 9 Table 3.5-2, Item 12
Reactor Coolant Pump Supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 27 Table 3.5-1, Item 28 Table 3.5-2, Item 13 Table 3.5-2, Item 14
Reactor Manway Covers	Provide pressure boundary and/or fission product barrier. Provide shelter/protection to safety-related equipment (including radiation shielding).	Table 3.5-1, Item 19 Table 3.5-2, Item 12
Reactor Vessel Missile Shield Frame	Serves as missile (internal or external) barrier. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions.	Table 3.5-1, Item 16

TABLE 2.4-1 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: CONTAINMENT STRUCTURE

Component/Commodity	Intended Function	AMR Results
Reactor Vessel Support	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-1, Item 27 Table 3.5-1, Item 28 Table 3.5-2, Item 13 Table 3.5-2, Item 14
Reinforced Concrete (Cylinder Wall, Dome, Basemat)	Provide structural and/or functional support to safety-related equipment. Provide shelter/protection to safety-related equipment (including radiation shielding). Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant. Serves as missile (internal or external) barrier. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO. Provide heat sink during SBO or design basis accidents.	Table 3.5-1, Item 7 Table 3.5-1, Item 8 Table 3.5-1, Item 9 Table 3.5-1, Item 10 Table 3.5-1, Item 15 Table 3.5-2, Item 4 Table 3.5-2, Item 10
Seals & Gaskets	Provide pressure boundary and/or fission product barrier.	Table 3.5-1, Item 6
Siding	Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions.	Table 3.5-2, Item 2
Slide Bearing Plates	Provide structural and/or functional support to safety-related equipment.	Table 3.5-1, Item 25 Table 3.5-1, Item 27 Table 3.5-1, Item 28 Table 3.5-2, Item 13 Table 3.5-2, Item 14

TABLE 2.4-1 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: CONTAINMENT STRUCTURE

Component/Commodity	Intended Function	AMR Results
Steam Generator Supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 27 Table 3.5-1, Item 28 Table 3.5-2, Item 13 Table 3.5-2, Item 14
Structural Steel (Beams, Plates, Connectors, Column)	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16
Sump Screens (supports)	Provide structural and/or functional support to safety-related equipment.	Table 3.5-1, Item 16 Table 3.5-2, Item 11 Table 3.5-2, Item 12
Threaded Fasteners	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-1, Item 27 Table 3.5-1, Item 28 Table 3.5-2, Item 11 Table 3.5-2, Item 12 Table 3.5-2, Item 13
Tube Track Supports	Provide structural and/or functional support to safety-related equipment. Provide shelter/protection to safety-related equipment (including radiation shielding). Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-1, Item 27 Table 3.5-2, Item 11 Table 3.5-2, Item 12

TABLE 2.4-1 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: CONTAINMENT STRUCTURE

Component/Commodity	Intended Function	AMR Results
Vibration Isolators	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions.	Table 3.5-1, Item 25 Table 3.5-1, Item 27

2.4.2 OTHER STRUCTURES

The following structures are included in this Subsection:

1. Reactor Auxiliary Building
2. Fuel Handling Building
3. Turbine Building
4. Dedicated Shutdown Diesel Generator Building
5. Radwaste Building
6. Intake Structure
7. North Service Water Header Enclosure
8. Emergency Operations Facility/Technical Support Center (EOF/TSC) Security Diesel Generator Building
9. Discharge Structures
10. Lake Robinson Dam
11. Pipe Restraint Tower
12. Yard Structures and Foundations
13. Refueling System

2.4.2.1 Reactor Auxiliary Building

The Reactor Auxiliary Building is a reinforced concrete, seismic Class I structure that houses safety related systems. It includes the Control Room, the Emergency Diesel Generator Rooms, the Residual Heat Removal Pump Pit, Boron Injection Tank Room, North and South Cable Vaults, piping penetration area, and the B Waste Evaporator enclosure installed on the roof of the building. A sump tank room and RHR Pit are located below grade.

The Reactor Auxiliary Building reinforced concrete foundation slab is at elevation 226’-0” and is supported on pilings (steel pipe, cast-in-place concrete pilings). The Auxiliary Building is constructed with reinforced concrete bearing walls and floor slabs. Water stops were used in the construction joints of the Reactor Auxiliary Building foundation slab. Also, waterproofing membrane was installed on the building sump and RHR pit exterior surfaces to inhibit the intrusion of ground water. The water stops and waterproofing are considered to be subcomponents of the concrete slabs and walls.

The Auxiliary Building is described in UFSAR Section 3.8.4.1 and is shown on UFSAR Figures 1.2.2-5 and 1.2.2-6. The RHR Pump Room Pit is discussed in UFSAR Section 6.3.2.2.5.1 and is shown on UFSAR Figures 1.2.2- 2 and 1.2.2-4. The electrical penetration areas are also shown on Figure 1.2.2-2.

The Control Room and Hagan Relay Room are located on elevation 254' in the south end of the RAB, above the Unit 2 Cable room and Safeguards Relay Room.

In the license renewal evaluation, common walls (and associated penetrations) between the Reactor Auxiliary Building and adjacent buildings were included in the scope of the Reactor Auxiliary Building, with the exception of the Containment walls. Also included in the scope of the Reactor Auxiliary Building are stairs and equipment supports located on the exterior walls of the building, and the area between the Containment, Fuel Handling Building, and Reactor Auxiliary Building in the vicinity of the RHR Pit. Floor drains in the Reactor Auxiliary Building are credited for minimizing flood levels following fire protection system pipe breaks or actuations. The floor drains are in scope for license renewal. The Motor Control Center (MCC) 5 water spray shield is in scope for license renewal, because it protects the MCC from water spray following a postulated pipe break.

The Auxiliary Building is in the scope of license renewal because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions

3. SCs that are relied on during postulated fires, anticipated transients without scram, and station blackout events.

The following table identifies the Reactor Auxiliary Building components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.4-2 COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: REACTOR AUXILIARY BUILDING

Component/Commodity	Intended Function	AMR Results
Anchorage/Embedments Exposed Surfaces	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-1, Item 27 Table 3.5-2, Item 1
Anchorage/Embedments (Embedded/Encased in Concrete)	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 9
Battery Rack	Provide structural and/or functional support to safety-related equipment. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Cable Tray and Conduit	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-2, Item 1 Table 3.5-2, Item 2 Table 3.5-2, Item 3
Concrete Sump	Provide structural and/or functional support to safety-related equipment. Provide spray shield or curbs for directing flow (such as safety injection flow to containment sump)	Table 3.5-1, Item 17 Table 3.5-2, Item 10 Table 3.5-1, Item 21 Table 3.5-1, Item 22 Table 3.5-1, Item 23

TABLE 2.4-2 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: REACTOR AUXILIARY BUILDING

Component/Commodity	Intended Function	AMR Results
Control Room Ceiling	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 16
Curb (Concrete)	Provide a protective barrier for internal/external flood event. Provide spray shield or curbs for directing flow (such as safety injection flow to containment sump).	Table 3.5-2, Item 10
Damper Mounting	Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-2, Item 2
Doors (includes Fire Doors)	Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant. Provide pressure boundary and/or fission product barrier. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.3-1, Item 19 Table 3.5-2, Item 1
Electrical & Instrument Panels and Enclosures	Provide shelter/protection to safety-related equipment (including radiation shielding). Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-2, Item 1 Table 3.5-2, Item 2 Table 3.5-2, Item 3 Table 3.5-2, Item 12

TABLE 2.4-2 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: REACTOR AUXILIARY BUILDING

Component/Commodity	Intended Function	AMR Results
Electrical Bus Duct (Enclosure)	Provide shelter/protection to safety-related equipment (including radiation shielding). Provide structural and/or functional support to safety-related equipment.	Table 3.5-1, Item 25
Electrical component supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-1, Item 27 Table 3.5-2, Item 1 Table 3.5-2, Item 3
Equipment Supports.	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-1, Item 27 Table 3.5-2, Item 1
Expansion Anchors	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-1, Item 27 Table 3.5-2, Item 1
Fire Barrier Assemblies (Composite structures that provide protection for equipment or separation of fire zones, composed of pyrocrete, ceramic fiber, masonry, etc.)	Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.	Table 3.3-1, Item 19

TABLE 2.4-2 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: REACTOR AUXILIARY BUILDING

Component/Commodity	Intended Function	AMR Results
Fire Barrier Penetration Seals	Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant. Provide structural and/or functional support to safety-related equipment. Provide pressure boundary and/or fission product barrier. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.3-1, Item 19
Fire Hose Station	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.3-1, Item 20 Table 3.5-1, Item 16
Fire Plugs/Fire Hatches	Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant. Provide structural and/or functional support to safety-related equipment. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 10
Floor Drains	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-2, Item 9
HVAC Duct Supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-2, Item 1 Table 3.5-2, Item 2
Instrument Line Supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-1, Item 27 Table 3.5-2, Item 1 Table 3.5-2, Item 2 Table 3.5-2, Item 12

TABLE 2.4-2 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: REACTOR AUXILIARY BUILDING

Component/Commodity	Intended Function	AMR Results
Instrument Racks and Frames	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-2, Item 1
Louvers	Provide structural and/or functional support to safety-related equipment.	Table 3.5-1, Item 16 Table 3.5-2, Item 5
Masonry Walls	Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant. Provide shelter/protection to safety-related equipment (including radiation shielding). Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 20
Miscellaneous Steel (Stairs & Ladders, Platforms & Connectors, Grating & Checker Plate)	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-2, Item 2
Pilings	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 6

TABLE 2.4-2 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: REACTOR AUXILIARY BUILDING

Component/Commodity	Intended Function	AMR Results
Pipe Supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-1, Item 27 Table 3.5-1, Item 28 Table 3.5-2, Item 1
Pipe Whip Restraints	Provide structural and/or functional support to safety-related equipment. Provide pipe whip restraint and/or jet impingement protection.	Table 3.5-1, Item 25
Protective Enclosure (Structures sheltering or enclosing plant equipment)	Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16
Raised Floor	Provide structural and/or functional support to safety-related equipment. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 16
Reinforced Concrete (Beams, Walls, Floors, Columns, etc.)	Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant Provide shelter/protection to safety-related equipment (including radiation shielding). Provide structural and/or functional support to safety-related equipment. Provide pressure boundary and/or fission product barrier. Serves as missile (internal or external) barrier Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-1, Item 17 Table 3.5-1, Item 21 Table 3.5-1, Item 22 Table 3.5-1, Item 23 Table 3.5-2, Item 10

TABLE 2.4-2 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: REACTOR AUXILIARY BUILDING

Component/Commodity	Intended Function	AMR Results
Roof (Membrane or Built Up)	Provide shelter/protection to safety-related equipment (including radiation shielding). Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 8
Seismic Joint Filler	Provide shelter/protection to safety-related equipment (including radiation shielding).	Table 3.5-2, Item 7
Siding	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-2, Item 2 Table 3.5-2, Item 5
Slide Bearing Plates	Provide structural and/or functional support to safety-related equipment.	Table 3.5-2, Item 13
Spray Shields	Provide spray shield or curbs for directing flow (such as safety injection flow to containment sump).	Table 3.5-1, Item 16
Structural Steel (Beams, Plates, Connectors, Column)	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16
Threaded Fasteners	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-2, Item 1
Tube Track Supports	Provide shelter/protection to safety-related equipment (including radiation shielding) Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-2, Item 1 Table 3.5-2, Item 2

TABLE 2.4-2 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: REACTOR AUXILIARY BUILDING

Component/Commodity	Intended Function	AMR Results
Vibration Isolators	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions.	Table 3.5-1, Item 25

2.4.2.2 Fuel Handling Building

The Fuel Handling Building is comprised of several adjacent structures and a superstructure that supports the Spent Fuel Cask Handling Crane. The Fuel Handling Building is further subdivided into structures, rooms, and functional areas as discussed in the following paragraphs.

The Fuel Handling Building includes the Spent Fuel Pit (including the Spent Fuel Pit structure, liner, spent fuel racks, and spent fuel cask storage area), the Gas Decay Tank room, Transfer Canal Structure, New Fuel Storage Room, Spent Fuel Pit Cooling Pump and Heat Exchanger Rooms, CVCS Holdup Tank room, Hot Machine Shop, Cask & Large Equipment Decontamination Area, Tool Room, and HVAC Fan Rooms. The Fuel Handling Building is supported on pilings with a higher density of pilings under the Spent Fuel Pit structure which consists of the gas decay tank room under the Spent Fuel Pit, the Spent Fuel Pit, and the superstructure above the Spent Fuel Pit. Water stops were used in the construction of the Fuel Handling Building sump pits. Water stops are considered to be subcomponents of the concrete sump pit slabs and walls.

The Spent Fuel Pit is designed for the underwater storage of spent fuel assemblies after their removal from the reactor. It is constructed of reinforced concrete. The entire interior basin face and transfer canal is lined with stainless steel plate. A spent fuel pool bridge crane is mounted on rails adjacent to the Spent Fuel Pit and is used to move components within the pit. The superstructure above the Spent Fuel Pit is constructed of structural steel with aluminum or fiberglass siding. The superstructure supports a 125-ton Spent Fuel Cask Handling Crane that is used to move the spent fuel cask and miscellaneous equipment between ground level and the spent fuel pit.

The New Fuel Building is a separate area whose location facilitates the unloading of new fuel assemblies from fuel containers. The New Fuel Storage Room includes new fuel racks, hoist/handling tool, and new fuel lift.

In the license renewal evaluation, the Hot Machine Shop, Tool Room, Cask & Large Equipment Decontamination Area, Spent Fuel Pit heat exchanger room and the pipe corridor beneath the Spent Fuel Pit pump room were determined to be in scope for license renewal. The Spent Fuel Pit, spent fuel racks, and Fuel Transfer Canal were determined to be in scope. (The Fuel Transfer Tube is in scope because it forms part of the pressure boundary for the Containment as discussed in Subsection 2.4.1.1.2.) The entire steel and reinforced concrete structure load path (including pilings) supporting the Spent Fuel Cask Handling Crane are included in scope. The Spent Fuel Cask Handling Crane itself as well as the Spent Fuel Bridge Crane were included in scope. However, the CVCS Holdup Tank Room structure was screened out, because it does not support any Fuel Handling Building structure intended function. Civil components and commodities in the New Fuel Storage Room were evaluated and determined not to support any Fuel Handling Building structure intended function.

The Fuel Handling Building is shown on UFSAR Figures 1.2.2-7 and 1.2.2-8. The Spent Fuel Pit is discussed in UFSAR Section 3.8.4.

The Fuel Handling Building is in the scope of license renewal because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events
2. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions
3. SCs that are relied on during postulated fires and station blackout events.

The following table identifies the Fuel Handling Building components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.4-3 COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: FUEL HANDLING BUILDING

Component/Commodity	Intended Function	AMR Results
Anchorage/Embedments Exposed Surfaces	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-1, Item 27

TABLE 2.4-3 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: FUEL HANDLING BUILDING

Component/Commodity	Intended Function	AMR Results
Anchorage/Embedments (Embedded/Encased in Concrete)	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 9
Bellows	Provide pressure boundary and/or fission product barrier.	Table 3.5-1, Item 19 Table 3.5-2, Item 11 Table 3.5-2, Item 12
Cable Tray and Conduit	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 2
Doors	Provide pressure boundary and/or fission product barrier.	Table 3.5-1, Item 16
Electrical & Instrument Panels and Enclosures	Provide shelter/protection to safety-related equipment (including radiation shielding). Provide structural and/or functional support to safety-related equipment.	Table 3.5-1, Item 16
Electrical Component Supports	Provide structural and/or functional support to safety-related equipment.	Table 3.5-1, Item 25 Table 3.5-1, Item 27
Expansion Anchors	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-1, Item 27

TABLE 2.4-3 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: FUEL HANDLING BUILDING

Component/Commodity	Intended Function	AMR Results
Fire Barrier Penetration Seals	Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant. Provide pressure boundary and/or fission product barrier. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.3-1, Item 19
HVAC Duct Supports	Provide structural and/or functional support to safety-related equipment.	Table 3.5-1, Item 25 Table 3.5-1, Item 27
Instrument Line Supports	Provide structural and/or functional support to safety-related equipment.	Table 3.5-1, Item 25 Table 3.5-1, Item 27
Instrument Racks and Frames	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Masonry Walls	Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 20
Miscellaneous Steel (Stairs & Ladders, Platforms & Connectors, Grating & Checker Plate)	Provide shelter/protection to safety-related equipment (including radiation shielding). Provide pressure boundary and/or fission product barrier.	Table 3.5-1, Item 16

TABLE 2.4-3 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: FUEL HANDLING BUILDING

Component/Commodity	Intended Function	AMR Results
Pilings	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions.	Table 3.5-2, Item 6
Pipe Supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-1, Item 27 Table 3.5-1, Item 28
Reinforced Concrete (Beams, Walls, Floors, Columns, etc.)	Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant. Provide shelter/protection to safety-related equipment (including radiation shielding). Provide structural and/or functional support to safety-related equipment. Serves as missile (internal or external) barrier. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provides heat sink during SBO or design basis accidents. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-1, Item 17 Table 3.5-1, Item 21 Table 3.5-1, Item 22 Table 3.5-1, Item 23 Table 3.5-2, Item 10
Seismic Joint Filler	Provide shelter/protection to safety-related equipment (including radiation shielding).	Table 3.5-2, Item 7
Spent Fuel Pool Liner	Provide structural and/or functional support to safety-related equipment. Provide pressure boundary and/or fission product barrier. Provides heat sink during SBO or design basis accidents.	Table 3.5-1, Item 19 Table 3.5-2, Item 9 Table 3.5-2, Item 11 Table 3.5-2, Item 12

TABLE 2.4-3 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: FUEL HANDLING BUILDING

Component/Commodity	Intended Function	AMR Results
Siding	Provide pressure boundary and/or fission product barrier.	Table 3.5-1, Item 16 Table 3.5-2, Item 5
Spent Fuel Bridge Crane	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions.	Table 3.3-1, Item 15
Spent Fuel Cask Crane	Provide structural and/or functional support to safety-related equipment.	Table 3.3-1, Item 15
Spent Fuel Storage Rack	Provide shelter/protection to safety-related equipment (including radiation shielding). Provide structural and/or functional support to safety-related equipment. Provides heat sink during SBO or design basis accidents.	Table 3.3-2, Item 1
Structural steel (Beams, Plates, Connectors, Column)	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16
Tube Track Supports	Provide shelter/protection to safety-related equipment (including radiation shielding). Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25

2.4.2.3 Turbine Building

The Turbine Building is primarily an open steel frame structure built on reinforced concrete foundations. The foundations are supported on pilings. In general, the Turbine Building is a Class III structure; Class III structures are not related to reactor operation or safety. However, the Turbine Building includes a Seismic Class I bay in the area that houses and supports the steam driven auxiliary feedwater pump and associated components. In addition, safety related piping is routed through a Class III portion of the Turbine Building in a concrete trench covered with a checkered plate on the bottom floor. The reinforced concrete turbine pedestal is the dominant structural feature of the building. The building is essentially rectangular in shape with the turbine centerline in the east/west direction. The building is located just south of the Reactor Containment Building. Spray/jet impingement shielding has been installed to protect the main steam pressure transmitters located in the Class I portion of the Turbine Building from a postulated main feedwater line break.

Enclosed areas in the Turbine Building are the entrance to the radiologically controlled area (RCA), the 4160 V/480 V switchgear room, water treatment enclosure, fire protection equipment room, condenser vacuum pump enclosure, chemical feed equipment room, and feedwater valve enclosures. The enclosed area between the Turbine Building, Reactor Auxiliary Building, and the Reactor Containment Building has been screened with the Turbine Building. This area contains a tool room, restrooms, and a dressout area, and encloses the Containment external structure for personnel access. The Turbine Building also is scoped to include the Turbine Gantry (TG) Crane System. The Turbine Gantry Crane is a 145-ton Whiting gantry crane with a 25-ton auxiliary hoist. The crane structural components are comprised of the crane structure, hoists, trolleys, end trucks, crane rails and end stops. Water stops were used in the construction of the Turbine Building foundations. Water stops are considered to be subcomponents of concrete slabs and walls. The Turbine Building is described in UFSAR Sections 3.2.1.2 and 3.8.4. UFSAR Figures 1.2.2-9 through 1.2.2-12 provide general arrangement drawings of the Turbine Building.

The Turbine Building is in the scope of license renewal because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions

3. SCs that are relied on during postulated fires, anticipated transients without scram, and station blackout events.

The following table identifies the Turbine Building components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.4-4 COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: TURBINE BUILDING

Component/Commodity	Intended Function	AMR Results
Anchorage/Embedments Exposed Surfaces	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-2, Item 1
Anchorage/Embedments (Embedded/Encased in Concrete)	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 9
Battery Rack	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Cable Tray and Conduit	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-2, Item 2 Table 3.5-2, Item 3
Doors	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16

TABLE 2.4-4 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: TURBINE BUILDING

Component/Commodity	Intended Function	AMR Results
Electrical & Instrument Panels and Enclosures	Provide shelter/protection to safety-related equipment (including radiation shielding). Provide structural and/or functional support to safety-related equipment. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-2, Item 2 Table 3.5-2, Item 3 Table 3.5-2, Item 12
Electrical Bus Duct (Enclosure)	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Electrical Component Supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-2, Item 2 Table 3.5-2, Item 3
Equipment Supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Expansion Anchors	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Instrument Line Supports	Provide structural and/or functional support to safety-related equipment. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25

TABLE 2.4-4 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: TURBINE BUILDING

Component/Commodity	Intended Function	AMR Results
Instrument Racks and Frames	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Louvers	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-2, Item 5
Masonry Walls	Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 20
Miscellaneous steel (Stairs & Ladders, platforms & connectors, grating & checker plate)	Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-2, Item 2
Pilings	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 6

TABLE 2.4-4 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: TURBINE BUILDING

Component/Commodity	Intended Function	AMR Results
Pipe Supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-1, Item 28
Pipe Whip Restraints	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Protective Enclosure (Structures sheltering or enclosing plant equipment)	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-2, Item 2 Table 3.5-2, Item 5
Reinforced Concrete (Beams, Walls, Floors, Columns, etc.)	Provide shelter/protection to safety-related equipment (including radiation shielding). Provide structural and/or functional support to safety-related equipment. Serves as missile (internal or external) barrier Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-1, Item 17 Table 3.5-1, Item 21 Table 3.5-1, Item 22 Table 3.5-1, Item 23 Table 3.5-2, Item 10
Siding	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-2, Item 5
Spray Shields	Provide spray shield or curbs for directing flow (such as safety injection flow to containment sump).	Table 3.5-1, Item 16

TABLE 2.4-4 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: TURBINE BUILDING

Component/Commodity	Intended Function	AMR Results
Structural Steel (Beams, Plates, Connectors, Column)	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16
Threaded Fasteners	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Tube Track Supports	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-2, Item 2
Turbine Gantry Crane	Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions.	Table 3.3-1, Item 15

2.4.2.4 Dedicated Shutdown Diesel Generator Building

Based on the fire protection safe shutdown analysis, certain postulated fires may cause multiple failures that could prevent safe plant shutdown; therefore, a dedicated shutdown (DS) system was installed to bring the plant to a safe shutdown condition. The DS Diesel Generator is part of the DS system.

The DS Diesel Generator Building structure is scoped to include: the reinforced concrete slab which supports the DS Diesel skid mounted structural steel enclosure, the DS Diesel battery charger, and the DS Diesel Cooling unit. The structure is located west of the Turbine Building as shown on Figure 2.2-1.

The DS Diesel Building is in the scope of license renewal because it contains SCs that are relied on during postulated fires and station blackout events.

The following table identifies the DS Diesel Generator Building components/ commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.4-5 COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: DEDICATED SHUTDOWN DIESEL GENERATOR BUILDING

Component/Commodity	Intended Function	AMR Results
Anchor Bolt Chair for Tank Foundation	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16
Anchorage/Embedments Exposed Surfaces	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Anchorage/Embedments (Embedded/Encased in Concrete)	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 9
Battery Rack	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Cable Tray and Conduit	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 3
Electrical & Instrument Panels and Enclosures	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16
Electrical Component Supports	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-2, Item 3
Equipment Supports	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Expansion Anchors	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Instrument Racks and Frames	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Louvers	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 5
Pipe Supports	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Protective Enclosure (Structures sheltering or enclosing plant equipment)	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 5

TABLE 2.4-5 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: DEDICATED SHUTDOWN DIESEL GENERATOR BUILDING

Component/Commodity	Intended Function	AMR Results
Reinforced Concrete (Beams, Walls, Floors, Columns, etc.)	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-1, Item 17 Table 3.5-1, Item 21 Table 3.5-1, Item 22 Table 3.5-1, Item 23 Table 3.5-2, Item 10
Structural Steel (Beams, Plates, Connectors, Column)	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16
Threaded Fasteners	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25

2.4.2.5 Radwaste Building

The Radwaste Building is a detached structure located adjacent to the east side of the Auxiliary Building as shown on Figure 2.2-1. The building is used for storage of contaminated materials, such as spent ion exchange resins; filters; anti-C clothing; and contaminated waste materials. An expansion joint assembly is installed at the Reactor Auxiliary Building-to-Radwaste Building pipe chase interface to prevent load transfer between buildings.

The Radwaste Building is a reinforced concrete structure supported on a concrete slab. The south and west walls support the grating providing missile and tornado protection for the North Service Water Header Enclosure (Refer to Subsection 2.4.2.7). The Radwaste Building walls provide protection for the safety related service water pipe.

The Radwaste Building was designed such that it would not fail during an earthquake. This was done to protect the Reactor Auxiliary Building. Components associated with Radwaste Building cranes and hoists and fire doors and fire penetrations were considered in the review of this structure. Except for structural steel mounted to the exterior of walls that are common with the North Service Water Header Enclosure, the structural commodities located on the interior and exterior are evaluated with the Radwaste Building. The Block Enclosure Annex surrounding fire protection piping and valves on the north wall of the Radwaste Building was included in the scope of review for the Radwaste Building. The enclosure is fabricated from concrete block walls and has a built-up metal roof.

The Radwaste Building is in the scope of license renewal because it contains:

1. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions

2. SCs that are relied on during postulated fires.

The following table identifies the Radwaste Building components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.4-6 COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: RADWASTE BUILDING

Component/Commodity	Intended Function	AMR Results
Anchorage/Embedments Exposed Surfaces	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Anchorage/Embedments (Embedded/Encased in Concrete)	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 9
Expansion Anchors	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Masonry Walls	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 20
Pipe Supports	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Reinforced Concrete (Beams, Walls, Floors, Columns, etc.)	Provide shelter/protection to safety-related equipment (including radiation shielding). Serves as missile (internal or external) barrier. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-1, Item 17 Table 3.5-1, Item 21 Table 3.5-1, Item 22 Table 3.5-1, Item 23 Table 3.5-2, Item 10
Seismic Joint Filler	Provide shelter/protection to safety-related equipment (including radiation shielding).	Table 3.5-2, Item 7
Structural steel (Beams, Plates, Connectors, Column)	Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions.	Table 3.5-1, Item 16
Threaded Fasteners	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25

2.4.2.6 Intake Structure

The Intake Structure is a Class I reinforced concrete structure consisting of three bays. The intake structure supports the four safety related Service Water Pumps, the three non-safety related circulating water pumps, and the three firewater pumps (booster pump, motor driven pump, engine driven pump). These pumps take suction from the bays and supply water to the plant via their respective systems. There are three traveling screens, one for each bay, to remove small debris from the intake water. The Intake Structure is in the scope of license renewal because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions

3. SCs that are relied on during postulated fires and station blackout events.

The following table identifies the Intake Structure components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.4-7 COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: INTAKE STRUCTURE

Component/Commodity	Intended Function	AMR Results
Anchorage/Embedments Exposed Surfaces	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Anchorage/Embedments (Embedded/Encased in Concrete)	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 9
Battery Rack	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25

TABLE 2.4-7 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: INTAKE STRUCTURE

Component/Commodity	Intended Function	AMR Results
Cable Tray and Conduit	Provide structural and/or functional support to safety-related equipment. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 3
Concrete Fill	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 17
Electrical & Instrument Panels and Enclosures	Provide shelter/protection to safety-related equipment (including radiation shielding). Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-2, Item 3 Table 3.5-2, Item 5
Electrical Component Supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-2, Item 3
Expansion Anchors	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-2, Item 5

TABLE 2.4-7 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: INTAKE STRUCTURE

Component/Commodity	Intended Function	AMR Results
Instrument Racks and Frames	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-2, Item 3
Manhole Covers	Provide shelter/protection to safety-related equipment (including radiation shielding). (Radiation shielding is not applicable to manhole covers.)	Table 3.5-1, Item 16
Miscellaneous Steel (Stairs & Ladders, Platforms & Connectors, Grating & Checker Plate)	Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 3
Pipe Supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Protective Enclosure (Structures sheltering or enclosing plant equipment)	Provide shelter/protection to safety-related equipment (including radiation shielding). Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-2, Item 5

TABLE 2.4-7 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: INTAKE STRUCTURE

Component/Commodity	Intended Function	AMR Results
Reinforced Concrete (Beams, Walls, Floors, Columns, etc.)	Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant. Provide shelter/protection to safety-related equipment (including radiation shielding). Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide source of cooling water for plant shutdown. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-1, Item 17 Table 3.5-1, Item 21 Table 3.5-1, Item 22 Table 3.5-1, Item 23 Table 3.5-2, Item 10
Siding	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-2, Item 5
Structural Steel (Beams, Plates, Connectors, Column)	Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16

2.4.2.7 North Service Water Header Enclosure

The North Service Water Header Enclosure provides support and protection for a portion of the North Service Water header that is routed above ground. The North Service Water header has been designed with protective barriers to ensure that this portion of the Service Water System is capable of withstanding the passage of a tornado without a loss of function. The protective barriers provided for the above-ground portion of the North Service Water header are a double layer of grating and a poured concrete wall in the area to the south and west of the Radwaste Building as shown on Figure 2.2-1. The Radwaste Building south and west walls also provide missile protection. The concrete structure is designed as Class I. Service Water Pit 3, south of the Radwaste Building, is surrounded by and included in the scope of the North Service Water Header Enclosure.

The North Service Water Header Enclosure is in the scope of license renewal because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions

3. SCs that are relied on during postulated fires and station blackout events.

The following table identifies the North Service Water Header Enclosure components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.4-8 COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: NORTH SERVICE WATER HEADER ENCLOSURE

Component/Commodity	Intended Function	AMR Results
Anchorage/Embedments Exposed Surfaces	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Anchorage/Embedments (Embedded/Encased in Concrete)	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 9
Cable Tray and Conduit	Provide structural and/or functional support to safety-related equipment. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 3

TABLE 2.4-8 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: NORTH SERVICE WATER HEADER ENCLOSURE

Component/Commodity	Intended Function	AMR Results
Concrete Fill	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions.	Table 3.5-1, Item 17
Curb (Concrete)	Provide spray shield or curbs for directing flow (such as safety injection flow to containment sump).	Table 3.5-2, Item 10
Electrical & Instrument Panels and Enclosures	Provide shelter/protection to safety-related equipment (including radiation shielding). Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-2, Item 3 Table 3.5-2, Item 5
Electrical Component Supports	Provide structural and/or functional support to safety-related equipment. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-2, Item 3
Expansion Anchors	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Instrument Line Supports	Provide structural and/or functional support to safety-related equipment. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-2, Item 3
Masonry Walls	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions.	Table 3.5-1, Item 20

TABLE 2.4-8 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: NORTH SERVICE WATER HEADER ENCLOSURE

Component/Commodity	Intended Function	AMR Results
Miscellaneous steel (Stairs & Ladders, Platforms & Connectors, Grating & Checker Plate)	Serves as missile (internal or external) barrier.	Table 3.5-2, Item 3
Pipe Supports	Provide structural and/or functional support to safety-related equipment. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Reinforced Concrete (Beams, Walls, Floors, Columns, etc.)	Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant. Provide shelter/protection to safety-related equipment (including radiation shielding). Provide structural and/or functional support to safety-related equipment. Serves as missile (internal or external) barrier. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-1, Item 21 Table 3.5-1, Item 22 Table 3.5-1, Item 23 Table 3.5-2, Item 10
Structural Steel (Beams, Plates, Connectors, Column)	Serves as missile (internal or external) barrier. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-2, Item 3
Threaded Fasteners	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25

2.4.2.8 EOF/TSC Security Diesel Generator Building

The Emergency Operations Facility/Technical Support Center (EOF/TSC) Security Diesel Generator Building houses equipment that is relied on to provide electrical power following postulated fires.

This structure consists of a reinforced concrete slab with walls constructed of concrete block and removable (from inside the structure) steel grating panels. The building is located west of the main power block near the Work Control Building. The building is shown on Figure 2.2-1.

The EOF/TSC Security Diesel Generator Building is in the scope of license renewal because it contains SCs that are relied on during postulated fires.

The following table identifies the EOF/TSC Security Diesel Generator Building components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.4-9 COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: EOF/TSC SECURITY EMERGENCY DIESEL GENERATOR BUILDING

Component/Commodity	Intended Function	AMR Results
Anchorage/Embedments Exposed Surfaces	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Anchorage/Embedments (Embedded/Encased in Concrete)	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 9
Battery Rack	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Cable Tray and Conduit	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 2
Doors	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16
Electrical & Instrument Panels and Enclosures	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16
Electrical Component Supports	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-2, Item 2

TABLE 2.4-9 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: EOF/TSC SECURITY EMERGENCY DIESEL GENERATOR BUILDING

Component/Commodity	Intended Function	AMR Results
Expansion Anchors	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Masonry Walls	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 20
Miscellaneous steel (Stairs & Ladders, Platforms & Connectors, Grating & Checker Plate)	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 2
Pipe Supports	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Protective Enclosure (Structures sheltering or enclosing plant equipment)	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16
Reinforced Concrete (Beams, Walls, Floors, Columns, etc.)	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-1, Item 17 Table 3.5-1, Item 21 Table 3.5-1, Item 22 Table 3.5-1, Item 23 Table 3.5-2, Item 10
Structural steel (Beams, Plates, Connectors, Column)	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16
Threaded Fasteners	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Vibration Isolators	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25

2.4.2.9 Discharge Structures

The structures associated with discharge of Circulating Water and Service Water to Lake Robinson are: Seal Well #2, the Discharge Canal, and the Canal Outlet Structure. Seal Well #2 is an underground/underwater reinforced concrete structure which receives water from the underground circulating water discharge conduit and injects the water into the discharge canal.

The Discharge Canal is an earthen structure that directs condenser cooling and service system water discharged from the plant to Lake Robinson via a channel. The Discharge Canal originates just east of the plant, parallels the west shore of the lake and terminates in the Lake near its upper end.

The Canal Outlet Structure is a reinforced concrete structure located at the intersection of the Discharge Canal and Lake Robinson. It contains a weir over which water is discharged, thereby promoting mixing with water in the lake.

In the scoping process, the Discharge Structures were conservatively assumed to contain SCs that are non-safety related whose failure could prevent satisfactory accomplishment of safety related functions. However, during screening, it was concluded that none of the structure components of the Discharge Structures could prevent the performance of any required safety related function. Therefore, the Discharge Structure components perform no intended functions and are not subject to an aging management review.

2.4.2.10 Lake Robinson Dam

Lake Robinson was constructed originally as a cooling water source for the Robinson Unit 1 fossil station. The lake was created by construction of the Lake Robinson Dam.

Lake Robinson Dam has a central vertical clay core and supporting shells of compacted sand. The dam has a maximum height of about 50 ft. Riprap protection is provided on the upstream face from the crest to Elevation 205 (5 ft below low water elevation) and on the downstream side for that portion of the slope below Elevation 195 ft. The dam includes a reinforced concrete spillway. Two large steel gates and steel valves are used to control water release from the reservoir.

Lake Robinson reservoir provides plant cooling water for normal and emergency situations and supplies fire protection water.

Lake Robinson Dam is in the scope of license renewal because it contains:

1. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions

2. SCs that are relied on during postulated fires.

The following table identifies the Lake Robinson Dam components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.4-10 COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: LAKE ROBINSON DAM

Component/Commodity	Intended Function	AMR Results
Anchorage/Embedments Exposed Surfaces	Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 18
Anchorage/Embedments (Embedded/Encased in Concrete)	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 9
Lake Dam	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide source of cooling water for plant shutdown. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 18
Spillway for Dam Structure	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide source of cooling water for plant shutdown. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 18 Table 3.5-2, Item 10

TABLE 2.4-10 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: LAKE ROBINSON DAM

Component/Commodity	Intended Function	AMR Results
Structural Steel (Beams, Plates, Connectors, Column)	Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 18
Gates/Valves (Dam)	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide source of cooling water for plant shutdown. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 18
Threaded Fasteners	Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 18

2.4.2.11 Pipe Restraint Tower

The Pipe Restraint Tower is a Seismic Class I structural steel frame structure supported by a reinforced concrete foundation. The foundation is supported on pilings. Grating platforms are located at various elevations. This structure provides access to components and is required for mitigation of pipe whip and jet impingement as a result of postulated high energy line breaks outside the containment. The location is due south of the reactor containment structure between Turbine Building column lines 11 and 12 (approximately). The Pipe Restraint Tower supports the Main Steam Safety Relief and Isolation Valves, the Feedwater Isolation Valves, and acts as a pipe whip restraint for the Main Steam and Feedwater Lines. The Pipe Restraint Tower is not physically attached to the Containment Building and is connected via platforms to the seismic Class I portion of the Turbine Building. Refer to UFSAR Figure 1.2.2-11.

The Pipe Restraint Tower is in the scope of license renewal because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions

3. SCs that are relied on during postulated fires, anticipated transients without scram, and station blackout events.

The following table identifies the Pipe Restraint Tower components/commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.4-11 COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: PIPE RESTRAINT TOWER

Component/Commodity	Intended Function	AMR Results
Anchorage/Embedments Exposed Surfaces	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Anchorage/Embedments (Embedded/Encased in Concrete)	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 9
Cable Tray and Conduit	Provide structural and/or functional support to safety-related equipment. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 3

TABLE 2.4-11 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: PIPE RESTRAINT TOWER

Component/Commodity	Intended Function	AMR Results
Electrical & Instrument Panels and Enclosures	Provide shelter/protection to safety-related equipment (including radiation shielding). Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-2, Item 3
Electrical Component Supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-2, Item 3
Instrument Line Supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Miscellaneous Steel (Stairs & Ladders, Platforms & Connectors, Grating & Checker Plate)	Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16

TABLE 2.4-11 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: PIPE RESTRAINT TOWER

Component/Commodity	Intended Function	AMR Results
Pilings	Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 6
Pipe Supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Pipe Whip Restraints	Provide pipe whip restraint and/or jet impingement protection.	Table 3.5-1, Item 25
Reinforced Concrete (Beams, Walls, Floors, Columns, etc.)	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-1, Item 17 Table 3.5-1, Item 21 Table 3.5-1, Item 22 Table 3.5-1, Item 23 Table 3.5-2, Item 10
Structural steel (Beams, Plates, Connectors, Column)	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16

TABLE 2.4-11 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: PIPE RESTRAINT TOWER

Component/Commodity	Intended Function	AMR Results
Threaded Fasteners	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25

2.4.2.12 Yard Structures and Foundations

Yard Structures and Foundations include concrete foundations and steel supports for miscellaneous in scope equipment, concrete trenches for in-scope piping and utilities, electrical enclosures and panels located in Personnel Access Portal (PAP) West supporting security lighting, and concrete duct banks and manholes. Portions of the PAP West structure were evaluated to be in scope during the screening process for security lighting when security lighting circuits were determined to be located there. The Yard Structures and Foundations classification includes Miscellaneous Yard Structures consisting of foundations (concrete and structural steel) for piping, cable trays, conduits, and electrical enclosures and panels located outside other structures and buildings. Concrete foundations and steel supports in the Yard Structures and Foundations category consist of the following. Most of these structures are shown on Figure 2.2-1.

1. Refueling Water Storage Tank (RWST) foundation. The structure includes the reinforced concrete ring foundation and steel anchorage for the tank.

2. Steam Generator Blowdown Tank foundation. The structure includes the tank skirt, supports, concrete foundation, and access platform.

3. Primary Water Storage Tank foundation. This structure was originally included in scope of license renewal; however, the tank was determined to be outside of the intended function boundary for license renewal. Consequently, the tank foundation structure does not support any license renewal intended functions.

4. Condensate Storage Tank foundation. The structure includes the reinforced concrete ring foundation and anchorage.

5. Diesel Generator Fuel Oil Storage Tank foundation. This structure includes the reinforced concrete tank foundation, anchorage, containment dike, and the supports for the fuel oil equipment.

6. Dedicated Shutdown Diesel Generator Fuel Oil Tank foundation. Included in this structure are the reinforced concrete foundation, saddle supports, and the concrete containment dike.

7. Diesel Fire Pump Fuel Oil Tank foundation. This structure consists of the reinforced concrete foundation, saddle supports, anchorage, access platform, and the concrete containment dike.

8. Unit 1 IC Fuel Oil Storage Facility structure. The structure includes three circular tank foundations, flange supports, and anchorage assemblies.

9. Dedicated Shutdown System Main Transformer structure. This structure includes the reinforced concrete foundation, protective masonry wall, and bus duct support.

10. Security Lighting structure. This structure includes portions of the PAP West and the foundations, poles, and electrical duct banks supporting two high mast lights.

11. Electrical Manholes and Duct Banks. This structure includes three electrical manholes associated with safety related electrical cable in the yard. Two of the manholes are located within the North Service Water Header Enclosure.

12. Concrete Trenches. This structure is the reinforced concrete trench in the area between the Turbine Building and the Condensate Storage Tank.

13. Miscellaneous Yard Structures. These structures are located outside the boundaries of other in-scope structures, such as the major buildings, and include, for example, supports for piping runs between or outside of the buildings.

Yard Structures and Foundations are in the scope of license renewal because they contain (individual structures may not perform all of the following functions):

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions

3. SCs that are relied on during postulated fires and station blackout events.

The following table identifies the Yard Structures and Foundations components/ commodities requiring aging management review (AMR), identifies their intended functions, and provides a reference to the results of the AMR for each component/commodity type.

TABLE 2.4-12 COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: YARD STRUCTURES AND FOUNDATIONS

Component/Commodity	Intended Function	AMR Results
Anchor Bolt Chair for Tank Foundation	Provide structural and/or functional support to safety-related equipment. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16
Anchorage/Embedments Exposed Surfaces	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Anchorage/Embedments (Embedded/Encased in Concrete)	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 9
Cable Tray and Conduit	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 2 Table 3.5-2, Item 3
Concrete Tank Foundation	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 17 Table 3.5-2, Item 10

TABLE 2.4-12 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: YARD STRUCTURES AND FOUNDATIONS

Component/Commodity	Intended Function	AMR Results
Doors	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 1
Electrical & Instrument Panels and Enclosures	Provide shelter/protection to safety-related equipment (including radiation shielding) Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-2, Item 1 Table 3.5-2, Item 3
Electrical Component Supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25 Table 3.5-2, Item 3
Electrical Manhole	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 17 Table 3.5-2, Item 10
Expansion Anchors	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-2, Item 1 Table 3.5-1, Item 25
Manhole Covers	Provide shelter/protection to safety-related equipment (including radiation shielding). Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16

TABLE 2.4-12 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: YARD STRUCTURES AND FOUNDATIONS

Component/Commodity	Intended Function	AMR Results
Masonry Walls	Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 20
Miscellaneous steel (Stairs & Ladders, Platforms & Connectors, Grating & Checker Plate)	Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16
Pipe Supports	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Protective Enclosure (Structures sheltering or enclosing plant equipment)	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16
Reinforced Concrete (Beams, Walls, Floors, Columns, etc.)	Provide shelter/protection to safety-related equipment (including radiation shielding). Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-1, Item 17 Table 3.5-2, Item 10
Siding	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16

TABLE 2.4-12 (continued) COMPONENT COMMODITY GROUPS REQUIRING AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS: YARD STRUCTURES AND FOUNDATIONS

Component/Commodity	Intended Function	AMR Results
Structural Steel (Beams, Plates, Connectors, Column)	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 16 Table 3.5-2, Item 3
Threaded Fasteners	Provide structural and/or functional support to safety-related equipment. Provide structural and/or functional support to non safety-related equipment where failure of this structural component could prevent satisfactory accomplishment of any of the required safety-related functions. Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 25
Underground Conduit Duct Bank	Provide structural support and/or shelter to components required for Fire Protection, ATWS and/or SBO.	Table 3.5-1, Item 17

2.4.2.13 Refueling System

The Refueling System contains components in the Containment and the Fuel Handling Building and provides a safe, effective means of transporting and handling fuel. There are no safety related components (except for the fuel transfer tube blind flange) in the Refueling System. The flange was transferred to the Containment Building and is screened there along with the Fuel Transfer Tube. No safety related functions are associated with this equipment, and no intended functions were assigned to the system other than for the Fuel Transfer Tube Flange. Therefore, all remaining components were screened as out of the evaluation boundary.

The flange on the fuel transfer tube is discussed in Subsection 2.4.1.1.2 as part of the Containment.

2.4.3 REFERENCES

2.4-1 CP&L letter to NRC, dated November 12, 1998, Response to Generic Letter 98-04, “Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-Of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment,” RNP-RA/98-0203.

2.5 SCOPING AND SCREENING RESULTS – ELECTRICAL AND INSTRUMENTATION AND CONTROLS (I&C) SYSTEMS

The methodology used to identify electrical/I&C components requiring an aging management review is discussed in Subsection 2.1.2.3. The screening for electrical/I&C components was performed on a generic component (commodity group) basis for the in-scope electrical/I&C systems listed in Table 2.2-3, as well as the electrical/I&C component types associated with in-scope mechanical systems and civil structures listed in Tables 2.2-1 and 2.2-2. The methodology employed is consistent with the guidance in NEI 95-10 [Reference 2.5-1].

The interface of electrical/I&C components with other types of components and the assessments of these interfacing components are provided in the appropriate mechanical or civil/structural sections. For example, the assessment of electrical racks, panels, frames, cabinets, cable trays, conduit, and their supports is provided in the civil/structural assessment documented in Section 2.4.

2.5.1 ELECTRICAL/I&C COMPONENT COMMODITY GROUPS

The electrical/I&C component types were identified from a review of plant documents, controlled drawings, the equipment database, and interface with the parallel mechanical and civil screening efforts. The in-scope electrical/I&C component commodity groups identified at RNP are listed in Table 2.5-1. This list includes all electrical/I&C component commodity groups listed in Appendix B of NEI 95-10 [Reference 2.5-1], with the exception of those types that did not meet the requirements of 10 CFR 54.4(a). Component types eliminated on this basis are:

1. **Electrical Bus** – The isolated-phase bus system (and associated isolated phase bus duct) and the switchyard and transformer system (and associated switchyard bus) are not relied on to meet the license renewal scoping requirements of 10 CFR 54.4(a).

2. **Transmission Conductors** – Transmission conductors are not relied on to meet the license renewal scoping requirements of 10 CFR 54.4(a).

3. **High Voltage Insulators** – High voltage insulators are not relied on to meet the license renewal scoping requirements of 10 CFR 54.4(a).

4. **High Voltage Surge Arrestors** – High voltage surge arrestors are not relied on to meet the license renewal scoping requirements of 10 CFR 54.4(a).

5. **Uninsulated Ground Cables** – Uninsulated ground cables are for personnel protection and not relied on to meet the license renewal scoping requirements of 10 CFR 54.4(a).

No additional electrical component commodity groups, beyond those listed in Appendix B of NEI 95-10, were identified.

2.5.2 APPLICATION OF SCREENING CRITERION 10 CFR 54.21(a)(1)(i) TO ELECTRICAL/I&C COMPONENT COMMODITY GROUPS

Following the identification of the electrical/I&C component commodity groups, the criteria of 10 CFR 54.21(a)(1)(i) was applied to identify component commodity groups that perform their intended functions without moving parts or without a change in configuration or properties. This evaluation was performed utilizing the guidance of 10 CFR 54.21(a)(1)(i) and NEI 95-10 [Reference 2.5-1].

The following electrical/I&C component commodity groups were determined to meet the screening criteria of 10 CFR 54.21(a)(1)(i) and were further evaluated against the criteria of 10 CFR 54.21(a)(1)(ii) as discussed in the following Subsection.

1. Bus Duct
2. Insulated Cables and Connections (including splices, connectors, and terminal blocks)
3. Electrical/I&C Penetration Assemblies

2.5.3 APPLICATION OF SCREENING CRITERION 10 CFR 54.21(a)(1)(ii) TO ELECTRICAL/I&C COMPONENT COMMODITY GROUPS

10 CFR 54.21(a)(1)(ii) allows the exclusion of those component commodity groups that are subject to replacement based on a qualified life or specified time period. The 10 CFR 54.21(a)(1)(ii) screening criterion was applied to the specific component commodity groups that were included by application of the 10 CFR 54.21(a)(1)(i) criterion. The results of this review are discussed below.

2.5.3.1 Bus Duct

The function of bus ducts is to electrically connect power supplies and load centers to deliver voltage and current. The bus ducts utilize pre-assembled raceway (enclosure) design with internal conductors installed on electrically insulated supports. Bus duct insulated copper conductors, their associated insulators, and electrical connections are reviewed as a single component commodity group. Bus ducts within scope of license renewal are: (1) non-segregated 480V bus duct connecting Emergency Diesel Generator A to Emergency Bus E1, (2) non-segregated 480V bus duct connecting Emergency Diesel Generator B to Emergency Bus E2, (3) non-segregated bus duct from the Dedicated Shutdown System Transformer to the Dedicated Shutdown Bus, (4) non-segregated bus duct connecting 480V Switchgear Bus 3 to the Dedicated Shutdown Bus, and (5) the cross-tie, non-segregated bus duct connecting Emergency Bus E1 and E2.

Bus ducts are not included in the RNP Environmental Qualification (EQ) Program. Equipment in the EQ Program has a documented qualified life. Components in the EQ Program that have a qualified life less than 40 years are replaced on the basis of a specified time period at the end of their qualified life. Components in the EQ Program that have a qualified life based on the 40-year current operating license term are the subject of Time-Limited Aging Analysis (TLAA). As no bus ducts are within the scope of the EQ Program, bus ducts in the scope of license renewal are considered to meet the criterion of 10 CFR 54.21(a)(1)(ii) and are subject to an aging management review.

2.5.3.2 Insulated Cables And Connections

The function of insulated cables and connections is to electrically connect specified sections of an electrical circuit to deliver voltage, current or signals. Electrical cables and their required terminations (i.e., connections) are reviewed as a single component commodity group. The types of connections included in this review are splices, connectors, and terminal blocks. Numerous insulated cables and connections are included in the EQ Program. The insulated cables and connections that are included in this program have a qualified life that is documented in the EQ Program. Components in the EQ Program that have a qualified life less than 40 years are replaced on the basis of a specified time period at the end of their qualified life. Components in the EQ Program that have a qualified life based on the 40-year current operating license term are the subject of Time-Limited Aging Analysis. Accordingly, all insulated cables and connections within the EQ Program are exempt from screening under 10 CFR 54.21(a)(1)(ii) and are not subject to an aging management review. Note that Time-Limited Aging Analyses associated with electrical/I&C components within the EQ Program are discussed in Subsection 4.4.1.

Insulated cables and connections that perform an intended function within the scope of license renewal, but are not included in the EQ Program, meet the criterion of 10 CFR 54.21(a)(1)(ii) and are subject to an aging management review.

The classification of electrical cables and connections into the above two categories, those subject to the EQ Program and those not subject to the EQ Program, is consistent with the classification used in Volume 2, Chapter VI, of the GALL Report, NUREG-1801 [Reference 2.5-2].

2.5.3.3 Electrical/I&C Penetration Assemblies

Electrical/I&C penetration assemblies included in the EQ Program have a qualified life that is documented. Therefore, electrical/I&C penetration assemblies in the EQ Program do not meet the criterion of 10 CFR 54.21(a)(1)(ii) and are not subject to an aging management review.

A review of the electrical/I&C penetrations determined that in addition to the electrical/I&C penetration assemblies included in the EQ Program, additional electrical penetration assemblies are employed at RNP. Except for spare penetrations and one penetration supporting a single, out-of-scope circuit, these additional electrical/I&C penetration assemblies were considered to be subject to an aging management review whether or not their associated cables are in the scope of license renewal. The penetration supporting the single, out of scope circuit is the same design as those covered by the EQ Program. Therefore, electrical penetrations not included in the EQ Program are considered to meet the criterion of 10 CFR 54.21(a)(1)(ii) and are subject to an aging management review except for spare penetrations and one non-EQ penetration containing a single, out-of-scope circuit.

2.5.4 ELECTRICAL/I&C COMPONENTS REQUIRING AN AGING MANAGEMENT REVIEW

The electrical/I&C component commodity groups subject to an aging management review include:

1. Bus Duct supporting Emergency Buses E1 and E2 and the Dedicated Shutdown System Bus

2. Insulated Cables and Connections (including splices, connectors, and terminal blocks) not included in the EQ Program

3. Electrical/I&C penetration assemblies not included in the EQ Program

A listing of electrical/I&C component commodity groups subject to aging management review, their intended functions, and reference to the AMR results is provided in tabular form below.

Electrical and Instrumentation and Control Systems		
Component/Commodity	**Intended Function**	**AMR Results**
Bus Duct	Electrically connect specified sections of an electrical circuit to deliver voltage, current, or signals	Table 3.6-2, Item 1
Non-EQ Insulated Cables and Connections	Electrically connect specified sections of an electrical circuit to deliver voltage, current, or signals	Table 3.6-1, Item 2 Table 3.6-1, Item 5 Table 3.6-2, Item 2
Non-EQ Electrical/I&C penetration assemblies	Electrically connect specified sections of an electrical circuit to deliver voltage, current, or signals	Table 3.6-1, Item 2

2.5.5 REFERENCES

2.5-1 NEI 95-10, “Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 – The license Renewal Rule,” Revision 3, Nuclear Energy Institute, March 2001.

2.5-2 NUREG-1801, Generic Aging Lessons Learned (GALL) Report, U. S. Nuclear Regulatory Commission, April 2001.

TABLE 2.5-1 ELECTRICAL/I&C COMPONENT COMMODITY GROUPS

ELECTRICAL/I&C COMPONENT COMMODITY GROUPS FOR IN-SCOPE SYSTEMS AND STRUCTURES AT RNP			
Alarm Units	Electrical/I&C Penetration Assemblies	Loop Controllers	Signal Conditioners
Analyzers	Elements	Meters	Solenoid Operators
Annunciators	Fuses	Motor Control Centers	Solid-State Devices
Batteries	Generators	Motors	Splices
Bus Duct	Heat Tracing	Power Distribution Panels	Surge Arresters
Chargers	Heaters	Power Supplies	Switches
Circuit Breakers	Indicators	Radiation Monitors	Switchgear
Converters	Insulated Cables and Connections	Recorders	Terminal Blocks
Communication Equipment	Inverters	Regulators	Thermocouples
Electrical Controls and Panel Internal Component Assemblies	Isolators	Relays	Transducers
	Light Bulbs	RTDs	Transformers
	Load Centers	Sensors	

3.0 AGING MANAGEMENT REVIEW RESULTS

For those structures and components that are identified as being subject to an aging management review, 10 CFR 54.21(a)(3) requires demonstration that the effects of aging will be adequately managed so that their intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.

This chapter describes the results of the aging management reviews of the components and structures determined, during the scoping and screening processes, to be subject to an aging management review. During the screening process, some structures and components (SCs) were incorporated into commodity groups based on similarity of their design or materials of construction. Use of commodity groups made it possible to address an entire group of SCs with a single evaluation. In the aging management reviews described in the following Sections, further definition of commodity groups was performed based on design, material, environmental, and functional characteristics in order to disposition an entire group with a single aging management review.

Organization of this chapter parallels Chapter 3, “Aging Management Review Results” of NUREG-1800, Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants (SRP-LR), dated April 2001. The major sections of this Chapter are:

3.1 Aging Management of Reactor Vessel, Internals, and Reactor Coolant System

3.2 Aging Management of Engineered Safety Features

3.3 Aging Management of Auxiliary Systems

3.4 Aging Management of Steam and Power Conversion Systems

3.5 Aging Management of Containments, Structures, and Component Supports

3.6 Aging Management of Electrical and Instrumentation and Controls

Components and structures subject to an aging management review were evaluated to demonstrate that the effects of aging will be managed so that the intended functions will be maintained consistent with CLB for the period of extended operation. The components, aging effects/mechanism, and aging management programs to be used for managing the effects of aging at RNP were compared to those listed in NUREG-1801, Generic Aging Lessons Learned (GALL), dated April 2001 (the GALL Report). The results are documented and discussed in the above Sections using the format suggested by Tables 3.1-1 through 3.6-1 of the SRP-LR. Aging management programs are described in Appendix B.

The RNP environments used in the aging management reviews are listed in the "Environment" column in Tables 3.0-1 and 3.0-2. Some of the internal environments have been subdivided into groups based on the fluid chemistry. The subgroups are identified in the "Description" column in Table 3.0-1.

The aging management review methodology for RNP did not credit the effects of aging management programs when determining if an aging effect requiring management may be applicable. The potential aging effects were evaluated assuming that any applicable aging management programs were not in effect. No credit was taken for coatings and linings, cathodic protection systems, corrosion inhibitors, biocides, inspections or other programs during the aging management reviews, because the entire set of aging effects requiring management may not be identified if these programs were credited a priori.

TABLE 3.0-1 - INTERNAL SERVICE ENVIRONMENTS

Environment	Description
Air and Gas	Includes atmospheric air, dry/ filtered instrument air, nitrogen, carbon dioxide, hydrogen, and helium
Treated Water (includes steam)	Treated water is demineralized water and is the base water for all clean systems. Two basic categories of treated water are: borated and unborated. Treated water can be subdivided into groups based in the chemistry of the water: <u>Treated water – primary</u>: Normal Reactor Coolant System makeup water quality <u>Treated water – borated, other</u>: Systems that contain borated water except those included in treated water – primary <u>Treated water – secondary</u>: Normal operating secondary chemistry <u>Treated water – inhibited</u>: Closed cooling water systems that use corrosion inhibitors and, in some cases, biocides.
Raw Water	Water drawn from Lake Robinson at the Intake Structure and well water drawn from deep wells at the site.
Fuel Oil	Fuel oil for the Emergency Diesel Generators, Diesel-driven Fire Pump, Emergency Operations Facility/ Technical Support Center Security Diesel, and the Dedicated Shutdown Diesel
Lubricating Oil	Lubricating oil for diesel engines, pumps, air compressors, the main turbine, oil storage tanks, and other components
Ohmic Heating	Internal resistance heating of the conductors of power cables

TABLE 3.0-2 - EXTERNAL SERVICE ENVIRONMENTS

Environment	Description
Outdoor [1]	Atmospheric air, temperature: 10–95°F, 40–95% humidity. Exposed to weather including precipitation and wind.
Indoor – Not Air Conditioned [1]	Atmospheric air, temperature: 104°F maximum, 20–80% humidity. Not exposed to weather.
Indoor – Air Conditioned [1]	Atmospheric air, specific temperature range/humidity dependent upon building/room. Typically, temperature: 40 - 85°F. Not exposed to weather.
Containment air [1]	Atmospheric air, Pressure ±1 psig, bulk temperature 120°F maximum, 20–95% humidity (35% average), Radiation – total integrated dose: $< 1.65 \times 10^6$ rad inside secondary shield wall and 4.6×10^3 rad outside secondary shield wall. Maximum neutron fluence outside the primary shield wall 1×10^{12} n/cm^2 (E>1 Mev). Not exposed to weather.
Buried	Exposed to soil/fill or ground water
Borated Water Leaks	Exposed to leakage from borated water systems
Embedded/Encased	Embedded/Encased in concrete

NOTE: 1. A component is considered susceptible to a wetted environment when it is submerged, has the potential to pool water, or is subject to external condensation. An outdoor environment alone does not constitute a wetted environment.

3.1 AGING MANAGEMENT OF REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

Systems and components included in this section are those associated with supporting the operability of the nuclear steam supply system to safely generate and remove heat from the reactor and provide a supply of steam for generating electricity.

Portions of the Incore Nuclear Instrumentation System (thimble tube pressure boundary), Chemical and Volume Control System (regenerative heat exchanger and Reactor Coolant Pump seal injection and return lines), Safety Injection System (piping and valves), Residual Heat Removal System (piping and valves), Reactor Vessel Level Instrumentation/Inadequate Core Cooling Monitor System, and the Primary Sampling System (piping and isolation valves) are included in this subsection, because they form part of the Class 1 Reactor Coolant System boundary.

3.1.1 AGING MANAGEMENT REVIEW

3.1.1.1 Methodology

Aging management review of Reactor Vessel, Internals, and Reactor Coolant System components and commodities involved consideration and evaluation of the materials, environments, and stressors that are associated with each structure, component, or commodity grouping under review, as discussed in NEI 95-10 [Reference 3.1-1], Section 4.2. The RNP AMR methodology follows the approach recommended in NEI 95-10 and is based on generic industry guidance for determining aging effects for both mechanical and civil/structural components. The guidance represents a set of rules that allow the evaluator to identify aging effects for a given material and environment combination. The material and environment-based rules, in the generic industry guidance documents, are derived from known age-related degradation mechanisms and industry operating experience. The guidance was reviewed for applicability to RNP materials of construction and component internal and external operating environments and was used to identify aging effects for components, structures, and commodities. The results of the evaluation of materials and environment combinations, using the RNP methodology, are aging effects; and, if the aging effects adversely affect intended functions, the results are aging effects requiring management for the applicable components and commodities. Aging effects that require management are correlated to aging management programs.

The aging management review identifies one or more aging management programs to be used to demonstrate that the effects of aging will be managed so that the intended functions will be maintained consistent with CLB for the period of extended operation. The programs to be used for managing the effects of aging were compared to those listed in the GALL Report [Reference 3.1-2] and evaluated for consistency with GALL Report programs that are relied on for license renewal. The results are documented and discussed in Subsection 3.1.2 using the format suggested by the NRC Standard

Review Plan for License Renewal (SRP-LR) [Reference 3.1-3]. Aging management programs are described in Appendix B.

3.1.1.2 Operating Experience

Operating experience (OE) through December 2001 was considered during the development of the RNP Integrated Plant Assessment. OE subsequent to that date will be reviewed and applicable OE will be updated in conjunction with the amendment to the application required by 10 CFR 54.21(b). The review consisted of the following:

Site: RNP site-specific operating experience was reviewed. The site-specific operating experience included a review of (1) Corrective Action Program, (2) Licensee Event Reports, (3) Maintenance Rule Data Base, and (4) interviews with Systems Engineers. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Subsection.

Industry: An evaluation of industry operating experience published since the effective date of the GALL Report was performed to identify any additional aging effects requiring management. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Subsection.

On-Going On-going review of plant-specific and industry operating experience is performed in accordance with the Corrective Action and Operating Experience Programs.

Attention has been directed to issues involving the potential for leakage from the reactor vessel head as documented in NRC Bulletin 2001-01, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,” dated August 3, 2001. By letter dated September 4, 2001, CP&L provided a response to that Bulletin and supplemented the response by letters dated October 2, October 19, November 2, and November 12, 2001. In addition, the NRC and CP&L held a telephone conference, on October 11, 2001, and a meeting on October 24, 2001. NRC letter, dated November 20, 2001, concluded that the visual examination of the VHP nozzles performed at RNP in April 2001 provides reasonable assurance of the structural integrity of the VHP nozzles until the next inspection scheduled for the fall 2002 outage of the plant, and that there is reasonable assurance that the public health and safety will be maintained. On March 18, 2002, the Nuclear Regulatory Commission (NRC) issued NRC Bulletin 2002-01, “Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," requesting further information related to the integrity of the reactor coolant pressure boundary, including the reactor pressure vessel (RPV) head, and the extent to which inspections have been undertaken to satisfy applicable regulatory requirements. By letter dated April 1, 2002, CP&L provided the requested information.

3.1.2 AGING MANAGEMENT PROGRAMS

3.1.2.1 Aging Management Programs Evaluated in the GALL Report that Are Relied on for License Renewal

Table 3.1-1 shows the component and commodity groups (combinations of materials and environments), and aging management programs evaluated in the GALL Report that are relied on for license renewal of the Reactor Vessel, Internals, and Reactor Coolant System. The table is based on Table 3.1-1 of the SRP-LR and provides a discussion of the applicability of the component commodity group and details regarding the degree to which RNP aging management programs are consistent with those recommended in the GALL Report. The discussion section includes (1) information regarding the applicability of the GALL Report component/commodity group to RNP, (2) any issues recommended in the GALL Report that require further evaluation, (3) details regarding RNP components to be included in the component/commodity group, and (4) a conclusion regarding consistency of the aging management review with the GALL Report.

3.1.2.2 Further Evaluation of Aging Management as Recommended by the GALL Report

Further evaluation of aging management as recommended by the GALL Report has been incorporated into the “Discussion” column of Table 3.1-1.

3.1.2.3 Aging Management Evaluations that Are Different from or Not Addressed in the GALL Report

Aging Management Evaluations that are different from or not addressed in the GALL Report are identified and discussed on Table 3.1-2.

3.1.3 CONCLUSIONS

Reactor Vessel, Internals, and Reactor Coolant System aging effects requiring management are adequately managed by the following programs:

1. ASME Section XI Subsection IWB, IWC, and IWD Program
2. Bolting Integrity Program
3. Boric Acid Corrosion Program
4. Fatigue Monitoring Program
5. Flow-Accelerated Corrosion Program
6. Flux Thimble Eddy Current Inspection Program
7. Nickel-Alloy Nozzles and Penetrations Program
8. One-Time Inspection Program
9. Preventive Maintenance Program
10. PWR Vessel Internals Program
11. Reactor Head Closure Studs Program
12. Reactor Vessel Surveillance Program
13. Steam Generator Tube Integrity Program
14. Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program
15. Water Chemistry Program

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that the intended functions of Reactor Vessel, Internals, and Reactor Coolant System components are maintained consistent with the current licensing basis for the period of extended operation.

3.1.4 REFERENCES

3.1-1 NEI 95-10, "Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule," Revision 3, Nuclear Energy Institute, March 2001.

3.1-2 NUREG-1801, "Generic Aging Lessons Learned (GALL)," U. S. Nuclear Regulatory Commission, April 2001.

3.1-3 NUREG-1800, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," U. S. Nuclear Regulatory Commission, April 2001.

TABLE 3.1-1 REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
1. Reactor coolant pressure boundary components	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	Thermal fatigue was identified in the GALL Report and in the RNP aging evaluation methodology as an aging effect requiring management. Fatigue of metal components is addressed as a TLAA in Section 4.3 for those components having evaluations based on time-limited assumptions defined by the current operating license term. This is consistent with the GALL Report. In accordance with the GALL Report, steam generator pressure boundary and reactor internals components are included although they are not part of the reactor coolant pressure boundary.
2. Steam generator shell assembly	Loss of material due to pitting and crevice corrosion (GALL Section IV.D1.1-c includes general corrosion)	Inservice inspection; water chemistry	Yes, detection of aging effects is to be further evaluated.	IN 90-04 states that “if general corrosion pitting of the SG shell is known to exist, the requirements of Section XI of the ASME Code may not be sufficient to differentiate isolated cracks from inherent geometric conditions.” The subject of IN 90-04 is cracking of the upper shell-to-transition cone girth welds in steam generators. AMPs applicable at RNP are the Water Chemistry and ASME Section XI, Subsection IWB, IWC, and IWD Programs. RNP has included the carbon steel steam and feedwater nozzles in this group, as they are attached to the steam generator secondary shell. (continued)

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
2. Steam generator shell assembly (continued)				For RNP, periodic augmented monitoring of steam generator upper shell to transition cone girth weld is performed in accordance with actions initiated in response to Information Notice 82-37. In 1990, the inside surface of this weld in Steam Generator A was visually examined. Only occasional minor surface pitting was observed. Similar conditions were reported and resolved during Steam Generator repairs in 1984 during which new lower portions of the steam generators (including the transition cones) were welded to existing upper steam generator top heads. Because no significant pitting occurred in Steam Generator A from 1984 to 1990 and all steam generators have been subjected to strict chemistry controls, pitting should not be a concern with respect to interfering with augmented inspections of the girth welds. Based on the above, management of pitting and crevice corrosion in the steam generator shell assemblies is consistent with the GALL Report.
3. Pressure vessel ferritic materials that have a neutron fluence greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of fracture toughness due to neutron irradiation embrittlement	TLAA, evaluated in accordance with Appendix G of 10 CFR 50 and RG 1.99	Yes, TLAA	Refer to Section 4.2 for TLAAs related to reactor pressure vessel integrity and loss of fracture toughness from neutron irradiation embrittlement. Consideration of this component/commodity group as a TLAA is consistent with the GALL Report. For RNP, the safety injection nozzles do not attach to the reactor vessel.

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
4. Reactor vessel beltline shell and welds	Loss of fracture toughness due to neutron irradiation embrittlement	Reactor vessel surveillance	Yes, plant specific	The RNP Reactor Vessel Surveillance Program, together with TLAA analyses, is used to manage the aging effects of reduction of fracture toughness due to neutron irradiation embrittlement for the Reactor Vessel beltline shell and welds. The Reactor Vessel Surveillance Program provides sufficient material data and neutron dosimetry information to predict irradiation embrittlement at the end of the period of extended operation and determine the need for operating restrictions to preserve Reactor Vessel fracture toughness. Nozzle and nozzle weld materials were evaluated and determined not to be controlling based on fracture toughness analyses. In addition, RNP has an active Reactor Surveillance Program with scheduled withdrawals extending into the license renewal period. Thus, the RNP Reactor Vessel Surveillance Program surveillance capsule withdrawal schedule provides for adequate vessel materials surveillance for the period of extended operation. Aging management of this component/commodity group is consistent with the GALL Report.

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
5. Westinghouse and B&W baffle/former bolts	Loss of fracture toughness due to neutron irradiation embrittlement and void swelling	Plant specific	Yes, plant specific	The RNP AMR identified reduction in fracture toughness due to neutron irradiation embrittlement and change in dimensions due to void swelling as two aging effects/mechanisms applicable to Baffle/Former Bolts. Both of these aging mechanisms will be managed by the PWR Vessel Internals Program. RNP will continue to participate in industry programs whose objectives include the investigation of aging effects applicable to Baffle/Former Bolts and identification of appropriate AMP(s) and aging management activities. Appropriate and applicable surveillance techniques will be incorporated as enhancements to the aging management activities applicable to Baffle/Former Bolts. Therefore, the plant-specific program called for in the GALL Report, consists of the PWR Vessel Internals Program as updated and enhanced in accordance with applicable results of ongoing industry activities. This is consistent with the GALL Report.

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
6. Small-bore reactor coolant system and connected systems piping [2]	Crack initiation and growth due to SCC, intergranular SCC, and thermal and mechanical loading	Inservice inspection; water chemistry; one-time inspection	Yes, parameters monitored/inspected and detection of aging effects are to be further evaluated	The Water Chemistry Program will manage the aging effect of cracking due to SCC in RCS branch lines < NPS 4; and, to determine the efficacy of the Water Chemistry Program in managing the effects of SCC prior to the period of extended operation, an inspection, per the One-Time Inspection Program, will be performed. The one-time inspection will be used to verify that service-induced weld cracking is not occurring by checking a representative sample of piping. Components to be examined will be selected based on accessibility, exposure levels, NDE techniques, and locations identified in NRC Information Notice (IN) 97-46. Besides the Water Chemistry Program, the GALL Report also cites the ASME Section XI, Subsections IWB, IWC, and IWD Program to address the effects of SCC. However, the GALL Report notes that volumetric inspections are not required by ASME Section XI for pipe < NPS 4. Therefore, the proposed combination of Water Chemistry Program and One-Time Inspection is consistent with the GALL Report for managing the effects of service-induced weld cracking for RCS Piping, Fitting, and Branch Connections < NPS 4.
7. Vessel shell	Crack growth due to cyclic loading	TLAA	Yes, TLAA	The evaluation of potential cracks in carbon or low alloy steel under austenitic stainless steel cladding is a TLAA. Refer to Section 4.3 for a summary of the TLAA evaluation of underclad cracking. This is consistent with the GALL Report.

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
8. Reactor internals	Changes in dimension due to void swelling	Plant specific	Yes, plant specific	The neutron flux thimble guide tubes were determined to be not susceptible to void swelling, because of the location of the guide tubes outside the reactor vessel. Void swelling can be potentially significant for components that can experience significant neutron irradiation while operating at elevated temperatures. This is not applicable to the flux thimble guide tubes. The GALL Report elicits a plant-specific program to manage the effects of void swelling. The RNP AMP applicable to this aging effect/mechanism is the PWR Vessel Internals Program. The PWR Vessel Internals Program is focused on managing several aging effects/mechanisms. These include changes in dimensions due to void swelling. As discussed in Item 5 above, RNP will continue to participate in industry programs whose objectives include the investigation of aging effects applicable to reactor internals and identification of appropriate AMP(s). RNP will incorporate the applicable results of industry initiatives related to void swelling into the PWR Vessel Internals Program. Based on the above, the AMP to be applied at RNP is consistent with the GALL Report.

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
9. PWR core support pads, instrument tubes (bottom head penetrations), pressurizer spray heads, and nozzles for the steam generator instruments and drains	Crack initiation and growth due to SCC and/or primary water stress corrosion cracking (PWSCC)	Plant specific	Yes, plant specific	The Pressurizer Spray Head performs no license renewal intended functions at RNP. The steam generator instrument nozzles (GALL item D1.1.10) are not fabricated from Alloy 600, so they do not meet the criteria of this group. The Reactor Pressure Vessel (RPV) flange leak detection line is fabricated from stainless steel and is included in the category of small bore piping. Management of crack initiation and growth for small-bore stainless steel piping is addressed in Item 6 above and is consistent with the GALL Report. The RNP Core Support Pads and Reactor Vessel Bottom Head Penetrations are fabricated of Nickel-based Alloy. The Water Chemistry Program is used to manage cracking from SCC for the support pads and both the ASME Section XI, Subsections IWB, IWC, and IWD Program and the Water Chemistry Program to manage cracking from SCC for the bottom head penetrations. As these AMPs differ from the plant-specific AMP recommended by the GALL Report, aging management for these components is addressed in Table 3.1-2, Items 9 and 10.

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
10. Cast austenitic stainless steel (CASS) reactor coolant system piping	Crack initiation and growth due to SCC	Plant specific	Yes, plant specific	The RNP pressurizer surge nozzle is not fabricated from CASS. It is carbon steel clad with stainless steel. RNP has included the CASS reactor coolant pump casing in this group. The RNP Water Chemistry Program is applicable. According to the GALL Report, Section IV.C, with respect to SCC of CASS components, a plant-specific program is required unless certain conditions apply. One of the conditions is maintaining water chemistry in accordance with EPRI TR-105714, Rev. 3 (or more recent). RNP meets this water chemistry requirement. Therefore, the aging management for CASS piping and reactor coolant pump casing in the RCS is consistent with the GALL Report.
11. Pressurizer instrumentation penetrations and heater sheaths and sleeves made of Ni-alloys	Crack initiation and growth due to PWSCC	Inservice inspection; water chemistry	Yes, AMP for PWSCC of Inconel 182 weld is to be evaluated	RNP Pressurizer instrument penetrations and heater sheaths and sleeves are made of stainless steel. No Alloy 600 components are used in the Pressurizer. However, the RNP AMPs (Water Chemistry Program and ASME Section XI, Subsection IWB, IWC, and IWD Program) are consistent with the GALL Report for stainless steel components, as discussed in Item 24 below. Therefore, aging management of this component/commodity group is consistent with the GALL Report.

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
12. Westinghouse and B&W baffle former bolts	Crack initiation and growth due to SCC and IASCC	Plant specific	Yes, plant specific	The GALL Report recommends a plant-specific AMP that assures these aging effects/mechanisms are adequately managed. For cracking due to SCC and IASCC, the PWR Vessel Internals Program together with the Water Chemistry Program are applicable. RNP is committed to incorporate into the PWR Vessel Internals Program additional aging management activities resulting from ongoing industry initiatives that are determined applicable for managing this aging effect and mechanisms. RNP will continue to participate in industry programs whose objectives include the investigation of aging effects applicable to Baffle/Former Bolts and identification of appropriate AMP(s). New AMP activities, or other surveillance techniques, will be incorporated as enhancements to the aging management activities applicable to Baffle/Former Bolts. In addition, the Water Chemistry Program has proven effective in managing cracking from SCC in general as indicated in the GALL Report for various other reactor vessel internals components. Together, the programs applied at RNP to manage cracking in baffle former bolts, with appropriate enhancements to be identified in ongoing industry programs, will adequately manage these aging effects and are considered to be consistent with the GALL Report.

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
13. Westinghouse and B&W baffle former bolts	Loss of preload due to stress relaxation	Plant specific	Yes, plant specific	Stress relaxation is a result of creep and/or irradiation induced creep. The RNP AMR identified loss of pre-load from irradiation creep as an applicable aging effect/mechanism for the Baffle/Former Bolts. The GALL Report calls for a plant-specific program to manage the effects of loss of preload/stress relaxation. The ASME Section XI, Subsection IWB, IWC, and IWD Program and the PWR Vessel Internals Program will be used to manage loss of pre-load from irradiation creep at RNP. Aging management activities, or surveillance techniques, resulting from the ongoing industry programs will be incorporated, as required, to enhance the aging management activities applicable to Baffle/Former Bolts. RNP will continue to participate in industry programs whose objectives include the investigation of aging effects applicable to Baffle/Former Bolts and identification of appropriate AMP activities. Based on the planned activities, aging management of loss of preload in Baffle/Former Bolts is consistent with the GALL Report.
14. Steam generator feedwater impingement plate and support	Loss of section thickness due to erosion	Plant specific	Yes, plant specific	The component/commodity group is not applicable to RNP. These components are not part of the RNP steam generators. The RNP Steam Generators employ feed rings with J-nozzles. In addition, the feed rings/J-nozzles perform no license renewal intended function.

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
15. (Alloy 600) Steam generator tubes, repair sleeves, and plugs	Crack initiation and growth due to PWSCC, outside diameter stress corrosion cracking (ODSCC), and/or intergranular attack (IGA) or loss of material due to wastage and pitting corrosion, and fretting and wear; or deformation due to corrosion at tube support plate intersections	Steam generator tubing integrity; water chemistry	Yes, effectiveness of a proposed AMP is to be evaluated	Steam generator tube sleeves have not been installed at RNP. Loss of material due to wastage and pitting corrosion owing to exposure to phosphate chemistry is not applicable, as phosphate chemistry is not used at RNP. However, pitting remains a possible aging mechanism in accordance with the RNP AMR. In addition, Bulletin No. 88-02 has been determined to be not applicable to RNP based upon the steam generator design and support plate material. Per the GALL Report, the effectiveness of the AMP for managing degradation in steam generator tubes and plugs is contingent on implementing the guidelines of NEI 97-06 in conjunction with the Steam Generator Tubing Integrity Program and the Water Chemistry Program for the steam generators. For RNP, a combination of the Steam Generator Tubing Integrity and Water Chemistry Programs will be used for management of potential cracking, loss of section thickness, loss of material, and denting for steam generator tubes and plugs. Per the guidelines of NEI 97-06, RNP Technical Specifications, Section 5.5.9, provide the requirements for SG degradation management. These requirements, including tube inspection scope and frequency, plugging, repair, and leakage monitoring, have been incorporated into plant administrative controls. The programs and guidelines for aging management of steam generator tubes and plugs at RNP are consistent with the GALL Report.

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
16. Tube support lattice bars made of carbon steel	Loss of section thickness due to FAC	Plant specific	Yes, plant specific	The GALL Report indicates that this component/commodity group is applicable to Combustion Engineering steam generators. Therefore, it is not applicable to RNP. In accordance with the RNP AMR, FAC is applicable to the steam generator steam nozzle, feedwater nozzle, and feedwater nozzle thermal sleeve. Refer to Item 21 below for a discussion of steam generator components susceptible to FAC.
17. Carbon steel tube support plate	Ligament cracking due to corrosion	Plant specific	Yes, effectiveness of a proposed AMP is to be evaluated	The tube support plates in the RNP steam generators are fabricated of stainless steel, not carbon steel. The GALL Report is not specific regarding the type of corrosion involved for this component/commodity group. At RNP, the AMR for this component identified cracking from SCC and loss of material from crevice corrosion, pitting corrosion, and erosion as applicable aging effects/mechanisms. For RNP, these effects/mechanism are managed by a combination of the Steam Generator Tubing Integrity Program and the Water Chemistry Program applicable to steam generators. This is in agreement with the AMPs cited for this component in Item IV.D1.2-k of the GALL Report. The cited AMPs applied at RNP are consistent with the GALL Report.

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
18. Reactor vessel closure studs and stud assembly	Crack initiation and growth due to SCC and/or IGSCC	Reactor head closure studs	No	The GALL Report considers cracking due to SCC of the reactor vessel stud assembly as an applicable aging effect. Leakage of primary coolant water could provide the aggressive environment needed for SCC in the bolting materials. For quenched and tempered low alloy steels used for closure bolting such as alloy 4140 steels (e.g., SA 193 Grade B7) material susceptibility to SCC is controlled by its yield strength. EPRI Report NP-5769 indicates that SCC should not be a concern for closure bolting such as alloy 4140 steel in nuclear power plant applications if the specified minimum yield strength is below 150 ksi. The yield strength limit specified in EPRI NP-5769 for SCC of bolting is not for actual yield strength, but for "minimum specified yield strength." As long as a material's "minimum specified yield strength" is less than 150 ksi, it is within the bounds of NP-5769 with regard to non-susceptibility to SCC. The RNP stud assemblies are fabricated from A540, Grade B23 or B24. This material is well within the 150-ksi limit of minimum specified yield strength with a minimum specified yield strength of 100 ksi. Accordingly, cracking due to SCC is not an aging effect requiring management for the reactor vessel studs. Nevertheless, the RNP Reactor Head Closure Studs Program is applied to the closure studs to manage other aging effects. This program includes inservice inspection capable of detecting cracking due to SCC. Therefore, the management of cracking, although not an applicable aging effect, is consistent with the GALL Report.

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
19. CASS pump casing and valve body	Loss of fracture toughness due to thermal aging embrittlement	Inservice inspection	No	For loss of fracture toughness, the RNP AMR applied the ASME Section XI, Subsection IWB, IWC, and IWD Program to reactor coolant pump casings and valve bodies in the RCS. This is consistent with the GALL Report for aging management of this component/ commodity group. In accordance with the GALL Report, Section XI.M12, the existing ASME Section XI inspection requirements, including Code Case N-481 for pump casings, are adequate for all pump casings and valve bodies. For Class 1 valves in systems connected the RCS, RNP applied the Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program. This is also consistent with the GALL Report, because the Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program depends on inservice inspection activities for these valves by means of the ASME Section XI, Subsection IWB, IWC, and IWD Program. In the AMP evaluation of the Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program, it is concluded that valves and pump casings are adequately covered by existing inspection requirements in Section XI of the ASME Code. Therefore, aging management for this component/ commodity group is consistent with the GALL Report.

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
20. CASS piping	Loss of fracture toughness due to thermal aging embrittlement	Thermal aging embrittlement of CASS	No	The RNP CRDM housings and pressurizer surge line and nozzle are not CASS, and the pressurizer spray head does not perform an intended function. RNP applies the Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program to CASS piping. This program refers to ASME Section XI requirements as well as to an updated evaluation of thermal aging using Leak Before Break (LBB) methods covering the period of extended operation. Management of thermal aging embrittlement for CASS piping can be accomplished by means of an evaluation of LBB using thermal aging conditions projected to the end of the period of extended operation. A reevaluation of LBB is the approach used for RNP, and an updated LBB evaluation has been performed. The LBB analysis is summarized in Section 4.6.1. Aging management for this component/ commodity group is consistent with the GALL Report.
21. BWR piping and fittings; steam generator components	Wall thinning due to flow-accelerated corrosion	Flow-accelerated corrosion	No	For RNP, the components susceptible to FAC include the carbon steel feedwater nozzle thermal sleeve, which is a component not specifically identified in the GALL Report. For all steam generator components subject to FAC, this aging effect/mechanism is managed by the Flow-Accelerated Corrosion Program. This is consistent with the GALL Report.

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
22. Reactor coolant pressure boundary (RCPB) valve closure bolting, manway and holding bolting, and closure bolting in high pressure and high temperature systems	Loss of material due to wear; loss of preload due to stress relaxation; crack initiation and growth due to cyclic loading and/or SCC	Bolting integrity	No	The RNP CRDM housings have no RCPB-related bolting. SCC is not an applicable aging mechanism for bolting in this group based on the specified minimum yield strength of the bolting, which is below 150 ksi; refer to the discussion in Item 18. Also, the program credited with managing Loss of Mechanical Closure Integrity from Loss of Material due to Aggressive Chemical Attack for bolting exposed to boric acid leakage is the Boric Acid Corrosion Program. The Bolting Integrity Program is applicable to all RCPB bolting except reactor vessel studs for which the Reactor Head Closure Studs Program applies (see Item 18 above and Item 34 below). The Bolting Integrity Program relies on the ASME Section XI, Subsection IWB, IWC, and IWD Program to assure that aging effects associated with wear and stress relaxation are managed for RCS Class 1 closure bolting (and Class 2 bolting greater than 2-inches in diameter). Also, the Preventive Maintenance Program includes activities to manage loss of preload for the RCP closure bolting at RNP. Therefore, aging management of RCPB bolting is consistent with the GALL Report, with exceptions identified in the Appendix B description of the Bolting Integrity Program. The RNP AMR for Steam Generator primary and secondary closure bolts determined that the only (continued)

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
22. (continued)				applicable aging effects/ mechanisms requiring management are cracking from thermal fatigue and loss of mechanical closure integrity from loss of material due to aggressive chemical (boric acid) attack. Since these aging effects differ from those identified in the GALL Report, they are discussed in Table 3.1-2 Item 12.
23. CRD nozzle	Crack initiation and growth due to PWSCC	Ni-alloy Nozzles and Penetrations Program and Water Chemistry Program	No	The Nickel-Alloy Nozzles and Penetrations Program and the Water Chemistry Program are applicable. Aging management of this component/commodity group is consistent with the GALL Report.
24. Reactor vessel nozzles safe ends and CRD housing; reactor coolant system components (except CASS and bolting)	Crack initiation and growth due to cyclic loading, and/or SCC, and PWSCC	Inservice inspection; water chemistry	No	Management of these aging effects/mechanisms for the Reactor Pressure Vessel and Pressurizer is consistent with the GALL Report. The Pressurizer Relief Tank is carbon steel and is not included in this group. Refer to Item 6 above for a discussion of small bore piping with respect to water chemistry and inservice inspection. RNP applies only the Water Chemistry Program for RCS piping and connected system piping based on the fact that primary chemistry controls have proven to be effective in preventing SCC in this piping. Nevertheless, the ASME Section XI, Subsection IWB, IWC and IWD Program is applicable to all RCS components in this group that are not small bore. Based on the above discussion, aging management of this component/commodity group is considered to be consistent with the GALL Report.

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
25. Reactor vessel internals CASS components	Loss of fracture toughness due to thermal aging, neutron irradiation embrittlement, and void swelling	Thermal aging and neutron irradiation embrittlement	No	The RNP AMP applicable to void swelling is discussed in Item 8 above. This component/commodity group consists of the RNP lower support plate column, upper support tube base, and bottom mounted instrumentation column cruciform. The RNP lower support forging is not CASS, and the upper support column is not subject to irradiation embrittlement because of its location away from the active fuel zone. RNP applies the PWR Vessel Internals Program to manage these aging effects/mechanisms. Since this is a difference from the GALL Report recommended program, the discussion is contained in Table 3.1-2, Item 14.

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
26. External surfaces of carbon steel components in reactor coolant system pressure boundary	Loss of material due to boric acid corrosion	Boric acid corrosion	No	The SRP-LR applies this AMP to components within the RCPB; the GALL Report applies it to components within the RCPB and supports. However, for all RCS components in scope for license renewal at RNP, the Boric Acid Corrosion Program is relied on to manage this aging effect/mechanism. There is no difference for external surfaces of components outside the RCPB. This is consistent with the GALL Report approach for carbon steel supports and components in systems other than the RCS. Component supports are addressed in Section 3.5. A visual inspection of the RCS, including the pressurizer, is performed periodically in accordance with the ASME Section XI, Subsection IWB, IWC, and IWD Program during pressure testing to detect evidence of leakage. This inspection would detect leakage that could cause boric acid corrosion degradation. Therefore, the ASME Inservice Inspection, as well as the Boric Acid Corrosion Program, is credited for managing boric acid corrosion of the Pressurizer. Aging management of this component/commodity group is consistent with the GALL Report.
27. Steam generator secondary manways and handholds (CS)	Loss of material due to erosion	Inservice inspection	No	The GALL Report indicates this item is applicable to Once-Through Steam Generators and, therefore, is not applicable to RNP. The aging effects applicable to RNP secondary manways and handholes do not include loss of material due to erosion.

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
28. Reactor internals, reactor vessel closure studs, and core support pads	Loss of material due to wear	Inservice inspection	No	Except for flux thimbles and reactor vessel studs, the ASME Section XI, Subsection IWB, IWC, and IWD Program is the applicable RNP program for this component/commodity group. This is consistent with the GALL Report. For the flux thimbles, the applicable RNP program is the Flux Thimble Eddy Current Inspection Program. The Program was implemented to satisfy NRC Bulletin 88-09 requirements that a tube wear inspection procedure be established and maintained for Westinghouse supplied reactors which use bottom mounted flux thimble tube instrumentation. This AMP is effective at managing the effects of wear of the thimble tubes and employs acceptance criteria based on ASME Code requirements. This Program is considered to be consistent with GALL item IV.B2.6-c, which discusses the AMP developed in response to IEB 88-09. For the vessel stud assembly, wear is managed by the Reactor Head Closure Studs Program. This is discussed in Component/Commodity Group 34 below. This is consistent with the GALL Report for the closure stud assembly, item IV.A2.1-d.

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
29. Pressurizer integral support	Crack initiation and growth due to cyclic loading	Inservice inspection	No	The RNP AMR of the Pressurizer identified the following aging effects/mechanisms applicable to the integral support: cracking from thermal fatigue (a TLAA) and loss of material from aggressive chemical attack (boric acid corrosion). However, the GALL Report identifies an additional aging effect of cracking initiation and growth due to cyclic loading. The RNP AMR recognized that the external surfaces of the Pressurizer Integral Support are covered by the ASME Section XI, Subsection IWB, IWC, and IWD Program and that the ASME Section XI, Subsection IWB, IWC, and IWD Program would identify the presence of cracks that may result from cyclic loading. Therefore, the postulated aging effect/mechanism in this component/commodity group would be acceptably managed consistent with the GALL Report.
30. Upper and lower internals assembly (Westinghouse)	Loss of preload due to stress relaxation	Inservice inspection; loose part and/or neutron noise monitoring	No	The GALL Report cites a combination of ASME Section XI and loose part and/or neutron noise monitoring programs. The applicable components identified in the GALL Report are the upper internals hold-down spring and lower internal assembly clevis insert bolts. RNP employs the ASME Section XI, Subsection IWB, IWC, and IWD and PWR Vessel Internals Programs to address stress relaxation for these components. Since this differs with the GALL Report, the discussion has been included in Item 15 of Table 3.1-2.

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
31. Reactor vessel internals in fuel zone region (except Westinghouse and Babcock & Wilcox [B&W] baffle bolts)	Loss of fracture toughness due to neutron irradiation embrittlement, and void swelling	PWR vessel internals; water chemistry	No	The information provided in Volume 2 of the GALL Report applied only the PWR vessel internals AMP and did not include water chemistry to manage this aging effect. The RNP evaluation determined some of the components listed in the GALL Report are not susceptible to this aging effect owing to their location away from the fuel zone region. To manage the effects of loss of fracture toughness due to neutron irradiation embrittlement and change in dimensions due to void swelling, RNP applied the PWR Vessel Internals Program. However, even those components that were determined to be located away from the fuel zone region have, at least, the RNP PWR Vessel Internals Program applied; and, of course, the Water Chemistry Program applies to the reactor vessel internals treated water environment. Therefore, management of this aging effect/mechanism is consistent with the GALL Report.

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
32. Steam generator upper and lower heads; tubesheets; primary nozzles and safe ends	Crack initiation and growth due to SCC, PWSCC, IASCC	Inservice inspection; water chemistry	No	The only component in this group from Volume 2 of the GALL Report applicable to RNP is item D1.1.9, primary nozzles and safe ends. The other GALL components are not applicable to Westinghouse steam generators. The Steam Generator lower head cladding is stainless steel and has been included in this component/ commodity group. The programs applied at RNP for primary nozzles and safe ends and Steam Generator lower head clad are the ASME Section XI, Subsection IWB, IWC, and IWD and Water Chemistry Programs. These are consistent with the GALL Report. RNP also has added the Steam Generator Primary Manway Insert to this component commodity group. The insert is fabricated of stainless steel but has no pressure boundary function; nor does it provide structural support to or maintain structural integrity of pressure boundary components. To manage possible cracking from SCC, RNP has applied the Water Chemistry Program. The Water Chemistry Program alone is considered to be acceptable, because the component has no pressure boundary-related function and, therefore, use of the inservice inspection program is not necessary.

TABLE 3.1-1 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
33. Vessel internals (except Westing-house and B&W baffle former bolts)	Crack initiation and growth due to SCC and IASCC	PWR vessel internals; water chemistry	No	This group includes additional RNP vessel internals components not identified in the GALL Report, but excludes RCCA Guide Tube Support Pins, which perform no intended functions for license renewal. The additional components are BMI columns; BMI column cruciforms; diffuser plate; head and vessel alignment pins; head cooling spray nozzles; secondary core support; and upper instrument column, conduit and supports. RNP applies the PWR Vessel Internals Program and the Water Chemistry Program. This is consistent with the GALL Report.
34. Reactor vessel closure studs and stud assembly	Loss of material due to wear	Reactor head closure studs	No	The RNP Reactor Head Closure Studs Program is applicable; this is consistent with the GALL Report.
35. Reactor internals (Westinghouse upper and lower internal assemblies; CE bolts and tie rods)	Loss of preload due to stress relaxation	Inservice inspection; loose part monitoring	No	The GALL Report cites a combination of inservice inspection and loose part monitoring programs. RNP considers the recommendations regarding loose part monitoring to be ineffective for the management of aging effects. Since this differs with the recommendations of the GALL Report, the discussion has been included in Item 15 of Table 3.1-2.

Notes: 1. Numbered Component/Commodity Groups consist of the components listed in NRC Standard Review Plan for License Renewal, NUREG-1800, Table 3.1-1, that are applicable to a PWR.
2. RNP Reactor Coolant System includes non-Class 1 components outside the Reactor Coolant Pressure Boundary.

TABLE 3.1-2 REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
1. Reactor Vessel Stud Assembly	Carbon Steel	Containment Air, Borated Water Leakage	Loss of Pre-Load from Stress Relaxation	Reactor Head Closure Studs Program	This aging effect was not identified in the GALL Report for the stud assembly. The RNP-assigned AMP would assure the effects of loss of preload were managed because the purpose of this program is to assure the continued performance of the intended function of the reactor head closure stud assembly.

TABLE 3.1-2 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
2. RCS Components (cladding, control rod drive housings, reactor vessel and pressurizer nozzle safe ends, core support pads, flux thimbles and guide tubes, pressurizer heaters, penetrations, seal table valves and fittings, valves, piping, tubing, fittings, SG divider plate, Pressurizer and SG primary manway insert, SG tubeplate cladding)	Stainless Steel, Nickel-based Alloy	Treated Water (including steam)	Loss of Material from Crevice Corrosion or Pitting Corrosion	Water Chemistry Program	These aging mechanisms are not specified for these components in the GALL Report. Except for cladding, RNP has applied the Water Chemistry Program alone. This is considered acceptable because, as discussed in the GALL Report, Chapter V, Section D.1, discussion of Systems, Structures, and Components, stainless steel is not subject to significant general, pitting, and crevice corrosion in borated water. Also, the hydrogen concentration established for the RCS ensures that corrosion is non-significant for the internal surfaces of the RNP pressurizer as well as other Class 1 components. Hydrogen concentration limits for the RCS are delineated in the Water Chemistry Program. Therefore, the proposed AMP is considered to be consistent with the GALL Report. For cladding in the lower head of the Steam Generator, the ASME Section XI, Subsection IWB, IWC, and IWD Program has been credited together with Water Chemistry Program. However, as discussed above, the Water Chemistry Program alone is sufficient to manage crevice and pitting corrosion.

TABLE 3.1-2 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
3. Steam Generator Anti-Vibration Bars	Nickel-based Alloy	Treated Water (including steam)	Cracking from SCC	Steam Generator Tube Integrity Program	This component is not specified in the GALL Report. The Water Chemistry Program is effective in preventing SCC and crevice corrosion, because it controls the aggressive chemical species required to promote these aging mechanisms. The Steam Generator Tube Integrity Program includes additional activities, such as secondary side inspections, beyond those dealing with tube integrity and will adequately manage the effects of cracking and loss of material. Also, the combination of Steam Generator Tube Integrity and Water Chemistry Programs is invoked by the GALL Report for management of loss of material from fretting. Refer to GALL Report Section IV.D1.2-e, for managing fretting and wear of steam generator tubes and sleeves.
				Water Chemistry Program	
			Loss of Material from Crevice Corrosion	Steam Generator Tube Integrity Program	
				Water Chemistry Program	
			Loss of Material Fretting	Steam Generator Tube Integrity Program	
				Water Chemistry Program	

TABLE 3.1-2 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
4. Steam Generator Components (feedwater nozzle thermal sleeve safe end, steam flow limiter)	Nickel-based Alloy	Treated Water (including steam)	Cracking from SCC	Water Chemistry Program	These Steam Generator secondary components are not specified in the GALL Report. The Water Chemistry Program has been proven effective in managing SCC and pitting and crevice corrosion, because it controls the aggressive chemical species required to promote these aging mechanisms.
			Loss of Material from Crevice, or Pitting Corrosion	Water Chemistry Program	
5. Steam Generator Components (feedwater nozzle thermal sleeve, secondary side manway and handhole covers, secondary side shell penetrations, tube bundle wrapper, tubeplate)	Carbon Steel	Treated Water (including steam)	Loss of Material from Crevice, General, or Pitting Corrosion	Water Chemistry Program	These Steam Generator secondary components are not specified in the GALL Report. The Water Chemistry Program has been proven effective in managing pitting and crevice corrosion, because it controls the aggressive chemical species required to promote these aging mechanisms.

TABLE 3.1-2 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
6. Steam Generator Components (Tube Bundle Wrapper, Tubeplate, Steam Flow Limiter)	Carbon Steel and Nickel-based Alloy	Treated Water (including steam)	Loss of Material from Erosion	Water Chemistry Program	The RNP AMR conservatively identified loss of material from erosion as an aging effect/mechanism for these components, which are not addressed in the GALL Report. During normal operation, the RNP Water Chemistry Program maintains strict controls on suspended solids in the feedwater system; this provides assurance that erosion will be managed. In addition, the Steam Flow Limiter is fabricated of Inconel; and, therefore, is highly resistant to loss of material from erosion.
7. Steam Generator Snubber Reservoir Components	Various piping components	Containment Air, Borated Water Leakage	Changes in Material Properties from Various Degradation Mechanisms	Preventive Maintenance Program (Plant Specific)	These components are not addressed in the GALL Report. The Preventive Maintenance Program continues to be effective in managing aging from various degradation mechanisms for these components.
			Cracking from Various Degradation mechanisms	Preventive Maintenance Program (Plant Specific)	
			Loss of Material from Various Degradation Mechanisms	Preventive Maintenance Program (Plant Specific)	

TABLE 3.1-2 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
8. Non-Class 1 Valves, Piping and Fittings	Carbon Steel	Air and Gas	See discussion	None Required	This component/commodity group consists of valves, piping, and fittings associated with piping connected to the Pressurizer Relief Tank. The Pressure Relief Tank is provided with a blanket of nitrogen gas; therefore, these components are subject to a dry, inert environment on their internal surfaces. The RNP AMR methodology determined that these valves, piping, and fittings have no aging effects resulting from this environment.
9. Reactor Vessel Components: (Core Support Pads)	Nickel-based Alloy	Treated Water (including steam)	Cracking from SCC	Water Chemistry Program	The RNP Core Support Pads are fabricated of Nickel-based Alloy. The applicable AMP for managing crack initiation and growth for support pads is the Water Chemistry Program. The Water Chemistry Program has been proven effective in managing SCC and other types of corrosion (e.g., pitting and crevice corrosion), because it controls the aggressive chemical species required to promote these aging mechanisms.

TABLE 3.1-2 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
10. Reactor Vessel Components: Penetrations Instrument Tubes (Bottom Head)	Nickel-based Alloy	Treated Water (including steam)	Cracking from SCC	Water Chemistry Program and ASME Section XI, Subsection IWB, IWC and IWD Program	The RNP Reactor Vessel Bottom Head Penetrations are fabricated of Nickel-based Alloy. The applicable AMPs for managing crack initiation and growth for bottom head penetrations are the ASME Section XI, Subsections IWB, IWC, and IWD Program and the Water Chemistry Program. The Water Chemistry Program has been proven effective in managing SCC, because it controls the aggressive chemical species required to promote this aging mechanism. And the ASME Section XI, Subsection IWB, IWC and IWD Program has been proven effective in detecting cracking in the pressure boundary of RCS components.
11. Steam Generator Lower Head Divider Plate, Steam Generator Tubeplate Cladding	Nickel-based Alloy	Treated Water (including steam)	Cracking from SCC	Water Chemistry Program	The RNP SG Lower Head Divider Plate and tubesheet cladding are fabricated of Nickel-based Alloy. The applicable AMP for managing crack initiation and growth for the divider plate and tubesheet cladding is the Water Chemistry Program. The Water Chemistry Program has been proven effective in managing SCC, because it controls the aggressive chemical species required to promote this aging mechanism.

TABLE 3.1-2 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
12. Secondary Manway and Handhole Bolting	Carbon Steel	Containment Air, Borated Water Leakage	Loss of Mechanical Closure Integrity from Loss of Material due to Aggressive Chemical Attack	Boric Acid Corrosion Program	The RNP AMR for Steam Generator primary and secondary closure bolts determined that the only applicable aging effects/ mechanisms requiring management are cracking from thermal fatigue and loss of mechanical closure integrity from loss of material due to aggressive chemical (boric acid) attack. SCC is not an applicable aging mechanism for bolting in this group based on the specified minimum yield strength of the bolting, which is below 150 ksi. Thermal fatigue is a TLAA and is addressed in Section 4.3, and the Boric Acid Corrosion Program manages boric acid wastage on these bolts.

TABLE 3.1-2 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
13. Pressurizer Relief Tank	Carbon Steel	Treated Water (including steam)	Loss of Material due to Aggressive Chemical Attack	Preventive Maintenance Program (Plant Specific)	The GALL Report does not address these aging effects/mechanisms for the internal surfaces of the Pressurizer Relief Tank. The RNP Pressurizer Relief Tank is a carbon steel tank with an internal corrosion resistant coating (lining). The Preventive Maintenance Program is applied to assure that degradation of the internal surfaces of the Pressurizer Relief Tank is identified and corrected, if required. The Preventive Maintenance Program activities include periodic visual inspections to monitor the condition of the internal surfaces of various components throughout the plant, including the Pressurizer Relief Tank.
			Loss of Material from Crevice Corrosion	Preventive Maintenance Program (Plant Specific)	
			Loss of Material from General Corrosion	Preventive Maintenance Program (Plant Specific)	
			Loss of Material from Pitting Corrosion	Preventive Maintenance Program (Plant Specific)	
14. Reactor Vessel Internals CASS components	Cast Austenitic Stainless Steel	Treated Water (including steam)	Reduction of Fracture Toughness from Thermal Embrittlement and Neutron Irradiation Embrittlement	PWR Vessel Internals Program	RNP applies the PWR Vessel Internals Program to manage thermal aging embrittlement of CASS components. As discussed previously, RNP will incorporate the applicable results of industry initiatives related to aging effects for reactor vessel internals into the PWR Vessel Internals Program. This includes information regarding thermal embrittlement and neutron irradiation embrittlement. The PWR Vessel Internals Program used at RNP will effectively manage the effects of loss of fracture toughness due to thermal aging and neutron irradiation embrittlement for CASS reactor internals components.

TABLE 3.1-2 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
15. Reactor Internals: Upper Support Column Bolts, Holddown Spring, Lower Support Plate Column Bolts, and Clevis Insert Bolts	Stainless Steel and Nickel-based Alloy	Treated Water (including steam)	Loss of Preload due to Stress Relaxation	ASME Section XI, Subsections IWB, IWC and IWD Program and PWR Vessel Internals Program	The GALL Report cites (1) a combination of ASME Section XI, Inservice Inspection and loose parts and/or neutron noise monitoring programs for the holddown spring and clevis insert bolts, and (2) a combination of ASME Section XI, Inservice Inspection, and loose parts monitoring for upper support column bolts and lower support plate column bolts. RNP employs both the ASME Section XI, Subsection IWB, IWC, and IWD Program and the PWR Vessel Internals Program to address stress relaxation for these components. RNP considers the recommendations regarding neutron or noise monitoring to be ineffective to the management aging effects. By the time neutron or noise monitoring indicate a concern, the aging degradation would have reached an unacceptable condition. As discussed previously, RNP will incorporate the applicable results of industry initiatives related to aging effects for reactor vessel internals into the PWR Vessel Internals Program. This includes information on loss of preload due to stress relaxation. The AMPs used at RNP will effectively manage the effects of loss of loss of preload for affected internals components.

TABLE 3.1-2 (continued) REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
16. Flux Thimbles	Nickel-based Alloy	Treated Water (including steam)	Cracking from SCC	Water Chemistry Program	The GALL Report does not identify this aging effect/mechanism combination for the flux thimbles. The RNP evaluation applied the Water Chemistry Program for managing the effects of SCC. The Water Chemistry Program has been proven effective in managing SCC and other types of corrosion (e.g., pitting and crevice corrosion), because it controls the aggressive chemical species required to promote this aging mechanism.
17. Seal Table Valves and Fittings; Valves, Piping, Tubing and Fittings; Flow Orifices/ Elements	Stainless Steel	Indoor – Not Air Conditioned, Containment Air, Borated Water Leakage	None	None Required	The RNP AMR determined that these components have no potentially significant aging effects requiring management in these environments. Boric acid is not an aggressive chemical species for stainless steel.
18. Valves, Piping, Tubing and Fittings (Non-Class 1 Reactor Vessel Level Instrument)	Stainless Steel	Treated Water (including steam)	None	None Required	The RNP AMR determined that these non-Class 1 components in the Reactor Vessel Level Instrumentation System would have no aging effects requiring management in the treated water environment, because they are isolated from the RCS in a closed portion of the system that is filled with purified, deionized water.

Note: 1. Environments used in the aging management review are listed on Tables 3.0-1 and 3.0-2.

3.2 AGING MANAGEMENT OF ENGINEERED SAFETY FEATURES

Engineered Safety Features Systems 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.

3.2.1 AGING MANAGEMENT REVIEW

3.2.1.1 Methodology

Aging management review of Engineered Safety Features System components and commodities involved consideration and evaluation of the materials, environments, and stressors that are associated with each structure, component, or commodity grouping under review, as discussed in Section 4.2 of NEI 95-10 [Reference 3.2-1]. The RNP AMR methodology follows the approach recommended in NEI 95-10 and is based on generic industry guidance for determining aging effects for both mechanical and civil/structural components. The guidance represents a set of rules that allow the evaluator to identify aging effects for a given material and environment combination. The material and environment-based rules in the generic industry guidance documents are derived from known age-related degradation mechanisms and industry operating experience. The guidance was reviewed for applicability to RNP materials of construction and component internal and external operating environments and was used to identify aging effects for components, structures, and commodities. The results of the evaluation of materials and environment combinations, using the RNP methodology, are aging effects; and, if the aging effects adversely affect intended functions, the results are aging effects requiring management for the applicable components and commodities. Aging effects that require management are correlated to aging management programs.

The aging management review identifies one or more aging management programs to be used to demonstrate that the effects of aging will be managed so that the intended functions will be maintained consistent with CLB for the period of extended operation. The programs to be used for managing the effects of aging were compared to those listed in the GALL Report [Reference 3.2-2] and evaluated for consistency with GALL Report programs that are relied on for license renewal. The results are documented and discussed in Subsection 3.2.2 using the format suggested by the NRC Standard Review Plan for License Renewal (SRP-LR) [Reference 3.2-3].

Aging management review results for the Containment Air Recirculation Cooling System are presented in Section 3.3, “Aging Management of Auxiliary Systems,” to conform with the format of the GALL Report. The GALL Report addresses Containment HVAC equipment with Auxiliary Systems.

3.2.1.2 Operating Experience

Operating experience (OE) through December 2001 was considered during the development of the RNP Integrated Plant Assessment. OE subsequent to that date will be reviewed and applicable OE will be updated in conjunction with the amendment to the application required by 10 CFR 54.21(b). The review consisted of the following:

Site: RNP site-specific operating experience was reviewed. The site-specific operating experience included a review of (1) Corrective Action Program, (2) Licensee Event Reports, (3) Maintenance Rule Data Base, and (4) interviews with Systems Engineers. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Subsection.

Industry: An evaluation of industry operating experience published since the effective date of the GALL Report was performed to identify any additional aging effects requiring management. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Subsection.

On-Going On-going review of plant-specific and industry operating experience is performed in accordance with the Corrective Action and Operating Experience Programs.

3.2.2 AGING MANAGEMENT PROGRAMS

3.2.2.1 Aging Management Programs Evaluated in the GALL Report that Are Relied on for License Renewal

Table 3.2-1 shows the component and commodity groups (combinations of materials and environments) and aging management programs evaluated in the GALL Report that are relied on for license renewal of the Engineered Safety Features Systems. The table is based on Table 3.2-1 of the SRP-LR [Reference 3.2-3] and provides a discussion of the applicability of the component commodity group and details regarding the degree to which RNP aging management programs are consistent with those recommended in the GALL Report. The discussion section includes (1) information regarding the applicability of the GALL Report component/commodity group to RNP, (2) any issues recommended in the GALL Report that require further evaluation, (3) details regarding RNP components to be included in the component/commodity group, and (4) a conclusion regarding consistency of the aging management review with the GALL Report.

3.2.2.2 Further Evaluation of Aging Management as Recommended by the GALL Report

Further evaluation of aging management as recommended by the GALL Report has been incorporated into the "Discussion" column of Table 3.2-1.

3.2.2.3 Aging Management Evaluations that Are Different from or Not Addressed in the GALL Report

Aging Management Evaluations that are different from or not addressed in the GALL Report are identified and discussed on Table 3.2-2.

3.2.3 CONCLUSIONS

Engineered Safety Features Systems aging effects requiring management are adequately managed by the following programs:

1. Boric Acid Corrosion Program
2. Closed-Cycle Cooling Water System Program
3. Fatigue Monitoring Program
4. Open-Cycle Cooling Water System Program
5. Preventive Maintenance Program
6. Selective Leaching of Materials Program
7. Systems Monitoring Program
8. Water Chemistry Program

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that the intended functions of Engineered Safety Features Systems components are maintained consistent with the current licensing basis for the period of extended operation.

3.2.4 REFERENCES

3.2-1 NEI 95-10, “Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule,” Revision 3, Nuclear Energy Institute, March 2001.

3.2-2 NUREG-1801, “Generic Aging Lessons Learned (GALL),” U. S. Nuclear Regulatory Commission, April 2001.

3.2-3 NUREG-1800, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” U. S. Nuclear Regulatory Commission, April 2001.

TABLE 3.2-1 ENGINEERED SAFETY FEATURES SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
1. Piping, fittings, and valves in emergency core cooling system	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	RHR pumps, heat exchanger tubing, and flow orifices have been included in this group. Evaluation of this component/commodity group is consistent with the GALL Report. Refer to Section 4.3 for the TLAA evaluations associated with metal fatigue.
2. Components in containment spray (PWR only), standby gas treatment (BWR only), containment isolation, and emergency core cooling systems (For PWRs, GALL limits this group to containment spray and containment isolation components. SRP-LR, Section 3.2.2.2.2.2, includes in this group the external surfaces of all ESF systems. External surfaces of ESF systems are addressed in Item 6 below.)	Loss of material due to general corrosion	Plant specific	Yes, plant specific	In accordance with the GALL Report, this aging effect/mechanism is applicable to internal and external surfaces of carbon and low alloy steel containment spray and containment isolation components. The RNP containment spray headers and valves are stainless steel. Therefore, this evaluation is limited to containment isolation components. The RNP AMR methodology assumed that external surfaces of carbon steel components would not be susceptible to corrosion if they were located in areas protected from the weather, were not subjected to condensation, and were not subjected to aggressive chemical attack (e.g., borated water leakage). Loss of material due to aggressive chemical attack is addressed in Item 11 below. Using the above criterion, RNP equipment in this component/commodity group is not subject to general corrosion.

TABLE 3.2-1 (continued) ENGINEERED SAFETY FEATURES SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
3. Components in containment spray (PWR only), standby gas treatment (BWR only), containment isolation, and emergency core cooling systems (For PWRs, GALL limits this group to containment isolation components and the RWST. SRP-LR, Section 3.2.2.2.3.2, includes in this group the containment spray system.)	Loss of material due to pitting and crevice corrosion	Plant specific	Yes, plant specific	In accordance with the GALL Report, this aging effect/mechanism is applicable to bottom surfaces of refueling water tanks and internal and external surfaces of containment isolation components. The RNP Refueling Water Storage Tank is fabricated of stainless steel. Pitting and crevice corrosion are not a credible aging mechanisms for the exterior bottom of the tank, because (1) the tank location is well above the groundwater elevation, (2) the area around the tank is well drained, and (3) the tank bottom sits on a layer of oiled sand. Loss of material from pitting and crevice corrosion was identified, in the RNP AMR, as an aging effect for stainless steel components in raw water associated with containment penetrations. For these components, the plant-specific Preventive Maintenance Program has been applied to manage loss of material from crevice and pitting corrosion. Aging management of the containment penetration components in this component/commodity group is consistent with the GALL Report.

TABLE 3.2-1 (continued) ENGINEERED SAFETY FEATURES SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
4. Containment isolation valves and associated piping	Loss of material due to microbiologically influenced corrosion	Plant specific	Yes, plant specific	In accordance with the GALL Report, this aging effect/mechanism is applicable only to containment isolation components exposed to a source of MIC. Applicable RNP components are containment penetration components in the Liquid Waste Processing and Isolation Valve Seal Water Systems conservatively assumed to be subjected to MIC. The RNP Preventive Maintenance Program, which is a plant-specific program, is used to manage this aging effect/mechanism for the affected components. Therefore, management of this aging effect and mechanism is consistent with the GALL Report.
5. High pressure safety injection (charging) pump miniflow orifice	Loss of material due to erosion	Plant specific	Yes, plant specific	This component/commodity group is not applicable. The RNP design does not include high head SI pumps. Charging is performed by positive displacement pumps in the Chemical and Volume Control System.

TABLE 3.2-1 (continued) ENGINEERED SAFETY FEATURES SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
6. External surface of carbon steel components (Note that this component/ commodity group is from the GALL Report. There is no corresponding group in SRP-LR Table 3.2-1.)	Loss of material due to general corrosion	Plant specific	Yes, plant specific	This discussion is applicable to the external surfaces of carbon and low alloy steel components per GALL Section V.E.1-b. The RNP AMR methodology assumed that external surfaces of carbon steel components would not be susceptible to corrosion if they were located in areas protected from the weather, were not subjected to condensation, and were not subjected to aggressive chemical attack (e.g., borated water leakage). The RNP AMR determined that carbon steel components in the Containment Spray System may be subject to corrosion due to aggressive chemical attack (leakage of NaOH). Also, carbon steel components may be subject to corrosion due to aggressive chemical attack (leakage of boric acid solution). Refer to Item 11 below for a discussion of the Boric Acid Corrosion Program. Under these conditions, the plant-specific Systems Monitoring Program (either by itself or together with the Boric Acid Corrosion Program) is used to manage the effects of aggressive chemical attack. This is consistent with the plant-specific program called for in GALL Report, Section VE.1-b, for corrosion of external surfaces of carbon steel components.
7. Piping and fittings of CASS in emergency core cooling system	Loss of fracture toughness due to thermal aging embrittlement	Thermal aging embrittlement of CASS	No	Portions of the ESF Systems that operate at sufficient temperature to effect thermal aging embrittlement of CASS components have been evaluated with the Class 1 RCS piping in Section 3.1.

TABLE 3.2-1 (continued) ENGINEERED SAFETY FEATURES SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
8. Components serviced by open-cycle cooling system	Local loss of material due to corrosion and/or buildup of deposit due to biofouling	Open-cycle cooling water system	No	According to the GALL Report, this group consists of heat exchangers cooled by an open cycle cooling water system. RNP does not have a heat exchanger that cools the containment spray to the containment. RNP applies the Open-Cycle Cooling Water System Program. Therefore, management of the aging effects of loss of material and buildup of deposit for this component/commodity group is consistent with the GALL Report for RNP heat exchangers cooled by the Service Water System. The RNP AMR determined that loss of material from galvanic corrosion and selective leaching also were applicable to certain components of heat exchangers cooled by the Service Water System. These aging mechanisms are addressed in Table 3.2-2, Item 4, as differences with the GALL Report.
9. Components serviced by closed-cycle cooling system	Loss of material due to general, pitting, and crevice corrosion	Closed-cycle cooling water system	No	RNP applies the Closed-Cycle Cooling Water System Program. Management of loss of material from general, pitting, and crevice corrosion for this component/ commodity group is consistent with the GALL Report. The RNP AMR determined that loss of material from galvanic corrosion and selective leaching and cracking from SCC also were applicable to certain components cooled by the Component Cooling Water System. These aging effects/mechanisms are discussed on Table 3.2-2, Items 5, 6, and 7.

TABLE 3.2-1 (continued) ENGINEERED SAFETY FEATURES SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
10. Pumps, valves, piping, and fittings in containment spray and emergency core cooling systems	Crack initiation and growth due to SCC	Water chemistry	No	SCC requires a combination of a susceptible material, a corrosive environment, and tensile stress. The minimum level of stress required for SCC is dependent not only on the material but also on temperature and the environment. For austenitic stainless steels in treated water, the relevant conditions required for SCC are the presence of oxygen in excess of 100 ppb, chlorides or fluorides in excess of 150 ppb, or sulfates in excess of 100 ppb, and elevated temperature. The generic industry guidance used to identify aging effects based on materials and environments established a minimum threshold value for stainless steel of 200 °F. The GALL Report does not identify a temperature threshold for this mechanism. At RNP, a temperature criterion of greater than 140°F is used as the threshold for susceptibility of austenitic stainless steels and nickel based alloys to SCC. Components of the RNP RHR system are susceptible to SCC, and the Water Chemistry Program is used to manage the effects. Therefore, aging management of this component/commodity group is consistent with the GALL Report.

TABLE 3.2-1 (continued) ENGINEERED SAFETY FEATURES SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
11. Carbon steel components	Loss of material due to boric acid corrosion	Boric acid corrosion	No	Application of the RNP Boric Acid Corrosion Program for carbon steel components is consistent with the GALL Report. In the RNP AMR, loss of material due to boric acid corrosion of closure bolting can lead to “loss of mechanical closure integrity from loss of material due to aggressive chemical attack.” This aging effect/ mechanism, while different than specified in the GALL Report, is considered to be consistent with the GALL Report; because it results from a loss of material due to boric acid corrosion.
12. Closure bolting in high pressure or high temperature systems	Loss of material due to general corrosion, loss of preload due to stress relaxation, and crack initiation and growth due to cyclic loading or SCC	Bolting integrity	No	Bolting in RNP ESF Systems is evaluated for loss of material due to aggressive chemical species (boric acid corrosion) and for SCC based on specified minimum yield strength. There are no bolts with specified minimum yield strength > 150ksi in the ESF Systems, and the Boric Acid Corrosion Program is used to assure that loss of material from boric acid corrosion is detected and managed. Therefore, the Bolting Integrity Program is not applicable to bolting for the RNP ESF Systems.

Note: 1. Numbered Component/Commodity Groups consist of the components listed in NRC Standard Review Plan for License Renewal, NUREG-1800, Table 3.2-1, that are applicable to a PWR.

TABLE 3.2-2 ENGINEERED SAFETY FEATURES SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
1. Boron Injection Tank; Eductors; Flow Orifices/ Elements; RWST; Pumps: SI, RHR, and CV Spray; Accumulators; SI Filters; Spray Additive Tank; Heat Exchanger Tubing: RHR, SI and CV Pump and RHR Heat Exchangers; Valves, Piping, Tubing and Fittings	Stainless Steel	Treated Water (including Steam)	Loss of Material from Crevice Corrosion	Water Chemistry Program	Except for valves, piping, and fittings in the IVSW System, these components have an internal environment of treated, borated water. The IVSW components contain treated, demineralized water. RNP has applied the Water Chemistry Program to manage crevice and pitting corrosion. As discussed in the GALL Report, Chapter V, Section D.1, discussion of Systems, Structures, and Components, stainless steel is not subject to significant general, pitting, and crevice corrosion in borated water. Also, the Water Chemistry Program controls chemical species that would promote crevice and pitting corrosion, i.e., chlorides, fluorides, sulfates, and dissolved oxygen in treated, demineralized water. In addition, RNP plant-specific operating experience supports the conclusion that crevice and pitting corrosion are not occurring in these systems.
			Loss of Material from Pitting Corrosion	Water Chemistry Program	
2. Heat Exchanger Tubing: CV Spray Pump, RHR pump, and SI Pump Seal Coolers and RHR Heat Exchangers	Stainless Steel	Treated Water (including Steam)	Loss of Heat Transfer Effectiveness from Fouling of Heat Transfer Surfaces	Water Chemistry Program	This mechanism is not addressed in the GALL Report. The RNP Water Chemistry Program assures the heat transfer effectiveness on the borated water side of the heat exchanger tubes. The Closed-Cycle Cooling Water System Program is applicable to the shell-side of the heat exchanger tubes.

TABLE 3.2-2 (continued) ENGINEERED SAFETY FEATURES SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
3. Valves, Tubing and Fittings	Aluminum	Indoor – Not Air Conditioned, Containment Air, Borated Water Leakage	Loss of Material from Aggressive Chemical Attack	Boric Acid Corrosion Program	The GALL Report does not address the aging effects/mechanisms associated with aluminum components subjected to borated water leakage. The RNP AMR methodology notes that aluminum alloys subjected to aggressive chemical species are subject to loss of material. The Boric Acid Corrosion Program provides for visual inspection of components subject to borated water leakage. Therefore, this program provides assurance that the aluminum components would maintain their intended function throughout the period of extended operation.
			Loss of Material from Crevice Corrosion	Boric Acid Corrosion Program	
			Loss of Material from Pitting Corrosion	Boric Acid Corrosion Program	
4. Safety Injection Pump Outboard Bearing Heat Exchanger Shell	Carbon Steel	Raw Water	Loss of Material from Galvanic Corrosion	Open-Cycle Cooling Water System Program	The Open-Cycle Cooling Water System Program is applied to manage galvanic corrosion, while the Selective Leaching of Materials Program is used to manage the effects of selective leaching. Applying these two AMPs is consistent with the management of loss of material for heat exchangers cooled by an open-cycle cooling system as discussed in GALL Report, Section VII.C1.3-a. For RNP, the management of effects of galvanic corrosion and selective leaching is consistent with the GALL Report with an exception related to the Selected Leaching of Materials Program discussed in Appendix B.
			Loss of Material from Selective Leaching	Selective Leaching of Materials Program	

TABLE 3.2-2 (continued) ENGINEERED SAFETY FEATURES SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
5. RHR Heat Exchanger Shell and Cover, RHR Pump Seal Heat Exchanger Shell, SI and CV Spray Pump Seal Heat Exchanger Shell and Cover	Carbon Steel	Treated Water (including steam)	Loss of Material from Galvanic Corrosion	Closed-Cycle Cooling Water System Program	The GALL Report does not identify galvanic corrosion as an applicable aging mechanism. The Closed-Cycle Cooling Water System Program is applied to manage galvanic corrosion. This is consistent with the GALL Report to the extent that the Closed-Cycle Cooling Water System Program is used to manage loss of material for other aging mechanisms. Therefore, loss of material from galvanic corrosion can be managed by the same program to assure these components maintain their intended function throughout the period of extended operation.
6. CV Spray Pump Seal Heat Exchanger Shell and Cover	Carbon Steel	Treated Water (including steam)	Loss of Material from Selective Leaching	Closed-Cycle Cooling Water System Program	The GALL Report applies the Closed Cycle Cooling Water System and Selective Leaching of Materials Programs to manage selective leaching (for example, refer to GALL Section VII.C2.3-a). However, RNP applies only the Closed-Cycle Cooling Water System Program. The RNP AMR methodology considers selective leaching of components exposed to treated water to be managed by cooling water chemistry. The chemistry of the CCW System utilizes corrosion inhibitors to protect base metal from electrochemical reactions and is maintained by the Closed-Cycle Cooling Water System Program. Therefore, aging management of selective leaching, although not consistent with the GALL Report, is effective in preventing the aging mechanism.

TABLE 3.2-2 (continued) ENGINEERED SAFETY FEATURES SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
7. RHR Heat Exchanger Tubing, RHR SEAL WTR Heat Exchanger Tubing	Stainless Steel	Treated Water (including steam)	Cracking from SCC	Closed-Cycle Cooling Water System Program	The RNP AMR determined that cracking due to SCC could be applicable to stainless steel heat exchanger tubing. The Closed-Cycle Cooling Water System Program was applied to manage the cracking due to SCC on the shell side of the tubing. This is appropriate because that Program limits the presence of chemical impurities required for SCC to occur. Use of water chemistry controls to manage cracking due to SCC is similar to its use in the GALL Report, Section V.D1.1-a.
8. Boron Injection Tank, SI Pumps, RHR Pumps, CV Spray Pumps, ECCS Screen Filters, ECCS Sump Hood Filter, Eductors, Flow Orifices/ Elements, Refueling Water Storage Tank, SI Pump Recirc Strainer Filters	Stainless Steel	Indoor – Not Air Conditioned, Containment Air, Air and Gas, Borated Water Leakage	None	None Required	The RNP AMR determined that these components have no aging effects requiring management for these environments. The applicable RNP environments do not promote concentration of contaminants or include exposure to aggressive chemical species. Boric acid is not an aggressive chemical species for stainless steel.

TABLE 3.2-2 (continued) ENGINEERED SAFETY FEATURES SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
9. Valves, Piping, Tubing and Fittings	Stainless Steel	Indoor – Not Air Conditioned, Containment Air, Air and Gas, Outdoor, Borated Water Leakage	None	None Required	The RNP AMR determined that these components have no aging effects requiring management for these environments. The applicable RNP environments do not promote concentration of contaminants or include exposure to aggressive chemical species. Boric acid is not an aggressive chemical species for stainless steel.
10. SI Accumulator Tanks	Stainless Steel	Air and Gas	None	None Required	The RNP AMR determined that the accumulator tank stainless steel internal cladding has no aging effects requiring management in an Air and Gas environment.
11. Safety Injection Pump Outboard Bearing Heat Exchanger Shell	Carbon Steel	Lubricating Oil	None	None Required	The RNP AMR determined that carbon steel has no aging effects requiring management in lubricating oil with no water contamination.
12. Valves	Copper Alloys	Indoor – Not Air Conditioned, Air and Gas, Borated Water Leakage	None	None Required	The RNP AMR determined that these components have no aging effects requiring management for these environments. The applicable RNP environments do not promote concentration of contaminants or include exposure to aggressive chemical species. Boric acid is not considered an aggressive chemical species for copper alloys.

TABLE 3.2-2 (continued) ENGINEERED SAFETY FEATURES SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
13. Valves	Aluminum	Air and Gas	None	None Required	The RNP AMR determined that these components have no aging effects requiring management in an Air and Gas environment. The applicable RNP environment does not promote concentration of contaminants or include exposure to aggressive chemical species.
14. Valves, Piping and Fittings	Carbon Steel	Indoor – Not Air Conditioned, Air and Gas	See discussion	None Required	The RNP AMR methodology assumed that external surfaces of carbon steel components would not be susceptible to corrosion if they were located in areas protected from the weather, were not subjected to condensation, and were not subjected to aggressive chemical attack (e.g., borated water leakage).

Note: 1. Environments used in the aging management review are listed on Tables 3.0-1 and 3.0-2.

3.3 AGING MANAGEMENT OF AUXILIARY SYSTEMS

Auxiliary Systems are those systems used to support normal and emergency plant operations. The systems provide cooling, ventilation, sampling and other required functions.

3.3.1 AGING MANAGEMENT REVIEW

3.3.1.1 Methodology

Aging management review (AMR) of Auxiliary Systems components and commodities involved consideration and evaluation of the materials, environments, and stressors that are associated with each structure, component, or commodity grouping under review, as discussed in Section 4.2 of NEI 95-10 [Reference 3.3-1]. The RNP AMR methodology follows the approach recommended in NEI 95-10 and is based on generic industry guidance for determining aging effects for both mechanical and civil/structural components. The guidance represents a set of rules that allow the evaluator to identify aging effects for a given material and environment combination. The material and environment-based rules in the generic industry guidance documents are derived from known age-related degradation mechanisms and industry operating experience. The guidance was reviewed for applicability to RNP materials of construction and component internal and external operating environments and was used to identify aging effects for components, structures, and commodities. The results of the evaluation of materials and environment combinations, using the RNP methodology, are aging effects; and, if the aging effects adversely affect intended functions, the results are aging effects requiring management for the applicable components and commodities. Aging effects that require management are correlated to aging management programs.

The aging management review identifies one or more aging management programs to be used to demonstrate that the effects of aging will be managed so that the intended functions will be maintained consistent with CLB for the period of extended operation. The programs to be used for managing the effects of aging were compared to those listed in the GALL Report [Reference 3.3-2] and evaluated for consistency with GALL Report programs that are relied on for license renewal. The results are documented and discussed in Subsection 3.3.2 using the format suggested by the NRC Standard Review Plan for License Renewal (SRP-LR) [Reference 3.3-3].

3.3.1.2 Operating Experience

Operating experience (OE) through December 2001 was considered during the development of the RNP Integrated Plant Assessment. OE subsequent to that date will be reviewed and applicable OE will be updated in conjunction with the amendment to the application required by 10 CFR 54.21(b). The review consisted of the following:

Site:	RNP site-specific operating experience was reviewed. The site-specific operating experience included a review of (1) Corrective Action Program, (2) Licensee Event Reports, (3) Maintenance Rule Data Base, and (4) interviews with Systems Engineers. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Subsection.
Industry:	An evaluation of industry operating experience published since the effective date of the GALL Report was performed to identify any additional aging effects requiring management. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Subsection.
On-Going	On-going review of plant-specific and industry operating experience is performed in accordance with the Corrective Action and Operating Experience Programs.

3.3.2 AGING MANAGEMENT PROGRAMS

3.3.2.1 Aging Management Programs Evaluated in the GALL Report that Are Relied on for License Renewal

Table 3.3-1 shows the component and commodity groups (combinations of materials and environments), and aging management programs evaluated in the GALL Report that are relied on for license renewal of the Auxiliary Systems. The table is based on Table 3.3-1 of the SRP-LR [Reference 3.3-3] and provides a discussion of the applicability of the component commodity group and details regarding the degree to which RNP aging management programs are consistent with those recommended in the GALL Report. The discussion section includes (1) information regarding the applicability of the GALL Report component/commodity group to RNP, (2) any issues recommended in the GALL Report that require further evaluation, (3) details regarding RNP components to be included in the component/commodity group, and (4) a conclusion regarding consistency of the aging management review with the GALL Report.

3.3.2.2 Further Evaluation of Aging Management as Recommended by the GALL Report

Further evaluation of aging management as recommended by the GALL Report has been incorporated into the “Discussion” column of Table 3.3-1.

3.3.2.3 Aging Management Evaluations that Are Different from or Not Addressed in the GALL Report

Aging Management Evaluations that are different from or not addressed in the GALL Report are identified and discussed on Table 3.3-2.

3.3.3 CONCLUSION

The aging effects requiring management for the Auxiliary Systems are adequately managed by the following programs:

1. Above Ground Carbon Steel Tanks Program
2. ASME Section XI, Subsection IWB, IWC, and IWD Program
3. Bolting Integrity Program
4. Boric Acid Corrosion Program
5. Buried Piping and Tanks Inspection Program
6. Buried Piping and Tanks Surveillance Program
7. Closed-Cycle Cooling Water System Program
8. Fatigue Monitoring Program
9. Fire Protection Program
10. Fire Water System Program
11. Fuel Oil Chemistry Program
12. Inspection of Overhead Heavy Load and Light Load Handling Systems Program
13. One-Time Inspection Program
14. Open-Cycle Cooling Water System Program
15. Preventive Maintenance Program
16. Selective Leaching of Materials Program
17. Systems Monitoring Program
18. Water Chemistry Program

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that the intended functions of the Auxiliary Systems components are maintained consistent with the current licensing basis for the period of extended operation.

3.3.4 REFERENCES

3.3-1 NEI 95-10, “Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule,” Revision 3, Nuclear Energy Institute, March 2001.

3.3-2 NUREG-1801, “Generic Aging Lessons Learned (GALL),” U. S. Nuclear Regulatory Commission, April 2001.

3.3-3 NUREG-1800, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” U. S. Nuclear Regulatory Commission, April 2001.

TABLE 3.3-1 AUXILIARY SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
1. Components in spent fuel pool cooling and cleanup	Loss of material due to general, pitting, and crevice corrosion	Water chemistry and one-time inspection	Yes, detection of aging effects is to be further evaluated	In scope components (filters and demineralizers) and material (carbon steel with lining) specified in the GALL Report are not applicable to the RNP Spent Fuel Pool Cooling System. For RNP, the in-scope components are limited to stainless steel valves, pipes, fittings, and flow elements. These are potentially susceptible to crevice and pitting corrosion as discussed in Table 3.3-2, Item 1.
2. Linings in spent fuel pool cooling and cleanup system; seals and collars in ventilation systems	Hardening, cracking and loss of strength due to elastomer degradation; loss of material due to wear	Plant specific	Yes, plant specific	In scope components in the Spent Fuel Pool Cooling System do not have elastomeric liners. For ventilation systems, this group includes the Containment Purge, Rod Drive Cooling, Containment Air Recirculation Cooling (CARC), Reactor Auxiliary Building (RAB) HVAC, Control Room Area HVAC, and Fuel Handling Building HVAC Systems. The RNP Diesel Generator HVAC is part of RAB HVAC System. The plant-specific Systems Monitoring Program is used to manage the aging effects for in-scope HVAC elastomeric components. Wear was not identified as an aging mechanism for these components; however, wear also would be managed by the Systems Monitoring Program, which includes visual inspections to detect degradation of various types. Aging management of the in scope components in this component/commodity group is consistent with the GALL Report.

TABLE 3.3-1 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
3. Components in load handling, chemical and volume control system (PWR), and reactor water cleanup and shutdown cooling systems(older BWR)	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	Primary Sampling System components have been included in this component/commodity group. Cumulative fatigue damage of in-scope cranes and thermal fatigue of CVCS and Primary Sampling System components are evaluated as TLAAs. Section 4.3 addresses TLAAs for metal components. Evaluation of this component/commodity group is consistent with the GALL Report.
4. Heat exchangers in reactor water cleanup system (BWR); high pressure pumps in chemical and volume control system (PWR)	Crack initiation and growth due to SCC or cracking	Plant specific	Yes, plant specific	SCC requires a combination of a susceptible material, a corrosive environment, and tensile stress. The generic industry guidance used to identify aging effects based on materials and environments established a minimum threshold value for stainless steel of 200 F. The GALL Report does not identify a temperature threshold for this mechanism. At RNP, a temperature criterion of > 140°F is used as the threshold for susceptibility of austenitic stainless steels and nickel based alloys to SCC. The RNP AMR concluded that SCC is not applicable to the positive displacement charging pumps, because the temperature of the pumped fluid is normally less than 140°F. In addition, the pump bolting is not susceptible to cracking. The bolting material has a minimum specified yield strength less than 150 ksi; therefore, it is within the bounds of EPRI NP-5769 with regard to non-susceptibility to SCC. Therefore, cracking due to SCC is not applicable to the RNP CVCS charging pumps.

TABLE 3.3-1 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
5. Components in ventilation systems, diesel fuel oil system, and emergency diesel generator systems; external surfaces of carbon steel components	Loss of material due to general, pitting, and crevice corrosion, and MIC	Plant specific	Yes, plant specific	RNP had included loss of material from galvanic corrosion and crevice and pitting corrosion of external surfaces of aluminum and stainless steel components and cracking from SCC of stainless steel components in this group; because the effects of these mechanisms on external surfaces of components were not addressed elsewhere. Since this GALL group does not recognize these mechanisms or materials, they are discussed in Table 3.3-2, Items 10, 11, 12, and 13. For components in this component/commodity group, the plant-specific Systems Monitoring Program is used to manage the applicable aging effects on external surfaces. The exception involves the external surfaces of above-ground tanks. For these tanks, the Above-Ground Carbon Steel Tank Inspection Program is applicable. In addition, the Preventive Maintenance (PM) Program, which is a plant-specific program, is used to manage the effects of aging for the internal surfaces of components of this component/commodity group. An inspection of the internal surfaces of emergency diesel exhaust silencers (mufflers) has been scheduled based on industry operating experience. This will be accomplished under the One-Time Inspection Program. Based on the above, aging management for components in this group is consistent with the GALL Report.

TABLE 3.3-1 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
6. Components in reactor coolant pump oil collect system of fire protection	Loss of material due to galvanic, general, pitting, and crevice corrosion	One-time inspection	Yes, detection of aging effects is to be further evaluated	RNP does not have an oil collection system consisting of a tank and collection piping. This component/ commodity group is not applicable to RNP.
7. Diesel fuel oil tanks in diesel fuel oil system and emergency diesel generator system	Loss of material due to general, pitting, and crevice corrosion, MIC, and biofouling	Fuel oil chemistry and one-time inspection	Yes, detection of aging effects is to be further evaluated	The GALL Report includes only tanks in this group. The RNP AMR included in this group the valves, piping, and fittings in systems connected to the tanks that are subject to the same fuel oil environment and subject to the same aging effects/mechanisms. RNP relies on the Fuel Oil Chemistry Program to manage loss of material in the fuel oil systems of the Diesel Fire Pump, Dedicated Shutdown Diesel, EOF/TSC Security Diesel, and Emergency Diesel Systems. Internal inspection of large fuel oil storage tanks is performed periodically. Internal surfaces are inspected for coating integrity; if coating integrity were found to be compromised, appropriate corrective action would be taken. A one-time inspection of the small, elevated, Diesel Fire Pump Fuel Oil Tank and diesel generator day tanks is not warranted. These small tanks have limited access to the tank internals making it impractical to clean and perform a meaningful inspection. Also, RNP operating experience indicates that degradation of these tanks is not occurring. The Fuel Oil Chemistry program ensures a high quality, non- (continued)

TABLE 3.3-1 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
7. (continued)				corrosive, non-biologically-contaminated fuel oil for use at RNP. Periodic measurements of bacteria as well as trending of sample results will be performed. Biofouling was not identified as an aging mechanism; however, the above program would detect biofouling, should it occur, as well as loss of material. Based on the above, the Fuel Oil Chemistry Program, supplemented with periodic inspections of large tanks, provides for aging management of fuel oil tank internals consistent with the GALL Report with exceptions as documented in the description of the Program in Appendix B.
8. Heat exchangers in chemical and volume control system	Crack initiation and growth due to SCC and cyclic loading	Water chemistry and a plant-specific verification program	Yes, plant specific	SCC is an applicable to the Seal Water, Excess Letdown, and Regenerative Heat Exchangers. The AMPs used to manage SCC in heat exchangers are the Water Chemistry Program, and, for heat exchangers cooled by the Component Cooling Water System, the Closed-Cycle Cooling Water System Program. To verify the effectiveness of the Water Chemistry Program in preventing cracking due to SCC, an inspection of small-bore piping will be performed under the One-Time Inspection Program in selected locations where degradation would be expected. Management of SCC for this group is consistent with the GALL Report with the exception that the one-time inspection will be used instead of the eddy current testing recommended in the GALL Report.

TABLE 3.3-1 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
9. Neutron absorbing sheets in spent fuel storage racks	Reduction of neutron absorbing capacity and loss of material due to general corrosion (Boral, boron steel)	Plant specific	Yes, plant specific	This aging effect/mechanism is not applicable because the RNP spent fuel racks do not use Boral or boron steel neutron absorbing materials. See Item 11 below.
10. New fuel rack assembly	Loss of material due to general, pitting, and crevice corrosion	Structures monitoring	No	The New Fuel Rack assembly is not in scope for License Renewal.
11. Neutron absorbing sheets in spent fuel storage racks	Reduction of neutron absorbing capacity due to Boraflex degradation	Boraflex monitoring	No	RNP uses Boraflex panels in spent fuel racks. As discussed in Subsection 4.6.4, prior to the period of extended operation the current Boraflex Monitoring Program will be evaluated against the 10 elements for an acceptable program documented in the GALL Report and used to manage the effects of Boraflex degradation through the period of extended operation.

TABLE 3.3-1 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
12. Spent fuel storage racks and valves in spent fuel pool cooling and cleanup	Crack initiation and growth due to stress corrosion cracking	Water chemistry	No	At RNP, a temperature criterion of > 140°F is used as the threshold for susceptibility of austenitic stainless steels and nickel based alloys to SCC. The RNP AMR concluded that SCC was not applicable to these components, because the temperature of the fluid is normally less than 140°F. Therefore, the aging effect/mechanism is not applicable to RNP. (The aging effects applicable are crevice and pitting corrosion for which the Water Chemistry Program is applicable. Refer to Table 3.3-2, Item 1.)
13. Closure bolting and external surfaces of carbon steel and low-alloy steel components	Loss of material due to boric acid corrosion	Boric acid corrosion	No	The RNP AMR determined that boric acid attack of aluminum components is a potential aging effect. Since aluminum is not recognized for this group in the GALL Report, it is discussed in Table 3.3-2, Item 14. For closure bolting, loss of material due to boric acid corrosion can lead to loss of mechanical closure integrity. Therefore, the aging effect/mechanism for bolting is defined as "Loss of Mechanical Closure Integrity from Loss of Material due to Aggressive Chemical Attack." The Boric Acid Corrosion Program is the applicable AMP for these aging effects requiring management. Aging management of this component/commodity group is consistent with the GALL Report.

TABLE 3.3-1 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
14. Components in or serviced by closed-cycle cooling water system	Loss of material due to general, pitting, and crevice corrosion, and MIC	Closed-cycle cooling water system	No	Components fabricated from copper alloys (tubing for coolers cooled by the Component Cooling Water System) are not in GALL; also, galvanic corrosion and fouling of heat exchanger tubing are aging mechanisms that are applicable but not in GALL. These issues are discussed in Table 3.3-2, Items 15 and 16. The RNP AMR determined that MIC was not applicable for the closed-cycle cooling water systems, because no source of microbial contamination was identified. The Closed-Cycle Cooling Water System Program manages aging for this group. This is consistent with the GALL Report for the applicable aging effects/mechanisms.
15. Cranes including bridge and trolleys and rail system in load handling system	Loss of material due to general corrosion and wear	Overhead heavy load and light load handling systems	No	The RNP AMR identified general corrosion, but not wear, as an aging mechanism for crane rails. For in-scope cranes, loss of material will be managed by the Inspection of Overhead Heavy Load and Light Load Handling Systems Program regardless of the aging mechanism. Therefore, the aging management results for this component/commodity group are consistent with the GALL Report.
16. Components in or serviced by open-cycle cooling water systems	Loss of material due to general, pitting, crevice, and galvanic corrosion, MIC, and biofouling; buildup of deposit due to biofouling	Open-cycle cooling water system	No	The RNP methodology identified an additional aging mechanism: loss of material from erosion for certain coolers cooled by the SWS. This is discussed on Table 3.3-2, Item 17. Aging management of this component/commodity group relies on the Open-cycle cooling Water System Program and is consistent with the GALL Report.

TABLE 3.3-1 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
17. Buried piping and fittings	Loss of material due to general, pitting, and crevice corrosion, and MIC	Buried piping and tanks surveillance	No	Systems containing buried components are the Service Water, Diesel Generator Fuel Oil, DS Diesel, and Fire Protection Systems. The aging mechanism of galvanic corrosion also is applicable but not addressed in GALL; see Table 3.3-2, Item 29. The Buried Piping and Tanks Surveillance Program is a cathodic protection system applied to components in the Fuel Oil System. Aging management is consistent with the GALL Report with exceptions detailed in the program description in Appendix B..
		or		
		Buried piping and tanks inspection	Yes, detection of aging effects and operating experience are to be further evaluated	The Buried Piping and Tanks Inspection Program is applied to portions of the Service Water, DS Diesel, and Fire Protection Systems. Based on operating experience, it was determined that periodic inspection of susceptible locations is not necessary. The number of leaks caused by external corrosion in buried pipe has been small and limited to service water piping. Three leaks have occurred in the North Service Water header, and were limited to pipe in a section of header that was re-routed for construction of the Radwaste Building in 1984. The cause of leakage has been identified as construction-related defects in the coating applied to the exterior of the pipe. No leaks have been detected in the undisturbed portion of the Service Water Piping. Therefore, additional measures to detect aging effects are not necessary. Management of aging effects is consistent with the GALL Report with exceptions detailed in the program description in Appendix B.

TABLE 3.3-1 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
18. Components in compressed air system	Loss of material due to general and pitting corrosion	Compressed air monitoring	No	The aging mechanisms in the GALL Report are based on internal air conditions. In the RNP AMR methodology, the mechanisms are not applicable to Instrument Air and Nitrogen Supply/ Blanketing Systems because moisture is controlled. The Instrument Air System contains clean, dried air and the Nitrogen Supply/Blanketing System uses dry bottled nitrogen. For internal surfaces of diesel air start systems, cracking and loss of material due to various aging mechanisms is managed by the Preventive Maintenance Program, which is a plant-specific program. This is accomplished mainly by assuring that moisture is removed from air receivers. Internal surfaces that are not wetted are not susceptible to loss of material. At RNP, the materials in air start systems include stainless steel and copper alloys as well as carbon steel. Based on this discussion, a compressed air monitoring program as recommended by the GALL Report is not necessary. External surfaces of air start systems are addressed in Item 5 above.

TABLE 3.3-1 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
19. Components (doors and barrier penetration seals) and concrete structures in fire protection	Loss of material due to wear; hardening and shrinkage due to weathering	Fire protection	No	The RNP Fire Protection Program manages the aging effects on the intended function of the penetration seals and all fire rated doors (automatic or manual) that perform a fire barrier function. The RNP AMR identified general corrosion as a mechanism applicable to fire doors. General corrosion of fire doors also is managed by the Fire Protection Program and is included in this component/ commodity group. Concrete structures (walls, ceilings, floors), that perform a fire barrier function, are addressed in Item 25 below. The aging effects, for barrier penetration seals identified at RNP, envelope those listed in the GALL Report. Also, the RNP Fire Protection Program provides inspection criteria for fire doors that would identify the effects of wear. Therefore, the aging management results for this component/commodity group are consistent with the GALL Report with certain exceptions regarding frequency of inspecting fire doors and barriers as detailed in the description of the Fire Protection Program in Appendix B.

TABLE 3.3-1 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
20. Components in water-based fire protection	Loss of material due to general, pitting, crevice, and galvanic corrosion, MIC, and biofouling	Fire water system	No	For RNP, this component group would include aluminum components in the Fire Water System and aluminum components in a portion of the Circulating Water System that supports the Fire Water System. Since this material is not addressed in the GALL Report, it is discussed in Table 3.3-2, Item 18. The Fire Water System Program is applicable to this component/commodity group. The aging management results for this component/ commodity group are consistent with the GALL Report with exceptions detailed in the program description in Appendix B.
21. Components in diesel fire system	Loss of material due to galvanic, general, pitting, and crevice corrosion	Fire protection and fuel oil chemistry	No	The GALL Report includes in this group the Diesel Driven Fire Pump components subject to fuel oil. However, GALL Section VII.G.8-a indicates that this group includes the diesel fire pump casing. At RNP, the pump casing is not exposed to fuel oil. The RNP Fire Protection Program includes pump testing in the aging management strategy for the Diesel Driven Fire Pump fuel supply line. The quality of the fuel oil supplied to the Diesel Fire Pump engine is maintained by the Fuel Oil Chemistry Program. Therefore, aging management of these components at RNP is consistent with the GALL Report.

TABLE 3.3-1 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
22. Tanks in diesel fuel oil system	Loss of material due to general, pitting, and crevice corrosion	Aboveground carbon steel tanks	No	The RNP AMR did not identify crevice and pitting corrosion as applicable to above ground external surfaces of carbon steel tanks. However, loss of material due to these mechanisms would be detectible by the identified AMP. The Above Ground Carbon Steel Tanks Program manages exterior corrosion of above ground Fuel Oil System tanks. An exception to the UT inspection recommended by GALL was evaluated. Bottoms of tanks mounted on the ground are protected by an impressed current, cathodic protection system; and the tanks are set on oiled sand. These preventive features are used in lieu of UT testing tank bottoms to ascertain loss of material from the exterior bottom of the tank. The cathodic protection system is addressed in the Buried Piping and Tanks Surveillance Program. Aging management of this component/commodity group is considered to be consistent with the GALL Report because the methods used to prevent corrosion of the tank bottoms are considered to be equal to or better than the monitoring provided by UT inspections.

TABLE 3.3-1 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
23. Closure bolting	Loss of material due to general corrosion; crack initiation and growth due to cyclic loading and SCC	Bolting integrity	No	The Bolting Integrity Program relies on other aging management programs to manage specific aging effects. Closure bolting in Auxiliary Systems was evaluated for loss of material due to aggressive chemical species (boric acid corrosion) and for SCC based on minimum specified yield strength greater than 150ksi. (Aggressive chemical attack is managed by the Boric Acid Corrosion Program; see Item 13 above.) The RNP AMR methodology concluded that only bolting with high specified yield strength is subject to SCC. High strength bolts installed in scope components are present in only one valve in one RNP Auxiliary System, and the Bolting Integrity Program is applied to manage potential cracking of these bolts consistent with the GALL Report. The high strength bolts will be evaluated for susceptibility for cracking in accordance with the Bolting Integrity Program. Loss of material from general corrosion of all accessible carbon steel components including bolting is managed by the plant specific Systems Monitoring Program. Based on this discussion, aging management activities under the Bolting Integrity Program for closure bolting are considered to be consistent with the GALL Report with the exceptions noted in the Program description in Appendix B.

TABLE 3.3-1 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
24. Components (aluminum bronze, brass, cast iron, cast steel) in open-cycle and closed-cycle cooling water systems, and ultimate heat sink	Loss of material due to selective leaching	Selective leaching of materials	No	The Selective Leaching of Materials Program addresses components that are buried or subject to raw water (i.e., fire protection water, service water) and are susceptible to loss of material by selective leaching Selective leaching of susceptible components in closed cooling water systems (Component Cooling Water System and diesel cooling systems) is managed by the Closed-Cycle Cooling Water System Program. Based on operating experience, the control of chemical additives in closed-cycle cooling water systems is effective in reducing the occurrence of selective leaching and other forms of corrosion to negligible levels. The Component Cooling Water System pumps at RNP are not fabricated of material susceptible to selective leaching. Management of aging for this component/commodity group is consistent with the GALL Report with the exception identified in the program description in Appendix B.

TABLE 3.3-1 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
25. Fire barriers, walls, ceilings and floors in fire protection	Concrete cracking and spalling due to freeze-thaw, aggressive chemical attack, and reaction with aggregates; loss of material due to corrosion of embedded steel	Fire protection and structures monitoring	No	With respect to the aging effects and mechanisms identified for structure concrete, the RNP AMR concluded that: The aging mechanisms of freeze-thaw and reaction with aggregates are not applicable to RNP concrete structures, and The aging mechanisms of aggressive chemical attack and corrosion of embedded steel are not applicable to above grade concrete walls, ceilings, and floors. However, fire barriers penetrations are subject to inspection under the Fire Protection Program. The Fire Protection Program currently provides inspection criteria for fire barrier penetrations and includes inspection criteria for concrete that address cracking, holes, voids, or gaps. In addition, the Structures Monitoring Program administrative controls will be enhanced to note that concrete structure inspections are credited in the Fire Protection Program for inspecting fire barrier walls, ceilings, and floors. Based on the above discussion, aging management of this component/commodity group is consistent with the GALL Report with exceptions regarding the Fire Protection Program as described in Appendix B.

Note: 1. Numbered Component/Commodity Groups consist of the components listed in NRC Standard Review Plan for License Renewal, NUREG-1800, Table 3.3-1, that are applicable to a PWR.

TABLE 3.3-2 AUXILIARY SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
1. Pumps, Valves, Tanks, Piping and Fittings (Primary Sampling, CVCS, Spent Fuel Pool Cooling), Spent Fuel Racks, CVCS Heat Exchangers, Pulsation Dampers, Flow Orifices/ Elements, Seal Injection Filter, Seal Return Filter, Volume Control Tank, Valves, Piping, Tubing, and Fittings	Stainless Steel	Treated Water (including steam)	Loss of Material from Crevice Corrosion	Water Chemistry Program	The RNP AMR identified crevice and pitting corrosion as potential aging mechanisms. It is assumed that oxygen and contaminants are present such that crevice corrosion is always possible and pitting corrosion is possible if low flow rate conditions exist. The GALL Report notes that stainless steel components are not subject to significant degradation in borated water and that effects of crevice and pitting corrosion on stainless steel components are not significant in chemically treated borated water. (Refer to the discussion of Systems, Structures, and Components in GALL Sections VII.E.1 and VII.A.3.) Therefore, the Water Chemistry Program alone is considered to be sufficient to manage the aging mechanisms.
			Loss of Material from Pitting Corrosion	Water Chemistry Program	

TABLE 3.3–2 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
2. Charging Pump Lube Tanks, CVCS and Sampling System Piping, Valves, Tubing, and Fittings, Flow Elements and Filters	Stainless Steel	Treated Water (including steam)	Cracking from SCC	Water Chemistry Program	The RNP AMR identified this aging mechanism. The AMP used to manage SCC is the Water Chemistry Program. An inspection of small-bore piping will be performed under the One-Time Inspection Program in selected locations where degradation would be expected and will verify the efficacy of the Water Chemistry Program in managing SCC for stainless steel in treated water. This is consistent with the GALL Report.
3. CVCS Excess Letdown, Seal Water, and Class 2 portion of Regenerative Heat Exchanger	Stainless Steel	Treated Water (including steam)	Loss of Heat Transfer Effectiveness from Fouling of Heat Transfer Surfaces	Water Chemistry Program	This item deals with the heat transfer function of heat exchanger tubing. This aging mechanism was not identified in the GALL Report. The AMP used to manage this aging effect/ mechanism is the Water Chemistry Program, which maintains the purity of the water and assures that significant degradation of heat transfer effectiveness would not occur.
4. Flexible Hoses and Couplings (CO_2, Halon, CARDOX)	Elastomers	Air and Gas	Change in Material Properties from Elevated Temperature	Fire Protection Program	These components (elastomeric hose and couplings) are not addressed in the GALL Report. For the Fire Protection CO_2, Halon Supply System, and the Emergency Diesel Generator CarDox System, the Fire Protection Program manages the aging of flexible hoses and couplings. This program has traditionally accomplished these aging management activities.
			Cracking from Elevated Temperature	Fire Protection Program	

TABLE 3.3–2 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
5. Flexible Hoses and Couplings (DG System, DS Diesel, EOF DG, Instrument Air, and Fuel Oil Systems)	Elastomers and Misc. Piping Components	Internal: Air and Gas, Lubricating Oil, Fuel Oil, and Treated Water (including steam) External: Indoor – Not Air Conditioned, Containment Air, Borated Water Leakage, and Outdoor	Change in Material Properties from Various Degradation Mechanisms	Preventive Maintenance Program	These components (elastomeric hose and couplings) are not addressed in the GALL Report. However, the GALL Report addresses other components fabricated of elastomeric materials and applies a plant-specific AMP. RNP applies the plant-specific Preventive Maintenance Program to manage the effects of aging for flexible hoses in diesel generator auxiliary and fuel oil systems.
			Cracking from Various Degradation Mechanisms	Preventive Maintenance Program	
			Loss of Material from Various Degradation Mechanisms	Preventive Maintenance Program	
6. Primary and Demineralized Water Valves, Piping, and Fittings	Stainless Steel	Treated Water (including steam)	Loss of Material from Crevice Corrosion	Water Chemistry Program and One-Time Inspection Program	The GALL Report did not cover this system. The environment is demineralized water from the Condensate Storage Tank. Consistent with the GALL Report, aging management will be accomplished by a combination of the Water Chemistry Program and a One-Time Inspection Program.
			Loss of Material from Pitting Corrosion	Water Chemistry Program and One-Time Inspection Program	

TABLE 3.3–2 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
7. Primary and Demineralized Water Valves, Piping, and Fittings	Carbon Steel	Treated Water (including steam)	Loss of Material from Crevice Corrosion	Water Chemistry Program and One-Time Inspection Program	The GALL Report did not cover this system. The environment is demineralized water from the Condensate Storage Tank. Consistent with the GALL Report, aging management will be accomplished by a combination of the Water Chemistry Program and a One-Time Inspection Program.
			Loss of Material from General Corrosion	Water Chemistry Program and One-Time Inspection Program	
			Loss of Material from Pitting Corrosion	Water Chemistry Program and One-Time Inspection Program	
			Loss of Material from Galvanic Corrosion	Water Chemistry Program and One-Time Inspection Program	
8. Piping and Fittings (Radioactive Equipment Drains)	Stainless Steel	Raw Water	Loss of Material from Crevice Corrosion	None	This system was not addressed in the GALL Report. The potential aging effects/ mechanisms that are applicable do not affect the ability of the components to perform their intended functions. Therefore no AMP is required.
			Loss of Material from General Corrosion	None	
			Loss of Material from Pitting Corrosion	None	
9. Valves, Piping, Tubing, and Fittings	Stainless Steel	Treated Water (including steam)	Reduction in Fracture Toughness from Thermal Embrittlement	ASME Section XI, Subsection IWB, IWC, and IWD Program	The GALL Report does not include this aging effect for Auxiliary Systems. This group consists of CASS components in the CVCS system exposed to temperature > 482°F.

TABLE 3.3–2 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
10. DS Diesel Air Volume Tank, Piping, Valves, Tubing, and Fittings	Carbon Steel	Indoor - Not Air Conditioned	Loss of Material from Galvanic Corrosion	Systems Monitoring Program	The GALL Report does not recognize this aging mechanism for these components. The Systems Monitoring Program, which is used to manage loss of material from external surfaces of components due to general, pitting, and crevice corrosion and MIC consistent with the GALL Report, as discussed in Table 3.3-1, Item 5, will be used to manage loss of material from galvanic corrosion, also.
11. Service Water and Fire Water Piping and Fittings	Aluminum	Indoor - Not Air Conditioned	Loss of Material from Pitting and Crevice Corrosion	Systems Monitoring Program	The GALL Report does not address this material. The Systems Monitoring Program, which is used to manage crevice and pitting corrosion on the external surfaces of other components consistent with the GALL Report, as discussed in Table 3.3-1, Item 5, will be used to manage loss of material from crevice and pitting corrosion for aluminum components, also.
12. Ventilation Equipment Frames and Housings and Heating/ Cooling Coils; Fuel Oil and Diesel Generator Valves, Piping Tubing and Fittings	Stainless Steel	Indoor - Air Conditioned; Indoor - Not Air Conditioned; Containment Air; Borated Water Leakage; Outdoor	Loss of Material from Pitting and Crevice Corrosion and MIC	Systems Monitoring Program and Preventive Maintenance Program	The Systems Monitoring Program, which is used to manage loss of material on the external surfaces of other components consistent with the GALL Report, as discussed in Table 3.3-1, Item 5, will be used to manage loss of material for the fuel oil and diesel generator components, also. Also, the Preventive Maintenance Program, which is used to manage loss of material on the internal surfaces of components, as discussed in Table 3.3-1, Items 5 and 18, (continued)

TABLE 3.3–2 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
12. (continued)					also will be used to manage loss of material for internal surfaces of the ventilation system equipment frames and housings and the heating/cooling coils.
13. Valves, Piping, Tubing and Fittings	Stainless Steel	Indoor - Not Air Conditioned, Outdoor	Cracking from SCC	Systems Monitoring Program	The Systems Monitoring Program, which is used to manage crevice and pitting corrosion on the external surfaces of other components consistent with the GALL Report, as discussed in Table 3.3-1, Item 5, will be used to manage cracking from SCC for the stainless steel components, also.
14. Valves, Piping, and Fittings	Aluminum	Indoor - Not Air Conditioned, Outdoor, Containment, Borated Water Leakage	Loss of Material from Aggressive Chemical Attack and crevice and pitting corrosion	Boric Acid Corrosion Program	The GALL Report does not address boric acid wastage of this material. The RNP AMP for loss of material caused by boric acid attack does not depend on the material of the affected component. Therefore, the aging effect is managed in the same way as recommended in the GALL Report for carbon steel components.
15. Components in or Serviced by a Closed-Cycle Cooling Water System	Carbon Steel	Treated Water (including steam)	Loss of Material from Galvanic Corrosion, Loss of Heat Transfer Effectiveness from Fouling of Heat Transfer Surfaces	Closed-Cycle Cooling Water System Program	These aging effects/mechanisms are not addressed in the GALL Report. The Closed-Cycle Cooling Water System Program is effective in managing these effects/ mechanisms, because it maintains the water chemistry conditions and purity such that loss of material is minimized and fouling cannot occur.

TABLE 3.3–2 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
16. Components in or Serviced by a Closed-Cycle Cooling Water System	Copper Alloys	Treated Water (including steam)	Loss of Material from Crevice, Pitting, and Galvanic Corrosion, Loss of Heat Transfer Effectiveness from Fouling of Heat Transfer Surfaces	Closed-Cycle Cooling Water System Program	This material is not addressed in the GALL Report. The Closed-Cycle Cooling Water System Program is effective in managing the effects/ mechanisms for copper alloys, because it maintains the water chemistry conditions and purity such that loss of material is minimized and fouling cannot occur.
17. Heat Exchangers Tubes and Tubesheet Serviced by the Open-Cycle Cooling Water System	Carbon Steel, Copper Alloys	Raw Water	Loss of Material from Erosion	Open-Cycle Cooling Water System Program and One-Time Inspection Program	The GALL Report did not address this aging mechanism. Similar to the other mechanisms managed by this program that cause loss of material, the Open-Cycle Cooling Water System Program will effectively manage loss of material due to erosion. Further, the RNP AMP determined that an inspection of CCW Heat Exchanger tubing would be prudent to assure that potential degradation due to erosion was managed. This inspection will be done under the One-Time Inspection Program.
18. Valves, Piping and Fittings	Aluminum	Raw Water	Loss of Material from Crevice, Pitting, MIC, and Galvanic Corrosion; Flow Blockage from Fouling	Fire Water System Program	The GALL Report applies this AMP to various materials including copper alloys and stainless steel in a raw water environment. Aging management for the aluminum components would be accomplished in like manner by the Fire Water System Program. Therefore, aging management of the aluminum components is considered to be consistent with the GALL Report.

TABLE 3.3–2 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
19. Instrument Air Filters and Regulators; Flow Orifices/ Elements; Air and Nitrogen Accumulator Tanks; Equipment Frames and Housings; D/G Auxiliaries Components; Closure Bolting; Fuel Oil System Components; Valves, Piping and Fittings	Carbon Steel	Indoor – Not Air Conditioned, Containment Air, Air and Gas	See discussion	None Required	The RNP AMR methodology assumed that external surfaces of carbon steel components would not be susceptible to corrosion if they were located in areas protected from the weather, were not subjected to condensation, and were not subjected to aggressive chemical attack (e.g., borated water leakage).
20. Damper Mounting, Equipment Frames and Housings, Ductwork and Fittings	Galvanized Steel	Indoor – Not Air Conditioned, Containment Air, Borated Water Leakage	None	None Required	The RNP AMR determined that these components would experience no age related degradation requiring management in these environments.

TABLE 3.3–2 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
21. Charging Pump Heat Exchanger Shell; Instrument Air Regulator; Heating/ Cooling Coils; Sprinklers; Emergency DG Air Start Strainers; Valves, Piping, Tubing and Fittings	Copper Alloys	Indoor – Not Air Conditioned, Containment Air, Air and Gas, Borated Water Leakage, Outdoor	None	None Required	The RNP AMR determined that these components have no aging effects requiring management for these environments. The applicable RNP environments do not promote concentration of contaminants or include exposure to aggressive chemical species. Boric acid is not an aggressive chemical species for copper alloys.
22. Charging Pump Heat Exchanger Shell, Tubing, and Waterbox; D/G Heat Exchangers; D/G Lube Oil Temperature Regulators; DG Pumps and Strainers; Valves, Piping, Tubing and Fittings	Carbon Steel, Stainless Steel, Copper Alloys	Lubricating Oil	None	None Required	The RNP AMR determined that these components have no aging effects requiring management in a lubricating oil environment without water contamination.

TABLE 3.3–2 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
23. SW Boost Pumps; Charging Pumps; Charging Pump Lube Tank; Charging Pump Suction Stabilizers and Pulsation Dampeners; Regen Heat Exchanger Shell and Cover; Seal Inj Filter; Seal Return Filter; Vol Control Tank; Equip Frames and Housings; Heat/ Cool Coils; Flow Orifices; Valves, Piping, Tubing and Fittings (various systems)	Stainless Steel	Indoor – Not Air Conditioned, Containment Air, Air and Gas, Borated Water Leakage, Outdoor	None	None Required	The RNP AMR determined that these components have no aging effects requiring management for these environments. The applicable RNP environments do not promote concentration of contaminants or include exposure to aggressive chemical species. Boric acid is not an aggressive chemical species for stainless steel.

TABLE 3.3–2 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
24. Instrument Air Filters; Valves	Aluminum	Indoor – Not Air Conditioned, Containment Air, Air and Gas Outdoor	None	None Required	The RNP AMR determined that these components have no aging effects requiring management for these environments, considering that the applicable RNP environments do not promote concentration of contaminants or include exposure to aggressive chemical species.
25. Rod Drive Cooling System Cooler Tubing	Copper Alloys	Indoor – Not Air Conditioned	Loss of Heat Transfer Effectiveness from Fouling of Heat Transfer Surfaces	Preventive Maintenance Program	The GALL Report does not address this aging effect/mechanism. Use of the Preventive Maintenance Program to manage this aging effect/mechanism is consistent with use of a plant- specific program to manage crevice and pitting corrosion of cooling coils of air handling units as recommended in the GALL Report, Section VII.F3.2-a.
26. Piping and Fittings	PVC	Buried	None	None Required	The RNP AMR methodology determined that these components have no aging effects requiring management in a buried environment.
27. Sight Glasses	Glass	Indoor – Not Air Conditioned, Containment Air, Borated Water Leakage, Outdoor	None	None Required	The RNP AMR methodology determined that these components have no aging effects requiring management in these environments.

TABLE 3.3–2 (continued) AUXILIARY SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
28. EOF/TSC Main Storage Tank, Piping and Fittings	Fiberglass Reinforced Polyester	Buried, Outdoor	None	None Required	The RNP AMR methodology determined that these components have no aging effects requiring management in the buried or outdoor environments.
29. Valves, Piping, and Fittings in Service Water System and Site Fire Protection System	Carbon Steel	Buried	Loss of Material from Galvanic Corrosion	Buried Piping and Tanks Inspection Program.	The RNP AMR methodology determined that galvanic corrosion was applicable to certain buried components. The Buried Piping and Tanks Inspection Program is applied to manage this aging mechanism in the same manner as the other applicable aging mechanisms noted in the GALL Report for buried carbon steel components.
30. Diesel- and Motor-Driven Fire Pump Casing	Carbon Steel	Raw Water	Loss of Material from General Corrosion	Preventive Maintenance Program	Based on RNP operating experience, the Diesel- and Motor-Driven Fire Pump Casings are replaced every 10 years in accordance with the Preventive Maintenance Program. This activity is used to manage degradation caused by corrosion of the external surface of the pumps in the “splash zone.”
31. Circulating Water System Piping and Fittings	Concrete	Raw Water, Buried	None	None Required	The concrete piping in the Circulating Water System provides a return path for Service Water to Lake Robinson. The RNP AMR determined that the concrete pipe has no aging effects requiring management for these environments.

Note: 1. Environments used in the aging management review are listed on Tables 3.0-1 and 3.0-2.

3.4 AGING MANAGEMENT OF STEAM AND POWER CONVERSION SYSTEMS

The Steam and Power Conversion Systems act as a heat sink to remove heat from the nuclear steam supply system and convert the heat generated in the reactor to the plant's electrical output.

3.4.1 AGING MANAGEMENT REVIEW

3.4.1.1 Methodology

Aging management review of Steam and Power Conversion Systems components and commodities involved consideration and evaluation of the materials, environments, and stressors that are associated with each structure, component, or commodity grouping under review, as discussed in Section 4.2 of NEI 95-10 [Reference 3.4-1]. The RNP AMR methodology follows the approach recommended in NEI 95-10 and is based on generic industry guidance for determining aging effects for both mechanical and civil/structural components. The guidance represents a set of rules that allow the evaluator to identify aging effects for a given material and environment combination. The material and environment-based rules in the generic industry guidance documents are derived from known age-related degradation mechanisms and industry operating experience. The guidance was reviewed for applicability to RNP materials of construction and component internal and external operating environments and was used to identify aging effects for components, structures, and commodities. The results of the evaluation of materials and environment combinations, using the RNP methodology, are aging effects; and, if the aging effects adversely affect intended functions, the results are aging effects requiring management for the applicable components and commodities. Aging effects that require management are correlated to aging management programs.

The aging management review identifies one or more aging management programs to be used to demonstrate that the effects of aging will be managed so that the intended functions will be maintained consistent with CLB for the period of extended operation. The programs to be used for managing the effects of aging were compared to those listed in the GALL Report [Reference 3.4-2] and evaluated for consistency with GALL Report programs that are relied on for license renewal. The results are documented and discussed in Subsection 3.4.2 using the format suggested by the NRC Standard Review Plan for License Renewal (SRP-LR) [Reference 3.4-3].

3.4.1.2 Operating Experience

Operating experience (OE) through December 2001 was considered during the development of the RNP Integrated Plant Assessment. OE subsequent to that date will be reviewed and applicable OE will be updated in conjunction with the amendment to the application required by 10 CFR 54.21(b). The review consisted of the following:

Site: RNP site-specific operating experience was reviewed. The site-specific operating experience included a review of (1) Corrective Action Program, (2) Licensee Event Reports, (3) Maintenance Rule Data Base, and (4) interviews with Systems Engineers. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Subsection.

Industry: An evaluation of industry operating experience published since the effective date of the GALL Report was performed to identify any additional aging effects requiring management. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Subsection.

On-Going On-going review of plant-specific and industry operating experience is performed in accordance with the Corrective Action and Operating Experience Programs.

3.4.2 AGING MANAGEMENT PROGRAMS

3.4.2.1 Aging Management Programs Evaluated in the GALL Report that Are Relied on for License Renewal

Table 3.4-1 shows the component and commodity groups (combinations of materials and environments), and aging management programs evaluated in the GALL Report that are relied on for license renewal of the Steam and Power Conversion Systems. The table is based on Table 3.4-1 of the SRP-LR [Reference 3.4-3] and provides a discussion of the applicability of the component commodity group and details regarding the degree to which RNP aging management programs are consistent with those recommended in the GALL Report. The discussion section includes (1) information regarding the applicability of the GALL Report component/commodity group to RNP, (2) any issues recommended in the GALL Report that require further evaluation, (3) details regarding RNP components to be included in the component/commodity group, and (4) a conclusion regarding consistency of the aging management review with the GALL Report.

3.4.2.2 Further Evaluation of Aging Management as Recommended by the GALL Report

Further evaluation of aging management as recommended by the GALL Report has been incorporated into the “Discussion” column of Table 3.4-1.

3.4.2.3 Aging Management Evaluations that Are Different from or Not Addressed in the GALL Report

Aging Management Evaluations that are different from or not addressed in the GALL Report are identified and discussed on Table 3.4-2.

3.4.3 CONCLUSION

The aging effects requiring management for the Steam and Power Conversion Systems are adequately managed by the following programs:

1. Boric Acid Corrosion Program
2. Closed-Cycle Cooling Water System Program
3. Fatigue Monitoring Program
4. Flow-Accelerated Corrosion Program
5. One-Time Inspection Program
6. Open-Cycle Cooling Water System Program
7. Preventive Maintenance Program
8. Selective Leaching Program
9. Systems Monitoring Program
10. Water Chemistry Program

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that the intended functions of the Steam and Power Conversion Systems components are maintained consistent with the current licensing basis for the period of extended operation.

3.4.4 REFERENCES

3.4-1 NEI 95-10, “Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule,” Revision 3, Nuclear Energy Institute, March 2001.

3.4-2 NUREG-1801, “Generic Aging Lessons Learned (GALL),” U. S. Nuclear Regulatory Commission, April 2001.

3.4-3 NUREG-1800, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” U. S. Nuclear Regulatory Commission, April 2001.

TABLE 3.4-1 STEAM AND POWER CONVERSION SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
1. Piping and fittings in main feedwater line, steam line and AFW piping (PWR only)	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	Evaluation of this component/commodity group is consistent with the GALL Report. Refer to Section 4.3 for the TLAA evaluations associated with metal fatigue.
2. Piping and fittings, valve bodies and bonnets, pump casings, tanks, tubes, tubesheets, channel head and shell (except main steam system)	Loss of material due to general (carbon steel only), pitting, and crevice corrosion	Water chemistry and one-time inspection	Yes, detection of aging effects is to be further evaluated	Pumps in the Feedwater, Condensate, and Steam Generator Blowdown Systems are not in scope for license renewal. Also, heat exchangers in the Condensate and Steam Generator Blowdown Systems are not in scope. Aging management for this component/ commodity group consists of the Water Chemistry Program and the One-Time Inspection Program. This is consistent with the GALL Report. Components in addition to those identified in the GALL Report are included in this group. These include flow elements, temperature elements, tubing and fittings, and feedwater heaters. These components are fabricated of carbon and stainless steel and are located in systems for which the Water Chemistry Program and the One-Time Inspection Program are applicable. The One-Time Inspection Program will be used to select inspection locations considering anticipated worst-case aging degradation. Therefore, although each component will not be inspected, the results of the One-Time Inspection Program will be applicable to each component in the inspected system.

TABLE 3.4 -1 (continued) STEAM AND POWER CONVERSION SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
3. Auxiliary feedwater (AFW) piping	Loss of material due to general, pitting, and crevice corrosion, MIC, and biofouling	Plant specific	Yes, plant specific	These aging effects/mechanisms are not applicable to the Auxiliary Feedwater System piping and fittings in the main flow path. The GALL Report assumes the occurrence of contamination in Auxiliary Feedwater System piping from backup water supplies and the consequent aging effects/ mechanisms of flow blockage from biofouling and loss of material from general, pitting, crevice, and MIC of carbon steel components. At RNP, backup supplies of raw water to the Auxiliary Feedwater System are available from the Service Water System and the Deepwell Pumps. The backup supplies are not normally aligned, and the normal internal environment for the piping and fittings is treated water from the Condensate Storage Tank. Contamination of the Auxiliary Feedwater System by raw water would be an extraordinary event and is not considered to be an applicable environment for license renewal. Therefore, the associated aging effects/mechanisms are not applicable.

TABLE 3.4 -1 (continued) STEAM AND POWER CONVERSION SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
4. Oil coolers in AFW system (lubricating oil side possibly contaminated with water)	Loss of material due to general (carbon steel only), pitting, and crevice corrosion and MIC	Plant specific	Yes, plant specific	The GALL Report considers the potential effects of oil contaminated with water on carbon steel components. These aging effects/ mechanisms are not applicable to the Auxiliary Feedwater System lubricating oil coolers. The RNP aging management review for Auxiliary Feedwater System pump lubricating oil coolers determined that water contamination of lube oil is not a credible environment, because the lube oil system is a closed system. The integrity of the Service Water side of the lubricating oil coolers is assured by the Open-Cycle Cooling Water System Program, which is addressed in Item 9 below.
5. External surface of carbon steel components	Loss of material due to general corrosion	Plant specific	Yes, plant specific	The Systems Monitoring Program is the plant specific program credited at RNP for managing loss of material due to general corrosion on the external surfaces of components. Therefore, aging management for the applicable components is consistent with the GALL Report.

TABLE 3.4 -1 (continued) STEAM AND POWER CONVERSION SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
6. Carbon steel piping and valve bodies	Wall thinning due to flow-accelerated corrosion	Flow-accelerated corrosion	No	RNP Steam Generators are not the Westinghouse preheater design. Also, the Turbine and Extraction Steam Systems are not in scope for license renewal. Carbon steel temperature elements in the Feedwater System have been included in this component/ commodity group. Management of aging effects for this group is by the Flow-Accelerated Corrosion Program and is consistent with the GALL Report.
7. Carbon steel piping and valve bodies in main steam system	Loss of material due to pitting and crevice corrosion	Water chemistry	No	The Water Chemistry Program is effective in managing loss of material due to crevice and pitting corrosion for carbon steel components. Aging management for this component/commodity group is consistent with the GALL Report. The RNP AMR determined that general and galvanic corrosion of internal surfaces of carbon steel steam system components is possible and that stainless steel components in the Main Steam System were susceptible to crevice and pitting corrosion. Since this GALL component/commodity group does not address crevice and pitting corrosion of stainless steel or galvanic and general corrosion of carbon steel components, these evaluations have been included in Table 3.4-2, Items 7 and 8.

TABLE 3.4 -1 (continued) STEAM AND POWER CONVERSION SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
8. Closure bolting in high-pressure or high-temperature systems	Loss of material due to general corrosion; crack initiation and growth due to cyclic loading and/or SCC.	Bolting integrity	No	Closure bolting in RNP Steam and Power Conversion Systems was evaluated for loss of material due to general corrosion or aggressive chemical species (boric acid corrosion), and for SCC based on minimum specified yield strength greater than 150ksi. The Boric Acid Corrosion Program is used to assure that loss of material from boric acid corrosion is detected and managed for components subjected to borated water leakage. Refer to Item 13, below. Also, there are no bolts in the Steam and Power Conversion Systems with sufficient specified minimum yield strength to be susceptible to SCC. Therefore, the Bolting Integrity Program is not applicable to bolting for the RNP Steam and Power Conversion Systems.

TABLE 3.4 -1 (continued) STEAM AND POWER CONVERSION SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
9. Heat exchangers and coolers/ condensers serviced by open-cycle cooling water	Loss of material due to general (carbon steel only), pitting, and crevice corrosion, MIC, and biofouling; buildup of deposit due to biofouling	Open-cycle cooling water system	No	RNP steam generator blowdown heat exchangers and condensate coolers are not in scope for license renewal. Aging management of carbon steel components relies on the Open-Cycle Cooling Water System Program and is consistent with the GALL Report except with respect to galvanic corrosion as discussed below. RNP employs the Open-Cycle Cooling Water System Program for lube oil coolers on the Steam-Driven and Motor-Driven Auxiliary Feedwater Pumps. The RNP AMR identified copper alloy tubing in these coolers. The GALL Report does not recognize this material for this component/commodity group. Therefore, the coolers are evaluated in Table 3.4-2, Item 9. The RNP AMR determined that galvanic corrosion was applicable to components in this group; however, since the GALL Report does not include this mechanism it is evaluated in Table 3.4-2, Item 10.
10. Heat exchangers and coolers/ condensers serviced by closed-cycle cooling water	Loss of material due to general (carbon steel only), pitting, and crevice corrosion	Closed-cycle cooling water system	No	Steam Generator Blowdown Sample Heat Exchangers are in scope only to maintain the pressure boundary function of the Component Cooling Water System. Aging management of the heat exchangers is via the Closed-Cycle Cooling Water System Program and is consistent with the GALL Report.
11. External surface of aboveground condensate storage tank	Loss of material due to general (carbon steel only), pitting, and crevice corrosion	Aboveground carbon steel tanks	No	The Condensate Storage Tank (CST) at RNP is fabricated of stainless steel. Therefore, this component/commodity group is not applicable. The internal environment is addressed in Item 2 above.

TABLE 3.4 -1 (continued) STEAM AND POWER CONVERSION SYSTEMS AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
12. External surface of buried condensate storage tank and AFW piping	Loss of material due to general, pitting, and crevice corrosion and MIC	Buried piping and tanks surveillance or Buried piping and tanks inspection	No Yes, detection of aging effects and operating experience are to be further evaluated	Neither the CST nor AFW piping is buried. Therefore, this component/commodity group is not applicable.
13. External surface of carbon steel components	Loss of material due to boric acid corrosion	Boric acid corrosion	No	Aging management of boric acid corrosion is accomplished by the Boric Acid Corrosion Program and is consistent with the GALL Report. For closure bolting, loss of material due to aggressive chemical attack can lead to loss of mechanical closure integrity.

Note: 1. Numbered Component/Commodity Groups consist of the components listed in NRC Standard Review Plan for License Renewal, NUREG-1800, Table 3.4-1, that are applicable to a PWR.

TABLE 3.4-2 STEAM AND POWER CONVERSION SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
1. Pumps, Valves, Piping and Fittings (includes temperature elements, flow elements/ orifices)	Carbon Steel	Treated Water (including steam)	Loss of Material from Galvanic Corrosion	Water Chemistry Program	These components are in the Steam Generator Blowdown, Steam Generator Chemical Addition, Feedwater, Auxiliary Feedwater, and Condensate Systems. The GALL Report does not identify this aging mechanism. The Water Chemistry Program is effective in managing loss of material due to galvanic corrosion, because it limits electrolytes in the treated water. An electrolytic solution is required for galvanic corrosion.
2. Piping, Valves, and Fittings (includes tubing, flow elements/ orifices, Heater Tubing)	Stainless Steel	Treated Water (including steam)	Cracking from SCC	Water Chemistry Program	The GALL Report does not identify these material/aging mechanism combinations. The Water Chemistry Program is effective in managing cracking due to SCC as well as the other aging mechanisms identified in Item 2 of Table 3.4-1.
3. Feedwater Heater Heat Exchanger Cover/ Tubesheet	Carbon Steel	Treated Water (including steam)	Loss of Material from Erosion and FAC	Preventive Maintenance Program	The GALL Report does not address this component. The Preventive Maintenance Program provides for periodic inspections and would detect loss of material from erosion or FAC.

TABLE 3.4–2 (continued) STEAM AND POWER CONVERSION SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
4. Pump Lube Oil Cooler Waterbox and Tubing	Carbon Steel, Copper Alloy	Raw Water	Loss of Material from Selective Leaching	Selective Leaching of Materials Program	The GALL Report does not address this aging mechanism for Steam and Power Conversion Systems. As discussed in Table 3.4-1, Item 9, this component also is covered by the Open-Cycle Cooling Water System Program, which addresses loss of material from various aging mechanisms but not selective leaching. To address selective leaching of materials, RNP will apply the Selective Leaching of Materials Program. Use of the Selective Leaching of Materials Program and the Open-Cycle Cooling Water System Program is consistent with other components in the GALL Report that are subject to this mechanism and are serviced by the open cycle cooling water system, e.g., GALL VII.C1.1-a. Therefore, management of this aging mechanism is consistent with the GALL Report.

TABLE 3.4–2 (continued) STEAM AND POWER CONVERSION SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
5. Condensate Storage Tank	Elastomers	Air and Gas	Change in Material Properties from Ultraviolet Radiation, Ozone Exposure, or Elevated Temperature	Preventive Maintenance Program	The GALL Report does not address this material. The Preventive Maintenance Program assures that the tank diaphragm (bladder) is inspected/replaced as necessary.
			Cracking from Ultraviolet Radiation, Ozone Exposure, or Elevated Temperature	Preventive Maintenance Program	
		Treated Water (including steam)	Change in Material Properties from Ultraviolet Radiation, Ozone Exposure, or Elevated Temperature	Preventive Maintenance Program	
			Cracking from Ultraviolet Radiation, Ozone Exposure, or Elevated Temperature	Preventive Maintenance Program	

TABLE 3.4–2 (continued) STEAM AND POWER CONVERSION SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
6. Valves, Piping, Fittings, and Temperature Elements in Main Steam, Feedwater, Auxiliary Feedwater, SG Blowdown, Condensate Systems	Carbon Steel	Treated Water (including steam)	Loss of Material from Erosion	Flow-Accelerated Corrosion Program	The RNP AMR determined that loss of material from erosion was applicable to these components. The RNP Flow-Accelerated Corrosion Program has been applied to these components. The Program is capable of detecting both loss of material from FAC and loss of material from erosion. Therefore, management of aging effects for this group is considered to be consistent with the GALL Report.
7. Valves, Piping, Tubing, Fittings, and Flow Elements in the Main Steam System; SDAFW Turbine	Carbon Steel	Treated Water (including steam)	Loss of Material from General, Galvanic, Pitting, and Crevice Corrosion	Water Chemistry Program	The RNP AMR determined that general, galvanic, pitting, and crevice corrosion of internal surfaces of carbon steel steam system components is possible. The Water Chemistry Program is effective in managing loss of material due to crevice and pitting corrosion for carbon steel, as discussed in Table 3.4-1, Item 7. Thus, it would be effective in managing loss of material from general and galvanic corrosion.
8. Valves, Piping, Tubing, Fittings, and Flow Elements in the Main Steam System	Stainless Steel	Treated Water (including steam)	Loss of Material from Crevice and Pitting Corrosion	Water Chemistry Program	The RNP AMR determined that stainless steel components in the Main Steam System were susceptible to crevice and pitting corrosion. The Water Chemistry Program is effective in managing loss of material due to crevice and pitting corrosion for carbon steel, as discussed in Table 3.4-1, Item 7. Thus, it would be effective in managing these aging mechanisms for stainless steel.

TABLE 3.4–2 (continued) STEAM AND POWER CONVERSION SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
9. Steam and Motor Driven AFW Pump Lube Oil Heat Exchanger Tubing	Copper Alloys	Raw Water	Loss of Material due to Pitting, Crevice Corrosion, and MIC; Flow Blockage from Fouling; Loss of Heat Transfer Effectiveness from Fouling of Heat Transfer Surfaces	Open-Cycle Cooling Water System Program	The GALL Report does not include copper alloy materials for these heat exchangers. Similar to the AMP applied in Table 3.4-1, Item 9, the Open-Cycle Cooling Water System Program is applied to manage these mechanisms for copper as well as carbon steel. (Note that the Steam-Driven Auxiliary Feedwater Pump is aligned to provide self-cooling for the lube oil coolers. In this cooling mode, the source of water is the Condensate Storage Tank. This alignment allows treated water from the pumped fluid to cool the lube oil. This alignment eliminates many of the aging mechanisms that are applicable if, as it was assumed above, the coolers are cooled by an open-cycle cooling system.)
10. Steam and Motor Driven AFW Pump Lube Oil Heat Exchanger Waterbox	Carbon Steel	Raw Water	Loss of Material from Galvanic Corrosion	Open-Cycle Cooling Water System Program	The GALL Report does not recognize this aging mechanism. Similar to the AMP applied in Table 3.4-1, Item 9, the Open-Cycle Cooling Water System Program is applied to manage this mechanism.

TABLE 3.4–2 (continued) STEAM AND POWER CONVERSION SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
11. AFW Pump and Turbine; AFW Lube Oil Heat Exchanger and Lube Oil Pump; Valves, Piping, Tubing and Fittings (various systems)	Carbon Steel	Indoor – Not Air Conditioned, Air and Gas	See discussion	None Required	The RNP AMR methodology assumed that external surfaces of carbon steel components would not be susceptible to corrosion if they were located in areas protected from the weather, were not subjected to condensation, and were not subjected to aggressive chemical attack (e.g., borated water leakage).
12. AFW Pump Lube Oil Heat Exchanger; Valves, Piping, Tubing and Fittings (various systems)	Copper Alloys	Indoor – Not Air Conditioned, Air and Gas, Outdoor	None	None Required	The RNP AMR determined that these components have no aging effects requiring management for this environment. The applicable RNP environment does not promote concentration of contaminants or include exposure to aggressive chemical species. Boric acid is not an aggressive chemical species for copper alloys.
13. Flow Orifices/ Elements; Valves, Piping, Tubing and Fittings (various systems)	Stainless Steel	Indoor – Not Air Conditioned, Containment Air, Air and Gas, Borated Water Leakage, Outdoor	None	None Required	The RNP AMR determined that these components have no aging effects requiring management for these environments. The applicable RNP environments do not promote concentration of contaminants or include exposure to aggressive chemical species. Boric acid is not an aggressive chemical species for stainless steel.

TABLE 3.4–2 (continued) STEAM AND POWER CONVERSION SYSTEMS AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
14. AFW Pump Lube Oil Heat Exchanger and Lube Oil Pump; AFW Valves, Piping, Tubing and Fittings	Carbon Steel, Stainless Steel, Copper Alloys	Lubricating Oil	None	None Required	The RNP AMR determined that these components have no aging effects requiring management in a lubricating oil environment without water contamination.
15. AFW Valve	Copper Alloys	Treated Water (including steam)	None	None Required	This check valve in the AFW System is Bronze ASTM B-62. The RNP AMR determined that the valve has no aging effects requiring management in the treated water pumped by the AFW System.

Note: 1. Environments used in the aging management review are listed on Tables 3.0-1 and 3.0-2.

3.5 AGING MANAGEMENT OF CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Structures, structural components, and commodities subject to aging management review include the Containment, Reactor Auxiliary Building, Fuel Handling Building, Turbine Building, Dedicated Shutdown Diesel Generator Building, Radwaste Building, Intake Structure, North Service Water Header Enclosure, Emergency Operations Facility/Technical Support Center Security Diesel Generator Building, Lake Robinson Reservoir and Dam, Pipe Restraint Tower, and Yard Structures and Foundations. The Control Room is located in the south end of the RAB. Component Supports include the support structures for the major Class 1 Components such as the Reactor Vessel, Steam Generators, Reactor Coolant Pumps, Pressurizer, and Reactor Coolant System Piping Supports and Restraints.

3.5.1 AGING MANAGEMENT REVIEW

3.5.1.1 Methodology

Aging management review of structures and structural components and commodities involved consideration and evaluation of the materials, environments, and stressors that are associated with each structure, component, or commodity grouping under review, as discussed in Section 4.2 of NEI 95-10 [Reference 3.5-1]. The RNP AMR methodology follows the approach recommended in NEI 95-10 and is based on generic industry guidance for determining aging effects for both mechanical and civil/structural components. The guidance represents a set of rules that allow the evaluator to identify aging effects for a given material and environment combination. The material and environment-based rules in the generic industry guidance documents are derived from known age-related degradation mechanisms and industry operating experience. The guidance was reviewed for applicability to RNP materials of construction and component internal and external operating environments and was used to identify aging effects for components, structures, and commodities. The results of the evaluation of materials and environment combinations, using the RNP methodology, are aging effects; and, if the aging effects adversely affect intended functions, the results are aging effects requiring management for the applicable components and commodities. Aging effects that require management are correlated to aging management programs.

The aging management review identifies one or more aging management programs to be used to demonstrate that the effects of aging will be managed so that the intended functions will be maintained consistent with CLB for the period of extended operation. The programs to be used for managing the effects of aging were compared to those listed in the GALL Report [Reference 3.5-2] and evaluated for consistency with GALL Report programs that are relied on for license renewal. The results are documented and discussed in Subsection 3.5.2 using the format suggested by the NRC Standard Review Plant for License Renewal (SRP-LR) [Reference 3.5-3]. Aging management programs are described in Appendix B.

3.5.1.2 Operating Experience

Operating experience (OE) through December 2001 was considered during the development of the RNP Integrated Plant Assessment. OE subsequent to that date will be reviewed and applicable OE will be updated in conjunction with the amendment to the application required by 10 CFR 54.21(b). The review consisted of the following:

Site: RNP site-specific operating experience was reviewed. The site-specific operating experience included a review of (1) Corrective Action Program, (2) Licensee Event Reports, (3) Maintenance Rule Data Base, and (4) interviews with Systems Engineers. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Subsection.

Industry: An evaluation of industry operating experience published since the effective date of the GALL Report was performed to identify any additional aging effects requiring management. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Subsection.

On-Going On-going review of plant-specific and industry operating experience is performed in accordance with the Corrective Action and Operating Experience Programs.

3.5.2 AGING MANAGEMENT PROGRAMS

3.5.2.1 Aging Management Programs Evaluated in the GALL Report that Are Relied on for License Renewal

Table 3.5-1 shows the component and commodity groups (combinations of materials and environments), and aging management programs evaluated in the GALL Report that are relied on for license renewal of the Containment, Structures, and Component Supports. The table is based on Table 3.5-1 of the SRP-LR [Reference 3.5-3] and provides a discussion of the applicability of the component commodity group and details regarding the degree to which RNP aging management programs are consistent with those recommended in the GALL Report. The discussion section includes (1) information regarding the applicability of the GALL Report component/commodity group to RNP, (2) any issues recommended in the GALL Report that require further evaluation, (3) details regarding RNP components to be included in the component/commodity group, and (4) a conclusion regarding consistency of the aging management review with the GALL Report.

3.5.2.2 Further Evaluation of Aging Management as Recommended by the GALL Report

Further evaluation of aging management as recommended by the GALL Report has been incorporated into the "Discussion" column of Table 3.5-1.

3.5.2.3 Aging Management Evaluations that Are Different from or Not Addressed in the GALL Report

Aging Management Evaluations that are different from or not addressed in the GALL Report are identified and discussed on Table 3.5-2.

3.5.3 CONCLUSION

Aging effects requiring management for Containments, Structures, and Component Supports are adequately managed by the following programs:

1. 10 CFR Part 50, Appendix J Program
2. ASME Section XI, Subsection IWE Program
3. ASME Section XI, Subsection IWF Program
4. ASME Section XI, Subsection IWL Program
5. Boric Acid Corrosion Program
6. Fatigue Monitoring Program
7. One-Time Inspection Program
8. Dam Inspection Program
9. Structures Monitoring Program
10. Water Chemistry Program

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that the intended functions of Containments, Structures, and Component Supports components are maintained consistent with the current licensing basis for the period of extended operation.

3.5.4 REFERENCES

3.5-1 NEI 95-10, “Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule,” Revision 3, Nuclear Energy Institute, March 2001.

3.5-2 NUREG-1801, “Generic Aging Lessons Learned (GALL),” U. S. Nuclear Regulatory Commission, April 2001.

3.5-3 NUREG-1800, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” U. S. Nuclear Regulatory Commission, April 2001.

TABLE 3.5-1 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
Containment				
1. Penetration sleeves, penetration bellows, and dis-similar metal welds	Cumulative fatigue damage (CLB fatigue analysis exists)	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	A CLB fatigue analysis exists for certain containment penetration bellows. These are evaluated as a TLAA in Section 4.3. This evaluation is consistent with the GALL Report.
2. Penetration sleeves, bellows, and dissimilar metal welds.	Cracking due to cyclic loading, or crack initiation and growth due to SCC	Containment ISI and Containment leak rate test	Yes, detection of aging effects is to be evaluated	SCC is not an applicable aging mechanism for sleeves/bellows. The RNP AMR requires both high temperature (>140^{o}F) and exposure to an aggressive environment for SCC to be applicable. Also, SCC is not applicable to carbon steel. The sole occurrence of SCC on penetration bellows at RNP involved a stainless steel bellows exposed to chlorides. This was corrected by changes to the penetration design and by replacing the piping insulation with a chloride-free type. Based on RNP experience with SCC, additional methods of detecting aging effects are not warranted. Some penetrations have two bellows or one bellows and one plate. One of these is located inside and one outside Containment. The inside bellows/plate is considered the IWE pressure boundary, and both the ASME Section XI, Subsection IWE Program, and the 10 CFR Part 50, Appendix J Program are applicable. In these cases, the outside bellows/plate is tested by the Appendix J program alone as part of the local penetration pressurization test boundary. Management of cracking for this component/commodity group is consistent with the GALL Report.

TABLE 3.5-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
Containment (continued)				
3. Penetration sleeves, penetration bellows, and dissimilar metal welds	Loss of material due to corrosion	Containment ISI and Containment leak rate test	No	The ASME Section XI, Subsection IWE Program, and the 10 CFR Part 50, Appendix J Program are applicable. In addition to the ISI and containment leak rate testing, the Water Chemistry Program is applied to stainless steel components in this component/commodity group subject to borated, treated water. This is consistent with the GALL Report, as discussed in Item 19 below. Also, the Boric Acid Corrosion Program would be credited to manage loss of material in addition to the IWE and Appendix J Programs, if the corrosion is caused by leakage of borated water onto carbon steel components. Protective coatings are not credited. Some penetrations have two bellows or one bellows and one plate. One of these is located inside and one outside Containment. The inside bellows/plate is considered the IWE pressure boundary, and both the ASME Section XI, Subsection IWE, and 10 CFR Part 50, Appendix J Programs are applicable. In these cases, the outside bellows/plate is tested by the Appendix J program alone as part of the local penetration pressurization test boundary envelope. Management of loss of material for this component/ commodity group is consistent with the GALL Report.

TABLE 3.5-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
Containment (continued)				
4. Personnel airlock and equipment hatch	Loss of material due to corrosion	Containment ISI and Containment leak rate test	No	The ASME Section XI, Subsection IWE Program and the 10 CFR Part 50, Appendix J Program are applicable to this component/commodity group. Protective coatings are not credited. In addition, the Boric Acid Corrosion Program would be credited to manage the degradation, if the corrosion is caused by leakage of borated water onto carbon steel components, Aging management of these components is consistent with the GALL Report.
5. Personnel airlock and equipment hatch	Loss of leak tightness in closed position due to mechanical wear of locks, hinges and closure mechanism	Containment leak rate test and Plant Technical Specifications	No	The RNP AMR applied the aging effect/mechanism of loss of material due to wear to this component/ commodity group. RNP Technical Specifications address operability of the equipment hatch and the personnel airlock. The 10 CFR 50, Appendix J Program is applicable to this equipment and references the Technical Specification requirements. Aging management of these components is consistent with the GALL Report.

TABLE 3.5-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
Containment (continued)				
6. Seals, gaskets, and moisture barriers	Loss of sealant and leakage through containment due to deterioration of joint seals, gaskets, and moisture barriers	Containment ISI and Containment leak rate test	No	The ASME Section XI, Subsection IWE Program and the 10 CFR Part 50, Appendix J Program are applicable. <u>Seals and Gaskets</u> Leak tightness of components that perform a containment pressure boundary function is by means of the 10 CFR Part 50, Appendix J Program consistent with the GALL Report. <u>Moisture Barriers</u> The ASME Section XI, Subsection IWE Program does not inspect inaccessible components. The moisture barrier between the containment liner and concrete floor at elevation 228 feet is included in the IWE program to be inspected whenever the containment liner insulation is removed for maintenance work. As noted in Section 10.0 of the 90-day ISI Summary Report submitted by letter RNP-RA/01-0125, dated 8/10/01, certain inaccessible areas in the Containment were identified which are required to be evaluated because conditions exist in accessible areas that could indicate the presence of or result in degradation to inaccessible areas. These areas include the moisture barrier at elevation 228 feet. These areas have been evaluated to be acceptable until 2005. An inspection, to be performed under the One-Time Inspection Program, will verify the results of the evaluation and identify any aging effects. Aging management of moisture barriers is considered to be consistent with the GALL Report.

TABLE 3.5-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
Containment (continued)				
7. Concrete elements: foundation, walls, dome.	Aging of accessible and inaccessible concrete areas due to leaching of calcium hydroxide, aggressive chemical attack, and corrosion of embedded steel	Containment ISI	Yes, if aging mechanism is significant for inaccessible areas	RNP concrete is not exposed to flowing water, is dense, well cured, has low permeability, and was constructed in accordance with ACI recommendations at the time of construction. Thus, leaching of calcium hydroxide is not applicable to RNP concrete structures. RNP ground-water values for chlorides and sulfates are much less than the threshold values necessary for aggressive chemical attack. However, the aging mechanisms associated with aggressive chemical attack and corrosion of embedded steel are potentially applicable to below-grade concrete structures owing to slightly acidic groundwater. Groundwater pH has a measured range of 3.7 to 6.0 (average of 4.4). The ASME Section XI, Subsection IWL Program is applicable to the Containment structure. However, RNP will enhance the inspection requirements to apply a special inspection provision for monitoring aging effects potentially caused by aggressive chemical attack and corrosion of embedded steel. This involves inspecting the condition of below grade concrete that is exposed during excavation. These aging management activities are consistent with the GALL Report.
8. Concrete elements: foundation	Cracks, distortion, and increases in component stress level due to settlement	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program	The RNP AMR determined that cracking due to settlement is not applicable. Monitoring for settlement was performed during construction of the plant. Based on the results of the monitoring program and 30 years of operating experience, settlement is not an applicable aging mechanism. A dewatering system is not used.

TABLE 3.5-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
Containment (continued)				
9. Concrete elements: foundation	Reduction in foundation strength due to erosion of porous concrete subfoundation	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program	Not applicable. The RNP AMR for concrete determined that RNP concrete foundations are not constructed of porous concrete and, therefore, are not susceptible to this aging mechanism.
10. Concrete elements: foundation, dome, and wall	Reduction of strength and modulus due to elevated temperature	Plant specific	Yes, for any portions of concrete containment that exceed specified temperature limits	Generally, RNP concrete elements do not experience temperatures that exceed the temperature limits associated with aging degradation due to elevated temperature. During an accident, uninsulated concrete may experience a temperature greater than $200^{o}F$ for less than 10 seconds, but this was considered to have minimal effects. Therefore, this aging effect is not applicable. However, a TLAA was evaluated to demonstrate the continuing capability of one containment penetration when subject to temperature cycles that exceed $200^{o}F$ in adjacent concrete. Refer to Section 4.6.
11. Prestressed containment: tendons and anchorage components	Loss of prestress due to relaxation, shrinkage, creep, and elevated temperature	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	Loss of prestress of the Containment structure post-tensioning system has been evaluated as a TLAA in Section 4.5. This evaluation is consistent with the GALL Report.

TABLE 3.5-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
Containment (continued)				
12. Steel elements: liner plate, containment shell	Loss of material due to corrosion in accessible and inaccessible areas	Containment ISI and Containment leak rate test	Yes, if corrosion is significant for inaccessible areas	Certain inaccessible areas in the Containment were identified which are required to be evaluated because conditions exist in accessible areas that could indicate the presence of or result in degradation to inaccessible areas. These areas include the containment liner plate at elevation 228 feet and the containment liner plate beneath the concrete floor below 228 feet. As noted in the 90-day ISI Summary Report submitted by letter RNP-RA/01-0125, dated 8/10/01, these areas have been evaluated to be acceptable until 2005. A One-Time Inspection Program action has been identified to verify the results of the evaluation and to manage any aging effects at these locations. At that time, the GALL-recommended AMPs will continue to manage the aging effects. This is consistent with the GALL Report. In addition, if the corrosion is caused by leakage of borated water onto carbon steel components, the Boric Acid Corrosion Program in addition to the ISI Program would be applied to manage the localized degradation caused by aggressive chemical attack. Therefore, the ASME Section XI, Subsection IWE, the 10 CFR 50, Appendix J, the Boric Acid Corrosion, and One-Time Inspection Programs are used to manage (continued)

TABLE 3.5-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
Containment (continued)				
12. Steel elements: liner plate, containment shell (continued)				corrosion in accessible and inaccessible areas. Aging management for this component/commodity group is consistent with the GALL Report.
13. Steel elements: protected by coating	Loss of material due to corrosion in accessible areas only	Protective coating monitoring and maintenance	No	Not applicable; protective coatings are not credited for aging management.
14. Prestressed containment: tendons and anchorage components	Loss of material due to corrosion of prestressing tendons and anchorage components	Containment ISI	No	Not applicable. The RNP containment tendons are embedded and cannot be accessed for inspection. Inspections of sample surveillance blocks at 5-year and 25-year intervals determined that grouting has proven to be an effective means of preventing corrosion of the tendons and anchorage components.
15. Concrete elements: foundation, dome, and wall	Scaling, cracking, and spalling due to freeze-thaw; expansion and cracking due to reaction with aggregate	Containment ISI	No	Not applicable. Aggregates were selected locally and were in accordance with specifications and materials conforming to ACI and ASTM standards at the time of construction. RNP structures are constructed of a dense, durable mixture of sound coarse aggregate, fine aggregate, cement, water, and admixture. Water/ cement ratios are within the limits provided in ACI 318-71, and air entrainment percentages were within the range prescribed in the GALL Report. Based on the design specifications for the concrete, the RNP AMR determined that the aging mechanisms of freeze-thaw and reaction with aggregates were not applicable to RNP concrete elements.

TABLE 3.5-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
Class I Structures (Note that this section includes Class I and other in-scope structures at RNP.)				
16. All Groups except Group 6: accessible interior/exterior concrete & steel components	All types of aging effects	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program	The Structures Monitoring Program is applied to components/ commodities in this group that have aging effects. <u>Concrete</u> The RNP AMR methodology concluded that above-grade concrete/grout structures have no aging effects. Refer to Table 3.5-2, Item 10. <u>Steel</u> In addition to the Structures Monitoring Program, the Boric Acid Corrosion Program is applicable for corrosion caused by leakage of borated water onto carbon steel components of this component/commodity group. Protective coatings are not credited for aging management of steel components. <u>Lubrite</u> (GALL Section III.A4.2-b) Reactor Pressure Vessel Supports use bearing plates of high strength, hard tool steel instead of Lubrite. Owing to the wear-resistant material used, the low frequency (number of times) of movement, and the slow movement between sliding surfaces, mechanical wear was determined not to be an aging mechanism. Similarly, lock-up due to wear is not considered to be an aging effect at RNP. Also, refer to the discussion in Item 28 below regarding the Reactor Pressure Vessel Supports. (continued)

TABLE 3.5-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
Class I Structures (continued)				
16. All Groups except Group 6: accessible interior/ exterior concrete & steel components (continued)				Based on the above information, aging management of this component/commodity group is consistent with the GALL Report for applicable aging effects/mechanisms.
17. Groups 1-3, 5, 7-9: inaccessible concrete components, such as exterior walls below grade and foundation	Aging of inaccessible concrete areas due to aggressive chemical attack, and corrosion of embedded steel	Plant-specific	Yes, if an aggressive below-grade environment exists	The aging mechanisms associated with aggressive chemical attack and corrosion of embedded steel are applicable only to below-grade concrete/grout structures owing to the slightly acidic pH of groundwater. The Structures Monitoring Program is applicable to these structures. RNP will apply a special, plant-specific inspection provision to monitor aging effects caused by aggressive chemical attack and corrosion of embedded steel for below grade concrete in this component/ commodity group. This will include inspection of below grade concrete and grout that is exposed during excavation. These aging management activities are consistent with the GALL Report.
18. Group 6: all accessible/inaccessible concrete, steel, and earthen components	All types of aging effects, including loss of material due to abrasion, cavitation, and corrosion	Inspection of Water-Control Structures or FERC/US Army Corps of Engineers dam inspections and maintenance	No	Lake Robinson Dam is an earthen structure with water control components fabricated of concrete and carbon steel. RNP applies a Dam Inspection Program that relies on the Recommended Guidelines for Safety Inspection of Dams, which is based on U.S. Army Corps of Engineers dam inspection guidance. Aging management of the Lake Robinson Dam in accordance with this guidance is consistent with the GALL Report.

TABLE 3.5-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
Class I Structures (continued)				
19. Group 5: liners	Crack initiation and growth from SCC and loss of material due to crevice corrosion	Water Chemistry Program and Monitoring of spent fuel pool water level	No	Consistent with the GALL Report, RNP applies the Water Chemistry Program. Monitoring of spent fuel pool water level is required by the RNP Technical Specifications. SCC is not applicable for reactor cavity or spent fuel pool liners. The RNP AMR methodology requires both high temperatures (> 140^{o}F) and exposure to an aggressive environment In order for SCC to be applicable. The normal temperatures in the fuel pool and reactor cavity do not exceed 140^{o}F. The RNP review identified pitting corrosion as an applicable aging mechanism; however, the Water Chemistry Program assures adequate management of pitting as well as crevice corrosion. In addition to the liner, other stainless steel components subject to borated treated water in the spent fuel pool or reactor cavity pool have aging effects managed by the Water Chemistry Program and have been included in this component/commodity group. These additional components include the fuel transfer tube and associated bellows, detector and manway covers, spent fuel racks, and reactor cavity seal ring plate. Aging management of this component commodity group is consistent with the GALL Report.
20. Groups 1-3, 5, 6: all masonry block walls	Cracking due to restraint, shrinkage, creep, and aggressive environment	Masonry Wall	No	The RNP AMR determined that no aging effects are applicable, based on the locations and design of the Masonry Walls at RNP. The locations are not subject to aggressive chemical environments, and the design does not restrain potential expansion or contraction of the walls.

TABLE 3.5-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
Class I Structures (continued)				
21. Groups 1-3, 5, 7-9: foundation	Cracks, distortion, and increases in component stress level due to settlement	Structures Monitoring	No, if within the scope of the applicant's structures monitoring	The RNP AMR determined that cracking due to settlement is not applicable. Monitoring for settlement was performed during construction of the plant. Based on the results of the monitoring program and 30 years of operating experience, settlement is not an applicable aging mechanism. RNP does not rely on a dewatering system.
22. Groups 1-3, 5-9: foundation	Reduction in foundation strength due to erosion of porous concrete subfoundation	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program	Not applicable. The RNP AMR for concrete determined that RNP concrete foundations are not constructed of porous concrete and, therefore, are not susceptible to this aging mechanism.
23. Groups 1-5: concrete	Reduction of strength and modulus due to elevated temperature	Plant-specific	Yes, for any portions of concrete that exceed specified temperature limits	RNP concrete elements in this component/commodity group do not exceed the temperature limits associated with aging degradation due to elevated temperature. Therefore this aging effect is not applicable.

TABLE 3.5-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
Class I Structures (continued)				
24. Groups 7, 8: liners	Crack Initiation and growth due to SCC; Loss of material due to crevice corrosion	Plant-specific	Yes	Not applicable. The RNP plant does not include tanks with liners. Aging management of in-scope tanks is addressed together with the systems in which the tanks are located in Sections 3.1 through 3.4.
Component Supports				
25. All Groups: support members: anchor bolts, concrete surrounding anchor bolts, welds, grout pad, bolted connections, etc.	Aging of component supports	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program	The Structures Monitoring Program is applicable to all components in this component/commodity group. This is consistent with the GALL Report. As noted in Appendix B, the Structures Monitoring Program will be enhanced. One enhancement is to assure that additional concrete structures that provide support to component support members are included in required monitoring. Carbon steel parts of slide bearing plates used for non-ASME components are included in this group. Aging management for this component/commodity group is consistent with the GALL Report.

TABLE 3.5-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
Component Supports (continued)				
26. Groups B1.1, B1.2, and B1.3: support members: anchor bolts, welds	Cumulative fatigue damage (CLB fatigue analysis exists)	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	Not applicable. No TLAAs exist for this component/ commodity group within the RNP current licensing basis.
27. All Groups: support members: anchor bolts, welds	Loss of material due to boric acid corrosion	Boric acid corrosion	No	The Boric Acid Corrosion Program is applicable to all RNP components in this component/commodity group. Aging management of this group is consistent with the GALL Report.
28. Groups B1.1, B1.2, and B1.3: support members: anchor bolts, welds, spring hangers, guides, stops, and vibration isolators	Loss of material due to environmental corrosion; loss of mechanical function due to corrosion, distortion, dirt, overload, etc.	ISI	No	RNP has no supports in the B1.3 Group (Class MC). Management of the aging effects for component supports in the B1.1 and B1.2 Groups is accomplished by the ASME Section XI, Subsection IWF Program, as recommended in the GALL Report. The RNP IWF Program determines if supports are functional and, if not, corrections are made prior to returning the supports to service. Corrosion, clearances, scoring, misalign-ment, and settings are examined. Aging management for supports that are in-scope for license renewal, but not addressed by the IWF Program, is addressed by the Structures Monitoring Program, as discussed in Item 25. The Reactor Vessel nozzle supports are inaccessible and not currently inspected under the RNP ASME Section XI, Subsection IWF Program. Therefore, an inspection, under the One-Time Inspection Program, will (continued)

TABLE 3.5-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
Component Supports (continued)				
28. Groups B1.1, B1.2, and B1.3: support members: anchor bolts, welds, spring hangers, guides, stops, and vibration isolators (continued)				be performed to verify effective management of potential environmental corrosion of the supports. Based on the above, management of aging effects for applicable supports in this component/commodity group is consistent with the GALL Report.
29. Group B1.1: high strength low-alloy bolts	Crack initiation and growth due to SCC	Bolting integrity	No	The RNP AMR, which included operating experience, determined that SCC is not an applicable aging mechanism for RNP bolting. In general, high strength structural bolting, i.e., bolting with specified yield strength >150 ksi, is not being used; and, for the one case where high strength bolts have been installed, the environment experienced by the bolts is considered benign with respect to SCC, i.e., the bolts are located in a dry environment high up on the steam generator above any source of leakage and, therefore, not exposed to an aggressive or aqueous environment. Based on these results, no AMP is required to manage cracking of bolting due to SCC.

Note: 1. Numbered Component/Commodity Groups consist of the components listed in NRC Standard Review Plan for License Renewal, NUREG-1800, Table 3.5-1, that are applicable to a PWR.

TABLE 3.5-2 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
1. Cable tray and conduit, Damper Mounting, Electrical and Instrument Panels and Enclosures, Doors, Equipment Supports, Expansion Anchors, Anchorage/ Embedments (exposed surfaces), Threaded Fasteners	Carbon Steel	Indoor – Air Conditioned	None	None Required	The RNP AMR determined that carbon steel, stainless steel, and galvanized steel would experience no aging effects requiring management when subjected to an air-conditioned environment.

TABLE 3.5-2 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
2. Cable tray and conduit, Electrical and Instrument Panels and Enclosures, Misc. Steel, Anchorage/ Embedments (exposed surfaces), Electrical Component Supports, Siding	Galvanized Steel	Indoor – Air Conditioned, Indoor – Not Air Conditioned, Containment Air, Borated Water Leaks	None	None Required	The GALL Report does not address this material for structures and supports. The RNP AMR determined that galvanized steel would experience no age related degradation in these environments.
3. Cable tray and conduit, Electrical & Instrument Panels and Enclosures, Misc. Steel, Structural Steel, Electrical Component Supports	Galvanized Steel	Outdoor	Loss of Material from General Corrosion	Structures Monitoring Program	The GALL Report does not address this material for structures and supports. Aging management will be via the Structures Monitoring Program. Use of the Structures Monitoring Program is in accordance with the GALL Report for structural (non-galvanized) steel. (For example, see GALL, Section III.A8.2-a.) Therefore, the application of this AMP to galvanized steel components is conservative.

TABLE 3.5-2 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
4. Reinforced Concrete (Containment cylinder wall dome, basemat)	Concrete	Indoor – Not Air Conditioned	Cracking and Change in Material Properties from Fatigue	ASME Section XI Subsection IWL Program	This aging mechanism is not defined in the GALL Report; it is applicable to concrete at Containment penetrations where pipe reaction forces are imposed on the penetration. The current ASME Section XI, Subsection IWL activities ensure concrete cracking and change in material properties due to fatigue are monitored.
		Outdoor	Cracking and Change in Material Properties from Fatigue	ASME Section XI Subsection IWL Program	
5. Electrical and Instrument Panels and Enclosures, Expansion Anchors, Siding, Louvers	Stainless Steel, Aluminum	Outdoor (Stainless Steel and Aluminum) Indoor – Not Air Conditioned (Aluminum)	Loss of Material from Crevice Corrosion	Structures Monitoring Program	These combinations of materials and aging mechanisms were not addressed in the GALL Report. The RNP AMR methodology determined that these aging mechanisms were applicable to aluminum and stainless steel components in the outdoor environment and to aluminum components in an indoor-not air conditioned environment. The Structures Monitoring Program has been applied and will assure management of these aging effects/mechanisms for aluminum and stainless steel.
			Loss of Material from Pitting Corrosion	Structures Monitoring Program	
6. Foundation Pilings	Carbon Steel	Buried	None	TLAA	The GALL Report does not address this component. The corrosion rate for the period of service has been evaluated as a TLAA. Refer to Section 4.6.

TABLE 3.5-2 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
7. Seismic Joint Filler	Elastomers	Indoor – Not Air Conditioned	Change in Material Properties from Elevated Temperature	Structures Monitoring Program	This component/material is not addressed in the GALL Report. The Structures Monitoring Program has been applied to manage the age related degradation of joint filler. The Structures Monitoring Program will effectively manage these aging effects/mechanism by visually determining the condition of the elastomeric material.
			Cracking from Elevated Temperature	Structures Monitoring Program	
8. Roof (Membrane or Built Up)	Elastomers	Outdoor	Change in Material Properties from Elevated Temperature	Structures Monitoring Program	This component was not addressed in the GALL Report. RNP applies the Structures Monitoring Program to manage age-related degradation. The Structures Monitoring Program will effectively manage these aging effects/mechanism by visually determining the condition of the elastomeric material.
			Cracking from Elevated Temperature	Structures Monitoring Program	
9. Penetration Sleeves, Liner Plate, Airlock and Hatch Penetrations, Anchorages/ Embedments, Floor Drains, Grouted Tendons	Carbon Steel, Stainless Steel, Galvanized steel	Embedded/ Encased in Concrete	See discussion	None Required	The RNP AMR determined that carbon, stainless, and galvanized steel components would experience no loss of material requiring management when completely encased in concrete.

TABLE 3.5-2 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
10. Reinforced Concrete (Contain. cyl. wall, dome, and basemat; columns, beams, walls, floors, pads, slabs, curbs, plugs, grout, etc.); Concrete Sump, Tank Foundation; Electrical Manhole	Concrete/ Grout	Containment Air, Indoor – Not Air Conditioned, Outdoor	See discussion	None Required	The RNP AMR determined that concrete and grout would experience no aging effects requiring management resulting from an aggressive environment in the listed environments. All above grade locations in these environments are considered to be non-aggressive.
11. Bellows; Component Supports, Panel Enclosures, Expansion Anchors, Fuel Transfer Tube, Protective Enclosures, Liner Plate, Mechanical Penetrations, Threaded Fasteners	Stainless Steel, Galvanized Steel	Borated Water Leaks	See discussion	None Required	The RNP AMR determined that stainless steel and galvanized steel components would experience no aging effects requiring management when subject to a boric acid leakage environment.

TABLE 3.5-2 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
12. Bellows, Cavity Seal Ring Plate, Containment Liner Plate, Reactor Cavity Liner Plate, Panels and Enclosures, Component Supports, Expansion Anchors, Fuel Transfer Tube/ Blind Flange, Mechanical Penetrations, Protective Enclosures, Manway Covers, NIS Detector Covers, Threaded Fasteners	Stainless Steel	Indoor – Air Conditioned, Indoor – Not Air Conditioned, Containment Air	See discussion	None Required	The RNP AMR determined that stainless steel components would experience no loss of material due to corrosion when subject to these environments.

TABLE 3.5-2 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
13. Slide Bearing Plates, Threaded Fasteners	Copper Alloys	Containment Air, Indoor, - Not Air Conditioned, Borated Water Leaks	See discussion	None Required	The slide bearing plate consists of a bronze material impregnated with "Lubrite." Manganese bronze threaded fasteners may be employed on the fuel transfer tube blind flange. The RNP AMR determined that copper alloy components would experience no aging effects requiring management when subject to these environments.
14. Slide Bearing Plate	Miscel-laneous	Containment Air, Borated Water Leaks	See discussion	None Required	The miscellaneous material is "Lubrite," a graphitic material embedded in bronze plates. The RNP AMR determined that the slide bearing plate material would experience no aging effects requiring management when subject to this environment.
15. Containment Liner and Penetration Insulation	Miscel-laneous	Containment Air, Indoor – Not Air Conditioned, Outdoor	See discussion	None Required	Containment liner insulation consists of PVC or Polyimide foam panels enclosed in stainless steel sheathing. Penetration insulation is fabricated from high density, BTU-BLOCL (a Johns-Manville product), fiberglass blankets, and ceramic fiber. The RNP AMR determined that the insulation material would experience no aging effects requiring management when subject to these environments.

TABLE 3.5-2 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
16. Control Room Ceiling and Control and Cable Spreading Room Raised Floors	Miscel-laneous	Indoor – Air Conditioned	See discussion	None Required	Miscellaneous materials for the control room ceiling are standard suspended ceiling components: carbon steel structural members and ceiling tile. Materials for raised floors consist of carbon steel structural support and floor tile. The RNP AMR concluded that these materials have no aging effects in an indoor – air-conditioned environment.

Note: 1. Environments used in the aging management review are listed on Tables 3.0-1 and 3.0-2.

3.6 AGING MANAGEMENT OF ELECTRICAL AND INSTRUMENTATION AND CONTROLS

The electrical/I&C component commodity groups subject to an aging management review are:

1. Bus Duct supporting Emergency Buses E1 and E2 and the Dedicated Shutdown System Bus

2. Insulated Cables and Connections (including splices, connectors, and terminal blocks) not included in the EQ Program

3. Electrical/I&C penetration assemblies not included in the EQ Program

3.6.1 AGING MANAGEMENT REVIEW

3.6.1.1 Methodology

The methodology used for aging management review employs the "plant spaces" approach in which the plant is segregated into areas (or spaces) where common bounding environmental parameters can be assigned. Each bounding environmental parameter is evaluated against the most-limiting (worst-case) material in the area to determine if the components will be able to maintain their intended functions through the period of extended operation.

The Department of Energy (DOE), "Aging Management Guideline for Commercial Nuclear Power Plants – Electrical Cable and Terminations," (the Cable AMG) [Reference 3.6-1], was used to identify aging effects for all electrical commodity groups within the scope of this review. As discussed in the Cable AMG, potential aging effects are based upon materials of construction and their exposure to environmental stressors, such as heat, radiation, and moisture.

The aging management review identifies one or more aging management programs to be used to demonstrate that the effects of aging will be managed to assure the intended functions will be maintained consistent with CLB for the period of extended operation. The programs to be used for managing the effects of aging were compared to those listed in the GALL Report [Reference 3.6-2] and evaluated for consistency with GALL Report programs that are relied on for license renewal. The results are documented and discussed in Subsection 3.6.2 using the format suggested by the NRC Standard Review Plan for License Renewal (SRP-LR) [Reference 3.6-3]. Aging management programs are described in Appendix B.

Based on a review of potential aging effects using the Cable AMG, the following stressors and aging effects were identified:

Applicable Stressor	Voltage Category[1]	Applicability	Potential Aging Effects
Heat, oxygen	Low & Medium	All insulation materials	Reduced insulation resistance (IR); electrical failure
Radiation, oxygen	Low & Medium	All insulation materials	Reduced IR; electrical failure
Moisture and voltage stress	Medium	All insulation materials exposed to standing water	Electrical failure (caused by a breakdown of the insulation)

Notes: 1. Low- and medium-voltage values are defined in Section 1.2 of Reference 3.6-1.

3.6.1.2 Operating Experience

Operating experience (OE) through December 2001 was considered during the development of the RNP Integrated Plant Assessment. OE subsequent to that date will be reviewed and applicable OE will be updated in conjunction with the amendment to the application required by 10 CFR 54.21(b). The review consisted of the following:

Site: RNP site-specific operating experience was reviewed. The site-specific operating experience included a review of (1) Corrective Action Program, (2) Licensee Event Reports, (3) Maintenance Rule Data Base, and (4) interviews with Systems Engineers. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Subsection.

Industry: An evaluation of industry operating experience published since the effective date of the GALL Report was performed to identify any additional aging effects requiring management. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Subsection.

On-Going On-going review of plant-specific and industry operating experience is performed in accordance with the Corrective Action and Operating Experience Programs.

3.6.2 AGING MANAGEMENT PROGRAMS

3.6.2.1 Aging Management Programs Evaluated in the GALL Report that Are Relied on for License Renewal

Table 3.6-1 shows the electrical/I&C component commodity groups and aging management programs evaluated in the GALL Report [Reference 3.6-2] that are relied on for license renewal.

The components of Non-EQ Electrical Penetration Assemblies subject to aging management review are the organic insulating materials associated with electrical conductors and connections. Therefore, the Non-EQ Electrical Penetration Assemblies are included with the electrical cables and connections not subject to 10 CFR 50.49 EQ requirements on Table 3.6-1. Considering cable systems to include penetration assemblies is consistent with program description XI.E1, Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements, in the GALL Report.

3.6.2.2 Further Evaluation of Aging Management as Recommended by the GALL Report

The GALL Report does not contain any issues for further evaluation for electrical/I&C components with the exception of addressing the TLAA aspects of electrical equipment subject to EQ. EQ-related TLAAs are addressed in Section 4.4.

3.6.2.3 Aging Management Evaluations that Are Different from or Not Addressed in the GALL Report

Aging Management Evaluations that are different from or not addressed in the GALL Report are identified and discussed on Table 3.6-2. This difference involves Bus Ducts, which are not addressed in the GALL Report.

3.6.3 CONCLUSION

Electrical/I&C component aging effects requiring management are adequately managed by the following programs:

1. Non-EQ Insulated Cables and Connections Program

2. Boric Acid Corrosion Program

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that the intended functions of Electrical/I&C components are maintained consistent with the current licensing basis for the period of extended operation.

3.6.4 REFERENCES

3.6-1 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.

3.6-2 NUREG-1801, “Generic Aging Lessons Learned (GALL),” U. S. Nuclear Regulatory Commission, April 2001.

3.6-3 NUREG-1800, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” U. S. Nuclear Regulatory Commission, April 2001.

TABLE 3.6-1 ELECTRICAL/I&C AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
1. Electrical equipment subject to 10 CFR 50.49 environmental qualification (EQ) requirements	Degradation due to various aging mechanisms	Environmental qualification of electric components	Yes, TLAA	TLAAs for electrical equipment in the EQ Program are discussed in Section 4.4
2. Electrical cables and connections not subject to 10 CFR 50.49 EQ requirements	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance (IR); electrical failure caused by thermal/ thermoxidative degradation of organics; radiolysis and photolysis (ultraviolet [UV] sensitive materials only) of organics; radiation-induced oxidation; moisture intrusion	Aging management program for electrical cables and connections not subject to 10 CFR 50.49 EQ requirements	No	RNP applies the Non-EQ Insulated Cables and Connections Aging Management Program. Based on an evaluation of insulating materials, the RNP AMR concluded that only PVC insulated cables were required to be subject to aging management. However, the scope of the Non-EQ Insulated Cables and Connections Program will include other accessible non-EQ insulated cables and connections within the scope of license renewal, not only those installed in an adverse, localized environment caused by heat or radiation. Management of aging effects for Non-EQ Insulated Cables and Connections is consistent with the GALL Report. This program also addresses accessible non-EQ cables used in instrumentation circuits. Use of the Non-EQ Insulated Cables and Connections Aging Management Program for this purpose is discussed below.

TABLE 3.6-1 (continued) ELECTRICAL/I&C AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
3. Electrical cables used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements that are sensitive to reduction in conductor insulation resistance (IR)	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced IR; electrical failure caused by thermal/ thermoxidative degradation of organics; radiation-induced oxidation; moisture intrusion	Aging management program for electrical cables used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements	No	The GALL Report recommends an AMP specifically for cables with sensitive, low-level signals. RNP applies the previously mentioned Non-EQ Insulated Cables and Connections Aging Management Program. Additional information is provided in Table 3.6-2, Item 3.
4. Inaccessible medium-voltage (2 kV to 15 kV) cables (e.g., installed in conduit or direct buried) not subject to 10 CFR 50.49 EQ requirements	Formation of water trees, localized damage leading to electrical failure (breakdown of insulation); water trees caused by moisture intrusion	Aging management program for inaccessible medium-voltage cables not subject to 10 CFR 50.49 EQ requirements	No	No AMP is required. After evaluation of potential medium voltage circuits, it has been determined that no medium voltage cables, that are potentially susceptible to wetting, provide any license renewal intended functions. Therefore, no aging management activities are required.

TABLE 3.6-1 (continued) ELECTRICAL/I&C AGING MANAGEMENT PROGRAMS EVALUATED IN THE GALL REPORT THAT ARE RELIED ON FOR LICENSE RENEWAL

Component/ Commodity Group (1)	Aging Effect/ Mechanism	Aging Management Program	GALL Further Evaluation Recommended	Discussion
5. Electrical connectors not subject to 10 CFR 50.49 EQ requirements that are exposed to borated water leakage	Corrosion of connector contact surfaces caused by intrusion of borated water	Boric acid corrosion	No	The Boric Acid Corrosion Program is applicable to items in this component/commodity group based on information from the GALL Report. The RNP AMR did not identify this aging effect, and crediting this program is not based on evidence that boric acid corrosion is occurring. Nevertheless, the scope of the program would address boric acid leakage onto these electrical components.

Note: 1. Numbered Component/Commodity Groups consist of the components listed in NRC Standard Review Plan for License Renewal, NUREG-1800, Table 3.6-1, that are applicable to a PWR.

TABLE 3.6-2 ELECTRICAL/I&C AGING MANAGEMENT EVALUATIONS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN THE GALL REPORT

Component Commodity	Material	Environment (1)	Aging Effect/ Mechanism	Aging Management Program	Discussion
1. Bus Duct Assembly (insulated copper bus bars, bus bar insulated supports, connections)	Various Organic Polymers, Fiberglass	Indoor – Air Conditioned Indoor – Not Air Conditioned Outdoor [2] Ohmic heating	None	None Required	A bus duct provides a means of connecting electrical power between equipment utilizing a pre-assembled, metal-enclosed raceway with conductors installed on insulated supports. Bus ducts were not evaluated in the GALL Report. Based on the RNP AMR, no applicable aging effects were identified for the bus duct. Therefore, it is concluded that no aging management activities are required for the extended period of operation.
2. Electrical cables used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements that are sensitive to reduction in conductor insulation resistance (IR)	Various Organic Polymers	Containment Air, Indoor – Not Air Conditioned. Indoor – Air Conditioned, Outdoor (Ohmic heating is not applicable)	See Aging Effect/Mechanism for Electrical cables and connections not subject to 10 CFR 50.49 EQ requirements in Table 3.6-1, Item 2	Non-EQ Insulated Cables and Connections Aging Management Program	The GALL Report contains an AMP specifically for cables with sensitive, low-level signals. RNP applies the Non-EQ Insulated Cables and Connections Aging Management Program (Table 3.6-1, Item 2). The inspection required by this program would be effective in identifying visual indications of insulation deterioration caused by environmental conditions, e.g., embrittlement, cracking, melting, discoloration, and swelling. This is considered to be a preferred alternative to the AMP identified in the GALL Report.

Notes: 1. Environments used in the aging management review are listed on Tables 3.0-1 and 3.0-2. All environments are external except ohmic heating, which is considered an internal environment.

2. Bus Duct exposed to an outdoor environment is totally enclosed and supplied with breather elbows that allow air, but not rain, to enter; therefore, the conductors and conductor supports effectively are exposed to indoor – not air conditioned parameters.

4.0 TIME-LIMITED AGING ANALYSES

Two areas of technical review are required to support an application for a renewed operating license. The first area of technical review is the Integrated Plant Assessment, described in Chapters 2 and 3. The second area of technical review is the identification and evaluation of plant-specific time-limited aging analyses and exemptions, provided in this chapter.

The evaluations included in this chapter meet the requirements contained in 10 CFR 54.21(c) and allow the NRC to make the finding contained in 10 CFR 54.29(a)(2).

4.1 IDENTIFICATION OF TIME-LIMITED AGING ANALYSES

10 CFR 54.21(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 10 CFR 54.3 as those licensee calculations an analyses that:

1. Involve systems, structures, and components within the scope of license renewal, as delineated in 10 CFR 54.4(a);

2. Consider the effects of aging;

3. Involve time-limited assumptions defined by the current operating term, for example, 40 years;

4. Were determined to be relevant by the licensee in making a safety determination;

5. Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in 10 CFR 54.4(b); and

6. Are contained or incorporated by reference in the current licensing basis.

4.1.1 TIME-LIMITED AGING ANALYSES IDENTIFICATION PROCESS

The process used to identify the RNP-specific time-limited aging analyses is consistent with the guidance provided in NEI 95-10, "Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule," [Reference 4.1-1]. Calculations and evaluations that meet the six criteria of 10 CFR 54.3 were identified by searching the Technical Specifications, UFSAR, Environmental Reports, and docketed licensing correspondence. Additionally, Westinghouse Commercial Atomic Power (WCAP) reports that contained potential TLAA analyses were identified. Industry-prepared documents that list generic time-limited aging analyses also were reviewed to provide additional assurance of the completeness of the plant-specific list. These documents included the Standard Review Plan, NEI 95-10, and Westinghouse Owners Group topical reports. The calculations and evaluations that meet all six criteria of 10 CFR 54.3 are the time-limited aging analyses for RNP and are listed in Table 4.1-1.

As required by 10 CFR 54.21(c)(1), an evaluation of RNP-specific time-limited aging analyses must be performed to demonstrate that:

(i) The analyses remain valid for the period of extended operation;

(ii) The analyses have been projected to the end of the period of extended operation; or

(iii) The effects of aging on the intended functions(s) will be adequately managed for the period of extended operation.

The results of these evaluations are provided in Table 4.1-1 and discussed in Sections 4.2 through 4.6.

4.1.2 IDENTIFICATION OF EXEMPTIONS

The requirements of 10 CFR 54.21(c) also requires that the application for a renewed license include a list of plant-specific exemptions granted pursuant to 10 CFR 50.12 and in effect that are based on time-limited aging analyses as defined in 10 CFR 54.3. This was performed by evaluating the basis for each active exemption, granted pursuant to 10 CFR 50.12, to determine whether the exemption was based on a time-limited aging analysis. None of the active 10 CFR 50.12 exemptions identified for RNP involve a time-limited aging analysis as defined in 10 CFR 54.3.

4.1.3 REFERENCES

4.1-1 NEI 95-10, “Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule,” Revision 3, March 2001.

TABLE 4.1-1 - TIME-LIMITED AGING ANALYSES

TLAA Category	Analysis	Resolution [10 CFR 54.21 (c)(1) Section]	Section
Reactor Vessel			
Neutron Embrittlement	Pressurized Thermal Shock	10 CFR 54.21(c)(1)(ii)	4.2.1
	Upper Shelf Energy	10 CFR 54.21(c)(1)(ii)	4.2.2
Metal Fatigue	Fatigue Analysis (Design)	10 CFR 54.21(c)(1)(i)	4.3.1
	Reactor Vessel Underclad Cracking	10 CFR 54.21(c)(1)(ii)	4.3.4
RV Internals			
Metal Fatigue	Thermal Fatigue of Reactor Internals Holddown Springs and Vessel Alignment Pins	10 CFR 54.21(c)(1)(i)	4.3.1
Pressurizer			
Metal Fatigue	Fatigue Analysis (Design)	10 CFR 54.21(c)(1)(i)	4.3.1
	Pressurizer Insurge/Outsurge	10 CFR 54.21(c)(1)(i)	4.3.1
Reactor Coolant Pumps			
Metal Fatigue	Fatigue Analysis (Design)	10 CFR 54.21(c)(1)(i)	4.3.1
Thermal Aging Embrittlement	Code Case N-481 Fracture Mechanics Analysis	10 CFR 54.21(c)(1)(ii)	4.6.1
RCS Piping			
Metal Fatigue	Fatigue Analysis (Design)	10 CFR 54.21(c)(1)(i)	4.3.2
	Pressurizer Surge Line Thermal Stratification (Bulletin 88-11)	10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii) (for environment assisted fatigue)	4.3.1 4.3.3
Thermal Aging Embrittlement	Primary Loop Leak-Before-Break Analysis	10 CFR 54.21(c)(1)(ii)	4.6.1
Steam Generators			
Metal Fatigue	Fatigue Analysis (Design)	10 CFR 54.21(c)(1)(i)	4.3.1
Secondary Systems			
Metal Fatigue	Auxiliary Feedwater Line Fatigue Analysis	10 CFR 54.21(c)(1)(i) (pending)	4.3.1
Containment Structure			
Containment Tendon Loss of Prestress	Containment Tendon Stress Relaxation	10 CFR 54.21(c)(1)(ii)	4.5
Penetration Bellows Fatigue	Penetration Mechanical Bellows Fatigue Analysis	10 CFR 54.21(c)(1)(i)	4.3.5
Concrete Temperature Cycles	Elimination of Containment Penetration Coolers	10 CFR 54.21(c)(1)(i)	4.6.3
Pile Corrosion	Foundation Piles	10 CFR 54.21(c)(1)(ii)	4.6.2
High Density Fuel Racks			
Depletion of Neutron Absorber	Boraflex Depletion Allowance	10 CFR 54.21(c)(1)(iii)	4.6.4

TABLE 4.1-1 (continued)
TIME-LIMITED AGING ANALYSES

TLAA Category	Analysis	Resolution [10 CFR 54.21 (c)(1) Section]	Section
Cranes			
Mechanical Fatigue	Crane Fatigue (Polar Crane and Spent Fuel Cask Crane)	10 CFR 54.21(c)(1)(ii)	4.3.6
Environmental Qualification			
Qualified Life	ASCO NP8316 and NP8321 Series Solenoid Valves	10 CFR 54.21(c)(1)(ii)	4.4.1.1
	ASCO Solenoid Valves – AQR Report	10 CFR 54.21(c)(1)(ii)	4.4.1.2
	Limitorque SMB MOV Actuators – Outside Containment	10 CFR 54.21(c)(1)(iii)	4.4.1.3
	Limitorque SB-3 and SMB-00 MOV Actuators – Inside Containment	10 CFR 54.21(c)(1)(ii)	4.4.1.4
	Rockbestos Cable – Firewall III	10 CFR 54.21(c)(1)(ii)	4.4.1.5
	Rockbestos – RSS-6-104/LE Series Coaxial Cable	10 CFR 54.21(c)(1)(ii)	4.4.1.6
	Rockbestos Cable – Firezone R	10 CFR 54.21(c)(1)(ii)	4.4.1.7
	GEMS Liquid Level Transmitters – Model XM-54853 & XM-54854	10 CFR 54.21(c)(1)(ii)	4.4.1.8
	B&W Valve Monitoring System	10 CFR 54.21(c)(1)(ii)	4.4.1.9
	Westinghouse Reactor Containment Fan Cooler Motors	10 CFR 54.21(c)(1)(ii)	4.4.1.10
	Westinghouse Motors – Frame 506UPZ, 509US, and SBDP – RHR, SI Pumps, HVA 6A, 6B, 8A, & 8B	10 CFR 54.21(c)(1)(ii)	4.4.1.11
	Westinghouse Motors – Model S068C20085 – Containment Spray Pumps	10 CFR 54.21(c)(1)(ii)	4.4.1.12
	Crouse-Hinds Electrical Penetration Assemblies	10 CFR 54.21(c)(1)(ii)	4.4.1.13
	Continental Shielded Instrument Cable – CC2115	10 CFR 54.21(c)(1)(ii)	4.4.1.14
	Continental/Anaconda Cable – Instrumentation	10 CFR 54.21(c)(1)(ii)	4.4.1.15
	Samuel Moore Dekoron Instrument Cables (EPDM & XLPO Insulations)	10 CFR 54.21(c)(1)(ii)	4.4.1.16
	Eaton Corporation Dekoron Cable 16 AWG	10 CFR 54.21(c)(1)(ii)	4.4.1.17
	Raychem WCSF-N Splices	10 CFR 54.21(c)(1)(ii)	4.4.1.18
	Raychem Splices – NPKV Stub Kits	10 CFR 54.21(c)(1)(ii)	4.4.1.19
	Raychem Splices – NPK Connection Kits	10 CFR 54.21(c)(1)(ii)	4.4.1.20

TABLE 4.1-1 (continued)
TIME-LIMITED AGING ANALYSES

TLAA Category	Analysis	Resolution [10 CFR 54.21 (c)(1) Section]	Section
Environmental Qualification			
Qualified Life (continued)	Raychem Splices – NMCK Connection Kits	10 CFR 54.21(c)(1)(ii)	4.4.1.21
	Raychem Splices – NESK End Seal Kits	10 CFR 54.21(c)(1)(ii)	4.4.1.22
	AMP Butt Splices	10 CFR 54.21(c)(1)(ii)	4.4.1.23
	AMP PIDG Terminals	10 CFR 54.21(c)(1)(ii)	4.4.1.24
	CM-303 Tape Splice Assemblies – Scotch 27 and Scotch 70	10 CFR 54.21(c)(1)(ii)	4.4.1.25
	Kerite HTK Power Cable	10 CFR 54.21(c)(1)(ii)	4.4.1.26
	Kerite FR2/FR3 Insulated Multiconductor Cable	10 CFR 54.21(c)(1)(ii)	4.4.1.27
	Thomas & Betts STA-KON Terminal	10 CFR 54.21(c)(1)(ii)	4.4.1.28
	Conax Electric Conductor Seal Assemblies – ECSA	10 CFR 54.21(c)(1)(ii)	4.4.1.29
	Conax Electrical Penetration Assemblies	10 CFR 54.21(c)(1)(ii)	4.4.1.30
	Westinghouse CET/CCM – Incore T/C Connectors and MI Cable Assemblies	10 CFR 54.21(c)(1)(ii)	4.4.1.31
	Westinghouse CET/CCM – Reference Junction Boxes and Potting Adaptors	10 CFR 54.21(c)(1)(ii)	4.4.1.32
	Westinghouse Intermediate Disconnect Box Connectors	10 CFR 54.21(c)(1)(ii)	4.4.1.33
	Gamma-Metrics Excore Neutron Detectors	10 CFR 54.21(c)(1)(ii)	4.4.1.34
	Pyco RTDs	10 CFR 54.21(c)(1)(ii)	4.4.1.35
	Buchanan Terminal Blocks	10 CFR 54.21(c)(1)(ii)	4.4.1.36
	Barton Press Switch – Model 580A	10 CFR 54.21(c)(1)(ii)	4.4.1.37
	NAMCO Connectors–Mod. EC210	10 CFR 54.21(c)(1)(ii)	4.4.1.38
	Victoreen High Range Radiation Detectors	10 CFR 54.21(c)(1)(ii)	4.4.1.39
	Brand Rex Cable - Instrumentation	10 CFR 54.21(c)(1)(ii)	4.4.1.40
	Brand Rex Cable – Control	10 CFR 54.21(c)(1)(ii)	4.4.1.41
	Raychem Cable - Flamtrol	10 CFR 54.21(c)(1)(ii)	4.4.1.42
	Cable – PVC and XLPE Outside Containment	10 CFR 54.21(c)(1)(ii)	4.4.1.43
	Greases – Motors and MOVs	10 CFR 54.21(c)(1)(ii)	4.4.1.44
	Target Rock Solenoid Valves	10 CFR 54.21(c)(1)(ii)	4.4.1.45
	Boston Insulated Wire – Cable	10 CFR 54.21(c)(1)(ii)	4.4.1.46
	Honeywell Model V4-21 Microswitch Assembly	10 CFR 54.21(c)(1)(ii)	4.4.1.47
	Ram-Q Connectors	10 CFR 54.21(c)(1)(ii)	4.4.1.48

4.2 REACTOR VESSEL NEUTRON EMBRITTLEMENT

The NRC has established regulations to address the implications of accumulated neutron irradiation on the structural integrity of reactor pressure vessels in the commercial nuclear industry. These regulations include:

10 CFR 50.60, Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation;

10 CFR 50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events;

10 CFR 50, Appendix G, Fracture Toughness Requirements; and

10 CFR 50, Appendix H, Reactor Vessel Material Surveillance Requirements.

10 CFR 50.60 requires licensees to comply with the reactor coolant pressure boundary requirements from 10 CFR 50, Appendix G and with the requirements from 10 CFR 50 Appendix H for surveillance of reactor pressure vessel (RPV) materials. Both 10 CFR 50.61 and 10 CFR 50, Appendix G, establish limits on embrittlement of the RPV resulting from neutron irradiation. 10 CFR 50, Appendix H establishes the requirements for developing plant-specific RPV surveillance data that are used for structural integrity assessments required by 10 CFR 50.61 and 10 CFR 50, Appendix G.

NUREG-1801 [Reference 4.2-1] also requires an evaluation of the inlet and outlet nozzles and safety injection nozzle, if applicable, to determine if they should be added to the reactor vessel surveillance program scope. If the 60-year fluence level for the nozzles is determined to be below 10^{17} n/cm^2 or the nozzle materials are shown to be not controlling, these components need not be added to the surveillance program for future monitoring. Nozzle materials were evaluated and determined not to be controlling based on fracture toughness analyses described below. For RNP, the safety injection nozzles do not attach to the reactor vessel.

The potential for neutron embrittlement of reactor vessel components was recognized at RNP during the 1970s, and corrective measures were taken in the design and management of the nuclear fuel to minimize the number and distribution of neutrons reaching sensitive locations of the reactor vessel. RNP implemented two methods that have resulted in a reduction in neutron leakage from the core by approximately a factor of 10 from the original levels. The first method implemented was the use of low-leakage fuel loading patterns, in which previously burned fuel or low enrichment fuel is located about the perimeter of the core to absorb and reflect neutrons back toward the core, in effect shielding the vessel from the remainder of the core. The second method implemented was the use of Part Length Shield Assemblies in the fuel bundles immediately adjacent to the reactor vessel. These fuel assemblies have stainless steel rods substituted for the fuel in the vertical portions of the fuel bundle nearest the welds of the reactor vessel, locally reducing neutron leakage to the areas most susceptible to neutron embrittlement. Natural uranium is also substituted for enriched uranium in the upper portion of these fuel assemblies to further reduce neutron leakage. The result of having taken these measures relatively early in the life of the plant is that the neutron fluence and neutron embrittlement have been greatly reduced, extending the useful life of the reactor vessel significantly.

4.2.1 PRESSURIZED THERMAL SHOCK

10 CFR 50.61 defines screening criteria for embrittlement of RPV materials in pressurized-water reactors, as well as actions that are required if these screening criteria are exceeded. The screening criteria limit the degree that a vessel material may increase in its reference temperature RT_{PTS}, following neutron irradiation of the RPV. For circumferential welds, the pressurized thermal shock (PTS) screening criterion is 300°F, maximum. For plates, forgings, and axial weld materials, the screening criterion is 270°F, maximum. The projected end-of-license (EOL) RT_{PTS} values must be shown to remain below the applicable screening temperature.

The calculated RT_{PTS} temperatures for reactor vessel beltline materials, including axial welds, circumferential welds and plates, have been demonstrated to remain below the applicable PTS screening criteria throughout the 60-year license renewal period. The limiting location is circumferential Weld 10-273, which has a 60-year RT_{PTS} reference temperature more than 25°F below the screening criteria (60-year RT_{PTS} = 275 °F vs. 300 °F, maximum for circumferential welds). These RT_{PTS} values were calculated using the methodology from 10 CFR 50.61.

Conservative 60-year RT_{PTS} reference temperatures were also calculated for the reactor vessel inlet and outlet nozzles and welds. The highest 60-year RT_{PTS} reference temperature for the nozzles was 35°F below the screening criteria (60-year RT_{PTS} = 235°F vs. 270°F, maximum for plates, forgings, and axial welds). Therefore, the nozzles and nozzle welds have been shown to meet the PTS criteria for 60 years and have been shown not to be the limiting components, since the beltline materials were closer to the limit. Therefore, the inlet and outlet nozzles and welds need not be added to the reactor vessel surveillance program.

The analysis associated with pressurized thermal shock has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

4.2.2 UPPER SHELF ENERGY

10 CFR 50, Appendix G contains screening criteria that limit the degree that the upper shelf energy (USE) value for an RPV material may be allowed to drop due to neutron irradiation exposure. The regulation requires the initial USE for an RPV material to be greater than 75 ft-lbs when the material is in the unirradiated condition, and for the USE to remain above 50 ft-lbs in the fully irradiated condition throughout the licensed life of the vessel, unless it is demonstrated that lower values of energy will provide margins of safety against fracture equivalent to those required by the ASME Code, Section XI, Appendix G.

Upper shelf energy (USE) values were calculated for a 60-year operating period using methodology from 10 CFR 50, Appendix G, and Regulatory Guide 1.99, Rev. 2, [Reference 4.2-2] and the 60-year fluence projections.

For welds and forgings exposed to EOL fluence, the USE screening criterion is 50 ft-lbs minimum. The projected 60-year USE values for reactor beltline welds, both axial and circumferential, were shown to be above the minimum USE screening criteria. The limiting location is Weld 2-273A, with a 60-year USE value of 56 ft-lbs, which is acceptable.

For reactor vessel plate materials, a 42 ft-lbs minimum USE acceptance criterion has been established, based upon WCAP-13587, Rev. 1, [Reference 4.2-3], which demonstrated equivalent margins of safety for RNP vessel plates with USE as low as 42 ft-lbs. The 60-year USE values were calculated for RNP vessel plates. The limiting plate location is Plate W 10201-4, with a 60-year USE value of 45 ft-lbs, which is acceptable.

The nozzle forgings have a 60-year USE value of 53 ft-lbs and the nozzle welds have a 60-year USE value of 52 ft-lbs, compared with the 50 ft-lbs minimum criterion for welds and forgings from 10 CFR 50, Appendix G, which is acceptable.

The analysis associated with upper-shelf energy has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

4.2.3 OTHER ANALYSES

Other analyses impacted by neutron embrittlement, i.e., those for heatup / cooldown curves and Low Temperature Overpressure Protection (LTOP), were determined not to be TLAAs because they are not based upon end-of-license fluence projections. Instead, these analyses are periodically updated as required by regulations based upon fluence projections that bound the current period of operation, but this period is not necessarily associated with the end-of-license. The analyses are also updated whenever new information is available that would significantly affect the projections, either from the Reactor Vessel Surveillance Program or from other industry sources. Therefore, these analyses do not require updating as a part of the license renewal process since they will be updated when required in accordance with applicable regulations.

4.2.4 REFERENCES

4.2-1 NUREG-1801, "Generic Aging Lessons Learned (GALL)," U. S. Nuclear Regulatory Commission, April 2001.

4.2-2 Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, February 1986.

4.2-3 WCAP 13587, Rev. 1, Reactor Vessel Upper Shelf Energy Bounding Evaluation for Westinghouse Pressurized Water Reactors, September, 1993.

4.3 METAL FATIGUE

Several thermal and mechanical fatigue analyses of plant mechanical components have been identified as time-limited aging analyses (TLAAs) for RNP. These are discussed in the following Subsections.

Subsection	Fatigue Issue
4.3.1	Explicit Fatigue Analysis (ASME Section III, Class A)
4.3.1.1	Pressurizer Surge Line Thermal Stratification
4.3.1.2	Pressurizer Insurge/Outsurge
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4.3.1 EXPLICIT FATIGUE ANALYSIS (ASME Section III, Class A)

Explicit fatigue analyses were prepared during the design process for the Class 1 RCS primary system components. Explicit fatigue analyses are those performed in accordance with ASME Section III, Class A (now Class 1) requirements, which required the analyses to demonstrate that the Cumulative Usage Factor (CUF) for the components would remain below 1.0, assuming the components were exposed to all of the postulated transient cycles. The list of transients used in these calculations were intended to envelope all foreseeable thermal and pressure cycles which could be expected to occur within a nominal 40-year operating period for the plant. The analyses are classified as time-limited aging analyses (TLAA's) due to the 40-year transient basis. The following reactor coolant system components have been designed using this methodology:

1. Reactor Vessel
2. Steam generators (original and replacement)
3. Reactor Coolant Pumps
4. Pressurizer

Additional explicit fatigue analyses have been prepared since original design to address certain metal fatigue issues, such as, thermal stratification of the pressurizer surge line, insurge/outsurge flow between the pressurizer and surge line, reactor vessel internal components, and thermal cycling of auxiliary feedwater to main feedwater connections. These analyses are addressed separately in Subsections 4.3.1.1 through 4.3.1.4.

A detailed review of the RNP Fatigue Monitoring Program was performed including a review of the transients selected, counting methods, and results to date. Adjustments to several of the cumulative cycle counts were recommended, due to past counting practices which were excessively conservative by counting partial temperature range cycles as full temperature range cycles. The actual frequency of occurrence for the design cycles was determined and compared to the design cycle set. The severity of the actual plant transients, e.g., partial cycles, was compared to the severity of the assumed design cycles. This comparison was performed in order to demonstrate that on an event-by-event basis, the design cycle profiles envelope actual plant operation. The result of this evaluation was the set of adjusted cumulative transient cycle counts. These data were used as a basis for 60-year projections, along with trending data from the past operational periods. Some projected cycle counts were adjusted to account for the decrease in number of cycles experienced currently versus the high number of cycles experienced during early years of plant operation. Using these methods, adjustments were made to the cumulative totals of cycles to date. These adjusted cycle counts, as well as plant operating history, were used as a basis for making 60-year transient cycle projections. These projections show that the original 40-year transient set is conservative and bounding for the 60-year operation of the plant.

Therefore, the analyses associated with verifying the structural integrity of the reactor vessel, steam generators, reactor coolant pumps, and pressurizer have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.3.1.1 Pressurizer Surge Line Thermal Stratification

The pressurizer surge line originally was designed to ANSI B31.1 rules; however, explicit detailed surge line fatigue analyses were performed to account for issues raised in NRC Bulletin 88-11 [Reference 4.3-1]. NRC Bulletin 88-11 requested licensees of all domestic, commercial pressurized water reactors to establish and implement a program to determine the impact of thermal stratification on pressurizer surge line integrity.

WCAP-12962 includes a fatigue analysis of the pressurizer performed for the Westinghouse Owner's Group (WOG) in 1991 to account for thermal stratification transients in the pressurizer surge line. This fatigue analysis was very conservative because it was based upon a significantly higher number of transients than the number specified in the RNP design basis.

Subsequently, a plant-specific evaluation for RNP was performed to confirm the WOG program results and to incorporate the results of temperature measurements performed on the RNP pressurizer surge line. The plant-specific analyses included WCAP-12962, Supplement 1, [Reference 4.3-2]. Supplement 1 to WCAP-12962 reports that the limiting location analyzed for the pressurizer surge line has a CUF less than 1.0, which includes consideration of thermal stratification and insurge/outsurge transients (as discussed in Subsection 4.3.1.2), and is based upon conservative numbers of transient cycles. The plant-specific stress and fatigue analyses were performed because the temperature monitoring data from sensors located at several locations on the surge line indicated that the temperature profile assumed in the previous analysis did not bound the observed data. However, the results of the stress and fatigue analyses demonstrated that stresses at the critical fatigue location (the surge line to Reactor Coolant System hot leg nozzle) remained below allowable. These results demonstrate acceptance to the requirements of the ASME Code, Section III, for the full current license term and allowed the resolution of the issues raised in Bulletin 88-11 as stated in Reference 4.3-3.

Since the number of transients projected to occur during the 60-year operational period is significantly less than the number or transients originally postulated for 40 years of operation and used in the fatigue analyses, the 40-year design transient set remains conservative for the 60-year license renewal period. Therefore, the analyses associated with pressurizer surge line thermal stratification have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.3.1.2 Pressurizer Insurge/Outsurge

Various Reactor Coolant System transient events can cause insurges and outsurges of water into the pressurizer. If a significant temperature difference exists between the pressurizer and the water entering the pressurizer via the surge line, the cooldown limits for the pressurizer may be exceeded.

In February 1994, a pressurizer transient occurred at RNP that exceeded the plant cooldown limits. WCAP-14209 [Reference 4.3-4] was prepared to complete the detailed evaluation of the transient. The detailed evaluation included the definition of a number of previous out-of-limit pressurizer transients, development of enveloping transients, determination of stresses in critical locations in the pressurizer lower head and surge nozzle, and evaluation of the effect of these stresses on the structural integrity of the pressurizer. The analysis conservatively calculated the fatigue usage that would result from 40 occurrences of each of the newly-defined transients. The fatigue usage values accounting for these insurge/outsurge transients (including the effects of thermal stratification fatigue, discussed in Subsection 4.3.1.1) are now included within the current RNP design and licensing basis. The analyses concluded that the highest fatigue location in the pressurizer is the carbon steel surge nozzle at the bottom of the vessel; and the 40-year CUF is below 1.0 which is acceptable.

The original fatigue analyses for the pressurizer are based upon 29,000 load/unload transient cycles, which equates to two cycles per day for 40 years. This number of cycles was originally postulated to account for daily load following, where power levels would be changed up to 2 times per each day. RNP does not operate the plant using daily load following, but instead operates as a base load plant, minimizing power level changes, thus minimizing load/unload transients. A review of past transient history was reported which showed fewer than 300 load/unload transients had occurred in the first 27 years of operation, and a projection was made which predicted fewer than 1,000 load/unload transients to occur in 60 years. Therefore, considerable margin exists with respect to the number of load/unload transients requiring consideration in fatigue analyses.

Since the number of transients projected to occur during the 60-year operational period is significantly less than the number or transients originally postulated for 40 years of operation and used in the fatigue analyses, the 40-year design transient set remains conservative for the 60-year license renewal period. Therefore, the analyses associated with fatigue of the surge line components from insurge/outsurge effects have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.3.1.3 Reactor Internals Holddown Spring And Alignment Pins

A Westinghouse topical report, WCAP-10322, Rev. 1, 10/84 [Reference 4.3-5], includes explicit fatigue analyses for reactor internals holddown spring and alignment pins. (It also includes fatigue analysis of control rod guide tube support pins; however, the RNP support pins do not support the performance of any intended functions for license renewal.) Since WCAP-10322, Rev. 1 has been incorporated by reference, the fatigue analyses for holddown springs and alignment pins are considered to be within the RNP design and licensing basis.

The analyses for the holddown springs and alignment pins were identified as TLAAs, and the fatigue analysis results in WCAP-10322, Rev. 1 show the holddown spring CUF = 0.073. The alignment pin CUF = 0.008. These 40-year CUF value are well below 1.0 and are acceptable. Since the number of transients projected to occur during the 60-year operational period is significantly less than the number or transients originally postulated for 40 years of operation and used in the fatigue analyses, the 40-year design transient set remains conservative for the 60-year license renewal period. Since the number of transients assumed in the existing fatigue analyses remain conservative for 60 years, each of the fatigue analyses based upon the number of transients in 40 years remain valid for the 60-year extended operating period.

Therefore, the analyses associated with fatigue of the reactor internals holddown spring and alignment pins have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.3.1.4 Auxiliary Feedwater Line Fatigue Analysis

In 1972, a leak was found on a 4 inch diameter Auxiliary Feedwater (AFW) System line at the connection to the 16 inch diameter Feedwater (FW) pipe upstream of the B Steam Generator. Indications were observed that were attributed to thermal fatigue cracking. As part of the corrective action for this event, AFW connections for each steam generator were replaced with a thermal-sleeved tee designed using ASME Code Section III, Subsection NB requirements; although this piping was designed originally using USAS B31.1. One of the replacement connections used a saddle-shaped reinforcement plate, and the other five were replaced using a pad plate reinforcement configuration. The saddle configuration was later determined to result in considerably more fatigue than the pad plate configuration, and it was replaced with a pad plate reinforcement design in 1995. In conjunction with that modification a fatigue calculation was performed for this feedwater branch connection reinforcement plate. This analysis is considered to be a TLAA.

The ASME Section III fatigue analysis, which was based in part upon the number of surveillance tests projected to occur prior to the 40-year end of license, resulted in an acceptable CUF value less than 1.0. As a result of the license renewal review, corrective action has been initiated to modify the CUF value to the correct, but still less than 1.0, value; and to revise or supplement the calculation as needed to evaluate the transients and resultant cycle count for the period of license renewal extended operation.

Therefore, the fatigue design analyses associated with the AFW to FW connections has been evaluated for the current 40-year operating period and, assuming the successful limitation of transient cycles for a 60-year period, will be determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i). The analysis will be updated, prior to the period of extended operation, to verify the fatigue CUF for the AFW-to-FW connections.

4.3.2 IMPLICIT FATIGUE DESIGN (ASME Section III, Class C, B31.1)

Components with implicit fatigue design comprise piping designed in accordance with USAS B31.1 Code (including RCS piping) and auxiliary heat exchangers designed in accordance with ASME Section III, Class C, or ASME Section VIII requirements.

Most RNP piping, including Reactor Coolant System Class 1 piping, has been designed to USAS B31.1, Power Piping, design rules. For piping components designed in accordance with the USAS B31.1, consideration of cyclic loading was required in the design process, but no formal fatigue analyses were required. These components are considered to have implicit fatigue analyses. USAS B31.1 design methods apply reduction factors to allowable stresses to allow for specified numbers of cyclic loadings. This effectively reduces the stress amplitude and prevents fatigue damage to the component. Since the 40-year design transient set has been demonstrated to be conservative for 60 years of operation for the RCS system (refer to Subsection 4.3.1), the number of thermal cycles imposed upon B31.1 piping systems and associated heat exchangers is not expected to exceed the original design assumptions. Therefore, the current design and licensing basis will be maintained for these systems throughout the license renewal period.

The conservatism of the implicit fatigue design methodology, as represented by the B31.1 code, has been documented many times. Recently, EPRI Report TR-102901 [Reference 4.3-6] compared the B31.1 fatigue strength reduction factor design calculations with ASME Section III, Class 1, explicit fatigue analysis for a Pressurized Water Reactor charging line. The analysis used the design basis transients for a recent-vintage Westinghouse unit. These events were converted into equivalent full temperature cycles for use in the B31.1 methodology. ANSI B31.1 requirements were met with all stresses less than 75% of allowable values. For ASME Class 1 explicit fatigue analysis, through-wall thermal gradient and differential thermal expansion stress terms dominate. These terms are not considered in the B31.1 analysis, which is limited to thermal stresses from piping thermal expansion bending moments. Even with these additional stresses, the ASME Class 1 cumulative usage factor at the most critical location was on the order of 0.1, using the 1986 Edition of NB-3600. Therefore, the B31.1 analysis was more limiting than the ASME analysis for the limiting component. This showed that the B31.1 methodology provided adequate margin for fatigue in service for this component. It is also an indication that this should be true for most B31.1 applications. The conclusions from the EPRI study indicate that piping systems designed to the requirements of ANSI B31.1 are adequate for continued service in nuclear plants. In the absence of stress risers (high stress indices or material discontinuities) and severe thermal transients, there is no reason to expect fatigue usage to approach unity (CUF = 1.0) in these systems. Even when stress risers are present, detailed analyses support the technical position that the current licensing basis for fatigue is adequate both for the original and extended license periods for piping constructed to the requirements of ANSI B31.1 and its predecessor standards, including those areas with geometric and loading discontinuities. The detailed analysis supporting this conclusion include fatigue analyses of the pressurizer surge line components, charging nozzles, safety injection nozzles, and RHR-to-RCS connection tee.

Auxiliary heat exchangers at RNP were designed in accordance with Westinghouse specification and ASME Section III, Class C, or ASME Section VIII requirements. These include the regenerative heat exchanger, residual heat exchanger, seal water heat exchangers, excess letdown heat exchangers, spent fuel pit heat exchangers, sample heat exchangers, and letdown heat exchangers. Each of the specified heat exchangers was designed for a specified number and magnitude of pressure and temperature cycles required by the specification in accordance with the implicit fatigue design rules in effect for the applicable codes, including ASME Section III, Class C. The fatigue design rules for ASME Section III, Class C are essentially identical to the B31.1 design rules described above which use a stress range reduction factor for components exposed to specified numbers of equivalent full temperature stress cycles. Therefore, any reductions in allowable stress needed for the components to safely withstand the specified thermal transients would have occurred during the original design of these heat exchangers in order to meet the code design requirements. No further reductions are needed because, as described previously, the number of pressure and temperature cycles projected for the 60-year license renewal period does not exceed the number of pressure and temperature cycles originally specified and analyzed for 40 years. Therefore, the current designs for the specified heat exchangers, including fatigue considerations, remain valid for the 60-year license renewal period.

Based on the above discussion, the implicit fatigue design analyses associated with ANSI B31.1 piping and auxiliary heat exchangers have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.3.3 ENVIRONMENTALLY ASSISTED FATIGUE EVALUATION

Generic Safety Issue (GSI)-190 [Reference 4.3-7], was identified by the NRC staff because of concerns about the potential effects of reactor water environments on Reactor Coolant System component fatigue life during the period of extended operation. GSI-190 was closed in December 1999 [Reference 4.3-8] on the basis that environmental effects have a negligible impact on core damage frequency. However, as part of the closure of GSI-190, the NRC concluded that license renewal applicants should address the effects of coolant environment on component fatigue life as part of their aging management programs. Reactor water environmental effects, as described in GSI-190, are not included in the RNP current licensing basis (CLB); therefore, the TLAA criterion specified in 10 CFR 54.3(a)(6) is not satisfied. Nevertheless, environmental effects on component fatigue have been evaluated to determine if any additional actions are required for the period of extended operation.

Plant-specific environmental fatigue calculations were performed for a sample of high fatigue locations to demonstrate that adequate conservatism exists within the fatigue TLAA's to account for reactor water environmental effects. These sample locations include the sample locations identified in NUREG/CR-6260 [Reference 4.3-9], for older-vintage Westinghouse plants. For RNP, four of these locations have ASME Section III specific fatigue analyses, and the remaining three have USAS B31.1 implicit fatigue analyses. Environmentally Assisted Fatigue (EAF) relationships developed in NUREG/CR-6583 [Reference 4.3-10], for carbon and low alloy steels, and NUREG/CR-5704 [Reference 4.3-11], for stainless steels, were used. The calculations use the environmental fatigue multiplier approach developed by General Electric and EPRI, as described in EPRI Report TR-105759 [Reference 4.3-12].

For the locations with a USAS B31.1, implicit fatigue evaluation, a comparison with the fatigue analyses in NUREG/CR-6260 was accomplished by comparing RNP plant-specific design attributes with those used in the fatigue analyses to show the similarity and to justify their use for RNP. Environmental fatigue multipliers were computed for each case and these factors were applied to the CUF values obtained from the NUREG/CR-6260 fatigue analyses. All EAF-adjusted CUFs were less than 1.0.

For the locations with an ASME Section III fatigue analyses, EAF factors were calculated and applied to the CUF from the fatigue analyses. The results showed that of the four locations, only the pressurizer surge line was not shown to have an EAF-adjusted CUF value below 1.0. As part of this analysis, the number of load/unload transients was reduced from 29,000 to 19,000 cycles in conjunction with an EAF-adjusted CUF analysis of pressurizer components. Reducing the number of load/unload transients is acceptable, because load/unload transients were originally postulated as an allowance for daily load following, which is not the manner of operation at RNP. The number of load/unload transients experienced to date is less than 300, and the 60-year projection is approximately 600. These values show that the adjusted value is still quite conservative. A revision will be made to the RNP design transient set in the UFSAR prior to the license renewal period to limit these transients to a maximum of 19,000 cycles. As noted previously in Subsection 4.3.1, the original 40-year transient set is conservative and bounding for the 60-year operation of RNP (without consideration of environmentally assisted fatigue effects).

In addition to the seven locations specified in NUREG/CR-6260, environmental fatigue calculations were performed for the seven RNP pressurizer locations that have an ASME Section III fatigue analysis in existence within the current design and licensing basis. The number of load/unload transients was reduced to 19,000 in this analysis to obtain a satisfactory EAF-adjusted CUF value. The acceptability of this adjustment was discussed above. The results of the pressurizer analyses concluded that all locations have an EAF-adjusted CUF value of less than 1.0 except for the pressurizer surge nozzle safe end.

The RNP surge line was fabricated from two 14 inch diameter stainless steel pipes, welded end-to-end. Each pipe has two large radius bends instead of welded elbows, limiting the number of welds significantly. The welds joining the surge line to the RCS hot leg and to the pressurizer surge nozzle are the limiting locations.

Based on the above discussion, the EAF-adjusted fatigue results for pressurizer surge line components indicated a CUF greater than 1.0 for the period of extended operation. While there is considerable conservatism both in the original CUF value and in the EAF adjustment factor, removal of these conservatisms may not result in an EAF-adjusted CUF value of less than 1.0. Therefore, fatigue of surge line components will be managed through the performance of periodic volumetric examinations in accordance with the ASME Section XI, Subsection IWB, IWC, and IWD Program. The frequency of these inspections is specified within the Program documents, and is subject to NRC review and approval. Each limiting location is inspected at least once during every 10-year interval. These inspections are considered adequate to detect the initiation of fatigue cracking prior to initiation of unstable crack growth. If unacceptable indications are identified, further evaluation, repair or replacement will be performed as required by ASME Section XI. This program is considered adequate to manage thermal fatigue of the surge line and adjacent components during the license renewal period.

Therefore, in accordance with 10 CFR 54.21(c)(1)(iii), an aging management program will be used to manage the effects of fatigue of the pressurizer surge line components. This will be accomplished through in-service inspections performed in accordance with the ASME Section XI, Subsection IWB, IWC, and IWD Program. If unacceptable indications are identified, further evaluation, repair or replacement will be performed as required by ASME Section XI.

4.3.4 REACTOR VESSEL UNDERCLAD CRACKING

A fracture mechanics analyses completed in 1971 concluded that fatigue growth of potential underclad flaws in reactor vessel base metal over a 40-year period would be insignificant and the structural integrity of the reactor vessels had not been compromised for their intended use for 40-year period.

The underclad cracking analysis has been updated by a topical report, WCAP-15338 [Reference 4.3-13], to justify operation for 60 years. The topical report results indicated that an assumed flaw, assumed to grow under the influence of transient cycles for a period of 60 years, would remain below the most critical allowable flaw depth. Since the estimated final flaw depth is smaller than the allowable flaw depth, it was concluded that a reactor vessel with postulated underclad cracks would be acceptable for operation for 60 years. An NRC Safety Evaluation [Reference 4.3-14] concludes that of WCAP-15338 is acceptable for referencing as a topical report, and RNP has verified that the report is applicable to the RNP reactor vessel. RNP has verified this by (1) concluding that the number of design cycles and transients assumed in the WCAP-15338 analysis bounds the number of cycles for 60 years of operation the RNP reactor vessel, and (2) including a summary description of the WCAP-15338 analysis in the RNP UFSAR Supplement.

WCAP-15338 evaluated 3-loop Westinghouse plants using the entire set of design transients with the number of cycles corresponding the RNP 40-year design transient set which in turn bounds the 60-year license renewal period for RNP as shown in Subsection 4.3.1.

Therefore, the TLAA for reactor vessel underclad cracking has been projected to the end of the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

4.3.5 CONTAINMENT PENETRATION BELLOWS FATIGUE

Fatigue of primary containment components was reviewed to identify potential TLAAs. Fatigue TLAAs were identified for three replacement bellows assemblies used for hot piping penetrations. The original bellows do not have analyses that fit the definition of TLAAs.

Fatigue analyses show that the specified bellows assemblies can withstand 4,000 cycles without fatigue cracking.

The significant thermal transients that result in flexure of the hot pipe penetration bellows are those that involve a full range temperature change in the piping system. This includes the plant heatup and cooldown cycles. The specified number of cycles for heatup and cooldown are 200 within the 40-year original design basis of the plant. As shown in Subsection 4.3.1, the 40-year transient counts remain conservative for 60 years of operation.

The 4,000 cycles analyzed in the three containment bellows fatigue calculations exceed the 200 heatup/cooldown cycles applicable for 60 years of operation; therefore, the calculations remain valid through the period of extended operation. The analyses associated with containment bellows fatigue remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.3.6 CRANE CYCLIC LOAD LIMITS

The load cycle limits for cranes was identified as a potential TLAA. The following RNP cranes are in the scope of license renewal and have been identified as having a TLAA which requires evaluation for 60 years:

- Polar Crane
- Spent Fuel Cask Crane

The method of review applicable to the crane cyclic load limit TLAA involves (1) reviewing the existing 40-year design basis to determine the number of load cycles considered in the design of each of the cranes in the scope of license renewal, and (2) developing 60-year projections for load cycles for each of the cranes in the scope of license renewal and compare with the number of design cycles for 40 years.

<u>The Containment Polar Crane</u>

The RNP Containment Polar Crane was designed in accordance with Electric Overhead Crane Institute (EOCI) Specification for Electric Overhead Traveling Cranes, 1961, (EOCI-61), and AISC 6th Edition, Steel Construction Manual. EOCI-61 did not require a reduction in allowable stresses for fatigue. However AISC 6th Edition permitted up to 10,000 complete stress reversals at maximum stress to occur for the life of the structure.

The total number of lift cycles for the Containment Polar Crane is directly dependent on the number of Refueling Outages. The total number of Refueling Outages for 60 years of operation has been established as 40. The total number of upper and mid-range lifts is 110 per outage for a total of 40 outages, which equates to a 60-year projection of 4,400 lift cycles. This is less than the 10,000 permissible lift cycles and is therefore acceptable. Thus, the RNP Polar Crane fatigue analysis has been successfully projected for 60 years of plant operation.

<u>Spent Fuel Cask Crane</u>

The maximum allowable stress for any member of the Spent Fuel Cask Crane under tension or compression subject to repeated loading was given as 17,600 psi. The basic allowable stress included the dead weight, live load and impact allowance as required by CMAA #70. This gives a minimum Safety Factor of 3.3 based on the maximum tensile strength of 58,000 psi for ASTM-A36. The crane is designed for 20,000 to 100,000 load cycles compared to actual load cycles of less than 2,500, which will take place over a 40-year life. Therefore, the RNP spent fuel cask crane is designed for a minimum of 20,000 cycles at up to 17,600 psi allowable stress with a Factor of Safety of 3.3.

The number of lift cycles originally projected for 40 years was 2,500. This can be multiplied by a factor of 1.5 to determine the number of cycles for 60-year life. Therefore, number of load cycles projected for 60 years is 3,750. This is less than the 20,000 permissible cycles and is therefore acceptable. Therefore, the RNP spent fuel cask crane fatigue analysis has been successfully projected for 60 years of plant operation.

Based on the above information, the analyses associated with fatigue of the Containment Polar Crane and the Spent Fuel Cask Crane have been projected to the end of the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

4.3.7 REFERENCES

4.3-1 NRC Bulletin 88-11, dated December 20, 1988: “Pressurizer Surge Line Thermal Stratification.”

4.3-2 WCAP-12962, Supplement 1, “Structural Evaluation of The H.B. Robinson Unit 2 and Shearon Harris Pressurizer Surge Lines, Considering the Effects of Thermal Stratification,” October 1995.

4.3-3 CP&L letter (R. Krich) to NRC, dated November 28, 1995: “Final Response to NRC Bulletin No. 88-11, Pressurizer Surge Line Thermal Stratification” RNP-RA/95-0211.

4.3-4 WCAP-14209, “Evaluation of the Effects of Insurge / Outsurge Transients on the Integrity of the Pressurizer at H.B. Robinson Unit 2” October 28, 1994.

4.3-5 WCAP 10322, Rev. 1, “Stress Report of 312 Standard Reactor Core Support Structures and Internal Structures Structural and Fatigue Analysis,” October 1984.

4.3-6 EPRI Report No. TR-102901S, “Fatigue Comparison of Piping Designed to ANSI B31.1 and ASME Section III, Class 1 Rules,” Electric Power Research Institute, December 1993.

4.3-7 Generic Safety Issue 190, “Fatigue Evaluation of Metal Components for 60-Year Plant Life,” U.S. Nuclear Regulatory Commission.

4.3-8 NRC Memorandum, A. Thadani, Director, Office of Nuclear Regulatory Research, to W. Travers, Executive Director of Operations: “Closeout of Generic Safety Issue 190, “Fatigue Evaluation of Metal Components for 60-Year Plant Life,” U.S. Nuclear Regulatory Commission, December 26, 1999.

4.3-9 NUREG/CR-6260 (INEL-95/0045), “Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components,” U.S. Nuclear Regulatory Commission, March 1995.

4.3-10 NUREG/CR-6583 (ANL-97/18), “Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels,” U.S. Nuclear Regulatory Commission, March 1998.

4.3-11 NUREG/CR-5704 (ANL-98/31), “Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels,” U.S. Nuclear Regulatory Commission, April 1999.

4.3-12 EPRI Report TR-105759, An Environmental Factor Approach to Account for Reactor Water Effects in Light Water Reactor Pressure Vessel and Piping Fatigue Evaluations, December 1995.

4.3-13 WCAP-15338, “A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants,” March 2000.

4.3-14 USNRC Safety Evaluation of WCAP-15338, A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants, October 15, 2001.

4.4 ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

The thermal, radiation, and wear cycle aging analyses of plant electrical and I&C components required to meet 10 CFR 50.49 have been identified as time-limited aging analyses for RNP.

The NRC has established nuclear station environmental qualification requirements for electrical equipment in 10 CFR 50.49. The requirements in 10 CFR 50.49 specify that an environmental qualification program be established to demonstrate that certain electrical and I&C components located in “harsh” plant environments (i.e., those areas of the plant that could be subject to the harsh environment effects of a loss-of-coolant accident, high energy line break, or post loss-of-coolant accident radiation) are qualified to perform their safety function in those harsh environments after the effects of in-service aging. Further, 10 CFR 50.49 requires that the effects of significant aging mechanisms be addressed as part of environmental qualification.

All operating plants must meet the requirements of 10 CFR 50.49 for certain electrical and I&C components important-to-safety. 10 CFR 50.49 defines the scope of components to be included, requires the preparation and maintenance of a list of in-scope equipment, and requires the preparation and maintenance of a qualification file that includes component performance specifications, electrical characteristics, and environmental conditions. The requirements in 10 CFR 50.49(e)(5) contain provisions for aging that require, in part, consideration of all significant types of aging degradation that can affect component functional capability. 10 CFR 50.49(e) also requires component replacement or refurbishment at the end of qualified life unless additional life is established through ongoing qualification. 10 CFR 50.49(f) establishes four methods of demonstrating qualification for aging and accident conditions. The requirements in 10 CFR 50.49(k) and (l) permit different criteria to apply based on plant and component vintage. Supplemental environmental qualification regulatory guidance for compliance with these different qualification criteria is provided in the DOR Guidelines, “Guidelines for Evaluating Environmental Qualification of Class 1E Electrical /Equipment in Operating Reactors” [Reference 4.4-1], NUREG-0588, “Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment” [Reference 4.4-2], and Regulatory Guide 1.89, Revision 1, “Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants” [Reference 4.4-3]. Compliance with 10 CFR 50.49 provides evidence that the component will perform its intended functions during accident conditions after experiencing the effects of in-service aging.

The RNP Environmental Qualification Program, which complies with regulatory requirements, includes three main elements: identifying applicable equipment and environmental requirements, establishing the qualification, and maintaining (or preserving) qualification.

Components included in the RNP Environmental Qualification Program have been evaluated to determine if existing environmental qualification aging analyses remain valid for the period of extended operation. Qualification for the license renewal period will be treated the same as for components currently qualified at RNP for 40 years or less. The Environmental Qualification Program manages component thermal, radiation, and wear cycle aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, environmentally qualified components must be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation. Aging evaluations for environmentally qualified components that specify a qualification of at least 40 years are considered time-limited aging analyses for license renewal.

4.4.1 ELECTRICAL AND I&C COMPONENT ENVIRONMENTAL QUALIFICATION ANALYSES

Age-related service conditions that are applicable to environmentally qualified components (i.e., 60 years of exposure versus 40 years) were evaluated for the period of extended operation to verify that the current environmental qualification analyses are bounding. Temperature and radiation values assumed for service conditions in the environmental qualification analyses are either the design operating values or measured values for RNP. The following paragraphs describe the thermal, radiation, and wear cycle aging effects that were evaluated.

THERMAL CONSIDERATIONS – The component qualification temperatures were calculated for 60 years using the Arrhenius method, as described in EPRI NP-1558, "A Review of Equipment Aging Theory and Technology" [Reference 4.4-4]. If the component qualification temperature bounded the service temperatures throughout the period of extended operation, then no additional evaluation was required.

RADIATION CONSIDERATIONS – The RNP Environmental Qualification Program has established bounding radiation dose qualification values for all environmentally qualified components. Typically, these bounding radiation dose values were determined by component vendors through testing. To verify that the bounding radiation values are acceptable for the period of extended operation, integrated dose values were determined and then compared to the bounding values. The total integrated dose through the period of extended operation is determined by adding the established accident dose to the normal operating dose for the component.

WEAR CYCLE AGING CONSIDERATIONS - Wear cycle aging is a factor for some equipment within the EQ program. In those cases where wear cycle aging was considered a credible aging mechanism, wear cycles were evaluated through the end of the new license term.

The following Subsections (4.4.1.1 through 4.4.1.47) provide a description for each of the environmental qualification analyses for the period of extended operation.

4.4.1.1 ASCO NP8316 and NP8321 Series Solenoid Valves

ASCO NP8316 and NP8321 Series Solenoid Valves are located inside containment, in the pressurizer cubicle, and in the Pipe Alley. Some of these solenoid valves are required for accident mitigation and are required to operate during the first five minutes of a LOCA, then fail safe. Some of these solenoid valves are intermediate components of post-accident monitoring control circuits and not required to operate past five minutes of a LOCA. However, their failure must not result in degradation of the circuit during their 30-day post-accident phase.

4.4.1.1.1 Thermal Analysis

The thermal aging test conditions qualify the solenoid valves for greater than 60 years at the current operating temperatures. The normally energized valves must have their coils replaced every 3.78 years inside containment and 6.82 years outside containment to maintain qualification. There are no EQ maintenance requirements for elastomers or for normally de-energized valves.

4.4.1.1.2 Radiation Analysis

The post accident dose of $1.3x10^{7}$ rads, gamma, plus $8.0x10^{7}$ rads, beta, plus $1.65x10^{6}$ rads, gamma normal aging, 60-year dose results in a TID requirement of $9.47x10^{7}$ rads. This projected TID is much less than the qualified value of $2.0x10^{8}$ rads, gamma.

4.4.1.1.3 Conclusion

The qualification of ASCO NP8316 and NP8321 Series Solenoid Valves has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii). The qualification of normally energized ASCO NP8316 and NP8321 Series Solenoid Valves will be maintained by periodic replacement of subcomponents (coils) in accordance with the EQ Program. Periodic replacement of coils does not involve a TLAA, because the period of replacement is not defined by the current 40-year operating term of the plant.

4.4.1.2 ASCO Solenoid Valves – AQR Report

ASCO NP 8314 and 8320 Series Solenoid Valves are located in the Pipe Alley and are used both to isolate primary containment in the event of a LOCA inside containment and for operation of the Post-Accident Sampling System (PASS). Both AC and DC valves are used, but only AC valves are used in continuously energized applications. All other valve applications are normally de-energized. The valves are to be qualified to a radiation-only environment as a result of a LOCA inside containment.

4.4.1.2.1 Thermal Analysis

The thermal aging test conditions qualify the solenoid valves for greater than 60 years at the current operating temperatures. The normally energized AC valves must have their elastomers replaced every 22 years and their coils replaced every 11 years to maintain qualification. There are no EQ maintenance requirements for normally de-energized valves.

4.4.1.2.2 Radiation Analysis

The post accident dose of 2.3×10^7 rads, gamma, plus a negligible normal aging, 60-year dose results in a projected dose of 2.3×10^7 rads, TID. This projected TID is much less than the qualified value of 2.05×10^8 rads.

4.4.1.2.3 Conclusion

The qualification of ASCO NP 8314 and 8320 Series Solenoid Valves has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii). The qualification of normally energized ASCO NP 8314 and 8320 Series Solenoid Valves will be maintained by periodic replacement of subcomponents (elastomers and coils) in accordance with the EQ Program. Periodic replacement of coils does not involve a TLAA, because the period of replacement is not defined by the current 40-year operating term of the plant.

4.4.1.3 Limitorque SMB Motor Operated Valve (MOV) Actuators – Outside Containment

Limitorque SMB MOV Actuators located outside containment are used in various safety-related systems at RNP. The operators must remain functional during and after a DBE for up to 30 days.

4.4.1.3.1 Thermal Analysis

The thermal aging test conditions qualify the SMB MOV Actuators for 60 years at the current operating temperature. However, the qualified life is limited by cycling to 40 years at the anticipated rate of usage.

4.4.1.3.2 Radiation Analysis

The post accident dose of 7.6×10^6 rads, gamma, plus the 1.5×10^6 rads, gamma normal aging, 60-year dose results in a TID requirement of 9.1×10^6 rads. This projected TID is much less than the qualified value of 2.0×10^7 rads.

4.4.1.3.3 Wear Cycle Aging Analysis

The test specimen was cycle aged for a total of 2,011 cycles, which limits the qualified life to 40 years at a conservative assumed rate of usage.

4.4.1.3.4 Conclusion

Limitorque SMB MOV Actuators located outside containment are qualified at RNP for 40 years, limited by the anticipated number of cycles during normal service. Aging effects during the period of extended operation will be managed by either (1) completing, prior to the period of extended operation, reanalysis of the wear aging using better cycling frequency data, or (2) replacing the actuators to maintain qualification in accordance with 10 CFR 54.21(c)(1)(iii).

4.4.1.4 Limitorque Model SB-3 and SMB-00 Motor-Operated Valve (MOV) Actuators – Inside Containment

Limitorque Model SB-3 and SMB-00 MOV Actuators are located inside containment and are required to operate during the initial 20 hours of a postulated LOCA or MSLB.

4.4.1.4.1 Thermal Analysis

The thermal aging test conditions qualify the MOVs for greater than 60 years at the current operating temperatures. There are no EQ maintenance requirements.

4.4.1.4.2 Radiation Analysis

The post accident dose of 1.3×10^7 rads gamma plus 8.0×10^7 rads beta, plus 1.65×10^6 rads, gamma normal aging, 60-year dose results in a TID requirement of 9.47×10^7 rads. This projected TID is much less than the qualified value of 2.04×10^8 rads, gamma.

4.4.1.4.3 Wear Cycle Aging Analysis

The test specimen was cycle aged for a total of 2,011 cycles, which exceeds the maximum, 60-year requirement of 773 cycles by a large margin.

4.4.1.4.4 Conclusion

The qualification of Limitorque Model SB-3 and SMB-00 MOV Actuators, located inside containment at RNP, has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.5 Rockbestos Cable – Firewall III

Rockbestos Firewall III Cables are used throughout RNP in safety-related instrumentation, control, and power circuits. The cables provide signal paths for various electrical equipment, including temperature elements, level transmitters, solenoid valves, and motors, and they must function in the event of a design basis accident for up to 30 days post-accident. The installed cables consist of both chemically and irradiation cross-linked polyethylene insulations.

4.4.1.5.1 Thermal Analysis

The thermal aging test conditions qualify the cables for greater than 60 years at the worst-case, normal conditions inside the pressurizer cubicle. There are self-heating effects for Rockbestos Firewall III cables in power circuits, but the cables are qualified for more than 60 years including these effects. There are no EQ maintenance requirements for these cables.

4.4.1.5.2 Radiation Analysis

The post accident dose of 1.3×10^7 rads gamma plus 8.0×10^7 rads beta, plus 1.65×10^6 rads, gamma normal aging, 60-year dose results in a TID requirement of 9.47×10^7 rads. This projected TID is much less than the qualified value of 1.84×10^8 rads, gamma.

4.4.1.5.3 Conclusion

The qualification of Rockbestos Firewall III Cables has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.6 Rockbestos RSS-6-104/LE Series Coaxial Cable

The Rockbestos RSS-6-104/LE Series Coaxial Cable provides a signal path for safety related circuits and must remain functional during and after a DBE. The cable is utilized by the Containment High Range Radiation Monitors.

4.4.1.6.1 Thermal Analysis

The thermal aging test conditions give a qualified life of greater than 95 years at a service temperature of 65^{o}C, which exceeds the maximum normal operating temperature of 120^{o}F. There is no heat rise for this instrumentation application (i.e., Containment High Range Radiation Monitors).

4.4.1.6.2 Radiation Analysis

The post accident dose of 1.3×10^{7} rads gamma plus 8.0×10^{7} rads beta, plus the 1.65×10^{6} normal aging, 60-year dose results in a TID requirement of 9.47×10^{7} rads. This projected TID is less than the qualified value of 2.0×10^{8} rads.

4.4.1.6.3 Conclusion

The qualification of Rockbestos RSS-6-104/LE Series Coaxial cable has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.7 Rockbestos Cable – Firezone R

The subject equipment provides a signal path for safety related circuits and must remain functional during and after a DBE. RNP Modification M-755I installed the Firezone R 4/C 14 cable from the containment penetration to the following resistance temperature detectors (RTDs): TE-410 and TE-413-1.

4.4.1.7.1 Thermal Analysis

The thermal aging test conditions qualify the Rockbestos Firezone R cable for greater than 77 years at 65^{o}C, which exceeds the maximum normal operating temperature of 120^{o}F. There is no heat rise for this instrumentation application (RTDs).

4.4.1.7.2 Radiation Analysis

The post accident dose of $1.3x10^{7}$ rads gamma plus the $1.65x10^{6}$ normal aging, 60-year dose results in a projected dose of $1.47x10^{7}$ rads, TID, which is less than the qualified value of $2.5x10^{7}$ rads. Since these cables are enclosed in conduit, beta radiation is not a consideration.

4.4.1.7.3 Conclusion

The qualification of Rockbestos Firezone R cable has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.8 GEMS Liquid Level Transmitters – Model XM-54853 and XM-54854

GEMS Liquid Level Transmitters are located inside containment and are required to operate in the event of a LOCA or MSLB for up to 30 days post-accident. Meter readings shall not vary from calibration more than +/- 3% full scale (+/- 6 microamperes). The high and low level alarm set points may not vary more than +/- 3% (+/- 2.7 inches).

4.4.1.8.1 Thermal Analysis

The thermal aging test conditions qualify the transmitters for 46 years inside containment. There are no self-heating effects for this instrument.

4.4.1.8.2 Radiation Analysis

The post-accident dose of $2.0x10^8$ rads, gamma plus $1.27x10^6$ rads, gamma normal aging, 46-year dose results in a TID requirement of $2.01x10^8$ rads. This projected TID value is met by the qualified value of $2.02x10^8$ rads, gamma.

4.4.1.8.3 Wear Cycle Aging Analysis

The only EQ maintenance requirement for this equipment is that their cycle life is limited to 220 cycles.

4.4.1.8.4 Conclusion

The qualification of GEMS Liquid Level Transmitters has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii). The qualification of GEMS Liquid Level Transmitters will be maintained by periodic refurbishment/replacement in accordance with the EQ Program. Periodic refurbishment/replacement does not involve a TLAA, because the period of refurbishment/replacement is not defined by the current 40-year operating term of the plant.

4.4.1.9 B&W Valve Monitoring System

B&W Valve Monitoring System (VMS) is located inside containment and is required to operate for up to 30 days following a postulated LOCA or MSLB. With an input of 1g, the output of the system will be 1VRMS ± 20 %.

4.4.1.9.1 Thermal Analysis

The thermal aging test conditions qualify the VMS for greater than 60 years at the current operating temperatures, with the exception of the Unholtz-Dickie RCA-2TR Preamplifier (11 years). The replacement of this component is considered an EQ maintenance requirement.

4.4.1.9.2 Radiation Analysis

The post accident dose of 1.5×10^7 rads, gamma plus 1.65×10^6 rads, gamma normal aging, 60-year dose results in a TID requirement of 1.67×10^7 rads. This projected TID is much less than the qualified value of 9.7×10^7 rads, gamma.

4.4.1.9.3 Conclusion

Qualification of the B&W Valve Monitoring System, located inside containment at RNP, has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii). The qualification of the B&W Valve Monitoring System will be maintained by periodic replacement of the Unholtz-Dickie RCA-2TR Preamplifier in accordance with the EQ Program. Periodic replacement of the preamplifier does not involve a TLAA, because the period of replacement is not defined by the current 40-year operating term of the plant.

4.4.1.10 Westinghouse Reactor Containment Fan Cooler (RCFC) Motors

Westinghouse RCFC Motors are located inside containment and are required to operate continuously during and after a DBE for up to 30 days post-accident.

4.4.1.10.1 Thermal Analysis

The thermal aging test conditions qualify the motors for greater than 60 years at the current operating temperatures.

4.4.1.10.2 Radiation Analysis

The post accident dose of 3.4×10^6 rads gamma plus 8.0×10^7 rads beta, plus 2.85×10^2 rads, gamma normal aging, 60-year dose results in a TID requirement of 8.34×10^7 rads. This projected TID is much less than the qualified value of 2.0×10^8 rads, gamma.

4.4.1.10.3 Conclusion

The qualification of Westinghouse RCFC Motors has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.11 Westinghouse Motors – Frame 506UPZ, 509US, and SBDP – RHR, SI Pumps, HVA 6A, 6B, 8A, & 8B

Westinghouse Pump and Fan Motors, Frame 506UPZ, 509U, and SBDP are located outside containment and are required to operate for up to 30 days following a postulated LOCA (radiation only).

4.4.1.11.1 Thermal Analysis

The installed motors are qualified for greater than 60 years under the DOR Guidelines using calculations comparing their nameplate ratings and the operating conditions under normal and accident conditions. The spare RHR pump motor is qualified using the thermal aging test conditions under NUREG-0588, Category I, for greater than 60 years at the operating conditions under normal and accident conditions. There are no EQ maintenance requirements.

4.4.1.11.2 Radiation Analysis

The post-accident dose of 7.6×10^6 rads, gamma plus 1.5×10^6 rads, gamma normal aging, 60-year dose results in a projected TID requirement of 9.1×10^6 rads, which is much less than the qualified value of 2.0×10^8 rads, gamma. There is no beta radiation requirement for the outside containment locations.

4.4.1.11.3 Conclusion

The qualification of Westinghouse Pump and Fan Motors, Frame 506UPZ, 509U, and SBDP located outside containment at RNP, has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.12 Westinghouse Motors – Model S068C20085 – Containment Spray Pumps

Westinghouse Model S068C20085 Pump Motors with PMR Insulation are located outside containment and are used to drive the containment spray pumps. Under normal conditions, the motors operate for 1% of the time. The motors are required to operate continuously during and after a DBE for up to 30 days post-accident. Since they are located outside containment, the accident environment is radiation-harsh only.

4.4.1.12.1 Thermal Analysis

The thermal aging test conditions qualify the motors for 60 years at the current operating temperatures.

4.4.1.12.2 Radiation Analysis

The post-accident dose of $1.3x10^6$ rads, gamma, plus $1.5x10^6$ rads, gamma normal aging, 60-year dose results in a projected TID requirement of $2.8x10^6$ rads, which is much less than the qualified value of $1.0x10^7$ rads, gamma.

4.4.1.12.3 Conclusion

The qualification of Westinghouse Model S068C20085 Pump Motors, with PMR Insulation, has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.13 Crouse-Hinds Electrical Penetration Assemblies

Crouse-Hinds Electrical Penetration Assemblies are located inside containment and supply power to safety-related induction motor feeders. The penetrations are required to operate for up to 30 days after a postulated LOCA or MSLB.

4.4.1.13.1 Thermal Analysis

The expected life of all of the component materials demonstrates that the penetrations are insensitive to time-temperature effects for over 60 years. There are no EQ maintenance requirements.

4.4.1.13.2 Radiation Analysis

The post accident dose of 1.3×10^7 rads, gamma plus 3.45×10^3 rads, gamma normal aging, 60-year dose results in a TID requirement of 1.3×10^7 rads TID. This projected TID requirement is less than the radiation damage threshold for all critical, non-metallic materials.

4.4.1.13.3 Conclusion

The qualification of Crouse-Hinds Electrical Penetration Assemblies has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.14 Continental Shielded Instrument Cable – CC2115

Continental Shielded Instrument Cables are used for 120 VAC/125 VDC control circuits outside containment, Rosemount transmitter circuits inside containment, and RTD circuits inside containment. The cables are required to function for up to 30 days after a LOCA or MSLB.

4.4.1.14.1 Thermal Analysis

The thermal aging test conditions qualify the cables for greater than 60 years at the current operating temperatures. There are no EQ maintenance requirements.

4.4.1.14.2 Radiation Analysis

The post-accident dose of 1.3×10^7 rads, gamma plus 4.0×10^6 rads, beta, plus 1.65×10^6 rads, gamma normal aging, 60-year dose results in a projected TID requirement of 1.87×10^7 rads, which is much less than the qualified value of 6.21×10^7 rads, gamma.

4.4.1.14.3 Conclusion

The qualification of Continental Shielded Instrument Cable, located inside and outside containment at RNP, has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.15 Continental/Anaconda Cable – Instrumentation

The Anaconda Instrumentation Cables are used in various safety-related instrumentation and control circuits. The cable must remain functional during and after a DBE.

4.4.1.15.1 Thermal Analysis

The thermal aging test conditions qualify the Anaconda Instrumentation Cables for greater than 60 years at the current operating temperatures. There is no temperature rise associated with the instrumentation and control applications of this cable (current less than 1 ampere).

4.4.1.15.2 Radiation Analysis

The post-accident dose of 1.3×10^7 rads gamma plus 8.0×10^7 rads, beta, plus the 1.65×10^6 rads normal aging, 60-year dose results in a projected dose of 9.47×10^7 rads TID, which is less than the qualified value of 2.0×10^8 rads.

4.4.1.15.3 Conclusion

The qualification of Anaconda Instrumentation Cables has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.16 Samuel Moore Dekoron Instrumentation Cables (EPDM & XLPO Insulations)

The Samuel Moore Dekoron Instrumentation Cables (EPDM & XLPO Insulations) are used in various safety related instrumentation and control circuits. The cable must remain functional during and after a DBE.

4.4.1.16.1 Thermal Analysis

The thermal aging test conditions qualify the Samuel Moore Dekoron Instrumentation Cables (EPDM & XLPO Insulations) for greater than 60 years at the current operating temperatures. Dekoron Instrument and Control Cable utilized at RNP is installed in low current instrument applications and will not experience significant self-heating. Therefore the qualified life calculation will be based on the maximum normal ambient temperature to which the cable will be exposed (120^{o}F).

4.4.1.16.2 Radiation Analysis

The post accident dose of $1.3x10^{7}$ rads gamma plus $8.0x10^{7}$ rads, beta, plus the $1.65x10^{6}$ rads, normal aging, 60-year dose results in a projected dose of $9.47x10^{7}$ rads TID, which is less than the qualified value of $2.0x10^{8}$ rads.

4.4.1.16.3 Conclusion

The qualification of Samuel Moore Dekoron Instrumentation Cables (EPDM & XLPO Insulations) has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.17 Eaton Corporation Dekoron Cable 16 AWG

The Eaton Corporation Dekoron Cable 16 AWG is used in various safety related instrumentation and control circuits. The cable must remain functional during and after a DBE.

4.4.1.17.1 Thermal Analysis

The thermal aging test conditions qualify the Eaton Corporation Dekoron Cable 16 AWG for greater than 60 years at the current operating temperatures. There is no temperature rise associated with the instrumentation and control applications of this cable (current less than 1 ampere).

4.4.1.17.2 Radiation Analysis

The post accident dose of $1.3x10^7$ rads gamma plus $8.0x10^7$ rads, beta, plus the $1.65x10^6$ rads, normal aging, 60-year dose results in a projected dose of $9.47x10^7$ rads TID, which is less than the qualified value of $2.0x10^8$ rads.

4.4.1.17.3 Conclusion

The qualification of Eaton Corporation Dekoron Cable 16 AWG has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.18 Raychem WCSF-N Splices

Raychem WCSF-N Splices are used throughout the plant to provide electrical and physical integrity to electrical connections in power, control, and shielded instrument cables. They are required to perform this function during and after a LOCA or MSLB for up to 30 days post-accident, with a minimum insulation resistance of 1.0×10^6 ohms.

4.4.1.18.1 Thermal Analysis

The thermal aging test conditions qualify the splices for greater than 60 years at the maximum operating temperature of 72°C. There are no EQ maintenance requirements for these splices.

4.4.1.18.2 Radiation Analysis

The post-accident dose of 1.3×10^7 rads, gamma plus 8.0×10^7 rads, beta, plus 1.65×10^6 rads, gamma normal aging, 60-year dose results in a projected dose of 9.47×10^7 rads TID, which is much less than the qualified value of 2.0×10^8 rads, gamma.

4.4.1.18.3 Conclusion

The qualification of Raychem WCSF-N splices has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.19 Raychem Splices – NPKV Stub Kits

Raychem NPKV Cable Splice Kits are used throughout the plant to provide a signal path for safety-related circuits. They must remain functional during and after a LOCA or MSLB for up to 30 days post-accident, with a minimum insulation resistance of $1.0x10^6$ ohms.

4.4.1.19.1 Thermal Analysis

The thermal aging test conditions qualify the splices for greater than 60 years at the maximum operating temperature of 72°C. There are no EQ maintenance requirements for these splice kits.

4.4.1.19.2 Radiation Analysis

The post-accident dose of $1.3x10^7$ rads, gamma plus $8.0x10^7$ rads, beta, plus $1.65x10^6$ rads, gamma normal aging, 60-year dose results in a projected dose of $9.47x10^7$ rads TID, which is much less than the qualified value of $2.15x10^8$ rads, gamma.

4.4.1.19.3 Conclusion

The qualification of Raychem NPKV Cable Splice Kits has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.20 Raychem Splices – NPK Connection Kits

Raychem NPK Cable Splice Kits are used throughout the plant to provide a signal path for safety related circuits. They must remain functional during and after a LOCA or MSLB for up to 30 days post-accident, with a minimum insulation resistance of 1.0×10^6 ohms.

4.4.1.20.1 Thermal Analysis

The thermal aging test conditions qualify the splices for greater than 60 years at the maximum operating temperature of 72°C. There are no EQ maintenance requirements for these splice kits.

4.4.1.20.2 Radiation Analysis

The post-accident dose of 1.3×10^7 rads, gamma plus 8.0×10^7 rads, beta, plus 1.65×10^6 rads, gamma normal aging, 60-year dose results in a projected dose of 9.47×10^7 rads TID, which is much less than the qualified value of 2.15×10^8 rads, gamma.

4.4.1.20.3 Conclusion

The qualification of Raychem NPK Cable Splice Kits has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.21 Raychem Splices – NMCK Connection Kits

The Raychem NMCKs must remain functional during and after a DBE. The Raychem NMCK product line is designed to insulate and environmentally seal Class 1E cable connections for low voltage motors (1000 volts or less). The kits can be purchased in three configurations, a V-stub configuration, an in-line configuration, and a Y-configuration. Sizes may also vary depending on the field cable and motor lead sizes. Regardless of the configuration or size, all NMCKs incorporate the same materials and insulating / sealing method.

4.4.1.21.1 Thermal Analysis

The thermal aging test conditions qualify the NMCK for 68 years at the maximum rated temperature of 90^oC.

4.4.1.21.2 Radiation Analysis

The post accident dose of $1.3x10^7$ rads gamma plus $8.0x10^7$ rads beta, plus the $1.65x10^6$ normal aging 60-year dose, results in a projected dose of $9.47x10^7$ rads TID, which is less than the qualified value of $2.0x10^8$ rads.

4.4.1.21.3 Conclusion

The qualification of Raychem NMCK Connection Kits has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.22 Raychem Splices – NESK End Seal Kits

The Raychem NESKs must remain functional during and after a DBE. Generally, the function of the Nuclear End Sealing Kit is to provide electrical and physical integrity when sealing unused conductors. NESKs are frequently utilized as subcomponents within Raychem splices.

NESKs are designed for low voltage applications (1000 VAC or less, Section 8, Attachment 2) and since RNP has no electrical distribution system between 480 VAC and 1000 VAC, there is a significant design margin in their application. Since the NESK is an end cap for an unused conductor, there is no current flow through the NESK.

4.4.1.22.1 Thermal Analysis

The thermal aging test conditions qualify the NMCK for 68 years at the maximum rated temperature of $90^{o}C$.

4.4.1.22.2 Radiation Analysis

The post accident dose of $1.3x10^7$ rads gamma plus $8.0x10^7$ rads beta plus the $1.65x10^6$ normal aging, 60-year dose results in a projected dose of $9.47x10^7$ rads TID, which is less than the qualified value of $2.0x10^8$ rads.

4.4.1.22.3 Conclusion

The qualification of Raychem NESK End Seal Kits has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.23 AMP Butt Splices

AMP Pre-Insulated Butt Connectors are required to function during and after a DBE. The three models addressed are identical except for wire size. The qualified life of the Amp connectors is predicated on their use as part of a splice. The connectors are used in various locations inside containment.

4.4.1.23.1 Thermal Analysis

The thermal aging test conditions qualify the AMP connectors for greater than 60 years at the normal operating temperatures.

4.4.1.23.2 Radiation Analysis

The post accident dose of 1.3×10^7 rads gamma plus the 1.65×10^6 rads gamma normal aging, 60-year dose results in a projected dose of 1.47×10^7 rads TID, which is less than the qualified value of 1.65×10^7 rads.

4.4.1.23.3 Conclusion

The qualification of AMP Pre-Insulated Butt Connectors has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.24 AMP PIDG Terminals

Amp PIDG Terminals are installed inside containment within sealed instrumentation or underneath Raychem heat shrink tubing. They must remain functional during and after a LOCA or MSLB for up to 30 days post-accident.

4.4.1.24.1 Thermal Analysis

The thermal aging test conditions qualify the splices for greater than 60 years at the maximum operating temperature of 72°C. There are no EQ maintenance requirements for these terminals.

4.4.1.24.2 Radiation Analysis

The post-accident dose of $1.3x10^7$ rads, gamma plus $8.0x10^7$ rads, beta, plus $1.65x10^6$ rads, gamma normal aging, 60-year dose results in a projected dose of $9.47x10^7$ rads TID, which is much less than the qualified value of $2.59x10^8$ rads, gamma.

4.4.1.24.3 Conclusion

The qualification of Amp PIDG Terminals has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.25 CM-303 Tape Splice Assemblies – Scotch 27 and Scotch 70

The CM-303 Tape Splice Assemblies, composed of Scotch #27 and #70 tapes and assembled per approved procedure, are installed inside and outside containment as tape splices for Class H motor leads and control circuits. The splices are required to function during and after a DBE for up to 30 days.

4.4.1.25.1 Thermal Analysis

The thermal aging test conditions qualify the splices for greater than 60 years provided that operating temperature limits are met. The user of the splice is responsible for determining the operating temperature based on the design of a specific installation.

4.4.1.25.2 Radiation Analysis

The post-accident dose of $1.3x10^7$ rads gamma plus $8.0x10^7$ rads, beta, plus the $1.65x10^6$ rads, normal aging, 60-year dose results in a projected dose of $9.47x10^7$ rads TID. This is less than the minimum qualified value of $2.0x10^8$ rads.

4.4.1.25.3 Conclusion

The qualification of CM-303 Tape Splice Assemblies, composed of Scotch #27 and #70 tapes, has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.26 Kerite HTK Power Cable

Kerite HTK Power Cables are installed inside containment to provide 480 VAC power to HVH motors. The cables must remain functional during and after a LOCA or MSLB for up to 30 days post-accident.

4.4.1.26.1 Thermal Analysis

The thermal life of the HTK insulation is calculated to be 194 years at 72°C, which is the worst-case operating temperature for any cable within the scope of LR. This is sufficient to demonstrate qualification for the entire period of extended operation. There are no EQ maintenance requirements for these cables.

4.4.1.26.2 Radiation Analysis

The post-accident dose of 1.3×10^7 rads, gamma plus 8.0×10^7 rads, beta, plus 3.45×10^3 rads, gamma normal aging, 60-year dose results in a projected dose of 9.30×10^7 rads TID, which is much less than the qualified value of 2×10^8 rads, gamma.

4.4.1.26.3 Conclusion

The qualification of Kerite HTK Power Cable has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.27 Kerite FR2/FR3 Insulated Multiconductor Cable

Kerite FR2/FR3 Insulated Multiconductor Cables are required to function during and after a DBE to provide 120 VAC or 125 VDC power for safety-related solenoids and limit switches inside and outside containment. The cables must function for up to 30 days after a DBE.

4.4.1.27.1 Thermal Analysis

The thermal aging test conditions qualify the cables for greater than 60 years at the maximum operating temperature of 120°F. Since the load current for all of the cables is less than 2 amperes, self-heating effects are insignificant and the operating temperature is equivalent to the room ambient temperature. There are no EQ maintenance requirements for these cables.

4.4.1.27.2 Radiation Analysis

The post-accident dose of 1.3×10^7 rads, gamma plus 4.0×10^6 rads, beta, plus 1.65×10^6 rads, gamma normal aging, 60-year dose results in a projected dose of 1.87×10^7 rads TID, which is much less than the qualified value of 1.05×10^8 rads, gamma.

4.4.1.27.3 Conclusion

The qualification of Kerite FR2/FR3 Insulated Multiconductor Cables has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.28 Thomas and Betts STA-KON Terminal

Thomas & Betts Tefzel Insulated STA-KON Terminals are required to function during and after a DBE. The terminals are used inside containment and are installed within sealed instrumentation or underneath Raychem heat shrink tubing. These particular applications result in isolation of the terminals from any moisture environments. They will not see the extreme pressures and humidity expected during LOCA, and will not be exposed to chemical spray.

4.4.1.28.1 Thermal Analysis

The thermal aging test conditions qualify the Thomas & Betts Tefzel Insulated STA-KON Terminals for greater than 60 years at a service temperature of $65^{o}C$ (to account for heat rise). This is much higher than the normal service temperatures, providing additional conservatism.

4.4.1.28.2 Radiation Analysis

The post accident dose of $2.3x10^{7}$ rads gamma plus the $1.65x10^{6}$ rads gamma normal aging, 60-year dose results in a projected dose of $2.47x10^{7}$ rads, TID, which is much less than the qualified value of $2.0x10^{8}$ rads.

4.4.1.28.3 Conclusion

The qualification of Thomas & Betts Tefzel Insulated STA-KON Terminals has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.29 Conax Electrical Conductor Seal Assemblies – ECSA

Conax ECSAs are required to maintain mechanical seal and electrical integrity before, during, and after a DBE. Conax ECSAs provide a mechanical seal and electrical integrity for RTDs TE-413-1 and TE-413-2 located inside containment.

4.4.1.29.1 Thermal Analysis

The thermal aging test conditions qualify the Conax ECSAs for greater than 60 years at the maximum conductor temperature rating of 90°C. Qualified life was calculated at 90°C for conservatism to account for proximity of the ECSA to other heat sources.

4.4.1.29.2 Radiation Analysis

The post accident dose of $1.5x10^7$ rads gamma plus the $1.65x10^6$ rads gamma normal aging, 60-year dose results in a projected dose of $1.67x10^7$ rads, TID, which is less than the qualified value of $2.25x10^8$ rads.

4.4.1.29.3 Conclusion

The qualification of Conax ECSAs has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.30 Conax Electrical Penetration Assemblies

Conax Electrical Instrumentation Penetration Assemblies are required to maintain mechanical seal and electrical integrity before, during, and after a DBE. Conax Electrical Instrumentation Penetration Assemblies provide a mechanical seal and electrical integrity for penetrations C1, C2, C5, C9, D9, E1, and E10.

4.4.1.30.1 Thermal Analysis

The thermal aging test conditions qualify the Conax Electrical Instrumentation Penetration Assemblies for greater than 60 years at the current operating conditions.

4.4.1.30.2 Radiation Analysis

The post accident dose of 1.5×10^7 rads gamma plus the 1.65×10^6 rads gamma normal aging, 60-year dose results in a projected dose of 1.67×10^7 rads, TID, which is less than the qualified value of 2.25×10^8 rads.

4.4.1.30.3 Conclusion

The qualification of Conax Electrical Instrumentation Penetration Assemblies has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.31 Westinghouse CET/CCM – Incore T/C Connectors and MI Cable Assemblies

Westinghouse CET/CCM – Incore T/C Connectors and MI Cable Assemblies are a part of the Core Exit Thermocouple/Core Cooling Monitor System, installed inside containment. The equipment must continuously maintain the output signals of the thermocouples during a LOCA or MSLB for up to 30 days post-accident, with a minimum insulation resistance of 0.1 megohm.

4.4.1.31.1 Thermal Analysis

The qualified life of the equipment is calculated to be 44 years. Based on the installation of this equipment in 1987, this qualified life is sufficient for the entire period of extended operation. There are no EQ maintenance requirements.

4.4.1.31.2 Radiation Analysis

The post-accident dose of 2.0×10^8 rads, gamma plus 2.0×10^8 rads, beta, plus 1.18×10^6 rads, gamma normal aging, 43-year dose results in a projected dose of 4.01×10^8 rads, TID, which is much less than the qualified value of 1.46×10^9 rads.

4.4.1.31.3 Conclusion

The qualification of Westinghouse CET/CCM – Incore T/C Connectors and MI Cable Assemblies has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.32 Westinghouse CET/CCM – Reference Junction Boxes and Potting Adaptors

Westinghouse CET/CCM – Reference Junction Boxes and Potting Adaptors are a part of the Core Exit Thermocouple/Core Cooling Monitor System, installed inside containment. The equipment must have continuity, maintain a compensated output of the thermocouples to within +/- 10°F of the actual temperature, and maintain enclosure seal integrity during accident conditions, for up to 30 days post-accident.

4.4.1.32.1 Thermal Analysis

The qualified life of the equipment is calculated to be greater than 47 years at the current operating conditions. Based on the installation of this equipment in 1987, this qualified life is sufficient for the entire period of extended operation. There are no self-heating effects for this instrumentation application (thermocouple signals). The potting adaptors have a qualified life of 29 years. Therefore, replacement of these adaptors before the end of their qualified life is an EQ maintenance requirement.

4.4.1.32.2 Radiation Analysis

The post-accident dose of 1.3×10^7 rads, gamma plus 8.0×10^7 rads, beta, plus 1.65×10^6 rads, gamma normal aging, 60-year dose results in a projected dose of 9.47×10^7 rads, TID, which is less than the qualified value of 1.65×10^8 rads.

4.4.1.32.3 Conclusion

The qualification of Westinghouse CET/CCM – Reference Junction Boxes and Potting Adaptors has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.33 Westinghouse CET/CCM – Intermediate Disconnect Box Connectors

Westinghouse CET/CCM Connectors in the Intermediate Disconnect Boxes (IDB) are located inside containment and are required to operate continuously during and after a DBE for up to 30 days post-accident. These components are part of the Core Exit Thermocouple/Core Cooling Monitor (CET/CCM) System.

4.4.1.33.1 Thermal Analysis

The thermal aging test conditions qualify the equipment for greater than 60 years at the current operating temperatures.

4.4.1.33.2 Radiation Analysis

The post accident dose of 1.3×10^7 rads, gamma plus 8.0×10^7 rads, beta, plus 1.65×10^6 rads, gamma normal aging, 60-year dose results in a projected dose of 9.47×10^7 rads, TID, which is much less than the qualified value of 1.6×10^8 rads, gamma.

4.4.1.33.3 Conclusion

The qualification of Westinghouse CET/CCM Connectors in the Intermediate Disconnect Boxes has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.34 Gamma – Metrics Excore Neutron Detectors

Gamma-Metrics Excore Neutron Flux Detectors are located inside containment and must remain functional during and after a DBE for up to 30 days. The equipment must function with a sensitivity of +/- 10% and insulation resistance of at least 1.0×10^8 ohms.

4.4.1.34.1 Thermal Analysis

The neutron flux detector and mineral-insulated cable do not contain non-metallics and as such are not susceptible to the effects of thermal aging. The thermal aging test conditions for the organic cable and the silicone rubber O-ring qualify these components for greater than 60 years at the current operating temperatures. Despite the calculated qualified life, the manufacturer recommends, and CP&L incorporated as EQ required maintenance, replacement of the silicone rubber O-ring every 10 years.

4.4.1.34.2 Radiation Analysis

For the organic cable, the post accident dose of 1.3×10^7 rads, gamma plus 8.0×10^7 rads, beta, plus 1.65×10^6 rads, gamma normal aging, 60-year dose results in a projected dose of 9.47×10^7 rads, TID, which is much less than the qualified value of 3.2×10^9 rads, gamma. The detector and MI cable assembly have no significant radiation aging mechanisms.

4.4.1.34.3 Conclusion

The qualification of Gamma-Metrics Excore Neutron Flux Detectors has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.35 Pyco Resistance Temperature Detectors (RTDs)

Pyco RTDs are installed both inside containment and in the Pipe Alley. The RTDs must function within an accuracy of +/- 1/2 % full scale during accident conditions, for up to 30 days post-accident.

4.4.1.35.1 Thermal Analysis

The qualified life of the RTDs inside containment is calculated to be 13 years at the current operating conditions. For TE-606 in the Pipe Alley, the qualified life is calculated to be greater than 60 years. The operating temperatures for these RTDs includes heating due to the process fluids where they are located. Based on their qualified life, the RTDs located inside containment must be replaced every 12 years as an EQ maintenance requirement to ensure that their qualified life is not exceeded. The cover gasket and grommet in the Patel Conduit Seal must also be replaced whenever the installations are opened.

4.4.1.35.2 Radiation Analysis

The post-accident dose of 1.3×10^7 rads, gamma plus 8.0×10^7 rads, beta, plus 1.65×10^6 rads, gamma normal aging, 60-year dose results in a projected dose of 9.47×10^7 rads, TID, which is less than the qualified value of 2.203×10^8 rads.

4.4.1.35.3 Conclusion

The qualification of Pyco RTDs has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii). The qualification of Pyco RTDs will be maintained by periodic replacement of RTDs located in containment and replacement of cover gaskets and grommets as necessary in accordance with the EQ Program. Periodic replacement of RTDs and gaskets and grommets does not involve a TLAA, because the period of replacement is not defined by the current 40-year operating term of the plant.

4.4.1.36 Buchanan Terminal Blocks

Buchanan NQB-108 Terminal Blocks are used inside NEMA 4 enclosures located outside containment as a part of various safety-related circuits. They are rated at 600 volts, 50 amperes and constructed of Durez 152 phenolic. The blocks are required to remain functional and maintain their structural integrity during and after a DBE (radiation only). They must maintain a minimum insulation resistance of $1.0x10^6$ ohms.

4.4.1.36.1 Thermal Analysis

The thermal aging test conditions qualify the Buchanan Terminal Blocks for greater than 60 years at the design ambient temperature of 104°F.

4.4.1.36.2 Radiation Analysis

The post accident dose of $2.6x10^6$ rads gamma plus the negligible, normal aging, 60-year dose results in a projected dose of $2.6x10^6$ rads, TID, which is much less than the qualified value of $2.0x10^8$ rads.

4.4.1.36.3 Conclusion

The qualification of Buchanan Terminal Blocks has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.37 Barton Pressure Switches – Model 580A

The installed Barton 580A-2 differential pressure indicating switches are required to function during and after a DBE (radiation only). Switch actuation is required to occur within an accuracy of +/- 4% full scale. The four (4) installed switches are all located in the Charging Pump Room of the Reactor Auxiliary Building at RNP.

4.4.1.37.1 Thermal Analysis

The thermal aging test conditions qualify the Barton switches for greater than 60 years at the design ambient temperature of 104°F. There is no temperature rise associated with this instrumentation application.

4.4.1.37.2 Radiation Analysis

The post-accident dose of 1.0×10^6 rads gamma plus the 5.26×10^4 rads, normal aging, 60-year dose results in a projected dose of 1.05×10^6 rads, TID, which is much less than the qualified value of 1.0×10^7 rads. Since this equipment is located outside containment, beta radiation is not a consideration.

4.4.1.37.3 Conclusion

The qualification of Barton 580A-2 differential pressure indicating switches has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.38 NAMCO Receptacle and Connector/Cable Assemblies – Model EC210

NAMCO EC210 Series Receptacle and Connector/Cable Assemblies are used inside and outside containment to provide an environmental seal and quick disconnect electrical connection for junction boxes, limit switches, and other enclosures. The equipment is required to function during and after a DBE. They must maintain a minimum insulation resistance of 1.0×10^6 ohms.

4.4.1.38.1 Thermal Analysis

The thermal aging test conditions qualify the NAMCO EC210 Series Receptacle and Connector/Cable Assemblies for greater than 60 years at the current operating temperatures. Components require replacement based on environmental conditions at the installed location.

4.4.1.38.2 Radiation Analysis

The post accident dose of 1.3×10^7 rads gamma plus 8.0×10^7 rads, beta, plus the 1.5×10^6 rads, gamma normal aging, 60-year dose results in a projected dose of 9.45×10^7 rads, TID, which is much less than the qualified value of 2.04×10^8 rads.

4.4.1.38.3 Conclusion

The qualification of NAMCO EC210 Series Receptacle and Connector/Cable Assemblies has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.39 Victoreen High Range Radiation Detectors

Victoreen High Range Containment Radiation Monitor (HRCRM) Model 875 is located inside containment and is required to operate continuously during and after a DBE for up to 30 days post-accident. The required minimum insulation resistance is $1.0x10^6$ ohms and the system accuracy must be within +/- 36% of the input radiation.

4.4.1.39.1 Thermal Analysis

The thermal aging test conditions qualify the HRCRM for 60 years at the current operating temperatures. Based on the installation date of the equipment, this qualified life is sufficient to reach the end of the extended period of operation.

4.4.1.39.2 Radiation Analysis

The post accident dose of $1.3x10^7$ rads, gamma plus $8.0x10^7$ rads, beta, plus $1.65x10^6$ rads, gamma normal aging, 60-year dose results in a projected dose of $9.47x10^7$ rads, TID, which is much less than the qualified value of $2.0x10^8$ rads, gamma.

4.4.1.39.3 Conclusion

The qualification of Victoreen HRCRM, Model 875, has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.40 Brand Rex Cable – Instrumentation

The subject equipment is required to remain functional during and after a DBE. The Brand-Rex cable is used in five applications at RNP: 1) 125 VDC control circuits inside containment, 2) 125 VDC / 120 VAC control circuits outside containment, 3) Rosemount transmitter circuits inside containment, 4) Rosemount transmitter circuits outside containment, and 5) RTD circuits inside containment.

4.4.1.40.1 Thermal Analysis

The thermal aging test conditions qualify the Brand Rex Instrumentation cable for greater than 60 years at the maximum operating temperature of 120^{o}F. Brand-Rex Instrument Cable utilized at RNP is installed in low current, instrument applications and will not experience significant self-heating. Therefore, the qualified life calculation was be based on the maximum operating temperature to which the cable could be exposed (120^{o}F).

4.4.1.40.2 Radiation Analysis

The post accident dose of $1.3 x 10^7$ rads gamma plus the $8.0 x 10^7$ rads beta plus the $1.65 x 10^6$ rads, normal aging, 60-year dose results in a projected dose of $9.47 x 10^7$ rads, TID, which is less than the qualified value of $2.0 x 10^8$ rads.

4.4.1.40.3 Conclusion

The qualification of Brand Rex Instrumentation cable has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.41 Brand Rex Cable –Control

The Brand Rex Control cable is required to function during and after a DBE. The cable is used only for Rosemount transmitter loops located in the SI Pump Room and the Pipe Alley. The subject cables are exposed to radiation only for post-accident environments. In the event of the failure of the fan cooler units in the SI pump room (due to the elevated radiation), the peak post-accident temperature postulated for this location is 101.7^oC (from motor operation).

4.4.1.41.1 Thermal Analysis

The thermal aging test conditions qualify the Brand Rex Control cable for greater than 60 years at a temperature of 104^oF. Brand-Rex Control Cable utilized at RNP is installed in low current, instrument applications (Rosemount transmitters) and will not experience significant self-heating. Therefore the qualified life calculation will be based on the design ambient temperature to which the cable will be exposed (104^oF).

4.4.1.41.2 Radiation Analysis

The post accident dose of $7.2x10^6$ rads gamma plus the negligible, normal aging, 60-year dose results in a projected dose of $7.2x10^6$ rads. This TID is less than the qualified value of $2.0x10^8$ rads.

4.4.1.41.3 Conclusion

The qualification of Brand Rex Control cable has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.42 Raychem Cable – Flamtrol

Raychem Flamtrol 1000V Control cable is used in various applications at RNP.

4.4.1.42.1 Thermal Analysis

The thermal aging test conditions qualify the cable for in excess of 350 years at 104°F. Since the cable is used in a control circuit and not continuously energized, the effects of self-heating are negligible.

4.4.1.42.2 Radiation Analysis

The post accident dose of 7.2×10^6 rads gamma plus the negligible, normal aging, 60-year dose results in a projected dose of 7.2×10^6 rads, TID, which is less than the qualified value of 1.1×10^8 rads.

4.4.1.42.3 Conclusion

The qualification of Raychem Flamtrol 1000V Control cable has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.43 Cable – PVC and XLPE Outside Containment

PVC and XLPE Cables are installed outside containment and are required for various safety-related, power, control, and instrumentation applications up to 480 VAC. The cables must function during accident conditions for up to 30 days post-accident.

4.4.1.43.1 Thermal Analysis

The qualified life of the cables is calculated to be greater than 60 years at the current operating conditions. There are no self-heating effects for these cables. There are no EQ maintenance requirements.

4.4.1.43.2 Radiation Analysis

The post-accident dose of 7.6×10^6 rads, gamma plus a negligible 60-year normal dose results in a projected dose of 7.6×10^6 rads, TID, which is less than the various tested values for these insulations.

4.4.1.43.3 Conclusion

The qualification of PVC and XLPE Cables, installed outside containment, has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.44 Greases – Motors and MOVs

Various lubricants are used inside containment and must remain functional during and after a DBE for up to 30 days. The lubricants are replaced and/or replenished in accordance with the Refueling Lubrication Data Sheet**.**

4.4.1.44.1 Thermal Analysis

The Arrhenius Theory cannot be applied to lubricants. Since the lubricants' temperature ratings are greater than their service temperatures, the lubricants are qualified for their normal service temperature. By performing regular lubrication maintenance as specified by the equipment manufacturers, the lubricants are judged qualified for a 60-year life.

4.4.1.44.2 Radiation Analysis

The post-accident dose of 1.3×10^7 rads, gamma, plus 1.65×10^6 rads, gamma normal aging, 60-year dose results in a projected dose of 1.47×10^7 rads, TID, which is less than the qualified values for the various lubricants. Beta radiation is not a consideration for these lubricants due to the attenuation of beta particles by metal enclosures.

4.4.1.44.3 Conclusion

Qualification of various motor and MOV lubricants has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii). The qualification of the lubricants will be maintained by regular lubrication maintenance in accordance with the Refueling Lubrication Data Sheet. Periodic lubrication maintenance does not involve a TLAA, because the period of lubrication maintenance is not defined by the current 40-year operating term of the plant.

4.4.1.45 Target Rock Solenoid Valves

Target Rock Solenoid Operated Globe Valves, Model Nos. 80B-001 and 1031210-2, are used inside containment to provide venting and isolation capabilities for a number of applications.

4.4.1.45.1 Thermal Analysis

The thermal aging test conditions qualify the Target Rock Solenoid Operated Globe Valves, Model Nos. 80B-001 and 1031210-2, for greater than 60 years at the current operating temperatures. Gaskets and O-rings must be replaced whenever they are disturbed by maintenance activities.

4.4.1.45.2 Radiation Analysis

The post accident dose of 1.3×10^7 rads gamma plus 8.0×10^7 rads, beta, plus the 1.65×10^6 rads, gamma normal aging, 60-year dose results in a TID requirement of 9.47×10^7 rads TID, which is much less than the qualified value of 2.7×10^8 rads.

4.4.1.45.3 Conclusion

Qualification of Target Rock Solenoid Operated Globe Valves, Model Nos. 80B-001 and 1031210-2, has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii). Valve qualification will be maintained by replacement of subcomponents (gaskets and o-rings) in accordance with the EQ Program. Replacement of these subcomponents does not involve a TLAA, because the period of replacement is not defined by the current 40-year operating term of the plant.

4.4.1.46 Boston Insulated Wire – Cable

BIW Model 15948-H-004 Cables are located inside containment and must remain functional during and after a DBE for up to 30 days. These cables are 4/C #16AWG twisted shielded and are used in the RVLIS RTD circuits.

4.4.1.46.1 Thermal Analysis

The cables are utilized in the RVLIS RTD circuits, where self-heating effects are negligible due to the low operating current of the RTDs. Using the current ambient conditions inside containment, the qualified life is greater than 60 years.

4.4.1.46.2 Radiation Analysis

The post accident dose of 1.3×10^7 rads, gamma plus 8.0×10^7 rads, beta, plus 1.65×10^6 rads, gamma normal aging, 60-year dose results in a projected dose of 9.47×10^7 rads, TID, which is much less than the qualified value of 2.0×10^8 rads, gamma.

4.4.1.46.3 Conclusion

The qualification of BIW Model 15948-H-004 Cables has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.47 Honeywell Model V4-21 Microswitch Assembly

A Honeywell Model V4-21 Microswitch Assembly is located on a flow-indicating controller in the pipe alley. The switch is required to operate during a LOCA inside containment for up to 30 days post-accident (radiation-only). A radiation and thermal analysis was performed to qualify the switch under the requirements of the DOR Guidelines.

4.4.1.47.1 Thermal Analysis

The lowest expected life of any of the component materials is 266 years, which is sufficient to justify installation for the entire period of extended operation.

4.4.1.47.2 Radiation Analysis

The post-accident dose of 7.2×10^6 rads, gamma, plus negligible normal aging, 60-year dose results in a projected dose of 7.2×10^6 rads, TID. This is less than the lowest radiation damage threshold of 7.4×10^6 rads, gamma.

4.4.1.47.3 Conclusion

The qualification of the Honeywell Microswitch Assembly has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.48 RAM-Q Connectors

RAM-Q Connectors are located inside containment and are required to operate during and after a DBE for up to 30 days post-accident. They are used in dual element RTD circuits in the hot and cold legs of the RCS.

4.4.1.48.1 Thermal Analysis

The thermal aging test conditions qualify the cable, connector, and Raychem sleeving for greater than 60 years at the current operating temperatures. There is no self-heating associated with these connectors in RTD circuits.

4.4.1.48.2 Radiation Analysis

The post-accident dose of 5.6×10^6 rads, gamma, plus 1.65×10^6 rads, gamma normal aging, 60-year dose results in a projected dose of 7.25×10^6 rads TID. This is much less than the qualified values of 9.94×10^7 rads, gamma for the connector and 1.84×10^8 rads, gamma for the cable and WCSF-N sleeving.

4.4.1.48.3 Conclusion

The qualification of the RAM-Q Connectors has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.2 GSI-168, ENVIRONMENTAL QUALIFICATION OF ELECTRICAL COMPONENTS

NRC guidance for addressing GSI-168 for license renewal is contained in the June 2, 1998, NRC letter to NEI [Reference 4.4-5]. 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 Statements of Consideration is to provide a technical rationale demonstrating that the current licensing basis for environmental qualification pursuant to 10 CFR 50.49 will be maintained in the period of extended operation. Although the Statements of Consideration 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 RNP. 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. The evaluations are provided in Subsection 4.4.1 of this Application. Consistent with the above NRC guidance, no additional information is required to address GSI-168 in a renewal application at this time.

4.4.3 REFERENCES

4.4-1 DOR Guidelines, “Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors.” U.S. Nuclear Regulatory Commission, June 1979.

4.4-2 NUREG-0588, “Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment,” U.S. Nuclear Regulatory Commission, July 1981.

4.4-3 Regulatory Guide 1.89, Revision 1, “Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, June 1984.

4.4-4 EPRI NP-1558, “A Review of Equipment Aging Theory and Technology,” Electric Power Research Institute, September 1980.

4.4-5 NRC letter (C. Grimes) to NEI (D. Walters), dated June 2, 1998, “Guidance on Addressing GSI 168 for License Renewal,” Project 690.

4.5 CONTAINMENT TENDON LOSS OF PRESTRESS

The RNP containment building is a steel lined concrete shell in the form of a vertical right cylinder with a hemispherical dome and a flat base. The dome and base are constructed of reinforced concrete. The cylinder walls are concrete - reinforced circumferentially and prestressed vertically.

Prestressing force is not constant; it decreases over time due to a variety of design conditions. The following design conditions were considered in the original evaluation of the containment prestressing tendons.

- Steel Relaxation
- Concrete Shrinkage
- Concrete Creep
- Elastic Shortening of Concrete
- 2% reduction for broken tendons

For license renewal, the calculation of prestress was updated to address potential losses through the period of extended operation. The new calculation considers the above factors that influence loss of prestress. However, the value for concrete shrinkage was marginally reduced based on a comparison to estimated shrinkage values used in the original calculation; as well as, reference to the time of application of loading compared to completion of the containment walls. Specifically, the original analysis used a shrinkage coefficient of 0.0003, and the original containment design information estimates the actual shrinkage to be 0.00005. The value used in the revised calculation is 0.0002. This is supported by the fact that shrinkage is a volume change in concrete that occurs with time rather than with load; as such, higher values are more realistic for pretensioned members where the prestress is transferred to the concrete at an early age, whereas the lower value is more appropriate for post-tensioned members. RNP tendons are considered to be post-tensioned because the tendons were not loaded until after the concrete was placed. This allowed a portion of the shrinkage to occur prior to tendon tensioning.

No prestress losses were considered for elastic shortening, due to the re-tensioning of the tendons approximately a month after the initial tensioning.

No reduction in prestress was taken for general corrosion based on review of the 5-year and 25-year surveillance tendon inspections. For example, visual examination of the 25-year tendon noted upon removal of the grout surrounding the tendon: “The surface of the bars were covered with a reddish-brown oxide that could be removed simply by wiping the surface clean by hand. No measurable metal loss or etching could be detected once the dust was removed.” Therefore, grouting the tendons has proven to be effective for the prevention of corrosion. The calculation projects the prestress losses over 60 years; however, the tendons were originally tensioned a few months prior to the original licensing date of the plant. As such, the actual prestress period for the tendons is more than 60 years. Based on comparison of the evaluated margin to the required minimum prestress the slight increase in duration will not allow the actual prestress to go below the required minimum.

Based on the above, analysis of tendon prestress has determined that the final effective prestress at the end of 60 years exceeds the minimum required value. Consequently, the post-tensioning system will continue to perform its intended function throughout the period of extended operation. Therefore, the analysis associated with containment tendon loss of prestress has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

4.6 OTHER PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES

4.6.1 THERMAL AGING EMBRITTLEMENT

Fracture mechanics analyses of Cast Austenitic Stainless Steel (CASS) components in the Reactor Coolant System are considered to be time-limited aging analyses because of the effects of thermal aging. For RNP, these analyses are the Leak-Before-Break analysis of Reactor Coolant System piping and welds and the analysis of Reactor Coolant Pumps in support of ASME Code, Section XI, Code Case N-481.

Leak-Before-Break

In accordance with the current licensing basis, a Leak-Before-Break (LBB) analysis was performed to show that any potential leaks that develop in the Reactor Coolant System loop piping can be detected by plant leak monitoring systems before a postulated crack causing the leak would grow to unstable proportions during the 40-year plant life.

LBB evaluations postulate a surface flaw at a limiting stress location, and demonstrate that a through-wall crack will not result following exposure to a lifetime of design transients. A separate evaluation assumes a through-wall crack of sufficient size, such that the resultant leakage can be easily detected by the existing leakage monitoring system, and then demonstrates that, even under maximum faulted loads, this crack is much smaller (with margin) than a critical flaw size that could grow to pipe failure. The aging effects to be addressed during the period of extended operation include thermal aging of the primary loop piping components and fatigue crack growth. Thermal aging refers to the gradual change in the microstructure and properties of a material due to its exposure to elevated temperatures for an extended period of time. The only significant thermal aging effect on the Reactor Coolant System loop piping is embrittlement of the cast austenitic stainless steel components. This effect results in a reduction in fracture toughness of the material.

WCAP-15628 [Reference 4.6-1] is a new leak-before-break (LBB) calculation applicable to RNP large bore Reactor Coolant System (RCS) piping and components that includes allowances for reduction of fracture toughness of cast austenitic stainless steel due to thermal embrittlement during a 60-year operating period. The new analysis meets the requirements for LBB required by 10 CFR 50, Appendix A, General Design Criterion 4, and uses the recommendations and criteria from the NRC Standard Review Plan for LBB evaluations. The new analysis uses the 40-year design basis thermal transients as input for the fracture mechanics analyses. These transients have been shown to be conservative for the 60-year operating period. Therefore, the RCS primary loop piping Leak-Before-Break analysis has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

Code Case N-481 Fracture Mechanics Analysis

Following ASME approval of Code Case N-481, “Alternate Examination Requirements for Cast Austenitic Pump Casings, Section XI, Division 1,” in March, 1990, the Westinghouse Owner’s Group sponsored WCAP-13045, which is a fracture mechanics analysis for the fleet of Westinghouse plants that demonstrated compliance with the code case on a generic basis. The code case permits substitution of surface examination in lieu of volumetric examination of the reactor coolant pump casing, provided a fracture mechanics analysis is prepared which meets specified requirements. The code case requires a plant-specific evaluation to demonstrate safety and serviceability of the pumps. Therefore, WCAP-15363, Rev. 0 was prepared in April 2000 as a plant-specific analysis for the Westinghouse Model 93 pumps at RNP to support using the alternate inspection techniques during the next ASME Section XI In-Service Inspection interval. Plant-specific loadings were compared to the generic loadings in the earlier evaluation, and plant-specific materials were compared to generic materials data used in the report, demonstrating the requirements of the code case were met for the 40-year operation of the plant.

In support of license renewal, a new report, WCAP-15363, Rev. 1 [Reference 4.6-2], was prepared. WCAP-15363, Rev. 1, supersedes WCAP-15363, Rev. 0, and includes an evaluation of the plant-specific pump casing material properties to account for reduced fracture toughness due to thermal embrittlement during the 60-year extended operational period. However, the new analysis uses the limiting transients from the 40-year design transient set. This is acceptable because the 40-year design transients have been shown to be conservative for 60 years of plant operation. WCAP-15363, Rev. 1, demonstrates that margin requirements for leakage and crack stability have been met. The new analysis permits the use of the surface examination of pump casings in lieu of volumetric examination in accordance with the Code Case throughout the period of extended operation. Therefore, the ASME Code Case N-481 analysis has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

4.6.2 FOUNDATION PILE CORROSION

Corrosion of Class 1 structure foundation piles was identified as a TLAA based on the evaluation of the piles for a 40-year corrosion loss. The original analysis determined corrosion losses would be negligible based on measured soil resistivity values that are so high the possibility of active corrosion is minimal.

The analysis relies on plant-specific data regarding soil resistivity and industry data from NUREG-1557 [Reference 4.6-3] and EPRI TR-103842 [Reference 4.6-4].

The RNP UFSAR states, "Any steel structure in soil (even without the protection afforded by concrete) is progressively less susceptible to corrosion as the electrical resistivity of the soil increases. Soil resistivity measurements taken in August 1958, prior to construction of Unit 1 and as reconfirmed by measurements taken at the construction site in December, 1966, have established that the soil resistivity is so high that the possibility of active corrosion is minimal".

NUREG-1557, "Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal" identifies corrosion of steel piles as a "Non-Significant ARDM," states "Steel piles driven in undisturbed soils have been unaffected by corrosion & those driven in disturbed soil experience minor to moderate corrosion to a small area of metal." EPRI TR-103842, "Class I Structures License Renewal Industry Report," states: "Romanoff examined corrosion data from 43 piling installations and on that basis drew some general conclusions regarding the corrosion of driven steel piles. These test installations had pile depths of up to 136 feet and time of exposure varying from 7 to 50 years in a wide variety of soil conditions. Romanoff's review of this data indicates that the type and amount of corrosion observed on steel pilings driven into undisturbed natural soil, regardless of the soil characteristics and properties, is not sufficient to significantly affect the strength of pilings as load bearing structures. The data also indicate that undisturbed soils are so deficient in oxygen at levels a few feet below the ground surface or below the water table, that steel piles are not appreciably affected by corrosion, regardless of the soil type or the soil properties."

A reanalysis of foundation pile corrosion for license renewal determined that corrosion losses would continue to remain non-significant for the period of extended operation and will not prevent the foundation piles from performing their license renewal intended functions. This conclusion is consistent with the recommendations and findings of NUREG-1557 and EPRI Report TR-103842 and is in accordance the estimated corrosion losses developed in the original analysis.

Therefore, the foundation pile corrosion analysis results have been projected to the end of the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

4.6.3 ELIMINATION OF CONTAINMENT PENETRATION COOLERS

In 1995, an evaluation was performed to justify eliminating the need for cooling water flow to the hot pipe containment penetration coolers to the maximum extent possible. As part of this effort, insulation was credited to reduce the temperature of the concrete surrounding the hot pipe penetrations. The performance requirement for the hot pipe penetrations was to maintain the surrounding concrete temperature below 200°F under normal operating conditions and other long term conditions.

Residual Heat Removal (RHR) system penetration S-15 did not require cooling water to be maintained because the concrete temperature around S-15 only exceeded 200°F during short duration transients and the temperature then was less than 350°F. In addition, the steady-state temperature without cooling water and continuous RHR flow at 380°F results in the temperature of the surrounding concrete of approximately 210°F

The analysis of concrete temperature determined that the allowable number of cycles of heatup and cooldown, at 40 hours or less per cycle, was 252 cycles. This is the total number of heatup/cooldown cycles the concrete surrounding the S-15 RHR penetration could experience temperatures greater than 200 °F over the balance of plant life figured from the year 1995. The balance of plant life was projected as 16 years (out of 40 years total plant life) when this calculation was issued in 1995. The allowable number of cycles was compared to the maximum number of heatup/cooldown cycles projected to the end of the period of extended operation.

Because the projected number of cycles for 60-years of operation (120 cycles) is less than the allowed number of cycles for penetration S-15 (252 cycles), the evaluation concluded that the analysis remains conservative and bounding for the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

4.6.4 AGING OF BORAFLEX

Boraflex is a boron carbide dispersion in an elastomeric silicone that is used in a portion of the RNP spent fuel storage racks as a neutron absorber. The base polymer of Boraflex has been demonstrated to degrade in the borated water environment of the spent fuel pool and under the influence of gamma radiation. Degradation effects include leaching of boron from the borosilicate matrix, and this results in diminished neutron absorption capability of the Boraflex panels. NRC Information Notice (IN) 87-43, “Gaps in Neutron Absorbing Material in High Density Spent Fuel Storage Racks,” IN 93-70, “Degradation of Boraflex Neutron Absorber Coupons,” and IN 95-38, “Degradation of Boraflex Neutron Absorber in Spent Fuel Storage Racks,” identified the issues regarding Boraflex degradation. These concerns resulted in issuance of NRC Generic Letter 96-04, “Boraflex Degradation in Spent Fuel Pool Storage Racks.

Continued monitoring and analyses of the Boraflex degradation was a commitment made by RNP to NRC Generic Letter 96-04. In order to assure that subcriticality margin limits can be maintained for the life of the spent fuel pool racks, the existing Boraflex coupon monitoring program will be continued into the period of extended operation. Spent fuel pool silica levels will continue to be monitored and silica evaluations will continue to be performed in order to confirm the subcriticality margin is maintained through the next evaluation period. These reanalyses 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.

Prior to the period of extended operation, either (1) an analysis will be performed to eliminate credit for the Boraflex sheets in the spent fuel racks in determining K_{eff} for the spent fuel array, or (2) the current Boraflex Monitoring Program will be evaluated against the 10 elements for an acceptable a license renewal aging management program documented in the GALL Report and used to manage the effects of Boraflex degradation through the period of extended operation.

Based on the above, unless the analysis to eliminate credit for the Boraflex demonstrates that it performs no intended functions for license renewal, the planned programmatic activities for continuing periodic evaluation of Boraflex condition will meet the requirements of 10 CFR 54.21(c)(1)(ii); and the ongoing sampling activities will demonstrate the continuing capability of the Boraflex panels to fulfill their intended functions in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

4.6.5 REFERENCES

4.6-1 WCAP-15628, “Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the H.B. Robinson Unit 2 Nuclear Power Plant for the License Renewal Program,” July 2001.

4.6-2 WCAP-15363, Rev. 1, “A Demonstration of Applicability of ASME Code Case N-481 to the Primary Loop Pump Casings of H.B. Robinson Unit 2 for the License Renewal Program,” July 2002.

4.6-3 NUREG-1557, Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal, dated October 1996.

4.6-4 EPRI TR-103842, Class I Structures License Renewal Industry Report, Rev. 1.

APPENDIX A

UPDATED FINAL SAFETY ANALYSIS REPORT SUPPLEMENT

TABLE OF CONTENTS

A.0 UPDATED FINAL SAFETY ANALYSIS REPORT SUPPLEMENT

A.1 INTRODUCTION

This appendix provides the information to be submitted in a Final Safety Analysis Report Supplement as required by 10 CFR 54.21(d) for the RNP License Renewal (LR) Application. The LR Application contains the technical information required by 10 CFR 54.21(a) and (c). Chapter 3 and Appendix B of the RNP LR Application provide descriptions of the programs and activities that manage the effects of aging for the proposed period of extended operation. Chapter 4 of the LR Application contains the evaluations of time–limited aging analyses for the period of extended operation. These LR Application sections have been used to prepare the program and activity descriptions that are contained in the RNP Updated Final Safety Analysis Report (UFSAR) Supplement information in this Appendix. The information presented here will be incorporated into the RNP UFSAR following issuance of the renewed operating license. Upon inclusion of the UFSAR Supplement in the RNP UFSAR, changes to the descriptions of the programs and activities will be made in accordance with 10 CFR 50.59.

The information in this Appendix is divided into two parts. In the first part, changes to the existing pages of the UFSAR are provided to incorporate information or identify areas of change made necessary by the license renewal review. The second part provides new information to be incorporated into the UFSAR. The changes shown in both parts will be incorporated into the UFSAR as part of the FSAR Supplement required by 10 CFR 54.21(d).

A.2 CHANGES TO EXISTING UFSAR INFORMATION

A.2.1.1 UFSAR Chapter 3 Changes

Section 3.5

- Change to be made to Section 3.5.1.1 on Page 3.5.1-1 (delete reference to "in 40 years" from reactor coolant pump flywheel fracture mechanics discussion):

A fracture mechanics evaluation was made on the reactor coolant pump flywheel. This evaluation considered the following assumptions:

a) Maximum tangential stress at an assumed overspeed of 125 percent

b) A crack through the thickness of the flywheel at the bore, and

c) 400 cycles of start up operation ~~in 40 years~~.

Using critical stress intensity factors and crack growth data attained on flywheel material, the critical crack size for failure was greater than 17 in. radially and the crack growth rate was 0.030 in. to 0.060 in. per 1000 cycles.

Section 3.6

- Information to be inserted into the leak-before-break discussion in Section 3.6.1 POSTULATED PIPING FAILURES IN FLUID SYSTEMS INSIDE CONTAINMENT on Page 3.6.1-1:

WCAP-15628 (Reference 3.6.1-1) provides a Leak-Before-Break (LBB) analysis of the H.B. Robinson Unit 2 large bore RCS piping and components which meets modified requirements for LBB required by 10 CFR 50, Appendix A, General Design Criterion 4, and which uses the recommendations and criteria from the NRC Standard Review Plan for LBB evaluations. The analysis uses material property values which account for thermal aging and thermal cycles appropriate for the 60-year operating period projected for license renewal.

- Reference 3.6.1-1 is to be added to the list of references:

3.6.1-1 WCAP-15628, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the H.B. Robinson Unit 2 Nuclear Power Plant for the License Renewal Program," dated July 2001.

Section 3.8

- Revised information to be inserted in Section 3.8.1.4.7 on Page 3.8.1-30:

b)	Final Prestress (Avg)	94.3 ksi ($0.59f_u$) at 60 years
c)	Tendon Relaxation Loss	8.5% of 120 ksi

- Revised information to be inserted at the end of the first paragraph on Page 3.8.1-56 (Section 3.8.1.7.2):

A value of shrinkage of 0.0002 in./in. was used to evaluate the containment tendon prestress value for 60 years of service.

- Information to be inserted at the end of the paragraph discussing 40-year corrosion loss in Section 3.8.5.4.3 on Page 3.8.5-13:

Also, on this basis, the loss of material due to corrosion on piles is considered to be negligible when projected to a service life of 60 years to cover the period of extended operation for license renewal. Thus, foundation pile corrosion is non-significant for the period of extended operation and will not prevent the foundation piles from performing their license renewal intended functions.

Section 3.9

- Delete "(40-year life)" from the fifth sentence in Section 3.9.1 SPECIAL TOPICS FOR MECHANICAL SYSTEMS on Page 3.9.1-1:

All components in the Reactor Coolant System (RCS) were designed to withstand the effects of cyclic loads due to reactor system temperature and pressure changes. These cyclic loads are introduced by normal power changes, reactor trip, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes and their bases are given in Table 3.9.1-1. During unit startup and shutdown, the rates of temperature and pressure changes are limited as indicated in Section 5.3.2. The cycles were estimated for equipment design purposes ~~(40-year life)~~, and were not intended to be an accurate representation of actual transients or actual operating experience. For example, the number of cycles for plant heatup and cooldown at 100°F per hour was selected as a conservative estimate based on an evaluation of the expected requirements. The resulting number, which averages five heatup and cooldown cycles per year, could be increased significantly; however, it was the intent to represent a conservative realistic number rather than the maximum allowed by the design.

Section 3.9 (continued)

- Revise Table 3.9.1-1 as shown:

TABLE 3.9.1-1

THERMAL AND LOADING CYCLES

TRANSIENT CONDITION	DESIGN CYCLES (1)
1. Plant heatup at 100°F per hour	200 ~~(5/yr)~~ (2)
2. Plant cooldown at 100°F per hour	200 ~~(5/yr)~~
3. Plant loading at 15% of full power per minute	29,000 ~~(2/day)~~ (3)
4. Plant unloading at 15% of full power per minute	29,000 ~~(2/day)~~ (3)
5. Step load increase of 20% of full power (but not to exceed full power)	2,000 ~~(1/week)~~
6. Step load decrease of 20% of full power	2,000 ~~(1/week)~~
7. Step load decrease of 95% of full power	80 ~~(2/year)~~
8. Reactor trip	400 ~~(10/year)~~
9. Hydrostatic test (4) pressure 3110 psig temperature 100°F	1 (pre-operational)
10. Hydrostatic test (4) pressure 2485 psig temperature 400°F	40 (post-operational)

11. Steady state fluctuations - the reactor coolant average temperature for purposes of design was assumed to increase and decrease a maximum of 5°F in one minute. The corresponding reactor coolant pressure variation is less than 100 psig. It was assumed that an infinite number of such fluctuations will occur.

(1) Estimated for equipment design purposes ~~(40 year life)~~ and not intended to be an accurate representation of actual transients nor to reflect actual operating experience.

(2) This transient includes pressurizer to 10 percent above the operating pressure.

(3) This value has been limited to 19,000 based on fatigue considerations for pressurizer components.

(4) The steam generator lower assemblies, replaced in 1984, were designed to meet these transient conditions with a variation in the temperatures.

Section 3.9 (continued)

- Revise the second paragraph on Page 3.9.3-5 (Section 3.9.3.3, Steam Generators):

When the plant was designed, no significant corrosion of the Inconel tubing was expected during the lifetime of the plant. The corrosion rate reported in Reference 3.9.3-8 shows "worst case" rates of 15.9 mg/dm^2 in the 2000 hr test under steam generator operating conditions. Conversion of this rate to a 40 year plant life gives a corrosion loss of less than 1.5 x 10^{-3} in., which is insignificant compared to the nominal tube wall thickness of 0.050 in. Likewise, the projected corrosion loss for 60 years of tube operation would be insignificant. Tube plugging is used to isolate a defective tube to preclude primary-to-secondary leakage.

Section 3.11

- Delete “40-year” from the first paragraph of 3.11.5.2 RADIATION ENVIRONMENT, on Page 3.11.5-1:

Safety related systems and components are designed to perform their safety related functions after the normal ~~40 year~~ operational exposure plus one accident exposure. The normal operation exposure is based on the design source terms presented in Section 11.1 and Section 12.2. Post-accident system and component radiation exposures are dependent on equipment location. Source terms and other accident parameters are presented in Section 12.2 and Chapter 15.

A.2.1.2 UFSAR Chapter 5 Changes

Section 5.3

- Information to be inserted into Section 5.3.1.5 Vessel Integrity on Page 5.3.1-3:

Projected 60-year Pressurized Thermal Shock (PTS) reference temperatures were calculated using 60-year fluence values, and these were demonstrated to remain below the PTS screening criteria throughout the 60-year license renewal period.

Projected 60-year Upper Shelf Energy (USE) values were calculated using 60-year fluence values. The projected USE values for plate materials exceed the 42 ft. lb. minimum acceptance criteria provided by WCAP-13587, Rev. 1 (Reference 5.3.1-19) for plate materials. The projected USE values for welds and for forged nozzles exceed the 50 ft. lb. screening criteria established by 10 CFR 50, Appendix G for forgings and weldments.

Furthermore, it was demonstrated that the beltline materials remain more limiting relative to PTS and USE than the inlet/outlet nozzles and adjoining welds.

- Reference 5.3.1-19 is to be added to the list of references:

5.3.1-19 WCAP-13587, Rev. 1, “Reactor Vessel Upper Shelf Energy Bounding Evaluation for Westinghouse Pressurized Water Reactors,” September 1993.

Section 5.3 (continued)

- Revise Note 2 in the fourth paragraph on Page 5.3.1-7 (Section 5.3.1.6, Material Surveillance) as shown below:

The following tabulates the remaining schedule for removal of the Reactor Vessel Materials Surveillance Capsules X, U, V, and W:

Capsule	Calendar Years (See Note 1)
X (See Note 2)	30
U (See Note 3)	40
V (See Note 4)	55
W (See Note 4)	70

Note 1: Capsules are to be removed during the refueling outage immediately after or prior to the end of the designated calendar year.

Note 2: The projected amount of irradiation exposure (fluence) on Capsule X will represent predicted reactor vessel plate and weld fluence values beyond the end of the current license (EOL) of 40 calendar years. Capsule X was removed for analysis in Spring 2001.

Note 3: The projected amount of irradiation exposure (fluence) on Capsule U will represent predicted reactor vessel plate and weld fluence values well beyond EOL to support extension of the current operating license.

Note 4: Capsules V and W will be repositioned after 40 calendar years to accelerated flux positions to provide additional support for license extension.

A.2.1.3 UFSAR Chapter 9 Changes

Section 9.3 Revise Table 3.9.1-1 as shown:

TABLE 9.3.4-1

CHEMICAL AND VOLUME CONTROL SYSTEM PERFORMANCE REQUIREMENTS

Plant design life, years . . . . ~~40~~ 60

Normal seal water supply flow rate, gpm* 24-39

Normal seal water return flow rate, gpm 9

Normal letdown flow rate, gpm** 45-60

Maximum letdown flow rate, gpm . . . , ... 120
Normal charging pump flow (one pump), gpm** 54-69

Normal charging line flow, gpm** 30-45

Maximum rate of boration with one transfer and one charging pump, ppm/min, (from initial RCS concentration of 1800 ppm)*** 23.8

Equivalent cooldown rate to above rate of boration, °F/min(EOC) 11.3

Maximum rate of boron dilution (two charging pumps) ppm/hr (from initial RCS concentration of 2500 ppm) 350

Two-pump rate of boration, using refueling water, ppm/min (from initial RCS concentration of 10 ppm) 4.5

Equivalent cooldown rate to above rate of boration, °F/min 1.1

Temperature of reactor coolant entering system at full power, °F (design) 555.0

Temperature of coolant return to RCS at full power, °F, (design) 493.0

Normal coolant discharge temperature to holdup tanks, °F, (design) 127.0

Amount of 12 percent boric acid solution maintained to meet cold shutdown requirements (EOL, 1 percent shutdown margin, xenon free) shortly after full power operation, gallons (including consideration for one stuck rod) 3,080

A.3 NEW UFSAR SECTION

The following information will be integrated into the UFSAR to document aging management programs and activities credited in the RNP license renewal review and time-limited aging analyses evaluated to demonstrate acceptability during the period of extended operation.

A.3.1 AGING MANAGEMENT PROGRAMS AND ACTIVITIES

The integrated plant assessment and the time-limited aging analyses for license renewal identified existing and new aging management programs necessary to provide reasonable assurance that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis (CLB) for the period of extended operation. This section describes the programs and their implementation activities. RNP will employ the Corrective Action and Document Control Programs to address the program elements of corrective action, confirmation process, and administrative (document) controls for both safety related and non-safety related structures and components that perform an intended function for license renewal.

A.3.1.1 ASME Section XI, Subsection IWB, IWC, and IWD Program

The ASME Section XI, Subsection IWB, IWC, and IWD program consists of periodic volumetric, surface, and/or visual examination of components for assessment, signs of degradation, and corrective actions. The RNP Fourth Ten-Year Interval Program was developed and prepared to meet the ASME Code, Section XI, 1995 Edition, through the 1996 Addenda.

A.3.1.2 Water Chemistry Program

To mitigate aging effects on component surfaces that are exposed to water as process fluid, chemistry programs are used to control water chemistry for impurities (e.g., dissolved oxygen, chloride, fluoride, and sulfate) that accelerate corrosion and contaminants that may cause loss of heat transfer effectiveness due to fouling of heat transfer surfaces. This program relies on monitoring and control of water chemistry to keep peak levels of various contaminants below the system-specific limits. Alternatively, chemical agents, such as corrosion inhibitors, oxygen scavengers, and biocides, may be introduced to prevent certain aging mechanisms. The RNP Water Chemistry Program is based on the current, approved revisions of EPRI PWR Primary Water Chemistry Guidelines and EPRI PWR Secondary Water Chemistry Guidelines as prescribed by NEI 97-06.

A.3.1.3 Reactor Head Closure Studs Program

The Reactor Head Closure Studs Aging Management Program is credited for aging management of the Reactor Head Closure Studs and Stud Components for the aging effects/mechanisms of concern: (1) Loss of Pre-load due to Stress Relaxation, and (2) Loss of Material due to Wear. The closure studs, nuts, and washers are included within the scope of the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program.

A.3.1.4 Steam Generator Tube Integrity Program

The Steam Generator (SG) Tube Integrity Program specifies inspection scope, frequency, and acceptance criteria for the plugging and repair of flawed SG tubes in accordance with the plant technical specifications and the guidance of NEI 97-06. Other components, in addition to SGs tubes, are inspected under this program.

A.3.1.5 Closed-Cycle Cooling Water System Program

The program relies on preventive measures to minimize corrosion by maintaining inhibitors and by performing monitoring consisting of inspection and performance evaluations. Concentrations of corrosion inhibitors are maintained in accordance with the guidelines of EPRI-TR-107396. Performance monitoring of diesel generator jacket water cooling systems is accomplished as part of regularly scheduled operation and testing of that equipment. Inspections and performance monitoring associated with the CCW heat exchangers are addressed in the Open Cycle Cooling Water System.

A.3.1.6 ASME Section XI, Subsection IWF Program

This program consists of periodic visual examination of component supports for signs of degradation. The examination requirements for the IWF portion of the RNP Inservice Inspection (ISI) program are taken from paragraph IWF-2500 (1995 edition), which are essentially the same as specified in Table IWF-2500-1 (1989 edition). The RNP Fourth Ten-Year Interval ISI Program was developed and prepared to meet the ASME Code, Section XI, 1995 Edition, 1996 Addenda, and is subject to the limitations and modifications of 10CFR 50.55a(b)(2), with the exception of design and access provisions and preservice examination requirements.

A.3.1.7 10 CFR Part 50, Appendix J Program

This program consists of inspections of accessible surfaces of containment and monitoring of leakage rates through containment liner/welds, penetrations, fittings, and access openings for detecting degradation of the containment pressure boundary. Corrective actions are taken if leakage rates exceed acceptance criteria. This program is implemented in accordance with 10 CFR Part 50, Appendix J, Regulatory Guide 1.163, and NEI 94-01, Rev. 0.

A.3.1.8 Flux Thimble Eddy Current Inspection Program

The Flux Thimble Eddy Current Inspection Program is a plant-specific program that determines the amount of wear on the flux thimbles and whether the amount of wear expected to occur during the next inspection interval will cause the total amount of wear to exceed the ASME standards specified for the examination. The Flux Thimble Eddy Current Inspection Program was implemented to satisfy NRC Bulletin 88-09 requirements that a thimble tube wear inspection procedure be established and maintained for Westinghouse-supplied reactors that use bottom mounted flux thimble tube instrumentation.

A.3.1.9 Fire Protection Program

The Fire Protection Program manages the aging effects as applicable for fire barriers (fire barrier walls, ceilings, and floors, penetration seals and fire rated doors) and non-water-based fire suppression systems (halon and carbon dioxide). It includes pump testing in the aging management strategy for the diesel engine fire water pump fuel supply line. The Fire Protection Program requires periodic visual inspection of fire barrier penetration seals and periodic visual inspection and functional tests of fire rated doors to ensure that their operability is maintained. The diesel-driven fire pump inspection requirements require that the pump be periodically tested to ensure that the fuel supply line can perform the intended function. The Program also includes periodic inspection and test of halon and carbon dioxide fire suppression systems.

Prior to the period of extended operation, the Fire Protection Program will be enhanced to note that concrete surface inspections performed under structures monitoring procedures will be credited for inspection of fire barrier walls, ceilings, and floors.

A.3.1.10 Boric Acid Corrosion Program

The Boric Acid Corrosion Program manages the aging effects for susceptible materials of structures and components that perform a license renewal intended function and that are exposed to the effects of borated water leaks. The program consists of (1) visual inspection of external surfaces that are potentially exposed to borated water leakage, (2) timely discovery of leak path and removal of the boric acid residues, (3) assessment of the damage, and (4) follow-up inspection for adequacy of corrective actions. This program is implemented in response to NRC Generic Letter 88-05.

Prior to the period of extended operation, the scope of the Boric Acid Corrosion Program will be expanded to (1) ensure the mechanical, structural, and electrical components in scope for license renewal are covered, and (2) identify additional areas in which components may be susceptible to exposure from boric acid (e.g. containment, auxiliary, and spent fuel buildings).

A.3.1.11 Flow-Accelerated Corrosion Program

The program consists of the following actions: (1) conduct appropriate analysis and baseline inspection, (2) determine extent of thinning, (3) replace/repair components, and (4) perform follow up inspections to confirm or quantify and take longer-term corrective actions as necessary. Originally, this program was prepared in response to NRC GL 89-08. The program relies on implementation of EPRI guidelines of NSAC-202L-R2.

Prior to the period of extended operation, the Flow-Accelerated Corrosion Program will be modified to (1) include additional components potentially susceptible to FAC and/or erosion, and (2) specify corrective actions be taken in accordance with the corrective action program when certain acceptance criteria are not met.

A.3.1.12 Bolting Integrity Program

This program consists of guidelines on materials selection, installation procedures, lubricants and sealants, corrosion considerations in the selection and installation of pressure-retaining bolting for nuclear applications, and inspection techniques. The Program relies on other Aging Management Programs for aging management of specific aging effects or specific components. The Section XI, Subsection IWB, IWC and IWD Aging Management Program is credited with inspections of bolting within Section XI boundaries. In addition, the Preventive Maintenance Program performs regular inspections of Reactor Coolant Pump bolting. Loss of Mechanical Closure Integrity Due to Loss of Material Due to Aggressive Chemical Attack (Boric Acid Wastage) has also been identified as a potential aging mechanism for mechanical system bolted closures at RNP. Aging Management Reviews have directly credited the Boric Acid Corrosion Program for management of this aging mechanism. Otherwise, from the standpoint of loss of material due to corrosion, bolting on mechanical components is treated as a subcomponent (i.e., a part of the parent component); and the Systems Monitoring Program is utilized to manage loss of material.

The Bolting Integrity Program is not utilized for aging management of structural bolting. The ASME Section XI, Subsection IWF Program is credited for aging management of all structural bolting associated with Class 1, 2, and 3 component supports, and the Structures Monitoring Program is credited for aging management of all structural bolting other than those associated with Class 1, 2, and 3 components.

Prior to the period of extended operation, Program administrative controls will be modified as necessary to specifically prohibit the use of MoS_2 compounds in high strength bolting applications. Also, a program enhancement will be implemented to inspect, and evaluate bolting on CVC-381 prior to the end of the current operating period to address susceptibility to cracking.

A.3.1.13 Open Cycle Cooling Water System Program

The program includes (1) surveillance and control of biofouling, (2) monitoring, inspecting, and cleaning to verify heat transfer, (3) routine inspection and maintenance program, (4) system walk down inspection, and (5) review of maintenance, operating, and training practices and procedures. The program provides assurance that aging effects for the open-cycle cooling water system can be managed for an extended period of operation. This program was originally developed in response to NRC Generic Letter 89-13.

Prior to the period of extended operation, an activity will be scheduled in the site Preventive Maintenance Program to replace cooling coils in the ECCS room coolers on a prescribed frequency.

A.3.1.14 Inspection of Overhead Heavy Load and Light Load Handling

The program provides guidelines and inspection attributes for monitoring the physical condition of the crane structures within the scope of License Renewal. Rails and girders are visually inspected on a routine basis for degradation. Functional testing requirements are specified. These cranes must also comply with the maintenance rule requirements provided in 10 CFR 50.65.

Administrative controls for Inspection of Overhead Heavy Load and Light Load Handling equipment will be enhanced, prior to the period of extended operation, to (1) include requirements for inspecting the Turbine Gantry Crane in addition to the other cranes that require inspection and (2) note that cranes are to be inspected using the attribute inspection checklist for structures.

A.3.1.15 Fire Water System Program

The Fire Water System Program manages the aging effects of loss of material and flow blockage due to fouling of Fire Protection System water flow paths. To ensure no significant corrosion, MIC, or biofouling has occurred in the water-based fire protection system, periodic full flow flush testing, system performance testing, and inspections during maintenance are conducted. Also, the system is normally maintained at required operating pressure and is monitored such that loss of system pressure would be detected and corrective actions initiated. The program relies on testing of water based fire protection system piping and components in accordance with applicable NFPA commitments.

In addition, this program will be modified to include (1) field service testing of sprinkler heads in accordance with NFPA 25 once prior to, and again 10 years into the period of extended operation, and (2) either full flow testing of portions of fire protection sprinkler systems that are not routinely subject to flow or inspecting/ UT testing of representative portions of fire protection sprinkler systems exposed to water, but not routinely subject to flow. The initial test or inspection would occur prior to the period of extended operation. Results from initial tests or inspections, reflecting 40 years of service, will be used to determine expansion of test/inspection scope and intervals.

A.3.1.16 Buried Piping and Tanks Surveillance Program

The Buried Piping and Tanks Surveillance Program manages the aging effect of loss of material for buried portions of the Fuel Oil System and bottoms of above ground fuel oil tanks. There are no buried tanks within this program. The program includes an impressed current, cathodic protection system. Preventive measures to mitigate corrosion by protecting the external surface of buried piping and components are performed under a different AMP: the Buried Piping and Tanks Inspection Program. The Buried Piping and Tanks Surveillance Program includes surveillance and monitoring of the cathodic protection system based on the guidance of NACE-RP-0169-76.

Prior to the period of extended operation, (1) a review will be performed to ascertain the need to update, as necessary, administrative controls to ensure consistency with NACE Standard RP-0169-96 regarding acceptance criteria for the cathodic protection system, and (2) incorporate additional leak testing provisions for underground piping.

A.3.1.17 Above Ground Carbon Steel Tanks Program

The Above Ground Carbon Steel Tanks Program manages aging effects of loss of material for external surfaces of Fuel Oil System tanks and appurtenances. The program includes preventive measures to mitigate corrosion by protecting the external surface of carbon steel components, per standard industry practice, with protective paint or coating and with sealant or caulking at the interface with soil or concrete. Visual inspections during periodic system walk downs are performed to monitor degradation of the protective paint, coating, caulking, or sealant. For tanks in contact with the ground, the tank sits on a layer of oily sand and a cathodic protection system is provided. These measures assure that degradation is not occurring and that the component intended function will be maintained during the extended period of operation.

Prior to the period of extended operation, the administrative controls for the Program will be revised to indicate that the external surfaces of the fuel oil tanks are to be inspected periodically and to incorporate corrective action requirements.

A.3.1.18 Fuel Oil Chemistry Program

The Fuel Oil Chemistry Program relies on a combination of surveillance and maintenance procedures. Monitoring and controlling fuel oil contamination in accordance with the guidelines of ASTM Standards, and other activities in accordance with the current licensing basis, maintains the fuel oil quality. Corrosion resulting from exposure to fuel oil contaminants such as water and microbiological organisms is minimized by periodic inspection and cleaning of tanks.

As a result of the license renewal review, administrative controls for the Program will be enhanced to (1) improve sampling and de-watering of selected storage tanks, (2) formalize existing practices for draining and filling the Diesel Fuel Oil Storage Tank periodically, (3) formalize bacteria testing for fuel oil samples from various tanks, and (4) incorporate quarterly trending of fuel oil chemistry parameters.

A.3.1.19 Reactor Vessel Surveillance Program

Periodic testing of metallurgical surveillance samples is used to monitor the progress of neutron embrittlement of the reactor pressure vessel as a function of neutron fluence, in accordance with Regulatory Guide 1.99, Rev. 2.

Prior to the period of extended operation, the administrative controls for the Program will be revised to require surveillance test samples to be stored in lieu of optional disposal.

A.3.1.20 Buried Piping and Tanks Inspection Program

The Buried Piping and Tanks Inspection Program manages the aging effect of loss of material for buried components in RNP systems. There are no buried tanks in this program. The program includes preventive measures to mitigate corrosion by protecting the external surface of buried piping and components by use of, for example, coating or wrapping. The program includes visual examinations of buried components when they are made accessible by excavation for maintenance or for some other reason.

Prior to the period of extended operation, the Program will be enhanced to (1) incorporate a requirement to ensure an appropriate as-found pipe coating and material condition inspection is performed whenever buried piping within the scope of this Program is exposed, (2) add precautions to ensure backfill with material that is free of gravel or other sharp or hard material that can damage the coating, (3) add a requirement that coating inspection shall be performed by qualified personnel to assess its condition, and (4) add a requirement that a coating engineer should assist in evaluation of any coating degradation noted during the inspection.

A.3.1.21 ASME Section XI, Subsection IWE Program

The ASME Section XI, Subsection IWE program consists of periodic visual, surface, and volumetric inspection of steel containment components for signs of degradation, assessment of damage, and corrective actions.

Prior to the period of extended operation, the administrative controls for the Program will be enhanced to (1) specify the requirements for conducting reexaminations, and (2) document that repairs meet the specified acceptance standards.

A.3.1.22 ASME Section XI, Subsection IWL Program

The ASME Section XI, Subsection IWL program consists of periodic visual inspection of concrete surfaces of reinforced and prestressed concrete containments for signs of degradation, assessment of damage and corrective actions. The RNP prestressing tendons are grouted in place. Therefore, ASME Section XI Subsection IWL rules regarding unbonded post-tensioning systems, are not applicable.

Prior to the period of extended operation, enhancements will be made to administrative controls to require supervisors to notify Civil/Structural Design Engineering of the location and extent of proposed excavations of foundation concrete and to require Civil/Structural Design Engineering to examine representative sample areas of below-grade concrete when excavated for any reason.

A.3.1.23 Structures Monitoring Program

The program consists of periodic inspection and monitoring the condition of structures and structure component supports to ensure that aging degradation leading to loss of intended functions will be detected and that the extent of degradation can be determined. The inspection criteria are based on ACI 349.3R-96 and ASCE 11-90; as well as INPO Good Practice document 85-033, "Use of System Engineers," NEI 96-03, "Guidelines for Monitoring the Condition of Structures at Nuclear Plants," and NUMARC 93-01, "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

Prior to the period of extended operation, administrative controls for the Program will be enhanced to (1) include buildings and structures, and associated acceptance criteria, in scope for license renewal but outside the scope of the Maintenance Rule, (2) identify interfaces between structures monitoring inspections of concrete surfaces and the Fire Protection Program requirements for barriers, (3) state clearly the boundary definition between systems and structures, (4) revise administrative controls to provide inspection criteria for portions of systems covered by structures monitoring and require a condition report be initiated for all inspection attributes found to be unacceptable, (5) expand system walkdown inspection criteria to include observation of adjacent components, and (6) revise personnel responsibilities to include (a) providing assistance in evaluating structural deficiencies when requested by the Responsible Engineer, (b) inspecting excavated concrete, and (c) notifying Civil/Structural Design Engineering of location and extent of proposed excavations.

A.3.1.24 Dam Inspection Program

The Dam Inspection Program manages the following aging effects for the Lake Robinson Dam and associated concrete and steel structures: (1) loss of material for steel structures, (2) loss of form for earthen structures, and (3) loss of material and change in material properties for concrete structures. Detection of aging effects is accomplished by an independent inspection using the FERC/U.S. Army Corps of Engineers “Recommended Guidelines for Safety Inspection of Dams.”

Prior to the period of extended operation, the following enhancements to the inspection program will be implemented. The system monitoring administrative controls will be revised to (1) identify the “Recommended Guidelines for Safety Inspection of Dams” as the required management program document for the dam, (2) require the responsible system engineer to review the inspection report and initiate corrective actions for any unacceptable attributes identified during the inspection process, and (3) include “Recommended Guidelines for Safety Inspections of Dams” as the applicable inspection guidance in the dam inspection procedure for RNP.

A.3.1.25 Systems Monitoring Program

The Systems Monitoring Program is based upon current plant activities delineated in administrative controls for performing system walkdowns.

Prior to the period of extended operation, administrative controls for the Program will be enhanced to: (1) include aging effects identified in the aging management reviews, (2) identify inspection criteria in checklist form, (3) include guidance for inspecting connected piping/components, (4) require that the extent of degradation to be recorded in the System Walkdown Report and that appropriate corrective action(s) are taken, and (5) add a section specifically addressing corrective actions.

A.3.1.26 Preventive Maintenance Program

The Preventive Maintenance (PM) Program assures that various aging effects are managed for a wide range of components. PM activities include periodic component replacement, inspections, and testing and may be used to manage aging effects and mechanisms.

Prior to the period of extended operation, administrative controls for the Program will be enhanced to: (1) include aging effects/mechanisms identified in the aging management reviews, and (2) incorporate specific aging management activities identified in the aging management reviews into the Program.

A.3.1.27 Metal Fatigue of Reactor Coolant Pressure Boundary (Fatigue Monitoring Program)

The Metal Fatigue of the Reactor Coolant Pressure Boundary (Fatigue Monitoring Program) monitors the bounding primary system transient cycles to assure that transient limits are not exceeded for in scope components. These monitoring results are considered bounding for most reactor coolant pressure boundary components and various secondary side components. The acceptance criteria for the Fatigue Monitoring Program are the transient limits, which are based upon the fatigue evaluations for RCS components. By maintaining the actual transient counts below the transient limits, the fatigue usage is kept below the design code limit.

As a result of the license renewal review, the plant load/unload transient limit will be reduced to provide the margin needed for consideration of reactor water environmental effects. This will be completed prior to the period of extended operation.

A.3.1.28 Nickel-Alloy Nozzles and Penetrations Program

The program includes (a) primary water stress corrosion cracking (PWSCC) susceptibility assessment to identify susceptible components, (b) monitoring and control of reactor coolant water chemistry to mitigate PWSCC, and (c) inservice inspection ISI of reactor vessel head penetrations to monitor PWSCC and its effect on the intended function of the component. For susceptible penetrations and locations, the program includes an industry wide, integrated, long-term inspection program based on the industry responses to NRC Generic Letter (GL) 97-01.

Prior to the period of extended operation, the Nickel-Alloy Nozzles And Penetrations Program will incorporate the following: (1) CP&L will perform evaluation of indications under the ASME Section XI program, (2) CP&L will perform corrective actions for augmented inspections to repair and replacement procedures equivalent to those requirements in ASME Section XI, (3) CP&L will maintain its involvement in industry initiatives (such as the Westinghouse Owners Group and the EPRI Materials Reliability Project) during the period of extended operation.

A.3.1.29 Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program

The Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program is credited for aging management of CASS components within Class 1 boundaries of the Reactor Coolant System and connected systems at RNP. The aging effect/mechanism of concern is loss of fracture toughness due to thermal embrittlement of cast austenitic stainless steel.

Flaw tolerance evaluations for RCP casings and primary loop CASS components have been done on the basis of fracture toughness methodology approved by the NRC.

Consistent with NRC guidance, the RNP Program does not include additional inspections of pump casings, valve bodies, or piping.

A.3.1.30 PWR Vessel Internals Program

The PWR Vessel Internals Program includes (a) participation in industry programs and initiatives to determine appropriate inspection techniques for use in managing aging effects, and (b) monitoring and control of reactor coolant water chemistry in accordance with the Water Chemistry Program to ensure the long-term integrity and safe operation of pressurized water reactor vessel internal components. This is a new program that will incorporate the following commitments (1) To address change in dimensions due to void swelling, RNP will continue to participate in industry programs to investigate this aging effect and determine the appropriate AMP, (2) To address baffle and former assembly issues, RNP will continue to participate in industry programs and will implement appropriate program enhancements to manage the aging effects associated with the Baffle and Former Assembly, (3) As WOG and EPRI Materials Reliability Project (MRP) research projects are completed, RNP will evaluate the results and factor them into the PWR Vessel Internals Program. The expected results include identification of components which are the most limiting and most susceptible and identification of appropriate inspection techniques, (4) RNP will implement an augmented inspection during the license renewal term. Augmented inspections, based on required program enhancements, will become part of the ASME Section XI program. Corrective actions for augmented inspections will be developed using repair and replacement procedures equivalent to those requirements in ASME Section XI.

A.3.1.31 One-Time Inspection Program

Special inspections of components within the scope of license renewal will be performed in accordance with the One-Time Inspection Program. The Program is used to verify the effectiveness of the aging management activities and to determine the present condition of components. One-Time Inspection Program activities consist of inspecting (1) the CCW heat exchanger tubes, (2) miscellaneous piping protected by the Water Chemistry Program, (3) small bore RCS and connected piping, (4) Emergency Diesel Generator exhaust silencers, (5) Reactor Vessel Supports, and (6) containment liner plate and moisture barrier.

A.3.1.32 Selective Leaching of Materials Program

The program includes mechanical testing to determine the properties of selected components that may be susceptible to selective leaching to determine whether loss of materials is occurring and whether the process will affect the ability of the components to perform their intended function for the period of extended operation. Mechanical means include resonance when struck by another object, scraping, or chipping. These techniques provide a valid method of identification and subsequent management of selective leaching.

A.3.1.33 Non-EQ Insulated Cables and Connections Program

The Non-EQ Insulated Cables and Connections Program is credited for aging management of cables and connections not included in the RNP EQ Program. The non-EQ insulated cables and connections managed by this program include those used for power, instrumentation, control and communication, including cables sensitive to reduction in insulation resistance such as radiation monitoring and nuclear instrumentation. The program involves periodic, visual inspections of accessible cables and connections installed in adverse localized environments to detect embrittlement, cracking, melting, discoloration or swelling that could lead to reduced insulation resistance or electrical failure. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service condition for the electrical cable or connection.

A.3.2 EVALUATION OF TIME LIMITED AGING ANALYSES

A.3.2.1 Reactor Vessel Neutron Embrittlement

A.3.2.1.1 Pressurized Thermal Shock

10 CFR 50.61 requires the reference temperature (RT_{PTS}) for reactor vessel beltline materials be less than the “PTS screening criteria” at the expiration date of the operating license unless otherwise approved by the NRC. The screening criteria limit the amount that the material reference temperature, RT_{PTS}, may increase following neutron irradiation. In support of license renewal, the neutron fluence values for the reactor vessel were projected to 60-years and were used as inputs to the 60-year PTS calculation prepared for license renewal. The calculated RT_{PTS} temperatures for reactor vessel beltline materials, including axial welds, circumferential welds and plates, have been demonstrated to remain below the applicable PTS screening criteria throughout the 60-year license renewal period. Therefore the TLAA for Pressurized Thermal Shock has been projected to the end of the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

Conservative 60-year RT_{PTS} values also were calculated for the reactor vessel inlet and outlet nozzles and welds. The highest 60-year RT_{PTS} reference temperature for the nozzles was below the screening criteria for plates, forgings, and axial welds. Therefore, the nozzles and nozzle welds have been shown to meet the PTS criteria for 60 years and not to be the limiting components, and the inlet and outlet nozzles and welds need not be added to the reactor vessel surveillance program.

A.3.2.1.2 Upper Shelf Energy

10 CFR Part 50, Appendix G, paragraph IV.A.1, requires that reactor vessel beltline materials have a Charpy upper-shelf energy (USE) of no less than 50 ft-lb (68 J) throughout the life of the reactor vessel unless otherwise approved by the NRC. In support of license renewal, the neutron fluence values for the reactor vessel were projected to 60-years and were used as inputs to the 60-year USE calculations prepared for license renewal. In accordance with these calculations, the projected 60-year USE values for reactor beltline welds (both axial and circumferential) and for nozzle forgings and nozzle welds were shown to be above the minimum USE screening criteria. For reactor vessel plate materials, the 60-year USE value remained above the minimum USE acceptance criterion that provides an equivalent margin of safety for RNP vessel plates.

Based on the foregoing discussion, the TLAA for reactor pressure vessel USE has been projected to the end of the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

A.3.2.2 Metal Fatigue

A.3.2.2.1 Fatigue Analysis (Design)

The reactor vessel, pressurizer, steam generators (primary side), and reactor coolant pumps have been designed to ASME Section III, Class A (now Class 1), requirements which include analyses to address fatigue and establish limits such that initiation of fatigue cracks is precluded. The fatigue analyses are contained in the stress reports for each of these components. Fatigue usage factors for critical locations in the NSSS components were determined using design cycles specified during the design process. These design cycles were intended to be conservative and bounding for all foreseeable plant operational conditions. The design cycles were utilized in the design stress reports for various NSSS components satisfying ASME fatigue usage design requirements.

Additional explicit fatigue analyses have been prepared since original design to address (1) thermal stratification of the pressurizer surge line, (2) reactor vessel internals holddown spring and alignment pins, (3) insurge/outsurge flow between the pressurizer and surge line, (4) containment bellows, and (5) thermal cycling of auxiliary feedwater to main feedwater connections. These analyses also determined fatigue usage factors using design cycles specified during the design process.

In addition, the RNP Class 1 piping was designed in accordance with the design rules from USAS B31.1 Power Piping Code – 1965 Edition. These rules provide an implicit fatigue design basis because cyclic loading is required to be considered when applying the code rules but explicit fatigue analyses are not required. Most RNP piping has been designed in accordance with USAS B31.1 rules. Auxiliary heat exchangers at RNP were designed in accordance with Westinghouse specifications and ASME Section III, Class C, or ASME, Section VIII, requirements that have fatigue design rules essentially identical to B31.1 design rules.

Experience has shown actual plant operation often is very conservatively represented by the assumed design cycle count. The use of actual operating history data allows the quantification of these conservatisms in the existing fatigue analyses. To demonstrate that the fatigue analyses remain valid for the period of extended operation, the operational cycle set for plant components was assembled. The actual frequency of occurrence for the design cycles was determined and compared to the design cycle set. The severity of the actual plant transients, e.g., partial cycles, was compared to the severity of the assumed design cycles. This comparison was performed in order to demonstrate that on an event-by-event basis, the design cycle profiles envelope actual plant operation. The result of this evaluation was the set of adjusted cumulative transient cycle counts. These data were used as a basis for 60-year projections, along with trending data from the past operational periods. Some projected cycle counts were adjusted to account for the decrease in number of cycles experienced currently versus the high number of cycles experiences during early years of plant operation. The resulting 60-year transient projections were then compared to the 40-year design transient limits, to determine the remaining margin above the projected values.

The evaluation concluded that, with one exception, the existing design cycles and cycle frequencies are conservative and bounding for the period of extended operation; and, therefore, the TLAAs for fatigue remain valid for the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(i). The exception involves a fatigue evaluation performed for feedwater line connection reinforcement pads. The fatigue analysis was based in part upon the number of surveillance tests performed during refueling cycles. Since the number of refueling cycles is increased during the license renewal period, this fatigue analysis requires projection for 60 years. A more detailed analysis of the connections between the auxiliary feedwater and main feedwater lines will be performed prior to the period of extended operation in order to justify the extension of this TLAA through the period of extended operation.

A.3.2.2.2 Environmentally Assisted Fatigue

Plant-specific environmental fatigue calculations were performed for a sample of high fatigue locations to demonstrate that adequate conservatism exists within the fatigue TLAA’s to account for reactor water environmental effects. These sample locations include the seven specified in NUREG/CR-6260 for older-vintage Westinghouse plants. Environmentally Assisted Fatigue (EAF) relationships developed in NUREG/CR-6583, for carbon and low alloy steels, and NUREG/CR-5704, for stainless steels, were used. Since the pressurizer surge line was not shown to have an EAF-adjusted CUF value below 1.0, an increased sample of high fatigue locations was evaluated for environmental fatigue, including seven additional locations in the pressurizer for which plant-specific fatigue analyses exist. All sample locations were shown to have an environmentally-adjusted CUF value below 1.0, indicating acceptability, except for the pressurizer surge line and the stainless steel pressurizer surge nozzle safe end and RCS hot leg pressurizer surge line nozzle.

Therefore, in accordance with 10 CFR 54.21(c)(1)(iii), an aging management program will be used to manage the effects of fatigue of the pressurizer surge line components. This will be accomplished through in-service inspections performed in accordance with the ASME Section XI, Subsection IWB, IWC, and IWD Program. If unacceptable indications are identified, further evaluation, repair or replacement will be performed as required by ASME Section XI.

A.3.2.2.3 Reactor Vessel Underclad Cracking

A fracture mechanics analyses completed in 1971 concluded that fatigue growth of potential underclad flaws in reactor vessel base metal over a 40-year period would be insignificant and the structural integrity of the reactor vessels had not been compromised for their intended use for 40-year period. The underclad cracking analysis has been updated by a topical report, WCAP-15338 [Reference A.3.2-1], to justify operation for 60 years. The topical report results indicated that an assumed flaw, assumed to grow under the influence of transient cycles for a period of 60 years, would remain below the most critical allowable flaw depth. Since the estimated final flaw depth is smaller than the allowable flaw depth, it was concluded that a reactor vessel with postulated underclad cracks would be acceptable for operation for 60 years. An NRC Safety Evaluation [Reference A.3.2-2] concludes that of WCAP-15338 is acceptable for referencing as a topical report, and RNP has verified that the report is applicable to the RNP reactor vessel. Therefore, the TLAA for reactor vessel underclad cracking has been projected to the end of the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

A.3.2.2.4 Containment Penetration Bellows Fatigue

Fatigue TLAA's were identified for certain the flexible bellows assemblies used for hot piping penetrations through containment. The current analysis assumes that the bellows assemblies will experience a number of transient cycles based on a 40-year life. The significant thermal transients that result in flexure of the hot pipe penetration bellows are those involving a full-range temperature change in the piping system. These are the plant heatup and cooldown cycles. As discussed above, an evaluation of operational transients shows that the number of heatup and cooldown cycles included within the 40-year design basis remain conservative for 60 years of operation. Therefore, the number of cycles assumed in the analysis of the penetration bellows exceed the heatup cooldown cycles applicable for 60 years of operation, and the bellows fatigue calculations remain valid through the license renewal period. Thus, the TLAA for containment bellows fatigue are conservative and bounding for the period of extended operation; and, therefore, the TLAAs for fatigue remain valid for the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

A.3.2.2.5 Crane Fatigue

Load lifting cranes within the scope of license renewal, have service life limitations based upon the number of load cycles they can safely withstand.

The polar crane and spent fuel cask crane have been identified as having 40 year TLAAs for structural fatigue considerations. In support of license renewal, the cranes have been evaluated for structural fatigue considerations for 60-year service period. The evaluations are summarized in the following paragraphs.

POLAR CRANE

The RNP polar crane is a low-cycle lifting device. While the plant is in operation, the polar crane is not operated. The polar crane is only operated during Refueling Outages; therefore, the total number of lift cycles is directly dependent on the number of Refueling Outages. The total number of Refueling Outages for 60 years of operation has been established as 40. The total number of upper and mid-range lifts has been determined to be 110 per outage; thus, for a total of 40 outages, the number of lift cycles is 4400. This is less than the 10,000 permissible lift cycles for this crane and is therefore acceptable. Based on the foregoing, the RNP Polar Crane has been evaluated for fatigue for 60 years, and the Polar Crane TLAA has been successfully projected for 60 years in accordance with 10 CFR 54.21(c)(1)(ii).

SPENT FUEL CASK CRANE

The number of lift cycles originally projected for the Spent Fuel Cask Crane during a 40-year period was 2,500. This can be multiplied by a factor of 1.5 to determine the number of cycles for 60-year life. Therefore, number of load cycles projected for 60 years is 3,750. This is less than the 20,000 permissible cycles and is therefore acceptable. Therefore, the RNP Spent Fuel Cask Crane has been evaluated for fatigue for 60 years, and the Spent Fuel Cask Crane TLAA has been successfully projected for 60 years in accordance with 10 CFR 54.21(c)(1)(ii).

A.3.2.3 Environmental Qualification

The thermal, radiation, and wear cycle aging analyses, as applicable, of plant electrical and I&C components required to meet 10 CFR 50.49 have been identified as TLAAs. Equipment in the RNP Environmental Qualification (EQ) Program has been evaluated to determine if existing environmental qualification aging analyses can be projected to the end of the period of extended operation. Qualification that extends into the period of extended operation is addressed in the same manner as qualification for the current operating term. Should an analysis fail to justify a qualified life to 60-years, the equipment or affected component will be replaced prior to exceeding its qualified life in accordance with the existing provisions of the EQ Program.

Age-related service conditions that are applicable to environmentally qualified equipment were evaluated for the period of extended operation to verify that the current analyses remain bounding. The evaluations considered thermal, radiation, and wear cycle aging effects as applicable.

Therefore, the analyses associated with environmental qualification of electrical equipment have been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii), or the effects of aging will be adequately managed by periodic replacement of aged components in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

A.3.2.4 Containment Tendon Loss of Prestress

The vertical prestressing tendons are used to impart compressive forces in the prestressed concrete containment to resist the internal pressure inside the containment that would be generated in the event of an accident. The prestressing forces generated by the tendons diminish over time due to losses in prestressing forces in the tendons and in the surrounding concrete. The prestressing force evaluation has been determined to remain valid to the end of the period of extended operation, and the final projected preload will remain above the minimum required preload for containment tendons to the end of this period. Consequently, the post-tensioning system will continue to perform its intended function throughout the period of extended operation.

Therefore, the analyses associated with containment tendon loss of prestress have been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

A.3.2.5 Thermal Aging Embrittlement

A.3.2.5.1 Leak-Before-Break Analysis of Reactor Coolant System Piping

WCAP-15628 [Reference A.3.2-3] is a new leak-before-break (LBB) calculation applicable to RNP large bore Reactor Coolant System (RCS) piping and components that includes allowances for reduction of fracture toughness of cast austenitic stainless steel due to thermal embrittlement during a 60-year operating period. The new analysis meets the requirements for LBB required by 10 CFR 50, Appendix A, General Design Criterion 4, and uses the recommendations and criteria from the NRC Standard Review Plan for LBB evaluations. The new analysis uses the 40-year design basis thermal transients as input for the fracture mechanics analyses. These transients have been shown to be conservative for the 60-year operating period. Therefore, the RCS primary loop piping Leak-Before-Break analysis has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

A.3.2.5.2 Fracture Mechanics Analysis for Reactor Coolant Pump

WCAP-15363, Rev. 1 [Reference A.3.2-4] is a new analysis that compares plant-specific loadings and materials to generic loadings and materials used in an earlier evaluation to support use of ASME Code Case N-481 for Westinghouse Model 93 pumps at RNP. WCAP-15363, Rev. 1, includes allowances for a reduction of fracture toughness of cast austenitic stainless steel during the 60-year operation period, but uses the limiting transients from the 40-year design transient set. This is acceptable because the 40-year design transients have been shown to be conservative for 60 years of plant operation. WCAP-15363, Rev. 1, uses plant-specific material property data instead of generic materials data and demonstrates that margin requirements for leakage and crack stability have been met. The new analysis permits the use of the surface examination of pump casings in lieu of volumetric examination in accordance with the Code Case throughout the period of extended operation. Therefore, the ASME Code Case N-481 analysis has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

A.3.2.6 Foundation Pile Corrosion

Corrosion of Class 1 structure foundation piles was identified as a TLAA based on the evaluation of the piles for a 40-year corrosion loss. The original analysis determined corrosion losses would be negligible based on measured soil resistivity values that are so high the possibility of active corrosion is minimal. Industry data from NUREG-1557, “Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal” and EPRI TR-103842, “Class I Structures License Renewal Industry Report” confirm that steel piles driven in undisturbed soils have been unaffected by corrosion and those driven in disturbed soil experience minor to moderate corrosion to a small area of metal.

A reanalysis of foundation pile corrosion for license renewal determined that corrosion losses would remain non-significant for the period of extended operation and will not prevent the foundation piles from performing their license renewal intended functions. This conclusion is consistent with the recommendations and findings of NUREG-1557 and EPRI Report TR-103842 and is in accordance the estimated corrosion losses developed in the original analysis. Therefore, the foundation pile corrosion analysis results have been projected to the end of the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

A.3.2.7 Elimination of Containment Penetration Coolers

In 1995, an evaluation was performed to justify eliminating the need for cooling water flow to the hot pipe containment penetration coolers to the maximum extent possible. As part of this effort, insulation was credited to reduce the temperature of the concrete surrounding the hot pipe penetrations. The performance requirement for the hot pipe penetrations was to maintain the surrounding concrete temperature below 200^{o}F under normal operating conditions and other long term conditions.

Residual Heat Removal (RHR) system penetration S-15 did not require cooling water to be maintained because the concrete temperature around S-15 only exceeded 200^{o}F during short duration transients and the temperature then was less than 350^{o}F. In addition, the steady-state temperature without cooling water and continuous RHR flow at 380^{o}F results in the temperature of the surrounding concrete of approximately 210^{o}F.

The analysis of concrete temperature determined that the allowable number of cycles of heatup and cooldown, at 40 hours or less per cycle, was 252 cycles. This is the total number of heatup/cooldown cycles the concrete surrounding the S-15 RHR penetration could experience temperatures greater than 200^{o}F over the balance of plant life figured from the year 1995. The balance of plant life was projected as 16 years (out of 40 years total plant life) when this calculation was issued in 1995. The allowable number of cycles was compared to the maximum number of heatup/cooldown cycles projected to the end of the period of extended operation. Because the projected number of cycles for 60-years of operation (120 cycles) is less than the allowed number of cycles for penetration S-15 (252 cycles), the evaluation concluded that the analysis remains conservative and bounding for the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

A.3.2.8 Aging of Boraflex in Spent Fuel Pool

The neutron absorber, Boraflex, in the spent fuel storage racks may experience degradation over time. Degradation includes leaching of boron from the borosilicate matrix, and this results in diminished neutron absorption capability of the Boraflex panels. Continued monitoring and analyses of the Boraflex degradation was a commitment in response to NRC Generic Letter 96-04. In order to assure that subcriticality margin limits can be maintained for the life of the spent fuel pool racks, the existing Boraflex coupon monitoring program will be continued into the period of extended operation. Spent fuel pool silica levels will continue to be monitored and silica evaluations will continue to be performed in order to confirm the subcriticality margin is maintained through the next evaluation period. These reanalyses 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. Prior to the period of extended operation, the current Boraflex monitoring program will be evaluated against the requirements for a license renewal aging management program, and the results of the evaluation will be documented in the UFSAR.

A.4 REFERENCES

A.3.2-1 WCAP-15338, A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants, March, 2000.

A.3.2-2 USNRC Safety Evaluation of WCAP-15338, A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants, October 15, 2001.

A.3.2-3 WCAP-15628, Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the H.B. Robinson Unit 2 Nuclear Power Plant for the License Renewal Program, July 2001.

A.3.2-4 WCAP-15363, Rev. 1, A Demonstration of Applicability of ASME Code Case N-481 to the Primary Loop Pump Casings of H.B. Robinson Unit 2 for the License Renewal Program, July 2001.

APPENDIX B

AGING MANAGEMENT PROGRAMS

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B.0 AGING MANAGEMENT PROGRAMS

B.1 INTRODUCTION

For those structures and components that are identified as being subject to an aging management review, 10 CFR 54.21(a)(3) requires demonstration that the effects of aging will be adequately managed so that their intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.

The NRC and the industry identified ten (10) program elements that would be useful in describing an aging management program and then demonstrating its effectiveness. These program elements are described in Appendix A.1, Section A.1.2.3 of the Standard Review Plan for License Renewal (SRP-LR) [Reference B-1]. The "Generic Aging Lessons Learned" (GALL) report [Reference B-2], uses these program elements to describe acceptable aging management programs.

RNP used the above-described elements in the review of RNP program activities to demonstrate that the effects of aging will be adequately managed in accordance with the License Renewal Rule.

There are two types of programs that may be evaluated this way. The first is a program that is being directly compared to one in the GALL Report. The second is a site-specific program.

For RNP aging management programs being directly evaluated against a defined program from the GALL Report, Section XI, the GALL Report criteria or activities delineated in each of the 10 elements are evaluated, and a conclusion is reached concerning consistency for the RNP activity with the GALL recommended program element. After all elements are evaluated, a demonstration of overall program effectiveness is made. Any program enhancements that are required are documented. Finally, an overall determination is made as to consistency with the program description in the GALL report.

For site-specific programs, an evaluation is performed to document how RNP meets the 10 generic program elements in the SRP-LR, identifies required program enhancements, and determines overall program effectiveness.

The RNP Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Section A.2 of the SRP-LR. A description of the RNP Quality Assurance Program is provided in UFSAR, Section 17.3. The Quality Assurance Program addresses three of the aging management program elements: (1) Corrective Action, (2) Confirmation Process (which is an integral part of the Corrective Action), and (3) Administrative Controls. The Quality Assurance Program applies these program elements via existing Corrective Action and Document Control

Programs. RNP will employ the Corrective Action and Document Control Programs to address the program elements of corrective action, confirmation process, and administrative (document) controls for both safety related and non-safety related structures and components that perform an intended function for license renewal.

The RNP programs described herein are credited for managing the effects of aging. These programs include existing programs, existing programs that have been enhanced to deal with specific aging effects, and new programs not currently defined in RNP administrative controls. The programs 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 of 10 CFR 54.21(a)(3). Along with the technical information contained in the body of this application, this appendix is intended to allow the NRC to make the finding required by 10 CFR 54.29(a)(1).

The results of RNP Aging Management Program evaluations are documented in the following sections for existing programs, existing programs that have been enhanced to deal with specific aging effects, and new programs. Table B-1 provides a correlation between the programs evaluated in the GALL Report and the RNP programs credited for aging management.

TABLE B-1 - CORRELATION OF AGING MANAGEMENT PROGRAMS BETWEEN THE GALL REPORT AND RNP

GALL Program		RNP Program	App. B Subsection
GALL Report Chapter X			
X.M1	Metal Fatigue of Reactor Coolant Pressure Boundary	Metal Fatigue of Reactor Coolant Pressure Boundary (Fatigue Monitoring Program)	B.3.19
X.S1	Concrete Containment Tendon Prestress	Not credited for aging management	
X.E1	Environmental Qualification (EQ) of Electrical Components	RNP Environmental Qualification Program (This is a CLB Program. EQ is a TLAA; refer to Section 4.4.)	
GALL Report Chapter XI			
XI.M1	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program	B.2.1
XI.M2	Water Chemistry	Water Chemistry Program	B.2.2
XI.M3	Reactor Head Closure Studs	Reactor Head Closure Studs Program	B.2.3
XI.M4	BWR Vessel ID Attachment Welds	Not applicable. RNP is a PWR	
XI.M5	BWR Feedwater Nozzle	Not applicable. RNP is a PWR	
XI.M6	BWR Control Rod Drive Return Line Nozzle	Not applicable. RNP is a PWR	
XI.M7	BWR Stress Corrosion Cracking	Not applicable. RNP is a PWR	
XI.M8	BWR Penetrations	Not applicable. RNP is a PWR	
XI.M9	BWR Vessel Internals	Not applicable. RNP is a PWR	
XI.M10	Boric Acid Corrosion	Boric Acid Corrosion Program	B.3.2
XI.M11	Nickel-Alloy Nozzles and Penetrations	Nickel-Alloy Nozzles and Penetrations Program	B.4.1
XI.M12	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program	B.4.2
XI.M13	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)	Not credited for aging management	
XI.M14	Loose Part Monitoring	Not credited for aging management	
XI.M15	Neutron Noise Monitoring	Not credited for aging management	
XI.M16	PWR Vessel Internals	PWR Vessel Internals Program	B.4.3
XI.M17	Flow-Accelerated Corrosion	Flow-Accelerated Corrosion Program	B.3.3
XI.M18	Bolting Integrity	Bolting Integrity Program	B.3.4
XI.M19	Steam Generator Tube Integrity	Steam Generator Tube Integrity Program	B.2.4
XI.M20	Open-Cycle Cooling Water System	Open-Cycle Cooling Water System Program	B.3.5
XI.M21	Closed-Cycle Cooling Water System	Closed-Cycle Cooling Water System Program	B.2.5
XI.M22	Boraflex Monitoring	Program to be evaluated; refer to Subsection 4.6.4.	

TABLE B-1 (continued) - CORRELATION OF AGING MANAGEMENT PROGRAMS BETWEEN THE GALL REPORT AND RNP

GALL Program		RNP Program	App. B Subsection
GALL Chapter XI (continued)			
XI.M23	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program	B.3.6
XI.M24	Compressed Air Monitoring	Not credited for aging management	
XI.M25	BWR Reactor Water Cleanup System	Not applicable. RNP is a PWR	
XI.M26	Fire Protection	Fire Protection Program	B.3.1
XI.M27	Fire Water System	Fire Water System Program	B.3.7
XI.M28	Buried Piping and Tanks Surveillance	Buried Piping and Tanks Surveillance Program	B.3.8
XI.M29	Aboveground Carbon Steel Tanks	Aboveground Carbon Steel Tanks Program	B.3.9
XI.M30	Fuel Oil Chemistry	Fuel Oil Chemistry Program	B.3.10
XI.M31	Reactor Vessel Surveillance	Reactor Vessel Surveillance Program	B.3.11
XI.M32	One-Time Inspection	One-Time Inspection Program	B.4.4
XI.M33	Selective Leaching of Materials	Selective Leaching of Materials Program	B.4.5
XI.M34	Buried Piping and Tanks Inspection	Buried Piping and Tanks Inspection Program	B.3.12
XI.S1	ASME Section XI, Subsection IWE	ASME Section XI, Subsection IWE Program	B.3.13
XI.S2	ASME Section XI, Subsection IWL	ASME Section XI, Subsection IWL Program	B.3.14
XI.S3	ASME Section XI, Subsection IWF	ASME Section XI, Subsection IWF Program	B.2.6
XI.S4	10 CFR Part 50, Appendix J	10 CFR Part 50, Appendix J Program	B.2.7
XI.S5	Masonry Wall Program	Not credited for aging management	
XI.S6	Structures Monitoring Program	Structures Monitoring Program	B.3.15
XI.S7	RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants	Not credited for aging management	
XI.S8	Protective Coating Monitoring and Maintenance Program	Not credited for aging management	
XI.E1	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Non-EQ Insulated Cables and Connections Program	B.4.6
XI.E2	Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	Non-EQ Insulated Cables and Connections Program	B.4.6
XI.E3	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Not credited for aging management	

TABLE B-1 (continued) - CORRELATION OF AGING MANAGEMENT PROGRAMS BETWEEN THE GALL REPORT AND RNP

GALL Program	RNP Program	App. B Subsection
GALL Chapter XI (continued)		
None	Flux Thimble Eddy Current Inspection Program	B.2.8
None	Dam Inspection Program	B.3.16
None	Systems Monitoring Program	B.3.17
None	Preventive Maintenance Program	B.3.18

B.2 EXISTING AGING MANAGEMENT PROGRAMS

B.2.1 ASME SECTION XI, SUBSECTION IWB, IWC AND IWD PROGRAM

The ASME Section XI, Subsection IWB, IWC and IWD Program is credited for aging management of selected components in several systems at RNP.

The aging effects/mechanisms of concern are as follows:

- Cracking due to SCC
- Loss of Pre-load due to Stress Relaxation
- Loss of Pre-load due to Irradiation Creep
- Loss of Material due to Wear
- Loss of Material due to Aggressive Chemical Attack
- Loss of Material due to Crevice Corrosion
- Loss of Material due to General Corrosion
- Loss of Material due to Pitting Corrosion
- Reduction of Fracture Toughness due to Thermal Embrittlement

The RNP ISI program also ensures early detection of cracking due to thermal fatigue in the pressurizer surge line through the routine performance of volumetric examination of the surge line welds as stated in the Thermal Fatigue TLAA.

The RNP Fourth Ten-Year Interval ASME Section XI Inservice Inspection Program began on February 19, 2002; and the Subsections IWB, IWC and IWD portion of the program includes all portions of the Class 1, 2, and 3 systems. The Fourth Ten-Year Interval Program was developed and prepared to meet the ASME Code, Section XI, 1995 Edition, 1996 Addenda.

Operating Experience

The RNP Fourth Ten-Year Interval ASME Section XI Inservice Inspection Program is implemented and maintained in accordance with the general requirements for engineering programs. This provides assurance that the programs:

- Are effectively implemented to meet regulatory, process, and procedure requirements, including periodic reviews
- Qualified personnel are assigned as program managers, and are given authority and responsibility to implement the program
- Adequate resources are committed to program activities
- Are managed in accordance with plant administrative controls

The program is continually upgraded based on industry experience and research. This aging management program has provided an effective means of ensuring the pressure integrity of the RNP Class 1, 2 and 3 systems. In addition to industry experience, plant-operating experiences are shared between CP&L and FPC sites through regular peer group meetings.

The ASME Section XI, Subsection IWB, IWC and IWD Program is subject to ongoing NRC inspections and self assessments; when weaknesses are noted actions are taken under the corrective action program (CAP) to initiate program improvements. The CAP has been effective in ensuring that the ISI program is continually improving.

Conclusion

Based on the above, the ASME Section XI, Subsection IWB, IWC and IWD Program is consistent with GALL Section XI.M1, ASME Section XI, Subsection IWB, IWC and IWD; and the continued implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

In the evaluation of the ASME Section XI, Subsection IWB, IWC and IWD Program against the program elements of the GALL Report, exceptions to Code requirements that have been granted by approved relief requests were not considered to be exceptions to the GALL criteria.

B.2.2 WATER CHEMISTRY PROGRAM

The Water Chemistry Program is credited for aging management of selected components in systems and structures at RNP.

The aging effects/mechanisms of concern are as follows:

- Loss of Material due to Crevice Corrosion
- Loss of Material due to Erosion
- Loss of Material due to Fretting
- Loss of Material due to Galvanic Corrosion
- Loss of Material due to General Corrosion
- Loss of Material due to Pitting Corrosion
- Cracking due to SCC
- Cracking due to IASCC
- Loss of Heat Transfer Effectiveness due to Fouling of Heat Transfer Surfaces

In various places in GALL Report, it is stated "The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI. M32, 'One-Time Inspection,' for an acceptable verification program." RNP has developed a new program to address one-time inspections to demonstrate the adequacy of water chemistry controls. Refer to the One-Time Inspection Program in Section B.4 below.

Operating Experience

A review of condition reports indicates that RNP had several applicable to the Chemistry Program. Most of these involved limits for parameters that would not affect component intended functions for license renewal or are considered suggestions for program improvements. In those instances where a chemistry action level was exceeded, prompt corrective actions were taken to re-establish proper chemistry.

NRC Inspection Reports of plant water chemistry were reviewed. For these inspections, no violations or deviations were identified. There was a Notice of Violation for "Failure to Take Adequate Corrective Action to an Out-of-Specification BAST [Boric Acid Storage Tank] Boron Concentration;" however, this item was closed out when the inspectors determined that RNP corrective actions had been adequately implemented.

The overall effectiveness of the Water Chemistry Program is supported by the excellent operating experience for systems, structures and components, which are influenced by the Water Chemistry Program. No chemistry related degradation has resulted in loss of component intended functions on any systems for which the fluid chemistry is actively controlled. The Water Chemistry Program has been subject to periodic internal assessment activities. These activities, as well as other external assessments help to maintain highly effective chemistry control, and facilitate continuous improvement through monitoring industry initiatives and trends in the area of corrosion control.

Conclusion

The Water Chemistry Program differs from GALL Section XI.M2, Water Chemistry, with respect to:

- An aging mechanism identified in the RNP AMR was not identified in the GALL Report (Loss of Heat Transfer Effectiveness due to Fouling of Heat Transfer Surfaces).
- The RNP Water Chemistry Program implements later revisions of the EPRI guidelines for Primary and Secondary Water Chemistry than recommended in the GALL Report. The RNP Water Chemistry Program is based on the current, approved revisions of EPRI Guidelines as prescribed by NEI 97-06.

These differences have no adverse effects on the ability of the program to manage aging effects, and they are not considered to be actual exceptions to the elements of the Water Chemistry Program described in the GALL Report.

Based on the aging management review of the pressurizer, hydrogen concentration limits for the RCS are maintained in accordance with EPRI guidelines for Primary Water Chemistry. This assures sufficient hydrogen overpressure to manage crevice corrosion of the internal surfaces of the pressurizer.

Based on the above, the Water Chemistry Program is consistent with GALL Section XI.M2, Water Chemistry, and implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.3 REACTOR HEAD CLOSURE STUDS PROGRAM

The Reactor Head Closure Studs Program is credited for aging management of Reactor Head Closure Studs and Stud Components.

The aging effects/mechanisms of concern are (1) Loss of Pre-load due to Stress Relaxation, and (2) Loss of Material due to Wear.

Operating Experience

The Reactor Head Closure Studs Program is implemented through the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program, which monitors the condition of the closure studs and stud components. The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program is implemented and maintained in accordance with the general requirements for engineering programs. This provides assurance that the programs are effectively implemented to meet regulatory, process, and procedure requirements, including periodic reviews, qualified personnel are assigned as program managers, and are given authority and responsibility to implement the program, and adequate resources are committed to program activities.

A search of condition reports was conducted, and no reports documenting deficiencies or problems with vessel head closure studs or stud components were found. Also, a review of the ISI history shows only two instances of stud or stud component deficiencies (thread damage detected by visual examination on stud 41 in May 1992, and replacement of stud 29 which showed a recordable indication during UT in April 2001). Based on these results, the operating experience provides evidence that the ISI program and maintenance practices are ensuring the continuing integrity of the Reactor Head Closure Studs and Stud Components.

Conclusion

While RNP is not committed to Regulatory Guide 1.65, head closure stud fabrication details and preventive measures are consistent with the recommendations of the regulatory guide.

The Reactor Head Closure Studs Program is consistent with GALL Section XI.M3, Reactor Head Closure Studs. Implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.4 STEAM GENERATOR TUBE INTEGRITY PROGRAM

The Steam Generator Tube Integrity Program is performed under the overall Steam Generator Program at RNP. The Steam Generator Tube Integrity Program is credited for aging management of the steam generator tube bundle, tube plugs, tube support plate and anti-vibration bars in the Steam Generators (SG) at RNP.

SG Component Aging Effects/Mechanisms		
Component	Aging Effect	Mechanism
Tube Bundle	Cracking Loss of Material Loss of Material Loss of Material	Stress Corrosion Cracking (SCC) Crevice Corrosion Fretting Pitting Corrosion
Tube Plugs	Cracking Loss of Material Loss of Material	SCC Crevice Corrosion Pitting Corrosion
Tube Support Plates	Cracking Loss of Material Loss of Material Loss of Material	SCC Crevice Corrosion Erosion Pitting Corrosion
SG Anti-Vibration Bars	Cracking Loss of Material Loss of Material	SCC Crevice Corrosion Fretting

Operating Experience

NRC Generic Letter (GL) 97-05, "Steam Generator Tube Inspection Guidelines," required PWR licensees to verify that licensee steam generator tube inspection practices were consistent with existing regulatory requirements and plant licensing bases. In response to the GL, RNP committed to implement the guidance of NEI 97-06, "Steam Generator Program Guidelines," with exceptions, as described in the RNP correspondence. By letter dated August 13, 1998, the NRC did not find any concerns relative to compliance with the RNP licensing basis for the steam generator tube inspection techniques in response to GL 97-05.

The RNP Steam Generator Tube Integrity Program is continually upgraded based on industry experience and research via the Operating Experience and Self-Assessment Programs. Continual improvement of the aging management program has provided an effective means of ensuring the integrity of the steam generator tubes.

The overall effectiveness of the Steam Generator Tube Integrity Program is supported by the excellent operating experience for systems, structures and components, which are influenced by the RNP Steam Generator Tube Integrity Program. No tube integrity related degradation has resulted in loss of component intended function.

Conclusion

Based on the above, the Steam Generator Tube Integrity Program is consistent with GALL Section XI.M19, Steam Generator Tube Integrity; and the continued implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.5 CLOSED-CYCLE COOLING WATER SYSTEM PROGRAM

The Closed-Cycle Cooling Water System Program is credited for aging management of selected components in the following systems at RNP:

- Component Cooling Water (CCW) System
- Diesel Generator System
- Dedicated Shutdown Diesel Generator System
- ESF/TSC Security Diesel Generator System

The aging effects/mechanisms of concern are as follows:

- Loss of Material due to Crevice Corrosion
- Loss of Material due to General Corrosion
- Loss of Material due to Pitting Corrosion
- Loss of Heat Transfer Effectiveness due to Fouling of Heat Transfer Surfaces
- Loss of Material due to Galvanic Corrosion
- Cracking due to SCC
- Loss of Material due to Selective Leaching

Chemistry is regularly monitored and maintained within standards in accordance with EPRI and/or manufacturer's recommendations. Component Cooling Water and Emergency Diesel Generator jacket water employs chromate chemistry. Dedicated Shutdown and EOF/TSC Security Diesel Generator jacket water utilizes a glycol solution supplemented with corrosion inhibitors.

Operating Experience

An operating experience review identified the occurrence of erosion on CCW piping downstream of the Spent Fuel Pool Heat Exchangers. This condition was addressed by replacing the thinned piping, and implementing periodic surveillance to monitor wall thickness in the future. This condition is considered to be caused by system configuration/alignment, and the erosion is managed by the Preventive Maintenance Program.

A detailed study of the Component Cooling Water System was performed in 1998 to assess the operational performance capability of the CCW System. This self-assessment followed the guidance for NRC Safety System Functional Inspections, and included consideration of significant interfacing systems and components. This assessment identified no issues with regard to Component Cooling Water chemistry or aging of Closed Cycle Cooling Water System components.

Conclusion

Based on the above, the Closed-Cycle Cooling Water System Program is consistent with GALL Section XI.M21, Closed-Cycle Cooling Water System; and the continued implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.6 ASME SECTION XI, SUBSECTION IWF PROGRAM

The ASME Section XI, Subsection IWF Program is credited for aging management of Class 1, 2, and 3 component supports (including piping supports) at RNP:

The aging effect/mechanism of concern is loss of material due to general corrosion.

In addition to general corrosion, the IWF Program examines hangers for loss of mechanical function; however, loss of mechanical function was not identified as an age-related degradation in the RNP aging management review.

Operating Experience

The ASME ISI program is implemented and maintained in accordance with the general requirements for engineering programs. This provides assurance that the programs:

- Are effectively implemented to meet regulatory, process, and procedure requirements, including periodic reviews
- Qualified personnel are assigned as program managers, and are given authority and responsibility to implement the program
- Adequate resources are committed to program activities
- Are managed in accordance with plant administrative controls

The RNP ASME Section XI, Subsection IWF Program is administered by two implementation documents that are administratively controlled by procedure. The first document is the Fourth Ten-Year Interval Inservice Inspection Program. The second document is the Fourth Ten-Year Interval Inservice Inspection Plan. The two ISI implementation documents combine to form the H.B. Robinson Fourth Ten-Year Interval Inservice Inspection Program/Plan.

The operating experience review determined that documentation exists demonstrating that discrepancies found during the visual examination of supports are transmitted to engineering personnel for evaluation.

Processes at RNP are continually being upgraded based upon industry experience and research. In addition to industry experience, plant-operating experiences are shared between CP&L and FPC sites through regular peer group meetings.

The ASME Section XI, Subsection IWF Program is subject to ongoing self-assessments; when weaknesses are noted actions are taken under the corrective action program (CAP) to initiate program improvements. The CAP has been effective in ensuring that the ISI program is continually improving.

Conclusion

Based on the above, the ASME Section XI, Subsection IWF Program is consistent with GALL Section XI.S3, ASME Section XI, Subsection IWF; and the continued implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

In the evaluation of the ASME Section XI, Subsection IWF Program against the program elements of the GALL Report, exceptions to Code requirements that have been granted by approved relief requests were not considered to be exceptions to the GALL criteria.

B.2.7 10 CFR PART 50, APPENDIX J PROGRAM

The 10 CFR Part 50, Appendix J Program is credited for aging management of selected components of the Reactor Containment Building at RNP.

The aging effects/mechanisms of concern are as follows:

- Cracking due to Elevated Temperature
- Cracking due to Thermal Fatigue
- Change in Material Properties due to Elevated Temperature
- Loss of Material due to General Corrosion
- Loss of Material due to Wear
- Loss of Material due to Aggressive Chemical Attack
- Loss of Material due to Crevice Corrosion
- Loss of Material due to Galvanic Corrosion
- Loss of Material due to Pitting Corrosion

Operating Experience

The Appendix J Program is implemented and maintained in accordance with the general requirements for engineering programs. This provides assurance that the programs:

- Are effectively implemented to meet regulatory, process, and procedure requirements, including periodic reviews
- Qualified personnel are assigned as program managers, and are given authority and responsibility to implement the program
- Adequate resources are committed to program activities
- Are managed in accordance with plant administrative controls

The RNP Appendix J Program is continually upgraded based on industry experience and research. This aging management program has provided effective means of ensuring the structural integrity and leak tightness of the RNP containment. In addition to industry experience, plant-operating experiences are shared between CP&L and Florida Power Corporation (FPC) sites through regular peer group meetings.

Based on a review of condition reports and inspection results, the corrective action program (CAP) has been effective in ensuring that the Appendix J program is continually improving. Several Condition Reports have been generated as a result of as-found conditions or as a result of assessments (site and corporate). When weaknesses are noted, actions are taken under the CAP to initiate program improvements. Program improvements were also made as a result of NRC Inspections.

Conclusion

Based on the above, the 10 CFR Part 50, Appendix J Program is consistent with GALL Section XI.S4, 10 CFR Part 50, Appendix J; and the continued implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.8 FLUX THIMBLE EDDY CURRENT INSPECTION PROGRAM

The Flux Thimble Eddy Current Inspection Program is credited for aging management of incore flux thimbles.

The aging effect/mechanism of concern is loss of material due to wear.

Evaluation

The following summarizes the results of an evaluation of the Program against the 10 program elements identified in Appendix A of the SRP-LR.

<u>Scope of Program</u>

The Flux Thimble Eddy Current Inspection Program is based upon current plant activities delineated in an existing procedure governing flux thimble eddy current inspection. The procedure was implemented to satisfy NRC Bulletin 88-09 requirements that a tube wear inspection procedure be established and maintained for Westinghouse supplied reactors which use bottom mounted flux thimble tube instrumentation.

<u>Preventive Actions</u>

The Flux Thimble Eddy Current Inspection Program is a condition-monitoring program; therefore, there are no preventive actions.

<u>Parameters Monitored/Inspected</u>

The aging effect to be managed by the Flux Thimble Eddy Current Inspection Program is loss of material due to wear in the double-walled, incore flux thimble tubes. This program is designed specifically to detect and manage that aging effect.

<u>Detection of Aging Effects</u>

The Flux Thimble Eddy Current Inspection Program is a periodic volumetric examination that allows a projection of the rate of wear of the double-walled, incore flux thimble tubes. It is performed at a variable frequency dependent on extrapolation of wear rates determined from previous inspections. This ensures that timely corrective action will be performed well before the projected failure of any of the tubes due to wear could occur.

Monitoring and Trending

The Flux Thimble Eddy Current Inspection Program projects the rate of wear of the double-walled, incore flux thimble tubes ensuring that timely corrective action will be performed well before failure of any of the tubes due to wear could occur.

Acceptance Criteria

The administrative controls for the Flux Thimble Eddy Current Inspection Procedure provide specific, objective acceptance criteria that ensure that any thimble tube that is expected to experience through wall wear greater than the ASME criteria specified for the examination prior to the next inspection is removed from service. No subjective analysis that might permit a marginal tube to be returned to service is permitted by the procedure.

Corrective Actions

Corrective actions including root cause determinations and prevention of recurrence are done in accordance with the corrective action program. Timeliness of corrective action is monitored.

Confirmation Process

Effectiveness of this AMP will be monitored using corrective action program and quality assurance procedures, review and approval processes, and administrative controls which are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B.

Administrative Controls

RNP corrective action and quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of extended operation.

Operating Experience

A review of condition reports identified two involving thimble tubes. Both of the condition reports identified thimble tubes with very small leak rates. The leaks were evaluated under the corrective action program; however, the root cause of leakage could not be determined. Corrective action involved replacing the thimble tubes.

Conclusion

The Flux Thimble Eddy Current Inspection Program is consistent with the 10 program elements identified in Appendix A of the SRP-LR. The program provides reasonable assurance that the aging effects will be managed such that the flux thimbles will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3 ENHANCED AGING MANAGEMENT PROGRAMS

B.3.1 FIRE PROTECTION PROGRAM

The Fire Protection Program is credited for aging management of selected components in fire protection–related systems at RNP.

The aging effects/mechanisms of concern are as follows:

- Loss of Material due to General Corrosion
- Cracking due to Differential Movement
- Cracking due to Vibration
- Delamination and Separation due to Differential Movement
- Delamination and Separation due to Vibration
- Loss of Material due to Abrasion
- Cracking Due to Elevated Temperature
- Change in Material Properties Due to Elevated Temperatures

As a result of the license renewal review, the program element *Parameters Monitored/ Inspected* will be enhanced to note that concrete surface inspections performed under administrative controls for structures monitoring are credited by the Fire Protection Program for inspections of fire barrier walls, ceilings, and floors.

Operating Experience

The last five years of self-assessment and external (triennial) inspections were reviewed for programmatic deficiencies that remained outstanding. These inspection reports noted a number of strengths and weaknesses, but generally found that the fire protection program was effective in fulfilling regulatory requirements and supporting the operation of RNP. Additionally, a 1996 NRC inspection of the Fire Protection Program was performed. In general, the self-assessments, external inspections, and NRC Inspection reports provide evidence that the fire protection program is not only effective, but also subject to ongoing observation/ assessment and continual improvement.

Although not formally credited for aging management of Fire Protection by this program, the Site Fire Protection System is a Maintenance Rule System and subject to periodic walkdowns, trending, and monitoring. This includes the bi-annual generation of formal system health reports, which are maintained in the system libraries under the Maintenance Rule Program. In general, self-assessments, external inspections, NRC Inspection reports and system health reports provide evidence that the fire protection program is not only effective, but also subject to ongoing observation/assessment and continual improvement.

Conclusion

The Fire Protection Program is consistent with GALL Report, Section XI.M26, Fire Protection, with the exceptions noted below. Both the exceptions and applicable aging program element(s) are identified.

- *Parameters Monitored/Inspected and Detection of Aging Effects:* Valve alignment and system status are not formally verified each month. Operator procedures check valve position and system status subsequent to any system realignments and as needed to support plant operation, and current procedures/ practices deemed acceptable for the current license period are considered to be sufficient for the period of extended operation. The current activities, in accordance with the existing Fire Protection Program are considered to be acceptable during the period of extended operation.
- *Detection of Aging Effects and Monitoring and Trending:* Inspection of fire barriers under systems and structures monitoring procedures are performed at a level of scrutiny deemed necessary by trained personnel to ensure operability, but are not specifically required on a level of detail commensurate with VT-1 inspections. The inspection interval is based on safety significance, not to exceed 10 years, as compared to the refueling frequency specified by GALL. The current activities, in accordance with the existing Fire Protection Program are considered to be acceptable during the period of extended operation, also.
- *Parameters Monitored/Inspected and Monitoring and Trending:* An exception is taken to GALL with regard to inspection frequency of fire doors. GALL specifies bi-monthly inspections, whereas RNP performs detailed inspections semi-annually, augmented by frequent inspections during operator rounds and additional inspections under system/structural monitoring procedures. The current activities, in accordance with the existing Fire Protection Program are considered to be acceptable during the period of extended operation, also.

These exceptions involve differences between the GALL Report and current Fire Protection Program activities (inspection frequencies and visual inspection criteria). The differences would have no adverse effect on aging management, since the current Fire Protection Program has proven to be effective based on operating experience. Note that the exception involving valve alignment and system status does not involve prevention or mitigation of aging effects applicable to the Fire Protection System.

Therefore, the Fire Protection Program, with the enhancement and the exceptions evaluated above, is considered to be consistent with GALL Section XI.M26, Fire Protection. Implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3.2 BORIC ACID CORROSION PROGRAM

The Boric Acid Corrosion Program is credited for aging management of components in systems and structures at RNP. The aging effects/mechanisms of concern are:

- Loss of material due to aggressive chemical attack
- Loss of mechanical closure integrity due to loss of material due to aggressive chemical attack
- Loss of material due to general corrosion
- Loss of material due to crevice corrosion
- Loss of material due to pitting corrosion

As a result of the license renewal review, the program element *Scope of Program* will be enhanced. This involves expanding the scope of the Boric Acid Corrosion Program to (1) ensure that mechanical, structural, and electrical components in scope for license renewal are covered, and (2) identify additional areas in which components may be susceptible to exposure from boric acid (e.g. containment, auxiliary, and spent fuel buildings).

Based on the aging management review for the pressurizer, boric acid leakage from the pressurizer is managed in accordance with ASME Section XI, Category B-P, as well as the Boric Acid Corrosion Program.

Operating Experience

A review of the condition report (CR) database determined that most plant operating events involving the Boric Acid Corrosion Program dealt with improvements to the inspection methods and acceptance criteria resulting from evaluations of leaks that occurred in plant systems. Also, NRC Inspection Reports for boric acid corrosion were reviewed. One violation was received: "Failure to Provide Adequate Work Instruction for Degraded Stud Inspection.” This involved failure to establish adequate work instructions (procedures) requiring direct or indirect visual inspection of the C Reactor Coolant Pump studs after the removal of boric acid residue and corrosion products. The RNP response to the violation resulted in a revision to improve the Boric Acid Corrosion Program.

Conclusion

The Boric Acid Corrosion Program, with the enhancements identified above, is consistent with GALL Section XI.M10, Boric Acid Corrosion, and implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3.3 FLOW-ACCELERATED CORROSION PROGRAM

The Flow Accelerated Corrosion (FAC) Program is credited for aging management of selected carbon steel and low-alloy steel piping and components in secondary systems at RNP.

The aging effects/mechanisms of concern are as follows:

- Loss of material due to FAC
- Loss of material due to erosion

As a result of the license renewal review, enhancements will be made to the program elements for *Scope of Program* and *Corrective Actions*. The review identified components that may be susceptible to FAC or to erosion. These components will be added to the Program scope. Also, administrative controls for the program will be revised to mandate that corrective actions be taken in accordance with the corrective action program when certain acceptance criteria are not met.

Operating Experience

The FAC Program is implemented and maintained in accordance with the general requirements for engineering programs. This provides assurance that the programs:

- Are effectively implemented to meet regulatory, process, and procedure requirements, including periodic reviews
- Qualified personnel are assigned as program managers, and are given authority and responsibility to implement the program
- Adequate resources are commited to program activities
- Are managed in accordance with plant administrative controls

Since the advent of Generic Letter 89-08, the corrective action program has been effective in ensuring that the FAC Program is continually improving. Several condition reports have been generated as a result of as-found conditions or as a result of assessments (site and corporate). These resulted in improvements to the FAC Program. Program improvements were also made as a result of NRC Inspections.

Conclusion

The Flow Accelerated Corrosion Program, with the enhancements identified above, is consistent with GALL Section XI.M17, Flow Accelerated Corrosion, and implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3.4 BOLTING INTEGRITY PROGRAM

The Bolting Integrity Program addresses aging management requirements for bolting on mechanical components within the scope of license renewal. The RNP Bolting Integrity Program is based on industry recommendations and EPRI guidance which considers material properties, joint/gasket design, service requirements and industry/site operating experience in specifying torque and closure requirements, with additional programmatic inspections and requirements as needed to adequately manage aging mechanisms.

The aging effects/mechanisms of concern specifically identified with regard to bolting integrity in applicable systems are:

- Loss of Material due to Wear
- Loss of Mechanical Closure Integrity due to SCC
- Loss of Pre-Load Due to Stress Relaxation
- Loss of Mechanical Closure Integrity Due to Loss of Material Due to Aggressive Chemical Attack (Boric Acid Wastage)

The Bolting Integrity Program relies on other aging management programs to manage specific aging effects. The Section XI, Subsection IWB, IWC and IWD Aging Management Program is credited with inspecting selected bolting within Section XI boundaries. In addition, the Preventive Maintenance Program performs regular inspections of Reactor Coolant Pump bolting. AMRs have credited the Boric Acid Corrosion Program for management of Loss of Mechanical Closure Integrity Due to Loss of Material Due to Aggressive Chemical Attack (Boric Acid Wastage) for mechanical system bolted closures subject to boric acid leakage. Otherwise, from the standpoint of loss of material due to general corrosion, bolting on mechanical components is treated as a sub-component (i.e., a part of the parent component), and the Systems Monitoring Program is utilized to manage this aging effect. The ASME Section XI, Subsection IWF Program, is credited for aging management of all structural bolting associated with Class 1, 2, and 3 component supports; and the Structures Monitoring Program is credited for aging management of all structural bolting other than those associated with Class 1, 2, and 3 components.

The Bolting Integrity Program described in GALL, Section XI.M18, Bolting Integrity, specifically addresses both mechanical and structural bolting. Although the Bolting Integrity Program is only credited at RNP for aging management of mechanical system bolting, both mechanical and structural bolting were reviewed to establish consistency with the GALL Report.

As a result of the license renewal review, enhancements will be made to the program administrative controls involving the program elements *Preventive Action* and *Parameters Monitored/Inspected*. The Bolting Integrity Program implementing documents will be enhanced to prohibit the use of molybdenum disulfide (MoS_2)

compounds in high strength bolting applications and to direct that high strength bolting used on one motor operated valve be inspected and evaluated prior to the end of the current operating period.

Operating Experience

The RNP implementation of NRC Bulletin 82-02, Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants, has been the subject of a number of NRC inspections. An NRC Inspection Report dated May 1987 notes that reviews of the maintenance history and program for lubricating threaded fasteners had been completed in 1982, and subsequent reviews were performed in 1983 and 1984, with no problems being identified.

The RNP Corrective Action Program has addressed several bolting issues:

- An instance boric acid wastage of bolting closure (studs and nuts) on one reactor coolant pump (RCP) main flange as evidenced by boric acid buildup during visual inspections. The cause of the leakage was attributed to stress relaxation over several operating cycles. A requirement for checking and retensioning of RCP flange bolts every other refueling outage has been incorporated into the Preventive Maintenance Program.
- Torque requirements associated with flange bolting on two motor operated valves were revised (one of which is within the scope of license renewal). This bolting was replaced with “high strength” bolting based on MOV operability weak link analysis and is the single instance of high strength pressure boundary bolting in plant systems. This bolting will be removed and inspected (or replaced) prior to the end of the current license period.
- A small number of corrective actions were identified which involved incorrect torque values or use of incorrect materials. These were self-identified events of an infrequent and isolated nature. These items were corrected by field changes, document updates and/or procedure revisions.

Conclusion

The Bolting Integrity Program is consistent with the GALL Report Section XI.M18, Bolting Integrity, with the following exceptions. Both the exceptions and applicable aging program element(s) are identified.

- *Scope of Program:* The Bolting Integrity Program is not utilized to address aging management requirements for structural bolting. This is not considered to be an exception with respect to mechanical system closure bolting.
- *Parameters Monitored/Inspected:* GALL specifies that high strength bolting used in NSSS component supports be inspected to the requirements for Class 1 components, examination category B-G-1. Exception is taken to these

requirements, since bolting in this application has been evaluated to be not susceptible to SCC owing to its location in a benign environment. Similarly, exception is taken regarding requirements for subjecting this bolting to an ongoing program for crack monitoring.

These exceptions were evaluated during RNP license renewal AMR. It has been concluded that the above differences will have no adverse effect on the intended functions of system closure bolting at RNP.

Therefore, the Bolting Integrity Program, with previously described enhancements and exceptions, is considered to be consistent with GALL Section XI.M18, Bolting Integrity. Continued implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3.5 OPEN CYCLE COOLING WATER SYSTEM PROGRAM

The Open Cycle Cooling Water System Program is credited for aging management of components in systems at RNP.

The aging effects/mechanisms of concern are as follows:

- Flow Blockage due to Fouling
- Loss of Heat Transfer Effectiveness due to Fouling of Heat Transfer Surfaces
- Loss of Material due to Crevice Corrosion
- Loss of Material due to Galvanic Corrosion
- Loss of Material due to General Corrosion
- Loss of Material due to MIC
- Loss of Material due to Pitting Corrosion
- Loss of Material due to Erosion

RNP maintains a formal plant program for oversight of the plant's commitments to GL 89-13. This program, identified as the "Cooling Water Reliability Program (GL 89-13)," generally corresponds to the Open Cycle Cooling Water System AMP described in the GALL Report, Section XI.M20. The RNP Cooling Water Reliability Program forms the basis for the Open Cycle Cooling Water System Program.

As a result of the license renewal review, an enhancement will be made to the program element of *Detection of Aging Effects.* The program will be enhanced to initiate an action under the site Preventive Maintenance Program to periodically replace cooling coils in certain room coolers. Also, a requirement to perform a one-time volumetric inspection of the CCW heat exchanger tubes prior to the end of the current license period will be incorporated into the One-Time Inspection Program. Results from this inspection will be used to determine the need for inspections/testing over the period of extended operation.

Operating Experience

The guidance of NRC GL 89-13 has been implemented for approximately 10 years and has been effective in managing aging effects due to biofouling, corrosion, erosion, protective coating failures, and silting in structures and components serviced by Open Cycle Cooling Water Systems. The RNP GL 89-13 program has been the subject of a number of assessments and inspections, including an NRC Service Water Operational Performance Inspection soon after program development to verify implementation of the GL requirements. This resulted in an "Integrated Service Water Action Plan" to address several NRC open items. The open items have been resolved.

A more recent NRC inspection, in July 2000, included a review of heat sink performance, specifically addressing methods and results of performance inspections of risk significant Service Water cooled heat exchangers. This inspection focused on the CCW heat exchangers, the EDG heat exchangers and the SI pump bearing coolers. The history of chemical treatment of the service water supplied to these heat exchangers was reviewed, as well as the closure of a number of condition reports generated as a result of previous Service Water self assessments. The inspection also included observations of the EDG B heat exchanger inspections and maintenance and verified that inspection results were appropriately categorized against pre-established acceptance criteria and that the acceptance criteria were adequately met. No findings were identified as a result of this inspection.

Conclusion

The Open Cycle Cooling Water System Program, with the enhancement identified above, is consistent with GALL Section XI.M20, Open-Cycle Cooling Water System, and the continued implementation of the program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3.6 INSPECTION OF OVERHEAD HEAVY LOAD AND LIGHT LOAD HANDLING SYSTEMS PROGRAM

The Overhead Heavy and Light Load Handling Systems Program is credited for aging management of the following crane lifting devices at RNP.

- Containment Polar Crane
- Spent Fuel Cask Crane
- Turbine Gantry Crane
- Spent Fuel Bridge Crane

The aging effect/mechanism of concern is loss of material due to corrosion.

As a result of the license renewal review, enhancements will be made to the program element for *Scope of Program*. Enhancements will be made to program administrative controls to: (1) add the Turbine Gantry Crane as a system requiring walkdown for license renewal purposes and (2) note that cranes are to be inspected using the attribute inspection checklist for structures.

Operating Experience

Three of the cranes that are in scope for license renewal have been addressed by the Maintenance Rule requirements provided in 10 CFR 50.65 and, therefore, have documented operating experience. The Maintenance Rule Program demonstrates that testing and monitoring programs have been implemented and have ensured that the structures, systems, and components of the cranes are capable of sustaining their rated loads. This is their intended function during the period of extended operation. It is noted that many of the systems and components of these cranes perform an intended function with moving parts or with a change in configuration, or subject to replacement based on qualified life. In these instances, these types of crane systems and components are not within the scope of this aging management program. This program is primarily concerned with structural components that make up the bridge and trolley. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," provides specific guidance on the control of overhead heavy load cranes.

The four cranes in scope for license renewal are periodically inspected to satisfy the ANSI B30.2 and NUREG-0612 requirements for inspection attributes such as steel member corrosion, damaged members or connections, baseplate or anchor bolt corrosion, damaged or degraded grout pads, structure geometry to include absence of excessive deflection, cross section distortion, or member misalignment, missing parts, coat deficiencies, and structural cracking. Inspections are documented on a System Walkdown report. The Work Management program schedules performance of crane maintenance and corrective actions.

Implementation of the above programs assures the continuing management of aging effects that may impact the ability of the in-scope cranes to perform their intended function.

Conclusion

The Overhead Heavy and Light Load Handling Systems Program, with the enhancements identified above, is consistent with GALL Section XI.M23, Overhead Heavy and Light Load (Related to Refueling) Handling Systems, and implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3.7 FIRE WATER SYSTEM PROGRAM

The Fire Water System Program is credited for aging management of selected components in fire protection related systems and structures at RNP:

The aging effects/mechanisms of concern are as follows:

- Flow Blockage due to Fouling
- Loss of Material due to Crevice Corrosion
- Loss of Material due to Galvanic Corrosion
- Loss of Material due to General Corrosion
- Loss of Material due to MIC
- Loss of Material due to Pitting Corrosion

As a result of the license renewal review, the following enhancement, involving the program element of *Acceptance Criteria* has been identified:

- NFPA 25 requires a program of field service testing of sprinkler heads be implemented for those in service for 50 years. The RNP Fire Water System Program will include requirements for an initial test effort implemented prior to the end of the current license period, and repeated 10 years into the period of extended operation.

Operating Experience

Plant operating experience documents corrosion related failure of the fire pump casings due to general corrosion and thinning in the “splash zones”. This aging mechanism is managed by replacing the pump casings periodically in accordance with the plant PM program. No pump casing failures have been documented since the implementation of this program.

The last five years of self-assessment and external (triennial) inspections were reviewed for programmatic deficiencies that remained outstanding. These inspection reports noted a number of strengths and weaknesses, but generally found that the fire protection program was effective in fulfilling regulatory requirements and supporting the operation of RNP. Additionally, a 1996 NRC inspection of the Fire Protection Program is documented in NRC Inspection Report No. 50-261/96-12. In general, the self-assessments, external inspections, and NRC Inspection reports provide evidence that the fire protection program is not only effective, but also subject to ongoing observation/assessment and continual improvement.

Conclusion

The Fire Water System Program differs from the GALL Report, Section XI.M27, Fire Water System, with respect to the following exceptions. Both the exceptions and applicable aging program element are identified.

- *Detection of Aging Effects:* GALL, Volume 2, Section XI.M27, specifies a program of full flow testing of portions of fire protection sprinkler systems which are not routinely subject to flow at maximum design flow and pressure prior to the period of extended operation, to be repeated at not more than 5 year intervals thereafter. The RNP Fire Water System Program will include requirements to either conduct full flow testing per these requirements, or as an alternative to conduct internal inspections or UT examination of a representative sampling of these systems. Results from initial tests and inspections, reflecting 40 years of service, will be used to determine expansion of scope and subsequent test/inspection intervals not to exceed 10 years. Note that full flow testing or internal inspections or UT examinations are not applicable to “dry pipe” portions of sprinkler systems, as they are not susceptible to biofouling.

The exceptions to the GALL-recommended program involve a proposed alternative to full flow testing, a potentially longer test/inspection interval, the use of representative sampling, and elimination of testing for dry pipe portions of sprinkler systems. Elimination of dry pipe portions from testing/inspection and use of internal inspections or UT examinations are considered to be as effective as full flow testing to detect degradation that would adversely affect system flow. Also, the potentially longer test interval (up to 10 years, instead of the GALL-recommended 5 years) and use of sampling would not adversely affect detection of aging effects, because the interval and scope would be adjusted, using the results of actual tests and inspections reflecting operating experience of 40 years of service. Employment of a program element that is based on plant experience would result in more efficient use of aging management assets than the GALL recommendation.

Based on the above discussion, the Fire Water System Program, with above described enhancement and exceptions, is considered to be consistent with GALL Section XI.M27, Fire Water System, and implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3.8 BURIED PIPING AND TANKS SURVEILLANCE PROGRAM

The Buried Piping and Tanks Surveillance Program is credited for aging management of selected components in the Fuel Oil System at RNP:

The aging effects/mechanisms of concern are as follows:

- Loss of Material due to Crevice Corrosion
- Loss of Material due to General Corrosion
- Loss of Material due to MIC
- Loss of Material due to Pitting Corrosion

An impressed current, Cathodic Protection System protects the buried Fuel Oil System piping and the bottoms of the connected, above-ground tanks. Aspects of underground Fuel Oil System Piping relating to coatings and visual inspections are included within the scope of the Buried Piping and Tanks <u>Inspection</u> Program. The license renewal review of this program evaluated the acceptance criteria associated with the Cathodic Protection System.

As a result of the license renewal review, the program elements of *Acceptance Criteria* and *Confirmation Process* of the Buried Piping and Tanks Surveillance Program will be enhanced to:

- Review and update, as necessary, Cathodic Protection procedures to ensure consistency with NACE Standard RP-0169, 1996. The review should focus on Acceptance Criteria to ascertain the condition of the Cathodic Protection System.
- Install pressure taps and perform leak testing on the underground fuel oil piping from Unit 1 to the Unit 2 Diesel Fuel Oil Storage Tank and the underground piping from the Diesel Fuel Oil Storage Tank to each Emergency Diesel Generator (EDG) Day Tank in the Reactor Auxiliary Building (RAB).

Operating Experience

A 1991 NRC inspection determined that the cathodic protection system was known to have been operating outside of its original specification. Therefore, the inspection concluded that only about 7 years of cathodic protection could be assured following the system's installation in 1981. Degradation of the cathodic protection system in 1988 appeared to have been caused by installation of concrete in the yard. Closure of this concern was based on an inspection of EDG fuel oil underground piping that demonstrated the piping coating was intact with no detectable piping degradation. RNP concluded from this sample that the underground fuel oil piping had not degraded by galvanic corrosion. Additionally, RNP completed a hardware upgrade of the Cathodic

Protection System and established base line operating parameters. The NRC found that RNP technical staff demonstrated a good knowledge level of the system operation and design. The inspector concluded the licensee had accomplished appropriate actions to verify the integrity of the underground fuel oil piping and had upgraded the cathodic protection system to an operable status. The net effect of this inspection was an increased emphasis was placed on operation of the cathodic protection system.

Anomalies in data recorded during the monitoring of the Cathodic Protection System resulted in system assessments by technical experts in the field of cathodic protection in 1996 and 2001. Both assessments recommended corrective action be taken to repair the system. Nevertheless, the reports concluded that the as-found condition for substantial portions of the buried fuel piping indicated they had some level of cathodic protection prior to system repairs. These evaluations demonstrate that identification of abnormal conditions is occurring as planned.

Conclusion

The Buried Piping and Tanks Surveillance Program differs from the GALL Report, Section XI.M28, Buried Piping and Tanks Surveillance, with respect to the following exceptions. Both the exceptions and applicable aging program elements are identified.

- *Scope of Program:* The RNP program uses the guidance in NACE RP-01-69-76 in lieu of the 1996 standard. The abovementioned enhancement to review and update, as necessary, cathodic protection procedures to ensure consistency with NACE Standard RP-0169, 1996, will address this exception.
- *Scope of Program:* There are no buried tanks in this program. The RNP Cathodic Protection System protects buried Fuel Oil System piping and the external, tank bottom surfaces of Fuel Oil System tanks in contact with the ground.
- *Scope of Program and Preventive Actions:* Aspects of underground Fuel Oil System piping relating to coatings and inspections are included within the scope of the Buried Piping and Tanks Inspection Program in lieu of this program.
- *Parameters Monitored/Inspected:* No documentation of initial coating conductance is available. In-situ measurement of coating conductance is not considered prudent due to the potential to cause coating damage during excavation and measurement, changing the local soil electrolytic conditions, or stressing the coatings due to changes in the local conditions of the supporting soil.
- *Acceptance Criteria:* The Buried Piping and Tanks Inspection Program, in lieu of this program, is used to determine the condition of pipe coatings when piping is exposed for any reason.

These differences have been evaluated. The exceptions involving the NACE standard will be addressed by the enhancement planned for this Program. The fact that there are no buried tanks has no effect on the capability of the Program to detect and manage aging effects. When considered together with the planned activities under the Buried Piping and Tanks Inspection Program, the exceptions involving buried piping coatings and inspections will be adequately addressed by the combined activities of the two Programs. The GALL Report recommends one of the two programs to manage aging of buried piping and tanks; RNP implements both. The Buried Piping and Tanks inspection Program addresses activities related to visual inspection of buried components. Additional assurance of coating integrity can be inferred by using cathodic protection current measurements. Also, the need for periodic inspections of buried components is reduced by the protection afforded by the impressed current cathodic protection system. Thus, the preventive measures of cathodic protection provided under the Buried Piping and Tanks Surveillance Program and the detection measures provided under the Buried Piping and Tanks inspection Program provide added assurance that aging effects for buried fuel oil piping and tank bottoms will be adequately managed.

Based on the above discussion, the Buried Piping and Tanks Surveillance Program, with previously identified enhancements and exceptions, is considered to be consistent with GALL Report, Section XI.M28, Buried Piping and Tanks Surveillance, and implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3.9 ABOVE GROUND CARBON STEEL TANKS PROGRAM

The Above Ground Carbon Steel Tanks Program is a collection of activities under administrative controls at RNP but considered to constitute a separate program. This Program is credited for aging management of exterior surfaces of tanks in the Fuel Oil System at RNP.

The aging effects/mechanisms of concern is loss of material due to general corrosion.

As a result of the license renewal review, the program elements of *Monitoring and Trending* and *Acceptance Criteria* for the Above Ground Carbon Steel Tanks Program will be enhanced to assure that the external surfaces of the fuel oil tanks are inspected periodically and to include, in the administrative controls for the Program, a section specifically addressing corrective actions.

Operating Experience

Based on a condition report review, a CR was issued to address several pits discovered in one vertical Unit 1 IC Turbine Fuel Oil Tank floor. Further investigation indicated a loss of diesel fuel had occurred from the tank. The CR then concluded, "The spill was due to an ineffective tank inspection program. The frequency at which past tank inspections had been performed could not be determined. Had the tanks been receiving inspections on an on going basis, predictive maintenance activities would have identified the potential for a leak. The tanks are now scheduled for inspections on a five year cycle." The leak occurred owing to pitting on the inside surface of the tank bottom. Therefore, this operating experience is applicable to internal tank corrosion. Corrosion of this type would be minimized by the Fuel Oil Chemistry Program, as opposed to the Above Ground Carbon Steel Tanks Program.

Conclusion

The Above Ground Carbon Steel Tanks Program differs from GALL Section XI.M29, Above Ground Carbon Steel Tanks, with respect to the following exception. Both the exception and applicable aging program elements are identified.

- *Detection of Aging Effects and Acceptance Criteria:* Thickness measurements are not performed on tank bottoms to detect exterior corrosion, because the tanks are protected from corrosion by the Cathodic Protection System (refer to the Buried Piping and Tanks Surveillance Program for a discussion of the Cathodic Protection System) and the tanks are located on a layer of oily sand.

This difference has been evaluated. The proposed use of cathodic protection and the oily sand used in the tank foundation provide protection against external corrosion of the tank bottoms. This provides better aging management of tank bottom degradation than detection of corrosion by thickness measurement.

The Above Ground Carbon Steel Tanks Program, with above described enhancements and exception, is considered to be consistent with GALL Section XI.M29, Above Ground Carbon Steel Tanks. Implementation of the Above Ground Carbon Steel Tanks Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3.10 FUEL OIL CHEMISTRY PROGRAM

The Fuel Oil Chemistry Program is credited for aging management of selected components in the Fuel Oil System at RNP.

The aging effect/mechanisms of concern are:

- Loss of Material due to Crevice Corrosion in Carbon Steel
- Loss of Material due to General Corrosion in Carbon Steel
- Loss of Material due to MIC in Carbon Steel, Copper Alloys, & Stainless Steel
- Loss of Material due to Pitting Corrosion in Carbon Steel

As a result of the license renewal review, the following enhancements affecting the program elements of *Preventive Actions, Parameters Monitored/Inspected,* and *Monitoring and Trending* will be made to the Program. The administrative controls for the Program will be enhanced to (1) improve sampling and de-watering of selected storage tanks, (2) formalize existing practices for draining and filling the Diesel Fuel Oil Storage Tank periodically, (3) formalize bacteria testing for fuel oil samples from various tanks, and (4) incorporate quarterly trending of fuel oil chemistry parameters.

Operating Experience

A number of condition reports (CRs) were initiated that resulted in self identified improvements to the fuel oil chemistry program, procedures, and training via self-assessments and other individual initiatives.

One CR summarizes a 1995 review of industry issues and how it relates to the RNP Fuel Oil System. The evaluation reinforced the conclusion that proper measures are being taken to ensure a high quality fuel supply is being delivered to Unit 1 (and consequently to Unit 2 from Unit 1). The CR provided a discussion of the measures taken to minimize biological growth in the Diesel Fuel Oil Storage Tank to reduce the potential for fouling and provided a basis for not requiring biocide addition.

As a follow-up to discovery of several through wall pits in the IC Turbine Lighting Oil Tank floor, the other three Unit 1 tanks were inspected. (These three Unit 1 tanks are within scope for license renewal.) One tank showed severe pitting. The other two tanks were found in excellent condition. The degraded tanks were repaired. The Unit 1 tanks are inspected periodically at a frequency from 5 years to 10 years based on tank condition and corrective actions taken.

Two additional events involved potential contamination of fuel oil. One involved receipt of contaminated fuel oil and resulted in a request for improved controls on carrier oil quality. The other event involved coating degradation and pitting corrosion to the Diesel

Fuel Oil Storage Tank bottom. Regarding the latter event, corrective actions are scheduled to be performed to repair the tank bottom.

The Maintenance Rule documentation for the system includes laboratory results from oil sample testing. Of note is that no adverse bacteria had been identified and results of chemical testing show bulk average oil conditions have always been within specifications.

Conclusion

The Fuel Oil Chemistry Program differs from GALL Section XI.M30, Fuel Oil Chemistry, with respect to the following exceptions. Both the exceptions and applicable aging program elements are identified.

- *Scope of Program:* In addition to storage tanks, the Program is used to manage aging effects on all system components "wetted" by fuel oil. This results in additional materials being in scope beyond those in the GALL Report.
- *Preventive Actions:* Based on operating history and fuel oil management activities, biocides, biological stabilizers and corrosion inhibitors are not necessary and are not used in the fuel oil at RNP.
- *Parameters Monitored/Inspected, Detection of Aging Effects and Acceptance Criteria:* Alternate standards and acceptance criteria are used for fuel oil sampling at RNP in place of the ASTM standards recommended in the GALL Report.
- *Detection of Aging Effects:* Ultrasonic thickness (UT) measurements of bottoms on large storage tanks are not typically performed at RNP unless warranted by the level of coating degradation and corrosion found during inspection.
- *Preventive Actions:* A one-time inspection of small, elevated, Diesel Fire Pump Fuel Oil Tank and diesel generator day tanks is not warranted. These small tanks have limited access to the tank internals making it impractical to clean and perform a meaningful inspection. Ultrasonic testing is also considered inappropriate to detect small amounts of pitting in tanks constructed of carbon steel that is measured in units of gauge thickness (e.g., Diesel Fire Pump Fuel Oil tank is made of hot rolled, 12 gauge steel). Based on operating history, external tank and structure inspections are considered sufficient to identify degradation in the tank walls.

These differences have been evaluated and determined to result in no significant adverse effects on the ability of the program to manage aging effects.

The Fuel Oil Chemistry Program, with the enhancements identified previously, is consistent with GALL Section XI.M30, Fuel Oil Chemistry, with acceptable exceptions, and implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3.11 REACTOR VESSEL SURVEILLANCE PROGRAM

The Reactor Vessel Surveillance Program is credited for aging management of the following components at RNP:

- Reactor Vessel Upper Shell, Intermediate Shell, and Lower Shell
- Reactor Vessel Inlet and Outlet Nozzles

The aging effect / mechanism of concern is a change in material properties due to irradiation embrittlement.

As a result of the license renewal review, the Reactor Vessel Surveillance Program will be enhanced to revise RNP procedures to require surveillance test samples to be stored in lieu of optional disposal.

Operating Experience

The RNP Reactor Vessel Surveillance Program is in compliance with 10 CFR 50, Appendix H. Surveillance capsules have been withdrawn and tested in the past, and the data from these surveillance capsules and data from other industry sources has been used to verify and predict the performance of RNP reactor vessel beltline materials with respect to neutron embrittlement. Calculations have been performed as required to project the degree of reduction of Upper Shelf Energy expected to result from projected neutron exposure in the future, including 60-year projections. Pressure/temperature limits have been imposed on operational parameters at RNP to assure that the vessel is operated within required safety margins. One surveillance capsule was removed after 30 years of operation, and is currently being tested. The data from these tests will be evaluated and incorporated into the database as required. Three additional surveillance capsules remain inside the reactor vessel. The program has been demonstrated to be effective in the past, and will continue to be effective throughout the 60-year license renewal period.

The transient data used in the RNP Reactor Vessel Surveillance Program has been collected since initial plant startup. The use of the program has been reviewed and approved by the NRC throughout this time. The current and most recently docketed projections have been included in the NRC's Reactor Vessel Integrity Database (RVID), as referenced in NRC Administrative Letter 95-03, Rev. 2.

Conclusion

The Reactor Vessel Surveillance Program differs from GALL Section XI.M31, Reactor Vessel Surveillance, with respect to:

- A short period of relatively low temperature operation of the RCS. However, this has been reviewed and accepted previously by the NRC, and the effects of the low temperature operation upon material property projections will be validated upon completion of testing and evaluation of Surveillance Capsule X, to be completed in 2002. Therefore, aging management concerns stemming from this occurrence will be managed, and this is not considered to be an exception.

Therefore, the Reactor Vessel Surveillance Program, with above described enhancement, is consistent with GALL Section XI.M31, Reactor Vessel Surveillance. Continued implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3.12 BURIED PIPING AND TANKS INSPECTION PROGRAM

The Buried Piping and Tanks Inspection Program is credited for aging management of selected components in systems at RNP.

The aging effects/mechanisms of concern are as follows:

- Loss of Material due to Crevice Corrosion
- Loss of Material due to General Corrosion
- Loss of Material due to MIC
- Loss of Material due to Pitting Corrosion
- Loss of Material due to Galvanic Corrosion

As a result of the license renewal review, the program elements associated with *Parameters Monitored/Inspected* and *Acceptance Criteria* for the Program will be enhanced to:

- Incorporate a requirement to ensure an appropriate as-found pipe coating and material condition inspection is performed whenever buried piping within the scope of this Program is exposed.
- Add precautions to ensure backfill with material that is free of gravel or other sharp or hard material that can damage the coating.
- Add a requirement that coating inspections be performed by qualified personnel to assess coating condition.
- Add a requirement that a coating engineer should assist in evaluation of any coating degradation noted during the Inspection.

Operating Experience

Leaks have occurred in the North Service Water header in pipe that was installed in 1984. The leaks were identified in July 1995 and in March and September 1998 and were repaired. Due to this adverse trend, a root cause evaluation was performed. The evaluation concluded:

- The environmental conditions found at the location of the North Service Water header are not especially harsh. The soil has high resistance, which restricts the current flow and consequent corrosion.
- The root cause of the March and September 1998 leaks was improper installation of the Tapecoat external wrapping. The root cause of the July 1995 leak was damage from misoperation of a backhoe during initial installation of the piping.
- Regarding similar situations/generic implications, other buried pipe on site has not exhibited exterior corrosion such as experienced on the North Service

Water header. The original service water piping has the same type of coating used in the North Service Water header but has not exhibited a similar tendency to leak. The reason for this is the assumption that the coating, when properly installed and not damaged, effectively prevents external degradation.

These conclusions demonstrate that the leaks can and have been detected on site and that appropriate corrective actions have been taken. Environmental conditions are not severe. If coating failures occur, there will be ample time to identify and repair leaks before catastrophic failure. Additionally, the number of leaks caused by external corrosion in buried pipe has been small and limited to service water piping.

Based on plant operating experience summarized above, scheduled, periodic excavations of buried piping for inspection are not warranted.

Conclusion

The Buried Piping and Tanks Inspection Program differs with the GALL Report, Section XI.M34, Buried Piping and Tanks Inspection with respect to the following exceptions. Both the exceptions and the affected program elements are listed.

- *Scope of Program:* The Program contains no buried tanks.
- *Scope of Program:* The Program includes additional components, i.e., underground Fuel Oil System piping, within the scope of the Buried Piping and Tanks Surveillance Program.
- *Scope of Program:* In addition to carbon steel components, buried cast iron piping and fittings were included in this program.
- *Scope of Program:* The Program includes galvanic corrosion as a potential aging mechanism.

These differences result in minimal impact on the ability of the Program to manage aging effects. Therefore, the Buried Piping and Tanks Inspection Program, with previously described enhancements, is considered to be consistent with GALL Section XI.M34, Buried Piping and Tanks Inspection, and implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3.13 ASME SECTION XI, SUBSECTION IWE PROGRAM

The ASME Section XI, Subsection IWE Program is credited for aging management of selected components in the Containment Building at RNP.

The aging effects/mechanisms of concern are as follows:

- Loss of Material due to General Corrosion
- Loss of Material due to Galvanic Corrosion
- Loss of Material due to Aggressive Chemical Attack
- Loss of Material due to Crevice Corrosion
- Loss of Material due to Pitting Corrosion
- Change in Material Properties due to Elevated Temperature
- Cracking due to Elevated Temperature
- Cracking due to Thermal Fatigue

As a result of the license renewal review, administrative controls associated with program element, *Confirmation Process*, for the program will be enhanced to (1) specify the requirements for conducting reexaminations, and (2) document that repairs meet the specified acceptance standards.

Operating Experience

The ASME Section XI, Subsection IWE Program is implemented and maintained in accordance with the general requirements for engineering programs. This provides assurance that the programs:

- Are effectively implemented to meet regulatory, process, and procedure requirements, including periodic reviews
- Qualified personnel are assigned as program managers, and are given authority and responsibility to implement the program
- Adequate resources are committed to program activities
- Are managed in accordance with plant administrative controls

Generic operating experience includes NUREG-1522, Assessment of Inservice Conditions of Safety Related Nuclear Plant. RNP was one of the six plants that were inspected in support of this document.

Plant-specific operating experience identifies many condition reports and engineering evaluations dealing with structures and components within the scope of ASME Section XI, Subsection IWE. Findings of note include:

- Corrosion was identified on the inside vertical face of the equipment hatch cylinder on the bottom of the cylinder. The insulation was removed at the low point of the equipment hatch, the equipment hatch inspected and recoated as necessary

- Based on the discovery of degraded protective insulation sheathing on the containment liner, administrative controls were upgraded to require a visual inspection of accessible interior and exterior surfaces of Containment Structures and components for evidence of deterioration.

- A steam generator blowdown penetration bellows failed due to a crack caused by TGSCC. Modifications were made eliminate the aging mechanism, including replacement of the penetration insulation with chloride free insulation.

- Localized bulging of the containment liner was identified during a walkdown. This was evaluated using criteria from the previous assessment of bulging that was discovered in 1974.

- Instances of containment liner corrosion have been identified. The affected areas have been evaluated and meet minimum wall thickness. Recent inspections identified the potential that boric acid leakage has penetrated the epoxy construction joint seal in the vicinity of the ECCS sump. The vertical liner plate at the external wall has been inspected as part of the IWE Program with some corrosion noted. However, it was concluded that the corrosion has not affected the intended function of the structure.

The RNP IWE Program is continually upgraded based on industry experience and research. This aging management program has provided effective means of ensuring the requirements of the Containment structure are met. In addition to industry experience, plant operating experiences are shared between CP&L sites through regular peer group meetings. Also, the corrective action program has been effective in ensuring that the IWE program is continually improving.

Conclusion

The ASME Section XI, Subsection IWE Program differs with the GALL Report, Section XI.S1, ASME Section XI, Subsection IWE, with respect to the following exceptions. Both the exceptions and the affected program elements are listed:

- *Scope of Program:* The GALL Report identifies ASME Section XI, Subsection IWE and the 10CFR50 Appendix J Programs for managing aging effects on the containment pressure boundary components. In addition to the AMPs recommended in the GALL Report, RNP has identified the One-Time

Inspection Program for inspecting inaccessible (due to permanent structures or ALARA concerns) portions of the containment liner, and the moisture barrier inside the containment at the liner plate/floor concrete interface.

- *Parameters Monitored/Inspected:* The RNP AMR methodology identified additional aging mechanisms not identified in the GALL Report, e.g., aggressive chemical attack for the containment liner plate and galvanic and general corrosion for penetration bellows. The AMPs used to manage these additional aging mechanisms are consistent with the GALL Report.
- *Parameters Monitored/Inspected:* GALL identifies SCC as a potential aging mechanism for the penetration sleeve and bellows. However, the RNP AMR methodology did not identify this mechanism, and resulting cracking, because the environmental stressors required to initiate cracking from SCC are not present at RNP.

These exceptions have been evaluated and would result in no adverse effects on the ability of the program to manage aging effects.

Based on the above discussion of differences, the ASME Section XI, Subsection IWE Program, with the enhancements identified above, is considered to be consistent with GALL Section XI.S1, ASME Section XI, Subsection IWE, and implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

In the evaluation of the ASME Section XI, Subsection IWE Program against the program elements of the GALL Report, exceptions to Code requirements that have been granted by approved relief requests were not considered to be exceptions to the GALL criteria.

B.3.14 ASME SECTION XI, SUBSECTION IWL PROGRAM

The ASME Section XI, Subsection IWL Program is credited for aging management of selected components in the Reactor Containment Building at RNP.

The aging effects/mechanisms of concern are as follows:

- Change in Material Properties due to Aggressive Chemical Attack
- Loss of Material due to Aggressive Chemical Attack
- Loss of Material due to Corrosion of Embedded Steel
- Change in Material Properties due to Fatigue
- Cracking due to Fatigue

As a result of the license renewal review, administrative controls associated with program element, *Scope of Program*, will be enhanced to notify Civil/Structural Design Engineering of the location and extent of proposed excavations and to require Civil/Structural Design Engineering to examine representative samples of below-grade concrete when excavated for any reason.

Operating Experience

The ASME Section XI, Subsection IWL Program is implemented and maintained in accordance with the general requirements for engineering programs. This provides assurance that the programs:

- Are effectively implemented to meet regulatory, process, and procedure requirements, including periodic reviews
- Qualified personnel are assigned as program managers, and are given authority and responsibility to implement the program
- Adequate resources are commited to program activities
- Are managed in accordance with plant administrative controls

Generic operating experience includes NUREG-1522, Assessment of Inservice Conditions of Safety Related Nuclear Plant. RNP was one of the six plants that were inspected in support of this document.

Plant-specific operating experience identifies many condition reports and engineering evaluations dealing with structures and components within the scope of ASME Section XI, Subsection IWL. Findings of note include:

- Degraded conditions in the containment structure concrete components for the North and South Cable Vault Rooms were evaluated. Staining, cracking, exposed aggregate, and spalling were identified. The indications were characterized as minor, and no signs of corrosion in the cracks were noted.

The spalling was from original construction. No active degradation was noted and the structural integrity of the containment structure was not affected.

- An evaluation was performed to address installing a cementious grout to fill the radial floor construction joint(s) and the construction joint(s) at the base of the crane wall in the area of the ECCS sump. This will replace the original filler material and the watertight epoxy material. This evaluates and allows the installation of cementious grout at the exterior wall to floor joint to replace the PVC foam material. Note, this is related to the concrete cover slab, not the concrete for the basemat or cylinder wall.

- An engineering evaluation was used to document that a repaired area of the concrete exterior wall met design requirements even though it did not meet the specified concrete strength. The area was located between elevations 226 and 232 and repaired using 5 Star Structural Grout. The repaired area was on the southwest side of containment between the equipment hatch and CV Access area. The degradation consisted of several areas of spalling at construction joints that had apparently been repaired during original construction. The particular depth of these areas did not extend beyond the centerline of the reinforcing steel. This type of spalling is not atypical of concrete construction and does not typically affect the structural integrity of the concrete structure. This repaired area has not exhibited any additional spalling since the 1992 repair and has not experienced any accelerated degradation and aging.

- An evaluation concluded that not providing cooling to the penetrations with hot piping does not degrade the concrete. Degradation has not occurred and does not require augmented examinations.

- Inspection of the exterior surface of the containment dome concrete revealed accelerated degradation of the grout covering in December of 1984. A work package was issued to repair this degradation and specified that an elastomeric coating system be applied after the grout was stripped and resurfaced. The evaluation indicated that grout deterioration did not affect the structural integrity of the associated concrete.

The RNP IWL Program is continually upgraded based on industry experience and research. This aging management program has provided effective means of ensuring the requirements of the Containment structure are met. In addition to industry experience, operating experiences are shared between CP&L sites through regular peer group meetings. The corrective action program has been effective in ensuring that the IWL program is continually improving.

Conclusion

The ASME Section XI, Subsection IWL Program differs with the GALL Report, Section XI.S2, ASME Section XI, Subsection IWL, with respect to the following exceptions. Both the exceptions and the affected program elements are listed:

- *Parameters Monitored/Inspected:* The RNP AMR methodology did not identify the aging effects of cracking and loss of bond due to corrosion of embedded steel. However, the RNP methodology identified loss of material due to the aging mechanism of corrosion of embedded steel and applies the ASME Section XI, Subsection IWL Program. The RNP approach would also require examination of concrete for cracking and loss of bond adequately.
- *Scope of Program and Parameters Monitored/Inspected:* The requirements of ASME Section XI, Subsection IWL, do not apply to the RNP prestressing system. The plant design includes a grouted tendon system, which is outside the scope of Subsection IWL. Therefore, aging management activities associated with ungrouted tendons are not applicable.
- *Parameters Monitored/Inspected:* RNP aging effects/mechanisms include cracking of concrete and change in material properties of concrete due to fatigue at penetration anchors. These are not addressed in the GALL. The ASME Section XI, Subsection IWL Program is applicable to these additional aging effects/mechanisms.
- *Parameters Monitored/Inspected:* Erosion of porous concrete subfoundation is not an applicable aging mechanism since porous concrete was not used at RNP under the containment building.
- *Parameters Monitored/Inspected:* GALL identifies “Increase in porosity, permeability” as aging effects for concrete in Section II.A1. The RNP effect of “change in material properties” includes the mechanisms identified in GALL.

These exceptions have been evaluated and would result in no adverse effects on the ability of the program to manage aging effects.

Based on the above discussion, the ASME Section XI, Subsection IWL Program, with the enhancements and exceptions identified above, is considered to be consistent with GALL Section XI.S2, ASME Section XI, Subsection IWL, and implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

In the evaluation of the ASME Section XI, Subsection IWL Program against the program elements of the GALL Report, exceptions to Code requirements that have been granted by approved relief requests were not considered to be exceptions to the GALL criteria.

B.3.15 STRUCTURES MONITORING PROGRAM

The Structures Monitoring Program is credited for aging management of civil structures and components within the scope of license renewal at RNP. The aging effects/mechanisms of concern are as follows:

<u>Steel aging effects/mechanisms:</u>

- Loss of Material due to General Corrosion
- Loss of Material due to Crevice Corrosion
- Loss of Material due to Pitting Corrosion

<u>Concrete (below-grade) aging effects/mechanisms</u>:

- Loss of Material due to Aggressive Chemical Attack
- Loss of Material due to Corrosion of Embedded Steel
- Change in Material Properties due to Aggressive Chemical Attack

<u>Elastomer aging effects/mechanisms</u>:

- Change in Material Properties due to Elevated Temperature
- Cracking due to Elevated Temperature

As a result of the license renewal review, administrative controls associated with program elements, *Scope of Program, Parameters Monitored/Inspected, Detection of Aging Effects, Acceptance Criteria,* and *Corrective Actions,* will be enhanced to:

- Include buildings and structures, and associated acceptance criteria, in scope for license renewal but outside the scope of the Maintenance Rule. (Structures addressed in the Maintenance Rule already are in the Program.)

- Identify interfaces between structures monitoring inspections of concrete surfaces and the Fire Protection Program requirements for barriers.

- State clearly the boundary definition between systems and structures. The physical structure is inspected as part of the structure/building walkdown and includes the concrete structure and all structural steel such as: main building structural steel, platform support steel, stairways, etc.

- Revise administrative controls to provide inspection criteria for portions of systems covered by structures monitoring. Provide acceptance categories similar to those used for Structures Monitoring, and require a condition report be initiated for all inspection attributes found to be unacceptable.

- Expand system walkdown inspection criteria to include observation of selected, adjacent components.

- Revise personnel responsibilities to include responsibilities to (1) provide assistance in evaluating structural deficiencies when requested by the Responsible Engineer, (2) inspect excavated concrete, and (3) notify Civil/Structural Design Engineering of location and extent of proposed excavations.

Operating Experience

The Structures Monitoring Program is a combination of the CP&L Corporate procedure for Condition Monitoring of Structures and the RNP plant procedure for system walkdown. The procedures were developed to support implementation of the Maintenance Rule. The primary function of Maintenance Rule is to provide reasonable assurance that structures, systems, trains, and components are capable of fulfilling their intended safety significant functions. The subject administrative controls have been proven effective for implementing the Maintenance Rule and are supported by the excellent operating experience for systems, structures and components. The Structures Monitoring Program relies on these existing programs with above enhancements, to manage aging effects.

The Structures Monitoring Program incorporates best practices recommended by the Institute of Nuclear Power Operations (INPO) and inspection guidance based on industry experience and recommendations from ACI and ASCE.

A review of condition reports and inspections performed has concluded that administrative controls are in effect and effective in identifying age related degradation, implementing appropriate corrective actions, and continually upgrading the administrative controls used for structural monitoring.

Conclusion

The Structures Monitoring Program, with the enhancements identified above, is consistent with GALL Section XI.S6, Structures Monitoring Program, and implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3.16 DAM INSPECTION PROGRAM

The Dam Inspection Program is credited for aging management of selected components for the Lake Robinson Dam.

The aging effects/mechanisms of concern are as follows:

Steel Structures Aging Effects/Mechanisms:

- Loss of Material due to Crevice Corrosion
- Loss of Material due to General Corrosion
- Loss of Material due to Pitting Corrosion
- Loss of Material due to MIC

Earthen Structures Aging Effects/Mechanisms:

- Loss of Form due to Settlement

Concrete Structures Aging Effects/Mechanisms:

- Loss of Material due to Aggressive Chemical Attack
- Loss of Material due to Corrosion of Embedded Steel
- Change in Material Properties due to Aggressive Chemical Attack

Evaluation

The following summarizes the results of an evaluation of the Program against the 10 program elements identified in Appendix A of the SRP-LR.

Scope of Program

FERC / US Army Corp of Engineers program, “Recommended Guidelines for Safety Inspection of Dams,” is credited by RNP for the aging management of the Lake Robinson earthen dam and associated concrete structures. The FERC / US Army Corp of Engineers program is one of the acceptable alternatives for managing aging effects for water control structures documented in the GALL Report, Section III.A6. The FERC/ US Army Corp of Engineers program is applicable to aging management only for the Lake Robinson Dam and associated structures.

Preventive Actions

The Dam Inspection Program is a condition monitoring program; therefore, preventive actions are not required.

Parameters Monitored/Inspected

Parameters monitored are addressed in detail under Appendix II of “Recommended Guidelines for Safety Inspection of Dams.” They include inspection of concrete structures, embankments, spillways, outlet works (gates, channels, sluices, etc.). The Recommended Guidelines for Safety Inspection of Dams adequately address the aging effects/mechanisms discussed in the GALL Report. The extent of aging effects is not quantified in the Program; the experience and judgment of qualified inspectors is relied upon for that determination. Provisions are included within the program for a more intrusive inspection if determined necessary by the inspector. An enhancement to the program will be made to ensure administrative controls will include “Recommended Guidelines for Safety Inspections of Dams” as the inspection guidance for RNP.

Detection of Aging Effects

The method of identifying aging effects is based on an independent inspection using the “Recommended Guidelines for Safety Inspection of Dams.”

The purpose of the independent inspection is to identify, within the limitations of visual field inspection and office review of available data, records and operating history, any actual or potential deficiencies, whether in the condition of the project works, the quality and adequacy of project maintenance, surveillance, or in the methods of operation, that might endanger public safety. The independent dam safety inspections are conducted at five year intervals.

Based on this inspection method and frequency, detection of aging effects will occur before there is a loss of structure or component intended function(s).

Monitoring and Trending

The Recommended Guidelines for Safety Inspection of Dams, Phase I, Appendix I, investigation report instructs the user to review the “history of previous failures or deficiencies and pending remedial measures for correcting known deficiencies and the schedule for accomplishing remedial measures should be indicated”. Additionally, review of inspection history, including the results of the last safety inspection is recommended. Based on the meticulous requirements of the Phase I inspection and the documented history of independent inspections; monitoring and trending will provide predictability of the extent of degradation, and timely corrective or mitigative actions.

Acceptance Criteria

Acceptance criteria for the inspection and monitoring of Lake Robinson Dam are in accordance with the requirements of the “Recommended Guidelines for Safety Inspection of Dams.” As such, the acceptance criteria will ensure the structure or component intended function(s) are maintained under CLB design conditions during the period of extended operation.

Corrective Actions

The inspection guidelines implement a two phased approach, Phase I performs a detailed field inspection. The Phase II investigation is supplementary to Phase I and is conducted when the results of the Phase I investigation indicate the need for additional in-depth studies, investigations or analyses. The Program will be enhanced by revising the plant system monitoring procedure to identify the “Recommended Guidelines for Safety Inspection of Dams” as the required management program and requiring the responsible system engineer to review the inspection report and initiate corrective actions for any unacceptable attributes identified during the inspection process.

Confirmation Process

“Recommended Guidelines for Safety Inspection of Dams,“ is a Corps of Engineers document and not subject to RNP quality assurance procedures; however, activities initiated in accordance with the system monitoring procedure, such as corrective actions, are subject to quality assurance program controls. Thus, the effectiveness of this AMP will be monitored using corrective action program procedures, review and approval processes, and administrative controls which are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. The Program will be enhanced by revising the plant system monitoring procedure as described in “Corrective Actions” above.

Administrative Controls

The system monitoring procedure is subject to RNP corrective action and quality assurance procedures, review and approval processes, and administrative controls. These are implemented in accordance with the requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of extended operation. The Program will be enhanced by revising the administrative controls for plant system monitoring as described in “Corrective Actions” above.

Operating Experience

The Lake Robinson Dam is an RNP Unit 1 (Fossil Plant) structure and therefore not typically subject to the corrective action process required for nuclear plant systems and structures. As such, the normal source of operating experience is not populated. However, five dam inspection reports, dating back to 1980 on five-year intervals from 2000, have been reviewed. In addition, a sample of Unit 1 visual inspection reports, yearly South Carolina dam inspections, and a year 2000 underwater visual inspection report for the spillway were reviewed. Recommendations are made in each report and photographs taken of typical areas and areas of concern. No significant issues have been identified; however, recommended maintenance activities have been performed as evidenced by succeeding inspection reports.

Conclusion

The Dam Inspection Program, with enhancements described above, is consistent with the 10 program elements identified in Appendix A of the SRP-LR. The program provides reasonable assurance that the aging effects will be managed such that Lake Robinson Dam 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.

B.3.17 SYSTEMS MONITORING PROGRAM

The Systems Monitoring Program is credited for aging management of selected components in the various plant systems at RNP.

The aging effects/mechanisms of concern are as follows:

- Loss of Material due to General Corrosion
- Loss of Material due to Crevice Corrosion
- Loss of Material due to Pitting Corrosion
- Loss of Material due to Galvanic Corrosion
- Loss of Material due to MIC
- Cracking due to SCC
- Change in Material Properties due to Elevated Temperatures
- Cracking due to Elevated Temperatures
- Loss of Heat Transfer Effectiveness due to Fouling of Heat Transfer Surfaces
- Change in Material Properties due to Irradiation Embrittlement
- Cracking due to Irradiation Embrittlement
- Loss of Material due to Aggressive Chemical Attack
- Loss of Mechanical Closure Integrity due to Loss of Material due to Aggressive Chemical Attack

Evaluation

The following summarizes the results of an evaluation of the Program against the 10 program elements identified in Appendix A of the SRP-LR.

Scope of Program

The System Monitoring Program is based on scheduled system walkdowns, system health reports, and performance monitoring of systems. This is done through administrative controls that implement the walkdowns, establish condition monitoring requirements and implement system and component trending and system libraries. The Program is conducted as part of the responsibilities of Systems Engineering. Current walkdown scope includes all maintenance rule systems and additional systems that encompass the License Renewal systems.

Preventive Actions

The Systems Monitoring Program is a condition monitoring program; thus, there is no preventive action.

Parameters Monitored/Inspected

The current systems monitoring procedures do not specifically describe aging effects identified in aging management reviews. Therefore, administrative controls for the Program will be enhanced to:

- Include aging effects identified in the aging management reviews,
- Identify inspection criteria in checklist form,
- Include guidance for inspecting connected piping/components,
- Require documenting identified degradation and initiating appropriate corrective action(s), and
- Add a section specifically addressing corrective actions.

Detection of Aging Effects

The Systems Monitoring Program relies on visual inspection of SSCs during system walkdowns to detect and qualify degradations. Degradations deemed to be “unacceptable” will have a condition report initiated and will be handled under the Corrective Action Program. Thus, the Systems Monitoring Program is designed to detect degradation not solely on detecting failure, but rather, on detection of aging effects prior to structure or component failure.

Accessible portions of maintenance rule and LR systems are walked down at least once per quarter. Walkdowns typically are scheduled and performed so the entire system is fully walked down within one operating cycle.

Monitoring and Trending

Administrative controls provide instructions for monitoring systems to permit early detection of degradation. Data from walkdowns is trended and evaluated to identify and correct problems. As stated earlier, the Systems Monitoring Program will include enhancements to ensure aging indicators are quantified so that trending can be done effectively.

Acceptance Criteria

The program administrative controls will be enhanced to include visual monitoring acceptance criteria: (ACC - Acceptable, AWD - Acceptable with Deficiencies, or UNA – Unacceptable) and guidelines for applying these criteria.

Corrective Actions

The program administrative controls will be enhanced to address corrective actions and to initiate a condition report for unacceptable degradation.

Corrective actions including root cause determinations and prevention of recurrence are performed in accordance with the Corrective Action Program. Timeliness of corrective action is monitored and is commensurate with the level of significance of the activity.

Confirmation Process

Effectiveness of this Program (as enhanced) will be monitored using site quality assurance (QA) procedures, review and approval processes, and administrative controls which are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B.

Administrative Controls

RNP QA procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of extended operation.

Operating Experience

Processes at RNP are continually being upgraded based upon industry experience and research via the ongoing Operating Experience Program. The processes that comprise this aging management program have provided effective means of ensuring the system health for the systems subject to periodic walkdown.

A review of condition reports and corrective actions has concluded that the Systems Engineering including its management and administrative controls, has been the subject of continuing assessment and improvement.

Conclusion

The Systems Monitoring Program, with the enhancements identified above, is consistent with the 10 program elements identified in Appendix A of the SRP-LR. Implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3.18 PREVENTIVE MAINTENANCE PROGRAM

The Preventive Maintenance Program is credited for aging management of selected components in the various plant systems at RNP.

The aging effects/mechanisms of concern are as follows:

- Change in Material Properties due to Elevated Temperature
- Change in Material Properties due to Irradiation Embrittlement
- Change in Material Properties due to Ultraviolet Radiation and Ozone Exposure
- Change in Material Properties due to Various Degradation Mechanisms
- Cracking due to Elevated Temperature
- Cracking due to Irradiation Embrittlement
- Cracking due to SCC
- Cracking due to Ultraviolet Radiation and Ozone Exposure
- Cracking due to Various Degradation Mechanisms
- Loss of Material due to Crevice Corrosion
- Loss of Material due to Erosion
- Loss of Material due to FAC
- Loss of Material due to Galvanic Corrosion
- Loss of Material due to General Corrosion
- Loss of Material due to MIC
- Loss of Material due to Pitting Corrosion
- Loss of Material due to Various Degradation Mechanisms
- Loss of Pre-load due to Stress Relaxation
- Reduced Insulation Resistance (IR) due to Thermal Embrittlement
- Loss of Material due to Aggressive Chemical Attack
- Loss of Heat Transfer due to Fouling of Heat Transfer Surfaces

Evaluation

The following summarizes the results of an evaluation of the Program against the 10 program elements identified in Appendix A of the SRP-LR.

<u>Scope of Program</u>

The PM Program assures that various aging effects are managed for a wide range of components. Although not part of the site maintenance procedures, the PM Program activities include activities performed during operations rounds and periodic operation of equipment, such as monitoring filter differential pressures and purging water from air receivers. This element will be enhanced to incorporate specific aging management activities identified in the aging management reviews into the program.

Preventive Actions

The PM Program includes periodic refurbishment or replacement of components, which could be considered to be preventive or mitigative actions. The inspections and testing activities used to identify component aging degradation effects do not constitute preventive actions in the context of this element. However, they are consistent with a monitoring approach to aging management.

Parameters Monitored/Inspected

The administrative controls that govern the PM Program provide instructions for monitoring structures, systems, and components to permit early detection of degradation. Inspection and testing activities monitor various parameters including surface condition, loss of material, presence of corrosion products, and signs of cracking. The current guidelines in operations, maintenance and surveillance test procedures and Model Work Orders may not specifically describe the aging effects applicable to LR. These documents will be enhanced, as necessary, to include aging effects/mechanisms identified in the aging management reviews into the PM Program.

Detection of Aging Effects

PM activities provide for periodic component replacement, inspections and testing to detect aging effects and mechanisms.

The extent and schedule of the inspections and testing assures detection of component degradation prior to the loss of their intended functions. Established techniques such as visual inspections are used. The PM Program controlling documents promote activities that are aimed at PM optimization and continual improvement. This includes evaluation of frequency and appropriateness of PMs to assess the effectiveness and to compare with typical industry practices.

Monitoring and Trending

PM activities provide for monitoring and trending of aging degradation. Inspection intervals are established such that they provide for timely detection of component degradation. Inspection intervals are dependent on the component material and environment and take into consideration industry and plant-specific operating experience and manufacturers recommendations. The PM Program administrative controls reference activities for monitoring structures, systems, and components to permit early detection of degradation. Data from walkdowns are trended and evaluated to identify and correct problems. An example of technique and parameters monitored and trended is visual examinations for coating failures, corrosion, cracking, erosion, leaking and physical condition, mechanical damage, loose or missing hardware, etc. As part of the conduct of maintenance at RNP, emphasis is placed on the responsibility of all station personnel to report equipment deficiencies on a Maintenance Work Request/Work Order or via the corrective action program. As stated above, the Preventive Maintenance Program ensures aging indicators are trended.

Acceptance Criteria

PM Program acceptance criteria are defined in the specific inspection and testing procedures. They confirm component integrity by verifying the absence of the aging effect or by comparing applicable parameters to limits based on the applicable intended function(s) as established by the plant design basis.

Degradations deemed to be unacceptable will have a condition report initiated and will be handled under the corrective action program.

Corrective Actions

Identified deviations are evaluated within the corrective action process, which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation. The corrective action process is in accordance with 10 CFR 50 Appendix B.

Confirmation Process

The corrective action process is in accordance with 10 CFR 50 Appendix B and includes (1) reviews to assure that proposed actions are adequate, (2) tracking and reporting of open corrective actions, and (3) for root cause determinations, reviews of corrective action effectiveness.

Administrative Controls

The review and approval process for procedures and procedure changes and the identification of maintenance activities that require maintenance procedures or instructions are described in formal plant administrative controls.

RNP QA procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of extended operation.

Operating Experience

Periodic surveillance and preventive maintenance activities have been in place at RNP Unit 2 since the plant began operation. These activities have proven effective at maintaining the material condition of systems, structures, and components and detecting unsatisfactory conditions. There is a demonstrated history of detecting damaged and degraded components and causing their repair or replacement in accordance with the site corrective action process.

The Preventive Maintenance and Surveillance Test Administration procedure describes the process used to ensure continuous improvement in the implementation of its governed processes. These activities, along with the requirements for self-assessments and periodic assessments, provide reasonable assurance that the Preventive Maintenance Program will continue to perform in an effective manner.

Conclusion

The Preventive Maintenance Program, with the enhancements identified above, is consistent with the 10 program elements identified in Appendix A of the SRP-LR. Implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3.19 METAL FATIGUE OF REACTOR COOLANT PRESSURE BOUNDARY (FATIGUE MONITORING PROGRAM)

The Fatigue Monitoring Program is credited for aging management of selected components in various NSSS and secondary systems at RNP.

The aging effect/mechanism of concern is cracking due to thermal fatigue.

As a result of the license renewal review, the program element, *Preventive Actions*, will be enhanced to reduce the plant load/unload transient limit to provide the margin needed for consideration of reactor water environmental effects.

Operating Experience

Since the original design of the RNP, fatigue failures at other plants worldwide led to the discovery of several additional thermal transients not originally considered in the design. This includes thermal stratification of the pressurizer surge line, identified by NRC Bulletin 88-08, and insurge/outsurge transients associated with operation of the pressurizer, identified by NRC Bulletin 88-11. Fatigue analyses were prepared to account for additional thermal transients associated with each of these issues. The analyses demonstrated compliance with ASME Section III fatigue requirements. More recently, cracking of unisolable branch lines attached to the RCS system has occurred due to thermal stratification and striping. Industry guidelines have been developed to determine susceptibility to cracking of these lines, and the RNP design has been evaluated against them. No susceptibility was identified. Industry experience was used in selecting the NUREG/CR-6260 locations evaluated for environmental fatigue, as well as the pressurizer locations evaluated for environmental fatigue.

Conclusion

The Fatigue Monitoring Program, with the enhancement identified above, is consistent with GALL Section X.M1, Fatigue Monitoring, with one exception. The exception involves the pressurizer surge line and impacts program elements for *Preventive Actions, Acceptance Criteria, and Corrective Actions*. The pressurizer surge line was not shown to have an environmentally-adjusted Cumulative Usage Factor (CUF) less than 1.0. Fatigue effects will be managed by periodic examinations in accordance with the ASME Section XI, Subsections IWB, IWC, and IWD Program. If unacceptable indications are identified, they will be evaluated for continued service or the component will be repaired or replaced in accordance with ASME Section XI Program.

Implementation of the Fatigue Monitoring Program, as described above, provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.4 NEW AGING MANAGEMENT PROGRAMS

B.4.1 NICKEL-ALLOY NOZZLES AND PENETRATIONS PROGRAM

The Nickel-Alloy Nozzles And Penetrations Program is credited for aging management of selected components in the Reactor Vessel and Internals System at RNP.

The aging effect/mechanism of concern is cracking due to SCC (includes PWSCC).

As a result of the license renewal review, the Nickel-Alloy Nozzles And Penetrations Program will incorporate the following enhancements that affect program elements for *Scope of Program, Acceptance Criteria,* and *Corrective Actions*.

- RNP will maintain its involvement in industry initiatives (such as the Westinghouse Owners Group and the EPRI Materials Reliability Project) during the period of extended operation.
- RNP will perform evaluation of indications under the ASME Section XI program.
- RNP will perform corrective actions for augmented inspections using repair and replacement procedures equivalent to those requirements in ASME Section XI.

Operating Experience

The Nickel-Alloy Nozzles And Penetrations Program is a new program with little operating experience history. The GALL report is based on industry operating experience through April 2001. Recent industry operating experience (2001) has been reviewed for applicability. Subsequent operating experience will be captured through the normal operating experience review process.

Recent events, as documented in NRC Bulletin 2001-01, have engendered heightened scrutiny of this issue beyond those recognized in Generic Letter 97-01. The RNP position regarding Bulletin 2001-01 is evolving based on correspondence with the NRC and industry research. Since this issue requires resolution during the initial licensing period, RNP will commit to continuing this resolution through the period of extended operation and will participate in industry initiatives (Westinghouse Owners Group and the EPRI Materials Reliability Program) to ensure that the components managed are maintained within the CLB during the period of extended operation.

Conclusion

The Nickel-Alloy Nozzles And Penetrations Program is consistent with GALL Section XI.M11, Nickel-Alloy Nozzles And Penetrations. Implementation of the program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.4.2 THERMAL AGING EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL (CASS) PROGRAM

The Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program is credited for aging management of CASS components within Class 1 boundaries of the Reactor Coolant System and connected systems at RNP.

The aging effect/mechanism of concern is loss of fracture toughness due to thermal embrittlement of cast austenitic stainless steel.

Operating Experience

The GALL Program description in Section XI.M12 notes that the program is based on research data using laboratory-aged and service-aged materials, and concludes that the program as defined is sufficient to manage the effects of thermal aging embrittlement on the intended function of CASS components. Flaw tolerance evaluations are based on an extensive test program performed by the Argonne National Laboratory (ANL) assessing the extent of thermal aging of CASS materials. ANL compiled an extensive database of compositions of CASS materials exposed to a temperature range of 550 – 750°F for up to 58,000 hours, and used this data to estimate the extent of thermal aging in developing fracture toughness determination procedures. The results of this study have been reviewed and approved by the NRC, and incorporated into plant specific analysis of Reactor Coolant System piping and Reactor Coolant Pump casings.

The ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD Program and procedures are generally credited with implementation of the Thermal Aging Embrittlement Program.

Conclusion

The Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program is consistent with GALL Section XI.M12, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS). Implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.4.3 PWR VESSEL INTERNALS PROGRAM

The PWR Vessel Internals Program is credited for aging management of selected components of the RNP Reactor Vessel and Internals.

The aging effects/mechanisms of concern are as follows:

- Cracking due to SCC
- Cracking due to IASCC
- Change in Dimensions due to Void Swelling
- Loss of Pre-load due to Irradiation Creep
- Loss of Pre-load due to Stress Relaxation
- Reduction of Fracture Toughness due to Thermal Embrittlement
- Reduction of Fracture Toughness due to Neutron Irradiation Embrittlement

The aging effect/mechanism cracking due to SCC is from the AMR for the Reactor Vessel. All other aging effect mechanisms are from the AMR for the Reactor Internals, which was based on a GTR.

The PWR Vessel Internals Program will incorporate the following enhancements that impact program elements for *Scope of Program* and *Corrective Actions.*

- To address change in dimensions due to void swelling, RNP will continue to participate in industry programs to investigate this aging effect and determine the appropriate AMP.
- To address baffle and former assembly issues, RNP will continue to participate in industry programs and will implement appropriate program enhancements to manage the aging effects associated with the Baffle and Former Assembly.
- As Westinghouse Owner’s Group (WOG) and EPRI Materials Reliability Project (MRP) research projects are completed, RNP will evaluate the results and factor them into the PWR Vessel Internals Program. The expected results include identification of components which are the most limiting and most susceptible and identification of appropriate inspection techniques.
- RNP will implement an augmented inspection during the license renewal term. Augmented inspections, based on required program enhancements, will become part of the ASME Section XI program. Corrective actions for augmented inspections will be developed using repair and replacement procedures equivalent to those requirements in ASME Section XI.

Operating Experience

The PWR Vessel Internals Program is a new program with little operating experience history. The program has been evaluated against the GALL Report, which incorporates industry operating experience through April 2001. Recent industry operating experience (2001) has been reviewed for applicability. Subsequent operating experience will be captured through the normal operating experience review process.

The two NRC Information Notices discussed in the GALL Report are dispositioned as follows:

- NRC Information Notice (IN) 84-18 dealt with two items, a) unacceptable levels of contaminants in purchased boric acid and b) collection of airborne contaminants on the free surface of the spent fuel pool. The first item was reviewed, and it was determined that the “pathway as described in IN 84-18 is not considered likely at H. B. Robinson Unit Two.” The second item was dispositioned similarly.
- NRC Information Notice 98-11 was reviewed and the following actions were proposed: 1) Continue active participation in the WOG materials subcommittee, 2) Confirm need for leak-before-break analyses, and 3) Update the Strategic Issue Action Plan for RPV internals to address the current issue status.

Conclusion

The PWR Vessel Internals Program is consistent with GALL Section XI.M16, PWR Vessel Internals, with the following exceptions. Both the exceptions and the affected program elements are listed:

- *Preventive Actions:* The PWR Vessel Internals Program relies on the Water Chemistry Program for maintaining high water purity to reduce susceptibility to cracking due to SCC. This program has been evaluated and was found to be consistent with GALL with exceptions that have no adverse effects on the ability of the program to manage aging effects. As stated in the description of the Water Chemistry Program, the differences from the GALL chemistry program were evaluated and determined not to be exceptions.
- *Parameters Monitored/Inspected* and *Detection of Aging Effects:* Augmented inspections will be performed based on the results of RNP’s participation in industry research. The GALL recommends that the program monitor the effects of cracking on the intended function of the component by detection and sizing of cracks by augmentation of ISI in accordance with the requirements of the ASME Code, Section XI, Table IWB 2500-1. The determination of consistency cannot be made at this time so this element is considered inconsistent.

Based on the above, implementation of the PWR Vessel Internals Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.4.4 ONE-TIME INSPECTION PROGRAM

The One-Time Inspection Program is credited for aging management of various structures/components at RNP as shown below:

Structure/Component	Building Structure/ System	Aging Effect/Mechanisms of Concern
CCW Heat Exchanger Tube Inspections	Component Cooling Water	Loss of Material due to erosion
Miscellaneous Piping Inspection to Demonstrate Water Chemistry Effectiveness	Spent Fuel, Steam Turbine System, Feedwater System, Condensate System, Steam Generator Blowdown System, Auxiliary Feedwater System	Cracking, Loss of Material
Small Bore RCS Class 1 Piping and Components	Reactor Coolant System and Connected Systems	Cracking
Diesel Generator Engine Exhaust Silencers	Diesel Generator System	Loss of Material due to Crevice Corrosion, General Corrosion, and Pitting Corrosion
Anchorage/Embedments – Exposed Surfaces	Reactor Containment	Loss of Material due to Aggressive Chemical Attack and General Corrosion.
Reactor Vessel Support (Structure)	Reactor Containment	Loss of Material due to Aggressive Chemical Attack and General Corrosion.
Slide Bearing Plates (Reactor Vessel Support)	Reactor Containment	Loss of Material due to Aggressive Chemical Attack and General Corrosion.
Moisture Barrier	Reactor Containment	Change in Material Properties and Cracking due to Elevated Temperature
Containment Liner Plate	Reactor Containment	Loss of Material due to Aggressive Chemical Attack, Crevice Corrosion, General Corrosion, and Pitting Corrosion.

The RNP One-time Inspection Program has been created to verify the effectiveness of existing programs as well as providing additional assurance that aging is not occurring or the evidence of aging is so insignificant that an aging management is not required for the license renewal period.

Operating Experience

The RNP One-Time Inspection Program is a new program and will be maintained in accordance with the general requirements for engineering programs. This provides assurance that the program:

- Is effectively implemented to meet regulatory, process, and procedure requirements, including periodic reviews.
- A qualified person is assigned as program manager and given the authority and responsibility to implement the program.
- Adequate resources are committed to program activities
- Are managed in accordance with plant administrative controls

The Program will invoke the corrective action program (CAP) when unacceptable results are identified. The CAP has been effective in ensuring that plant programs are continually improving.

Conclusion

The One-Time Inspection Program is consistent with GALL Section XI.M32, One-time Inspection. Implementation of the program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.4.5 SELECTIVE LEACHING OF MATERIALS PROGRAM

The Selective Leaching of Materials Program is credited for aging management of components susceptible to selective leaching in various systems at RNP.

The aging effect/mechanism of concern is loss of material due to selective leaching.

The Selective Leaching of Materials Program is a new program that involves a one-time inspection and mechanical test to be applied at RNP.

Operating Experience

The program has been evaluated against the GALL Report, which incorporates industry operating experience through April 2001. Recent industry operating experience (2001) has been reviewed for applicability. Subsequent operating experience will be captured through the normal operating experience review process. Since this program is new, there is no plant operating experience to evaluate. Nevertheless, a Condition Report search was performed and revealed no plant-specific concerns regarding selective leaching at RNP.

Conclusion

The Selective Leaching of Materials Program is consistent with GALL Section XI.M33, Selective Leaching of Materials, with one exception that affects the program elements for *Scope of Program, Preventive Actions, Parameters Monitored/Inspected, Detection of Aging Effects,* and *Monitoring and Trending.* The exception involves the use of mechanical means, other than Brinell hardness testing identified in the GALL Report, to identify the presence of selective leaching of material. The exception is justified, because (1) hardness testing cannot be reliably performed for most components due to form and configuration and (2) other mechanical means, i.e., resonance when struck by another object, scraping, or chipping, provide an equally valid method of identification. Implementation of the Program provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.4.6 NON-EQ INSULATED CABLES AND CONNECTIONS PROGRAM

The Non-EQ Insulated Cables and Connections Program is credited for aging management of cables and connections not included in the RNP EQ Program.

The aging effects/mechanisms of concern are as follows:

- Reduced Insulation Resistance
- Electrical Failure

The technical basis for selecting a sample of cables to be inspected will be defined prior to the period of extended operation. The sample locations will consider the location of PVC cables inside and outside containment as well as any known adverse localized environments. (PVC was determined to be the limiting insulation material.)

Operating Experience

The Non-EQ Insulated Cables and Connections Program is a new program with no operating experience history. However, as noted in the GALL Report, industry operating experience has shown that adverse localized environments caused by heat or radiation for electrical cables and connections have been shown to exist and have been found to produce degradation of insulating materials that is visually observable.

Conclusion

Upon defining the technical basis for the sample of cables to be inspected under the Non-EQ Insulated Cables and Connections Program, the program will be consistent with GALL Section XI.E1, Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements.

The *Scope of Program* for the Non-EQ Insulated Cables and Connections Program will also be applied to instrument cable insulation, as addressed in Section XI.E2 of the GALL Report; however, the calibration of instrument circuits for the purpose of detecting insulation degradation, as called for in Section XI.E2, is not part of the RNP program. This is acceptable because the visible effects of localized adverse environments caused by heat or radiation would be manifest on all electrical cables, including instrument cables, prior to significant insulation resistance degradation.

Implementation of the Non-EQ Insulated Cables and Connections Program will provide reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.5 REFERENCES

B-1 NUREG-1800, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," U. S. Nuclear Regulatory Commission, April 2001.

B-2 NUREG-1801, "Generic Aging Lessons Learned (GALL)," U. S. Nuclear Regulatory Commission, April 2001.

APPENDIX C

IDENTIFYING AGING EFFECTS BY MATERIAL AND ENVIRONMENT

Appendix C is not being used in this application.

APPENDIX D

TECHNICAL SPECIFICATION CHANGES

10 CFR 54.22 requires that an application for license renewal include any technical specification changes of additions necessary to manage the effects of aging during the period of extended operation. The RNP license renewal review determined that no changes to the plant Technical Specifications are required.