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.
		or		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..
		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. Compo-nents 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°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 groundwater 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°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°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. Concrete The RNP AMR methodology concluded that above-grade concrete/grout structures have no aging effects. Refer to Table 3.5-2, Item 10. Steel 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. Lubrite (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°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°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, misalignment, 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
4.3.1.3	Reactor Internals Holddown Spring and Alignment Pins
4.3.1.4	Auxiliary Feedwater Line Fatigue Analysis
4.3.2	Implicit Fatigue Design (ASME Section III, Class C, B31.1)
4.3.3	Environmentally Assisted Fatigue Evaluation
4.3.4	Reactor Vessel Underclad Cracking
4.3.5	Containment Penetration Bellows
4.3.6	Crane Cyclic Load Limits

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 1.3×10^7 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 2.0×10^8 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 $2.3x10^7$ rads, gamma, plus a negligible normal aging, 60-year dose results in a projected dose of $2.3x10^7$ rads, TID. This projected TID is much less than the qualified value of $2.05x10^8$ 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 $1.3x10^7$ 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 $2.04x10^8$ 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°C, which exceeds the maximum normal operating temperature of 120°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°C, which exceeds the maximum normal operating temperature of 120°F. There is no heat rise for this instrumentation application (RTDs).

4.4.1.7.2 Radiation Analysis

The post accident dose of 1.3×10^{7} rads gamma plus the 1.65×10^{6} 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 2.5×10^{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 2.0×10^8 rads, gamma plus 1.27×10^6 rads, gamma normal aging, 46-year dose results in a TID requirement of 2.01×10^8 rads. This projected TID value is met by the qualified value of 2.02×10^8 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 1.3×10^6 rads, gamma, plus 1.5×10^6 rads, gamma normal aging, 60-year dose results in a projected TID requirement of 2.8×10^6 rads, which is much less than the qualified value of 1.0×10^7 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°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 1.3×10^7 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.

4.4.1.17.3 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 1.0×10^{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 1.3×10^{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.

4.4.1.19.3 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°C.

4.4.1.21.2 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 projected dose of 9.47×10^7 rads TID, which is less than the qualified value of 2.0×10^8 rads.

4.4.1.21.3 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°C.

4.4.1.22.2 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 projected dose of 9.47×10^7 rads TID, which is less than the qualified value of 2.0×10^8 rads.

4.4.1.22.3 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 1.3×10^7 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 2.59×10^8 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 1.3×10^7 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. This is less than the minimum qualified value of 2.0×10^8 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 $1.3x10^7$ rads, gamma plus $8.0x10^7$ rads, beta, plus $3.45x10^3$ rads, gamma normal aging, 60-year dose results in a projected dose of $9.30x10^7$ rads TID, which is much less than the qualified value of $2x10^8$ 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°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 $1.3x10^7$ 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 less than the qualified value of $1.65x10^8$ 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 $1.3x10^7$ 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 $1.6x10^8$ 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 $1.0x10^8$ 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 $1.3x10^7$ 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 $3.2x10^9$ 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 1.0×10^6 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 2.6×10^6 rads gamma plus the negligible, normal aging, 60-year dose results in a projected dose of 2.6×10^6 rads, TID, which is much less than the qualified value of 2.0×10^8 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 1.0×10^{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 1.3×10^{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.

4.4.1.39.3 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°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°F).

4.4.1.40.2 Radiation Analysis

The post accident dose of 1.3×10^7 rads gamma plus the 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.

4.4.1.40.3 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°C (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°F. 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°F).

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 $7.2x10^6$ rads gamma plus the negligible, normal aging, 60-year dose results in a projected dose of $7.2x10^6$ 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 $7.6x10^6$ rads, gamma plus a negligible 60-year normal dose results in a projected dose of $7.6x10^6$ 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 $1.3x10^7$ rads, gamma, plus $1.65x10^6$ rads, gamma normal aging, 60-year dose results in a projected dose of $1.47x10^7$ 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 $1.3\text{x}10^7$ rads gamma plus $8.0\text{x}10^7$ rads, beta, plus the $1.65\text{x}10^6$ rads, gamma normal aging, 60-year dose results in a TID requirement of $9.47\text{x}10^7$ rads TID, which is much less than the qualified value of $2.7\text{x}10^8$ 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 $1.3x10^7$ 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.

4.4.1.46.3 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.