Progress Energy

Serial: RNP-RA/03-0103

SEP 1 6 2003

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

RESPONSE TO OPEN AND CONFIRMATORY ITEMS

Ladies and Gentlemen:

By letter dated June 14, 2002, Carolina Power & Light (CP&L) Company, now doing business as Progress Energy Carolinas, (PEC) Inc., submitted an application for renewal of the Operating License for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, which is also referred to as the Robinson Nuclear Plant (RNP).

By letter dated August 25, 2003, the NRC provided a "Safety Evaluation Report with Open Items Related to the License Renewal of the H. B. Robinson Steam Electric Plant, Unit 2," and identified Open and Confirmatory Items for resolution. A partial response to these items was provided by letter dated August 14, 2003. The response to the remainder of these Open and Confirmatory Items, with the exception of Confirmatory Item 4.6.4-1, is provided in Attachment III to this letter.

As a result of these items, certain RNP license renewal commitments have been revised. A listing of license renewal commitments is provided in Attachment II to this letter.

If you have any questions concerning this matter, please contact Mr. C. T. Baucom.

Sincerely,

J. F. Lucas

J. F. Lucas
Manager - Support Services - Nuclear

JSK/jsk

Attachments:
I. Affirmation
II. Robinson Nuclear Plant License Renewal Commitments
III. Response to Open and Confirmatory Items

Progress Energy Carolinas, Inc.
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville, SC 29550

A001

c: Mr. T. P. O'Kelley, Director, Bureau of Radiological Health (SC)
Mr. L. A. Reyes, NRC, Region II
Mr. C. P. Patel, NRC, NRR
NRC Resident Inspectors, HBRSEP
Attorney General (SC)
Mr. S. K. Mitra, NRC, NRR
Mr. R. L. Emch, NRC, NRR
Mr. R. M. Gandy, Division of Radioactive Waste Management (SC)

AFFIRMATION

The information contained in letter RNP-RA/03-0103 is true and correct to the best of my information, knowledge and belief; and the sources of my information are officers, employees, contractors, and agents of Progress Energy Carolinas, Inc. I declare under penalty of perjury that the foregoing is true and correct.

Executed on: SEP 1 6 2003

C L Burton

C. L. Burton
Director of Site Operations, HBRSEP, Unit No. 2

Robinson Nuclear Plant License Renewal Commitments

Item	Commitment	Updated Final Safety Analysis Report (UFSAR) Supplement Location	Frequency	Source
1.	Quality Assurance Program. Existing program is credited. No changes required. See note below.	A.3.1		
2.	Upon issuance of the renewed license, guidance will be incorporated into administrative control procedures that manage the RNP configuration control process to ensure that the requirements of 10 CFR 54.37(b) are met. New commitment	A.3.1	On-going activity following issuance of renewed license	Request for Additional Information (RAI) 2.1.1-2
3.	Prior to the period of extended operation, a statement will be incorporated into the UFSAR Supplement description of the programs to document consistency of RNP Aging Management Programs with NUREG-1801 "Generic Aging Lessons Learned (GALL) Report"-defined programs. For RNP programs that are consistent with NUREG-1801, the program description will be revised to state: "This program is consistent with the corresponding program described in the GALL Report." Revised commitment	A.3.1	Prior to the period of extended operation	RAI B.1-1
4.	ASME Section XI, Subsection IWB, IWC, and IWD Program. Existing program is credited. No changes required. See note below.	A.3.1.1		
5.	Water Chemistry Program. Existing program is credited. No changes required. See note below.	A.3.1.2		
6.	ASME Section XI, Subsection IWB, IWC, and IWD Program. Existing program is credited. No changes required. See note below.	A.3.1.3		
7.	Steam Generator Tube Integrity Program. Existing program is credited. No changes required. See note below.	A.3.1.4		

8.	Closed-Cycle Cooling Water System Program. Existing program is credited. No changes required. See note below.	A.3.1.5		
9.	ASME Section XI, Subsection IWF Program. Existing program is credited. No changes required. See note below.	A.3.1.6		
10.	10 CFR 50, Appendix J Program. Existing program is credited. No changes required. See note below.	A.3.1.7		
11.	Flux Thimble Eddy Current Inspection Program. Existing program is credited. No changes required. See note below.	A.3.1.8		
12.	The Fire Protection Program will be enhanced to note that concrete surface inspections performed under structures monitoring procedures are credited for inspection of fire barrier walls, ceilings, and floors. No change to original commitment	A.3.1.9	Prior to the period of extended operation	LR Application Appendix B, Section B.3.1
13.	The scope of the Boric Acid Corrosion Program will be expanded to: (1) ensure the mechanical, structural, and electrical components in scope for license renewal are addressed, and (2) identify additional areas in which components are susceptible to exposure from boric acid. No change to original commitment	A.3.1.10	Prior to the period of extended operation	LR Application Appendix B, Section B.3.2
14.	The Flow-Accelerated Corrosion (FAC) Program will be modified to: (1) include additional components potentially susceptible to FAC and/or erosion, and (2) clarify when condition reports shall be initiated. No change to original commitment	A.3.1.11	Prior to the period of extended operation	LR Application Appendix B, Section B.3.3
15.	The following will be implemented: (1) administrative controls for bolting will be modified to prohibit the use of MoS_2 compounds in high strength bolting applications, and (2) an inspection and evaluation will be performed on high strength bolting used on one motor-operated valve to determine susceptibility for cracking. No change to original commitment	A.3.1.12	Prior to the period of extended operation	LR Application Appendix B, Section B.3.4
16.	An activity will be scheduled in the site Preventive Maintenance Program to replace cooling coils in the Emergency Core Cooling System room coolers on a prescribed frequency. No change to original commitment	A.3.1.13	Prior to the period of extended operation	LR Application Appendix B, Section B.3.5

17.	Administrative controls for inspection of overhead heavy load and light load handling will be enhanced to: (1) include requirements for inspecting the turbine gantry crane in addition to the other cranes that require inspection, (2) note that cranes are to be inspected using the attribute inspection checklist for structures, and (3) revise the attribute inspection checklist for structures to include GALL terminology such as wear. Revised commitment	A.3.1.14	Prior to the period of extended operation	LR Application Appendix B, Section B.3.6 RAI B.3.6-2
18.	The Fire Water System Program will be modified to include: Fire Protection Sprinkler Systems (1) For sprinkler heads in service for 50 years, either sprinkler head replacement or sampling/field service testing of heads in accordance with National Fire Protection Association (NFPA) 25 requirements based on the in-service date of the affected systems, and (2) prior to the period of extended operation, either full flow testing of portions of fire protection wet pipe sprinkler systems through the system cross mains, which are not routinely subject to flow, at the greatest flow and pressure allowed by the design of the systems or, alternatively, inspections or ultrasonic (UT) testing of a representative sample of these systems. Results from initial tests or inspections, reflecting 40 years of service, will be used to determine the scope and subsequent test/inspection intervals. The intervals are not expected to exceed 10 years. Fire Protection Suppression Piping Prior to the period of extended operation, UT examination on a representative sampling of the above-ground fire protection piping normally containing water will be performed. Each sampling will include different sections of piping. Alternatively, internal inspections may be conducted on a representative sampling of these piping systems. Results from initial tests or inspections, reflecting 40 years of service, will be used to determine the scope and subsequent test/inspection intervals. The intervals are not expected to exceed 10 years. Halon/Carbon Dioxide Fire Suppression Systems The NRC staff guidance with respect to halon/carbon dioxide fire suppression systems will be implemented prior to the period of extended operation. The guidance is documented in a letter from C. Grimes (NRC) to A. Nelson (Nuclear Energy Institute) and D. Lochbaum (Union of Concerned Scientists): Proposed Staff Guidance on Aging Management of Fire Protection Systems for License Renewal, dated January 28, 2002. Revised commitment	A.3.1.15	As noted in the commitment	LR Application Appendix B, Section B.3.7 CP&L letter to NRC, RNP-RA/02-0159: Supplement to Application for Renewal of Operating License, dated October 23, 2002

19.	A review will be performed to ascertain the need to update, as necessary, administrative controls for the Buried Piping and Tanks Surveillance Program to ensure consistency with National Association of Corrosion Engineers (NACE) Standard RP-0169-96 regarding acceptance criteria for the cathodic protection system, and additional leak testing provisions for underground piping will be incorporated. No change to original commitment	A.3.1.16	Prior to the period of extended operation	LR Application Appendix B, Section B.3.8
20.	Administrative controls for the Above Ground Carbon Steel Tanks Program will be revised to indicate that the external surfaces of the fuel oil tanks are to be inspected periodically and to incorporate corrective action requirements. No change to original commitment	A.3.1.17	Prior to the period of extended operation	LR Application Appendix B, Section B.3.9
21.	Administrative controls for the Fuel Oil Chemistry Program will be enhanced to: (1) improve sampling and de-watering of selected storage tanks, (2) formalize existing practices for periodically draining and filling the Diesel Fuel Oil Storage Tank, (3) formalize bacteria testing for fuel oil samples from various tanks, and (4) incorporate quarterly trending of fuel oil chemistry parameters. No change to original commitment	A.3.1.18	Prior to the period of extended operation	LR Application Appendix B, Section B.3.10
22.	Reactor Vessel Surveillance Program administrative controls will be revised to require surveillance test samples to be stored in lieu of optional disposal. No change to original commitment	A.3.1.19	Prior to the period of extended operation	LR Application Appendix B, Section B.3.11
23.	The Buried Piping and Tanks Inspection Program will be enhanced to: (1) require that an appropriate as-found pipe coating and materiel condition inspection is performed whenever buried piping within the scope of this program is exposed, (2) add precautions to ensure backfill with material that is free of gravel or other sharp or hard material that can damage the coating, (3) require that coating inspection be performed by qualified personnel to assess its condition, and (4) require that a coating engineer assist in evaluation of any coating degradation noted during the inspection. No change to original commitment	A.3.1.20	Prior to the period of extended operation	LR Application Appendix B, Section B.3.12

24.	ASME Boiler & Pressure Vessel Code, Section XI, Subsection IWE Program administrative controls will be enhanced to: (1) specify the requirements for conducting reexaminations, and (2) document that repairs meet the specified acceptance standards. No change to original commitment	A.3.1.21	Prior to the period of extended operation	LR Application Appendix B, Section B.3.13
25.	ASME Boiler & Pressure Vessel Code, Section XI, Subsection IWL Program enhancements will be made to require supervisors to notify Civil/Structural Design Engineering of the location and extent of proposed excavations of foundation concrete, to require inspection of below-grade concrete when excavated for any reason to monitor for potential effects and to inspect above grade accessible concrete, and include trending requirements for structures based on aggressive ground water. Revised commitment	A.3.1.22	Prior to the period of extended operation	LR Application Appendix B, Section B.3.14 CP&L letter to NRC, RNP-RA/02-0159: Supplement to Application for Renewal of Operating License, dated October 23, 2002 Confirmatory Item 3.5-1

26.	Structures Monitoring Program administrative controls will be enhanced to: (1) include buildings and structures, and associated acceptance criteria, in scope for license renewal but outside the scope of the Maintenance Rule, (2) identify interfaces between structures monitoring inspections of concrete surfaces and the Fire Protection Program requirements for barriers, (3) state clearly the boundary definition between systems and structures, (4) revise administrative controls to provide inspection criteria for portions of systems covered by structures monitoring and require corrective action(s) be initiated for unacceptable inspection attributes, (5) expand system walkdown inspection criteria to include observation of adjacent components, (6) inspect above grade accessible concrete, and (7) revise personnel responsibilities to include providing assistance in evaluating structural deficiencies when requested by the Responsible Engineer, inspecting excavated concrete to monitor for potential aging effects, and notifying Civil/Structural Design Engineering of the location and extent of proposed excavations, (8) include trending requirements for structures based on aggressive ground water and lake water. Revised commitment	A.3.1.23	Prior to the period of extended operation	LR Application Appendix B, Section B.3.15 CP&L letter to NRC, RNP-RA/02-0159: Supplement to Application for Renewal of Operating License, dated October 23, 2002 Confirmatory Item 3.5-1

27.	To enhance the Dam Inspection Program, the system monitoring administrative controls will be revised to: (1) identify the "Recommended Guidelines for Safety Inspection of Dams" as the required management program document for the dam, (2) require the responsible system engineer to review the inspection report and initiate corrective actions for any unacceptable attributes, (3) include "Recommended Guidelines for Safety Inspections of Dams" as the applicable inspection guidance in the inspection procedure for RNP, (4) inspect above grade accessible concrete, (5) inspect submerged spillway concrete on a frequency not to exceed (10) ten years and (6) include trending requirements for structures based on aggressive ground water and lake water. Revised commitment	A.3.1.24	Prior to the period of extended operation	LR Application Appendix B, Section B.3.16 CP&L letter to NRC, RNP-RA/02-0159: Supplement to Application for Renewal of Operating License, dated October 23, 2002 RAI B.3.14-1 Confirmatory Item 3.5-1
28.	Systems Monitoring Program administrative controls will be enhanced to: (1) include aging effects identified in the aging management reviews, (2) identify inspection criteria in checklist form, (3) include guidance for inspecting connected piping/components, (4) require that the extent of degradation be recorded and that appropriate corrective action(s) be taken, (5) add a section specifically addressing corrective actions, and (6) ensure "Loss of Material due to Wear" is specifically included as an aging effect/mechanism identified in the system walkdown checklist. Revised commitment	A.3.1.25	Prior to the period of extended operation	LR Application Appendix B, Section B.3.17 RAI B.3.17-1
29.	Preventive Maintenance Program administrative controls will be enhanced to: (1) include aging effects/mechanisms identified in the aging management reviews, and (2) incorporate specific aging management activities identified in the aging management reviews into the program. No change to original commitment	A.3.1.26	Prior to the period of extended operation	LR Application Appendix B, Section B.3.18

30.	The Fatigue Monitoring Program load/unload transient limit will be reduced to provide the margin needed for consideration of reactor water environmental effects. No change to original commitment	A.3.1.27	Prior to the period of extended operation	LR Application Appendix B, Section B.3.19
31.	The Nickel-Alloy Nozzles and Penetrations Program is a new program that will incorporate the following: (1) evaluations of indications will be performed under the ASME Boiler & Pressure Vessel Code, Section XI program, (2) corrective actions for augmented inspections will be performed in accordance with repair and replacement procedures equivalent to those requirements in ASME Boiler & Pressure Vessel Code, Section XI, (3) RNP will maintain its involvement in industry initiatives and will implement any actions, unless impracticable, that are agreed upon between the NRC and the nuclear power industry to monitor for, detect, evaluate, and correct cracking in the Vessel Head Penetration (VHP) nozzles, specifically as the actions relate to ensuring the integrity of VHP nozzles in the RNP upper reactor vessel head during the extended period of operation, and (4) RNP will submit, for review and approval, its inspection plan for the Nickel-Alloy Nozzles and Penetrations Program, as it will be implemented from the applicant's participation in industry initiatives, prior to July 31, 2009. Revised commitment	A.3.1.28	As noted in the commitment	LR Application Appendix B, Section B.4.1 RAI B.4.1-1
32.	The Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program is a new program applied to CASS components within Class 1 boundaries of the Reactor Coolant System and connected systems where operating temperature exceeds the threshold criterion. No change to original commitment	A.3.1.29	The program will be implemented prior to the period of extended operation.	LR Application Appendix B, Section B.4.2

33.	The PWR Vessel Internals Program is a new program that will incorporate the following: (1) RNP will continue to participate in industry programs to investigate aging effects and determine the appropriate aging management program activities to address baffle and former assembly issues, and to address change in dimensions due to void swelling, (2) As WOG and EPRI MRP research projects are completed, RNP will evaluate the results and factor them into the PWR Vessel Internals Program as appropriate, and (3) RNP will implement an augmented inspection during the license renewal term. Augmented inspections, based on required program enhancements resulting from industry programs, will become part of the ASME Boiler & pressure Vessel Code, Section XI program. Corrective actions for augmented inspections will be developed using repair and replacement procedures equivalent to those requirements in ASME Boiler & Pressure Vessel Code, Section XI. RNP will submit, for review and approval, its inspection plan for the PWR Vessel Internals Program, as it will be implemented from the applicant's participation of industry initiatives, 24 months prior to the augmented inspection. Revised commitment	A.3.1.30	As noted in the commitment	LR Application Appendix B, Section B.4.3 RAI B.4.3-2
34.	One-Time Inspection Program activities consist of inspections of the following: (1) The Aging Management Program (AMP) determined that an inspection of CCW Heat Exchanger tubing would be prudent to assure that potential degradation due to erosion was managed. (2) Miscellaneous piping in steam and power conversion systems protected by the Water Chemistry Program will be inspected. The One-Time Inspection Program will be used to select representative inspection locations. (3) Small bore Reactor Coolant System and connected piping will be inspected to verify effectiveness of the Water Chemistry Program. Components to be examined will be selected based on accessibility, exposure levels, Non-Destructive Examination (NDE) techniques, and locations identified in NRC Information Notice 97-46. (4) Emergency Diesel Generator exhaust silencers. (5) Certain inaccessible areas of the containment liner plate and containment structure moisture barrier are required to be inspected to determine their materiel condition. (6) The Diesel Fire Pump Fuel Oil Tank. (7) Steam Generator feed ring/J-nozzles. Revised commitment	A.3.1.31	Prior to the period of extended operation	LR Application Appendix B, Section B.4.4 RAI 3.5.1-1 RAI B.3.10-6 Open Item 2.3.1.6-1

35.	The Selective Leaching of Materials Program is a new program to determine the properties of selected components that may be susceptible to selective leaching. The program will ascertain whether loss of material is occurring and whether the process will affect the ability of the components to perform their intended function for the period of extended operation. No change to original commitment	A.3.1.32	The program will be implemented prior to the period of extended operation.	LR Application Appendix B, Section B.4.5
36.	The Non-Environmentally Qualified Insulated Cables and Connections Program is a new program and involves inspecting accessible power and instrument and control cables at least once every 10 years. The technical basis for selecting a sample of cables to be inspected will be defined prior to the period of extended operation. The sample locations will consider the location of cables inside and outside containment, as well as any known adverse localized environments. Revised commitment	A.3.1.33	As noted in the commitment The program will be implemented prior to the period of extended operation.	LR Application Appendix B, Section B.4.6 RAI 3.6.1-2 B4.6-3 Confirmatory Item 3.6.2.3.1.2-1
37.	The Aging Management Program For Non-EQ Electrical Cables Used in Instrumentation Circuits is a new program that uses calibration or surveillance testing programs to identify the potential existence of aging degradation of cables. This program applies to the cables used in containment high-range radiation monitoring instrumentation circuits. The program has a 10-year frequency. New commitment	A.3.1.34	As noted in the commitment The program will be implemented prior to the period of extended operation.	RAI 3.6.1-2 RAI B.4.6-3
38.	The Aging Management Program For Neutron Flux Instrumentation Circuits is a new program that will employ insulation resistance or other testing to identify the potential existence of aging degradation of cables in neutron monitoring circuits. The program has a 10-year frequency. New commitment	A.3.1.35	As noted in the commitment The program will be implemented prior to the period of extended operation.	RAI 3.6.1-2 RAI B.4.6-3

39.	The Aging Management Program for Fuse Holders is a new program applicable to fuse holders located outside of active devices. The program utilizes thermography or other appropriate test methods to identify the potential existence of aging degradation. The program has a 10-year frequency. New commitment	A.3.1.36	As noted in the commitment The program will be implemented prior to the period of extended operation.	RAI 2.5.2-1
40.	The Aging Management Program for Bus Duct is a new program for inspecting bus duct for signs of cracks, corrosion, foreign debris, excessive dust buildup, or discoloration which may indicate overheating, loosening of bolted connections, or water intrusion. The program applies to the iso-phase bus duct as well as all non-segregated 4.16 KV and 480 V bus duct within the scope of license renewal. The program has a 10-year frequency. New commitment	A.3.1.37	As noted in the commitment The program will be implemented prior to the period of extended operation.	RAI 2.5.2-2
41.	Credit is taken for existing Environmental Qualification (EQ) of Electric Equipment activities. EQ is an ongoing program. EQ packages are undergoing revision to incorporate increased radiation values resulting from power uprate and will be updated prior to the end of the current license term. Revised commitment	A.3.1.38	As noted in the commitment	RAI 4.4-2 RAI 4.4.1-2
42.	Time-Limited Aging Analysis (TLAA) - Reactor Vessel Neutron Embrittlement. Existing program is credited. No changes required. See note below.	A.3.2.1		

43.	TLAA – Metal Fatigue. Based upon the most recent fatigue analysis performed to date for the three Auxiliary Feedwater (AFW)-to-Feedwater (FW) line connections downstream of the steam-driven pump, transient limits have been reduced in the RNP Fatigue Monitoring Program. These reduced limits are based upon inputs used in the analysis, and are more conservative than the original limits. The reduced limits will remain in effect until the connections are further analyzed, repaired, or replaced to assure the connections remain within their design basis through the period of extended operation. Based upon the fatigue analyses performed to consider environmentally assisted fatigue, the load/unload transient limit has been reduced in the RNP Fatigue Monitoring Program. The reduced limits are based upon inputs used in the analyses, and will remain in effect permanently unless the components are reanalyzed. The reduced limit is not expected to be approached through the period of extended operation, because the original limit was established at a high value to account for load following, which is not necessary at RNP. Further action is required for management of environmental fatigue of the surge line for the period of extended operation. Therefore, fatigue of the surge line will be managed using one or more of the following options: 1. Further refinement of the fatigue analyses to maintain the EAF-adjusted CUF below 1.0. 2. Repair of the affected locations. 3. Replacement of the affected locations. 4. Manage the effects of fatigue through the use of an augmented inservice inspection program that has been reviewed and approved by the NRC. This includes periodic surface and volumetric examinations of the limiting locations at inspection intervals to be determined by a method accepted by the NRC. If this option is selected, the scope, qualification, method, and frequency will be provided to the NRC for review and approval prior to the period of extended operation. Revised commitment	A.3.2.2	As noted in the commitment	LR Application, Section 4.3 RAI 4.3-2 RAI 4.3-7 RAI 4.3-10

44.	TLAA – Environmental Qualification. In accordance with the requirements of the Environmental Qualification Program, any component that is not qualified through the period of extended operation will be refurbished or replaced prior to exceeding its qualified life. Prior to the period of extended operation, certain motor operated valve actuators will either be reevaluated to demonstrate acceptable wear-cycle qualification or they will be replaced. No change to original commitment	A.3.2.3	On-going activity	LR Application, Sections 4.4 and 4.4.1.3
45.	TLAA-Containment Tendon Loss of Prestress. To provide additional assurance of the tendons design capacity, testing at Integrated Leak Rate Test pressure, similar to the Structural Integrity Test performed in 1992, will be scheduled to coincide with Appendix J containment Integrated Leak Rate Testing conducted during the period of extended operation (required frequency in accordance with 10 CFR 50, Appendix J). The monitoring criteria for these tests will be limited to deformations and cracking associated with the vertical prestressed tendons, and will not include radial monitoring. Guidelines for performing the IWL examinations for these tests will include additional emphasis on looking for a pattern of horizontal cracks, and additional cracking in the discontinuity areas. New commitment	A.3.2.4	As noted in the commitment	RAI 4.5-2
46.	TLAA-Containment Tendon Loss of Prestress. Information from the response to RAI 4.5-1 will be incorporated into Section 3.8.1.4.7 of the UFSAR. This will include initial average prestressing force, losses, and final average prestressing force at 50 and 60 years as discussed in the response to RAI 4.5-1. This commitment supercedes the proposed changes shown on LRA Page A-6 for UFSAR Section 3.8.1.4.7. New commitment	A.3.2.4	Prior to the period of extended operation	RAI 4.5-3
47.	TLAA – Aging of Boraflex in Spent Fuel Pool. The current Boraflex monitoring program will be evaluated against the requirements for a license renewal aging management program, and the results of the evaluation will be documented in the UFSAR. This commitment may be withdrawn if planned analysis successfully eliminates credit for the Boraflex sheets in the spent fuel racks in determining K_{eff} for the spent fuel array. No change to original commitment	A.3.2.8	Prior to the period of extended operation	LR Application, Section 4.6.4

Note: Consistent with guidance provided by letter from Pao-Tsin Kuo (NRC) to Alan Nelson (NEI) and David Lochbaum (Union of Concerned Scientists), "Consolidated List Of Commitments For License Renewal," dated December 16, 2002.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
RESPONSE TO OPEN AND CONFIRMATORY ITEMS

Open Item 2.3.1.6-1:

(steam generator feedrings)

The staff believes that the steam generator (SG) feedrings should be included in the scope of license renewal (Open Item 2.3.1.6-1). Since this component is completely enclosed by safety-related, pressure-boundary components, it is important to show that failures of this component could not impede certain safety-related functions of the components in which it is contained (10 CFR 54.4(a)(2)).

Open Item 2.3.1.6-1 Response:

This response provides supplemental information (see the response to Request for Additional Information (RAI) 2.3.1.6-1 in Robinson Nuclear Plant (RNP) letter RNP-RA/03-0031, dated April 28, 2003, and the response to Clarification M to RAI 2.3.1.6-1 in RNP letter RNP-RA/03-0074, dated June 13, 2003) regarding the evaluation process for scoping against 10 CFR 54.4(a)(2).

In the response to Clarification M to RAI 2.3.1.6-1 it was determined that flow blockage of the feed rings sufficient to impede auxiliary feedwater (AFW) flow was implausible based on the ability of the feed rings to pass fifty (50) times the required feedwater (FW) flow during normal operation versus that required for AFW in response to various events (i.e., Anticipated Transient Without Scram (ATWS), Station Blackout (SBO), etc.). An evaluation was also performed to determine the potential for a loose part to be generated by the RNP J-nozzles. It was determined that the generation of a loose part would not be plausible based on feedback from industry contacts that showed that the carbon steel portions of the feed ring with greater than 0.10% chromium content would not be susceptible to flow erosion/corrosion.

Subsequent to the response to Clarification M to RAI 2.3.1.6-1, a more focused review of operating experience for steam generator feed rings was performed in order determine if a failure that generated loose parts has occurred. Searches were not restricted to plants in the United States. The operating experience information compiled by both the Institute of Nuclear Power Operations (INPO) and World Association of Nuclear Operators (WANO) was extensively reviewed and the pertinent information found is as follows:

- On April 25, 1989, a Westinghouse Model 51 steam generator was undergoing periodic testing and J-nozzles were found to be eroded/corroded. However, there was no discussion of loose parts. J-nozzles were replaced and the equipment was returned to service.
- On January 14, 1995, a Westinghouse Model 51 steam generator was undergoing a special leak test of the feed ring and water appeared to be flowing from a fault

in either the piping or plug located at the support bracket, 180 degrees from the feedwater inlet. Further investigation revealed that the leak was in the feed ring piping. The leak occurred at a previously plugged bottom spray hole. There was no significant effect on the reactor coolant system or on the plant. The cause of the weld joint having a small leak was attributed to rusting and corrosion of the plug and weld. This was considered normal wear due to age and usage. Two J-nozzles were removed from the top of the feed ring, the area was cleaned, and the damaged weld in the bottom of the feed ring was weld repaired. Although there was no failure on the J-nozzles, two existing J-nozzles were replaced with new style Inconel J-nozzles as a preventive measure. It was also verified that there was no additional leakage. Again, there was no discussion of loose parts.

A search of the NRC website yielded Information Notice No. 91-19, "Steam Generator Feedwater Distribution Piping Damage." While this Information Notice required no specific action or response, RNP performed an applicability evaluation in 1991, which is summarized as follows:

- The feedwater ring configuration shown in Attachments 1 and 2 of the Information Notice were compared to the drawing representing the design at RNP and were found to be very different. Because of the difference in design, it is not likely that RNP would have the same failure.

Based on the above review of industry operating experience, no instances of the generation of a loose part from the feed ring/J-nozzles in steam generators similar to RNP's were identified. Based on this review, it was determined that a failure of this type is highly unlikely. This is based not only on the number of reactor-years of operating experience reviewed, but also on the type and nature of potential aging effects. This provides additional assurance that the intended function of the steam generators will be maintained during the period of extended operation.

RNP also performed an additional review of the current licensing basis to identify any potential requirement for the feed ring/J-nozzles to provide throttling of feedwater flow that could be construed as a license renewal intended function. The RNP Updated Final Safety Analysis Report (UFSAR) in Section 15.2.8 describes the Feedwater System Pipe Break. The UFSAR states:

"This event is postulated to be caused by the instantaneous severance of a feedwater line.

The H. B. Robinson steam generators are fed by a single 16-inch line. Auxiliary feedwater enters the same nozzle. The sparger is approximately 3 feet above the top of the tube bends and approximately 7 feet below the top of the downcomer. The sparger is approximately 2 feet above the low range liquid level tap. The feedwater sparger is of the J tube type. Upon

> *rupture, some liquid may initially blow down; however, substantial liquid will remain.*
>
> *In many PWRs, feedwater is introduced at the bottom of the steam generator and a feedwater pipe break potentially results in a major or total loss of steam generator liquid inventory and subsequent primary system heatup. In the case of H. B. Robinson, however, this event will be a cooldown event and will be bounded by the steam line break results as the feedwater pipe is much smaller in area than the minimum area for flow in the new steam generator integral flow restrictors."*

The feed rings/J-nozzles are not required to limit blowdown from the steam generator associated with this event.

As far as throttling flow to the steam generators, the Main Feedwater Regulating Valves (MFRVs) regulate the flow of feedwater to the steam generators to maintain a programmed level. The MFRVs are used from approximately 15% load to 100% load. The main feedwater regulating valve bypass valves are used during low load conditions for finer feedwater flow control. The feed ring/J-nozzles play no part in throttling the inlet flow. The Main Feedwater Isolation Valves (MFIVs) isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). The safety related function of the MFRVs is to provide the second isolation of MFW flow to the secondary side of the steam generators following an HELB. Closure of the MFIVs or MFRVs, and bypass valves terminates flow to the steam generators, terminating the event for feedwater line breaks (FWLBs) occurring upstream of the MFIVs or MFRVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream from the MFIVs will be mitigated by their closure. Closure of the MFIVs or MFRVs, and bypass valves effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks or feedwater system pipe breaks inside containment, and reducing the cooldown effects for steam line breaks.

Based on the above analysis, the feed rings/J-nozzles do not have a throttling function.

In summary, based on the scoping criteria of the rule, the postulation of hypothetical failures, and a focused review of operating experience, RNP concludes that the feed rings/J-nozzles do not meet the scoping requirements of 10 CFR 54.4.

However, to facilitate the NRC review process, RNP will include the feed rings/J-nozzles in scope for license renewal.

The results of the revised aging management evaluations are shown in the following updated License Renewal Application (LRA) tables:

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

Component/Commodity	Intended Function	AMR Results
	Steam Generator (continued)	
Steam Generator Support Pad	Provide structural support to pressure boundary components. Maintains structural integrity of pressure boundary components.	Table 3.1-1, Item 1 Table 3.1-1, Item 26
Steam Generator Tube Bundle Wrapper System	Provide structural support to pressure boundary components. Maintains structural integrity of pressure boundary components. Provides flow restriction or distribution.	Table 3.1-1, Item 1 Table 3.1-2, Item 5 Table 3.1-2, Item 6
Steam Generator Tubeplate	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-2, Item 5 Table 3.1-2, Item 6
Steam Generator Tubeplate Cladding	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide corrosion protection of pressure boundary components.	Table 3.1-1, Item 1 Table 3.1-2, Item 2 Table 3.1-2, Item 11
Steam Generator Feedwater Distribution Ring	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-2, Item 5 Table 3.1-2, Item 6 Table 3.1-2, Item 19
Steam Generator J-Nozzles	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.1-1, Item 1 Table 3.1-2, Item 4 Table 3.1-2, Item 6

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, J-nozzles)	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, feedwater distribution ring)	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, Feedwater Distribution Ring, J-Nozzles)	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 and J-Nozzles are fabricated of Inconel and, therefore, are 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
19. Steam Generator Feedwater Distribution Ring	Carbon Steel	Treated Water (including steam)	Loss of Material from FAC	One-Time Inspection Program	As noted in GALL discussion of item D 1.3.1, this form of degradation has not been detected in Westinghouse steam generators. RNP will perform a visual inspection of the feed ring/J-nozzles before entering into the extended period of operation to confirm that evidence of aging is so insignificant an aging management program is not required for the license renewal period.

B.4.4 ONE-TIME INSPECTION PROGRAM

The One-Time Inspection Program is credited for aging management of various structures/components at RNP as shown below:

Structure/Component	Building Structure/ System	Aging Effect/Mechanisms of Concern
Steam Generator Feedwater Distribution Ring/J-Nozzles	Steam Generators	Loss of material due to flow-accelerated corrosion

The information in LRA Subsection A.3.1.31, One-Time Inspection Program, has been previously revised as a result of RAI 3.5.1-1 and RAI B.3.10-6 (see RNP letter RNP-RA/03-0031, dated April 28, 2003).

As a result of the previous revision and the above response to Open Item 2.3.1.6-1, the LRA Subsection A.3.1.31, One-Time Inspection Program, now reads:

> *"Special inspections of components within the scope of license renewal will be performed in accordance with the One-Time Inspection Program. The Program is used to verify the effectiveness of the aging management activities and to determine the present condition of components. One-Time Inspection Program activities consist of inspecting (1) the CCW heat exchanger tubes, (2) miscellaneous piping protected by the Water Chemistry Program, (3) small bore RCS and connected piping, (4) EDG exhaust silencers, (5) containment liner plate and moisture barrier, (6) the Diesel Fire Pump Fuel Oil Tank and (7) Steam Generator feed ring/J-nozzles."*

Commitment 34 has been revised to reflect the above Open Item response.

Open Item 2.3.3.8-1:

(exclusion of deepwell pumps, piping, and valves from an AMR)

The staff requested the applicant to provide adequate justification for the exclusion of the deepwell pumps and associated piping from an AMR. The staff found that the applicant has not adequately justified the referred exclusion. The context of Section 10.4.8 of the UFSAR does not link dam failure to any particular set of initiating events, and seismic events and age-related degradation do not encompass all credible causes of dam failure. Dam failure results in loss of the ultimate heat sink and loss of the normal backup supply of feedwater from the service water system through the auxiliary feedwater system. Following dam failure and depletion of the condensate storage tank inventory, failure of the deepwell pumps would cause failure of the safety-related auxiliary feedwater system and prevent the residual heat removal (RHR) necessary to maintain a safe shutdown condition. Therefore, the deepwell pumps and associated piping are within the scope of LR in accordance with 10 CFR 54.4(a)(2). Therefore, the staff found that the applicant has not adequately justified excluding the deepwell pumps and associated piping and valves from an AMR, and this issue remains as Open Item 2.3.3.8-1.

Open Item 2.3.3.8-1 Response:

RNP agrees to include in the scope of license renewal the three deepwell pumps and associated piping required to provide a backup source of water for the auxiliary feedwater system. The deepwell pumps are vertical turbine-type pumps with integral carbon steel suction piping connected to the pump suction case. This suction piping is integral to the pump and therefore is not shown on the flow diagram. The suction piping is in the well and extends below the pump case. The revised boundary includes the suction piping, deepwell pumps, and piping up to and including the first isolation valve in each branch line. The flow path will connect with valve DW-21 which was included in the original scope of license renewal (refer to boundary drawing G-190202LR, Sheet 3, H-3).

The deepwell pumps and associated piping have been added to the RNP screening and AMR evaluations. The revised aging management evaluations resulted in the identification of material/environment combinations not previously identified in the LRA for the Primary and Demineralized Water Makeup System. The deepwell pumps are carbon steel/cast iron and are exposed to a raw water environment. The deepwell pump stations are fabricated with carbon steel, stainless steel, and copper alloy valves, piping, and fittings exposed internally to raw water and externally to outdoor air. The piping connected to the pump stations is plastic coated carbon steel which is run underground. This underground carbon steel piping makes up the majority of the piping in the deepwell system. The suction piping and remaining above ground piping is carbon steel.

AMR results applicable to the deepwell pumps and associated piping include LRA Table 3.3-1, Items 5 and 17, and LRA Table 3.3-2, Items 21 and 23. With the exception of Table 3.3-1, Item 17, these evaluations can be applied to the deepwell components as written and, therefore, do not require revision as a result of the expanded scope. Table 3.3-1, Item 17, requires only the identification of the Primary and Demineralized Water Makeup System in the discussion. Table 3.3-2 requires the addition of two items (i. e., rows) to describe the aging management evaluation of the deepwell pumps and associated piping not previously evaluated.

The results of the revised aging management evaluations are shown in the following updated LRA tables:

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

Component/Commodity	Intended Function	AMR Results
Valves, Piping, and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered. Provide structural support to safety-related components.	Table 3.3-1, Item 5 Table 3.3-1, Item 17 Table 3.3-2, Item 6 Table 3.3-2, Item 7 Table 3.3-2, Item 21 Table 3.3-2, Item 23 Table 3.3-2, Item 32 Table 3.3-2, Item 33
Deepwell Pumps	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.3-2, Item 32

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, Fire Protection, and Primary and Demineralized Water Makeup 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 3.3-2 (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
32. Deepwell Pumps, Valves, Piping, and Fittings within the Primary and Demineralized Water Makeup System	Carbon Steel	Raw Water	Loss of Material from Crevice Corrosion	Preventive Maintenance Program	The RNP AMR identified MIC, crevice, pitting, and general corrosion as potential aging mechanisms. Testing and inspections will be done under the Preventive Maintenance Program to ensure the aging effects are adequately managed for these components.
			Loss of Material from Pitting Corrosion	Preventive Maintenance Program	
			Loss of Material from MIC	Preventive Maintenance Program	
			Loss of Material from General Corrosion	Preventive Maintenance Program	
33. Valves, Piping, and Fittings within the Primary and Demineralized Water Makeup System	Copper Alloys, Stainless Steel	Raw Water	Loss of Material from Crevice Corrosion	Preventive Maintenance Program	The RNP AMR identified MIC, crevice, pitting, and general corrosion as potential aging mechanisms. Testing and inspections will be done under the Preventive Maintenance Program to ensure the aging effects are adequately managed for these components.
			Loss of Material from Pitting Corrosion	Preventive Maintenance Program	
			Loss of Material from MIC	Preventive Maintenance Program	

Confirmatory Item 2.3.1.3-1:

(pressurizer spray head)

The staff believed that the pressurizer spray head should be included in the scope of license renewal (RAI 2.3.1.3-1). Since this component is completely enclosed by safety-related, pressure-boundary components, it is important to show that its failure could not impede certain safety-related functions of the components in which they are contained (10 CFR 54.4(a)(2)). The possibility of a failure in the pressurizer spray head, affecting the functioning of the PORVs or pressurizer safety valves was noted. The applicant surveyed operating experience and concluded that such a failure had not occurred anywhere. The applicant provided supplemental information in support of a revised response to RAI 2.3.1.3-1. Pending the applicant's formal submittal of this information and the NRC staff's review of the acceptability of the supplemental information, RAI 2.3.1.3-1 will be considered to be Confirmatory Item 2.3.1.3-1.

Confirmatory 2.3.1.3-1 Response:

This response provides supplemental information (see the response to Request for Additional Information (RAI) 2.3.1.3-1, in Robinson Nuclear Plant (RNP) letter RNP-RA/03-0031, dated April 28, 2003, and the response to Clarification M to RAI 2.3.1.3-1 in RNP letter RNP-RA/03-0074 dated June 13, 2003) regarding the evaluation process for scoping against the criterion of 10 CFR 54.4(a)(2).

In the response to Clarification M to RAI 2.3.1.3-1 it was determined that clogging of the pressurizer spray head does not affect the pressure boundary component intended function of the pressurizer. An evaluation was also performed to determine the consequences of a hypothetical failure whereby a loose part is generated by the spray head. It was determined that it would be highly unlikely that the loose part generated could challenge the pressure boundary function of the pressurizer and/or the connected piping. That evaluation concluded that the spray head does not meet the scoping requirements of 10 CFR 54.4(a)(2).

Subsequent to the response to Clarification M to RAI 2.3.1.3-1, a more focused review of operating experience for pressurizers was performed in order determine if a failure of the type hypothesized has occurred. Searches were not restricted to plants in the United States. The operating experience information compiled by both the Institute of Nuclear Power Operations (INPO) and World Association of Nuclear Operators (WANO) was extensively reviewed. This review found no instances of the hypothetical failure. Based on this review, it was determined that a failure of the type postulated is highly unlikely. This is based not only on the number of reactor-years of operating experience reviewed, but also on the type and nature of potential aging effects. This provides additional assurance that the pressure boundary intended function of the pressurizer will be maintained during the period of extended operation.

In summary, based on the scoping criteria of 10 CFR 54, the postulation of hypothetical failures, and a focused review of operating experience, the pressurizer spray head does not meet the scoping requirements of 10 CFR 54.4(a)(2).

Confirmatory Item 2.3.2.5-1:

(hydrogen recombiners and supporting components)

The staff considered the applicant's responses to RAIs 2.3.2.5-1, 2.3.2.5-2, and 2.3.2.5-3 to be unacceptable because they are incomplete. Although the responses provided sufficient information to demonstrate that 10 CFR 54.4(a)(1) and (a)(3) did not apply to the hydrogen recombiners and supporting components, they did not adequately demonstrate that these components were not within the scope of license renewal in accordance with 10 CFR 54.4(a)(2). Specifically, although ample time is available to effect hydrogen control, 10 CFR 54.4 does not explicitly permit components required for accident mitigation to be excluded from the scope of license renewal on that basis. In addition, although the response states that sufficient time exists to ensure that all components of the recombiner system are operable before its operation is required, UFSAR Section 6.2.5.2.2 indicates that the majority of the lines associated with this system cannot be repaired due to the high radiation rates present during post-accident conditions. As described further in Section 2.3.2.5.2 of this SER, the applicant has transmitted a revised draft response to these RAIs that would bring within scope the components of the hydrogen recombiner system that are necessary to fulfill the hydrogen control intended function. Pending the applicant's formal submittal of this information and the NRC staff's review of the acceptability of the aging management results for the components that would be added within scope, RAIs 2.3.2.5-1, 2.3.2.5-2, and 2.3.2.5-3 are considered to be Confirmatory Item 2.3.2.5-1.

Confirmatory Item 2.3.2.5-1 Response:

Other than the safety related containment isolation piping and components, the Post Accident Hydrogen System (PAHS) is comprised of non-safety related permanently installed piping and components, and the external hydrogen recombiner and connecting temporary flexible piping.

As stated in the response to RAI Clarification F in RNP letter RNP-RA/03-0074, dated June 13, 2003, the safety related PAHS containment isolation components required for containment isolation are already in scope, as discussed in the RNP License Renewal Application (LRA). Also, the balance of the permanently installed hydrogen recombiner piping is considered to be in scope to perform the function associated with post-Loss of Coolant Accident (LOCA) hydrogen concentration control. In addition, RNP has placed the recombiner, associated temporary flexible piping, and passive components required to open the PAHS containment isolation valves in scope for license renewal.

Portions of the non-safety related permanently installed piping are carbon steel, while the balance of the piping and recombiner skid components are constructed of stainless steel. The nitrogen cylinders are carbon steel with copper alloy outlet valves and pressure regulators, with stainless steel tubing connecting to the PAHS containment isolation valves aluminum and carbon steel valve actuators.

As noted in the response to RAI Clarification F, carbon steel connecting piping within the Reactor Auxiliary Building will be included in the Boric Acid Corrosion Program to manage aging effects associated with the potential for boric acid corrosion of the carbon steel portion of the permanently installed PAHS. Susceptible nitrogen pressure boundary components required to open the PAHS containment isolation valves will also be included in the Boric Acid Corrosion Program to manage aging effects associated with the potential for boric acid corrosion. The stainless steel recombiner and flexible piping components do not require aging management as there are no anticipated aging effects associated with stainless steel in the air-gas environments (both internal and external) in which these components are located. LRA Tables 2.3-6, 3.2-1, and 3.2-2 are revised to reflect this response.

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

Post Accident Hydrogen System		
Component/Commodity	**Intended Function**	**AMR Results**
Closure Bolting	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.2-1, Item 11
Valves, Piping, and Fittings	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered.	Table 3.2-1, Item 11 Table 3.2-2, Item 3 Table 3.2-2, Item 9 Table 3.2-2, Item 13 Table 3.2-2, Item 15

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.

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	
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.
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.

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
15. Valves, Tubing, and Fittings	Copper alloy	Indoor – Not Air Conditioned, Air and Gas	None	None Required	The RNP AMR determined that these components have no aging effects requiring management based on their location. The applicable RNP environment does not promote concentration of contaminants or include exposure to aggressive chemical species.

Confirmatory Item 3.0.3.2.2-1:

(commitment inspections for the steam generator upper shell-to-transition cone weld)

Confirm that CP&L will commit to performing augmented inspections of the steam generator upper shell-to-transition cone weld during the two 10-year inservice inspection intervals for the extended period of operation for RNP.

Confirmatory Item 3.0.3.2.2-1 Response:

RNP will continue to perform examinations of the steam generator transition girth welds as required by ASME Section XI during the period of extended operation.

Confirmatory Item 3.1.2.1-1, Parts 1 and 2:

(issued with regard to the staff's assessment of AMR Item No. 22 of LRA Table 3.1-1, as evaluated in Section 3.1.2.1 of the SER)

The staff seeks confirmation as to whether or not there is any plant-specific or generic industry experience that supports the conclusion that crack initiation and growth due to stress corrosion cracking (SCC) is an applicable aging effect for carbon steel bolting materials in the RCS. If industry experience does support that crack initiation and growth due to SCC is an applicable aging effect for carbon steel bolting, the applicant should propose an AMP to manage this effect. This is Confirmatory Item 3.1.2.1-1, Part 1.

The applicant's response to RAI 3.1.2.1-3 states that stress relaxation is not applicable to valve closure bolting in the reactor coolant pressure boundary (i.e., RCPB valve bolting) and "other closure bolting in high pressure and high temperature systems." However, the applicant's discussion for AMR 22 to LRA Table 3.1-1 states that the Bolting Integrity Program is applicable to all RCPB bolting except reactor vessel studs for which the Reactor Head Closure Studs Program applies, and that 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 for Class 2 bolting greater than 2 inches in diameter. The applicant's discussion of AMR 22 in LRA Table 3.1-1 did not indicate that the applicant was exempting stress relaxation as an applicable aging effect for the RCPB valve bolting or "other closure bolting in high pressure and high temperature systems." Therefore, the staff concludes that the applicant's response to RAI 3.1.2.1-3, as it pertains to the management of stress relaxation in the RCPB valve bolting or "other closure bolting in high pressure and high temperature systems," contradicts the applicant's discussion of AMR 22 in LRA Table 3.1-1. The staff requests confirmation that, other than SCC, the aging effects identified in AMR 22 to LRA Table 3.1-1 are still applicable to the RCS bolting within the scope of the commodity group, other than the steam generator primary and secondary manway and handhole bolting. The applicant must explain the contradiction in the RAI response and the information in AMR 22 of LRA Table 3.1-1. This is Confirmatory Item 3.1.2.1-1, Part 2.

Confirmatory Item 3.1.2.1-1, Part 1 Response:

The RNP Aging Management Review (AMR) has not identified plant-specific or generic industry experience which supports a conclusion that crack initiation and growth due to Stress Corrosion Cracking (SCC) is an applicable aging effect for carbon steel or low-alloy steel bolting materials in the reactor coolant system (RCS). This is supported by operating experience and existing data which indicate that SCC failure should not be a significant issue for closure bolting within the RCS.

Confirmatory Item 3.1.2.1-1, Part 2 Response:

RAI 3.1.2.1-3 requested a technical basis for concluding why loss of preload due to stress relaxation is not applicable to steam generator manways. Accordingly, the response to RAI 3.1.2.1-3 relative to "loss of pre-load due to stress relaxation" should have been applied to the steam generator manways only. RNP confirms that, other than SCC, the aging effects identified in AMR 22 to LRA Table 3.1-1 are still applicable to the RCS bolting within the scope of the commodity group.

Confirmatory Item 3.1.2.1-1, Part 3:

(issued with regard to the staff's assessment of AMR Item No. 22 of LRA Table 3.1-1, as evaluated in Section 3.1.2.1 of the SER)

In its response to RAI 3.1.2.1-3, the applicant states that it recognizes that stress relaxation can occur in the SG manway and handhole bolting, at least for the bolting on the secondary side of the SGs, and states that it has a bolting and torque program to determine the closure and torque requirements for reactor cooling system (RCS) closure bolting. However, in its response to RAI 3.1.2.1-3, the applicant did not identify loss of preload as an aging effect and did not identify an aging management program (AMP) to manage the aging effect associated with SG bolting. GALL IV.D.1.1.7 identifies that loss of pre-load due to stress relaxation is an aging effect for the steam generator secondary manway and handhole bolting, and GALL XI.M18, "Bolting Integrity," is the AMP to manage this aging effect. According to 10 CFR 54.21(1), license renewal applicants must perform AMRs and identify all applicable aging effects for passive components within the scope of license renewal. The SG primary and secondary manway and handhole bolts are passive components within the scope of license renewal. The applicant has stated that stress relaxation is an applicable aging effect for the SG secondary manway and handhole bolting; therefore, the applicant is required by 10 CFR 54.21(a)(3) to propose an AMP to manage the aging effect. The staff also requests the applicant to provide technical justification as to why loss of preload stress relaxation does not have to be managed for the primary SG manway bolts in the manner required for the management of the SG secondary side bolting. In subsequent discussions with the NRC staff to resolve this issue, the applicant stated that the RNP bolting integrity program in LRA Section B.3.4 will be applied to the pressure retaining bolting for the primary and secondary side of the steam generators because the RNP bolting integrity program can be relied upon to prevent the loss of preload and that the RNP bolting integrity program will not take exception to the Scope of Program in GALL XI.M18, "Bolting Integrity". The staff evaluates the RNP bolting integrity program in Section 3.0.3 of this SER. The staff finds the applicant's resolution of the issue acceptable because the applicant credits its bolting integrity program to manage loss of preload due to stress relaxation in the SG primary and secondary manway and handhole bolts. However, the applicant needs to submit its resolution under oath and affirmation; therefore, this is Confirmatory Item 3.1.2.1-1, Part 3.

Confirmatory Item 3.1.2.1-1, Part 3 Response:

GALL relies upon the GALL XI.M18 Bolting Integrity Program to manage the effects of loss of preload/stress relaxation (when identified as in GALL IV.D.1.1.7).

The scope of the GALL XI.M18 Bolting Integrity Program is described in element 1, Scope of Program:

> "The program covers all bolting within the scope of license renewal including safety-related bolting, bolting for NSSS component supports, bolting for other

pressure retaining components, and structural bolting. The program covers both greater than and smaller than 2-in. diameter bolting."

The RNP Bolting Integrity Program, as described in LRA B.3.4, is consistent with GALL XI.M18 with two exceptions:

- *Scope of Program:* The Bolting Integrity Program is not utilized to address aging management requirements for structural bolting. This is not considered to be an exception with respect to mechanical system closure bolting.
- *Parameters Monitored/Inspected:* GALL specifies that high strength bolting used in NSSS component supports be inspected to the requirements for Class 1 components, Examination Category B-G-1. Exception is taken to these requirements, since bolting in this application has been evaluated to be not susceptible to SCC owing to its location in a benign environment. Similarly, exception is taken regarding requirements for subjecting this bolting to an ongoing program for crack monitoring.

The RNP Bolting Integrity Program is applied to pressure retaining bolting for the primary and secondary side of the SG (no exception to Scope of Program was taken). The RNP Bolting Integrity Program can be relied upon to prevent the loss of preload.

Confirmatory Item 3.1.2.1-2:

(issued with regard to the staff's assessment of AMR Item No. 26 of LRA Table 3.1-1, as evaluated in Section 3.1.2.1 of the SER)

In order to provide reasonable assurance that general corrosion is not an applicable aging effect for the Class 1 carbon steel or low steel components in containment air or indoor air environments, the staff seeks confirmation that the Class 1 carbon steel or lower alloy steel components operate at temperatures that are equivalent to or hotter than the ambient temperature for the surrounding containment air or indoor air environments. This is Confirmatory Item 3.1.2.1-2.

Confirmatory Item 3.1.2.1-2 Response:

RNP confirms that Class 1 carbon steel or low alloy steel components operate at temperatures that are equivalent to or hotter than the ambient temperature for the surrounding containment air or indoor air environments.

Confirmatory Item 3.1.2.1-3, Parts 1 and 2:

(issued with regard to the staff's assessment of AMR Item No. 31 of LRA Table 3.1-1, as evaluated in Section 3.1.2.1 of the SER)

The staff seeks confirmation that the reactor vessel (RV) thermal shield is adjacent to the fuel zone region of the RV, receives a neutron fluence greater than $1x10^{17}$ n/cm^2, is within the scope of the commodity group in AMR 31 to LRA Table 3.1-1, and will be managed by the Pressurized Water Reactor Internal Program. This is Confirmatory Item 3.1.2.1-3, Part 1.

The staff seeks confirmation whether or not the RV internal lower support and lower support plate columns are fabricated from cast austenitic stainless steel (CASS) materials and are within the scope of AMRs (i.e., within the scope of AMR Item 8 of LRA Table 3.1-1, AMR Item 33 of LRA Table 3.1-1, and AMR Item 14 of LRA Table 3.1-2). This is Confirmatory Item 3.1.2.1-3, Part 2.

Confirmatory Item 3.1.2.1-3, Parts 1 and 2 Response:

This response is related to the responses to RAI 3.1.2.1-9, Parts 1 and 2, RAI B.4.3-1, and RAI B.4.3-2 in RNP letter RNP-RA/03-0031, dated April 28, 2003.

Part 1

The reactor vessel thermal shield is adjacent to the fuel region of the reactor vessel; its projected neutron fluence will exceed 10^{17} n/cm^2. The reactor vessel thermal shield is specifically within the scope of Table 3.1-1, AMR Item 1, AMR Item 8, and AMR Item 33. It is not specifically within the scope of Table 3.1-1, AMR Item 31. However, this component is managed by the same Pressurized Water Reactor (PWR) Vessel Internals Program that is referenced by that AMR item.

In Subsection A.3.1.30 of the LRA, PWR Vessel Internals Program, as revised by subsequent responses to RAI B.4.3-2, RNP commits to the following for the PWR Vessel Internals Program:

> *"The Pressurized Water Reactor (PWR) Vessel Internals Program is a new program that will incorporate the following-(1) RNP will continue to participate in industry programs to investigate aging effects and determine the appropriate AMP activities to address baffle and former assembly issues, and to address change in dimensions due to void swelling, (2) as Westinghouse Owners Group and Electric Power Research Institute MRP research projects are completed, RNP will evaluate the results and factor them into the PWR Vessel Internals Program as appropriate, and (3) RNP will implement an augmented inspection during the license renewal term. Augmented inspections, based on required program enhancements resulting from industry programs, will become part of the*

> *ASME Boiler & Pressure Vessel Code, Section XI program. Corrective actions for augmented inspections will be developed using repair and replacement procedures equivalent to those requirements in ASME Boiler & Pressure Vessel Code, Section XI. RNP will submit, for review and approval, its inspection plan for the PWR Vessel Internals Program, as it will be implemented from the applicant's participation in industry initiatives, 24 months prior to the augmented inspection."*

<u>Part 2</u>

The reactor vessel lower support is equivalent to GALL Item IV.B2.5.3. As stated on page 2.3-6 of the RNP LRA:

> *"However, the lower support forging is A182 Type 304."*

The lower support columns are equivalent to GALL Item IV.B2.5.4. They are described on page 2.3-7 of the RNP LRA:

> *"Support column assemblies are stainless steel, A213, A276, or A182 Type 304. Control rod guide tube assemblies are A240, A249, and A479 Type 304. CASS (A351 Gr. CF8) components are the upper support tube base, lower support plate columns, and bottom mounted instrumentation column cruciform."*

The cast austenitic stainless steel (CASS) lower support columns are within the scope of Table 3.1-1, AMR Item 8, AMR Item 33, and Table 3.1-2, AMR Item 14. The lower support forging is within the scope of Table 3.1-1, AMR Item 8 and AMR Item 33. The lower support forging is not within the scope of Table 3.1-2, AMR Item 14, because this component is not fabricated from CASS.

Confirmatory Item 3.1.2.2.7-1:

(issued with regard to the staff's assessment of AMR Item No. 9 of LRA Table 3.1-1, as evaluated in Section 3.1.2.2.7 of the SER)

The staff seeks confirmation that the welds used to join the SG instrumentation nozzles to the SG shells were fabricated using Alloy 600 weld material (i.e., Alloy 82/182 filler metals). If Alloy 600 weld materials are utilized, the applicant should discuss whether the welds are within the scope of and managed by the Nickel-Alloy Nozzles and Penetrations Program. This is Confirmatory Item 3.1.2.2.7-1.

Confirmatory Item 3.1.2.2.7-1 Response:

The welds used to join the carbon steel shell to the carbon steel nozzles were not fabricated from Alloy 600 weld material.

Confirmatory Item 3.1.2.4.4.3-1:

(issued with regard to the staff's assessment of AMR Item No. 10 to LRA Table 3.1-2, as evaluated in Section 3.1.2.4.4.3 of the SER)

The staff seeks confirmation that CP&L is crediting the Nickel-Alloy Nozzles and Penetrations Program as an additional AMP for managing primary water stress corrosion cracking (PWSCC) in the RNP bottom head instrumentation tube nozzles. This is Confirmatory Item 3.1.2.4.4.3-1.

Confirmatory Item 3.1.2.4.4.3-1 Response:

This response is related to the responses to RAI 3.1.2.4.4-1, RAI B.4.1-1, and RAI B.4.1-2 in RNP letter RNP-RA/03-0031, dated April 28, 2003, and RAI Clarification G (RAI B.4.1-1, RAI 3.1.2.1-4, RAI 3.1.2.1-5) in RNP letter RNP-RA/03-0074, dated June 13, 2003.

In the response to RAI Clarification G, RNP amended part (3) of the commitment associated with the Nickel-Alloy Nozzles and Penetration Program to the following:

> *"(3) RNP will maintain its involvement in industry initiatives and will implement any actions, unless impracticable, that are agreed upon between the NRC and the nuclear power industry to monitor for, detect, evaluate, and correct cracking in the VHP nozzles, specifically as the actions relate to ensuring the integrity of VHP nozzles in the RNP upper reactor vessel head during the extended period of operation."*

RNP will add items detailed in Table 3.1-2, AMR Item 10, to the program if required by the results of the commitment stated above. Also, note that in the response to RAI Clarification G, RNP also agreed to submit, for review and approval, its inspection plan for the Nickel-Alloy Nozzles and Penetrations Program, as it will be implemented from participation in industry initiatives prior to July 31, 2009.

Confirmatory Item 3.1.2.4.5.2-1:

(issued with regard to the staff's assessment of AMR Item No. 9 to LRA Table 3.1-2, as evaluated in Section 3.1.2.4.5.2 of the SER)

The staff seeks confirmation that CP&L is crediting the Nickel-Alloy Nozzles and Penetrations Program as an additional AMP for managing PWSCC in the RV core support pads. This is Confirmatory Item 3.1.2.4.5.2-1.

Confirmatory Item 3.1.2.4.5.2-1 Response:

This response is related to the response to RAI 3.1.2.4.5-1, B.4.1-1, and B.4.1-2 in RNP letter RNP-RA/03-0031, dated April 28, 2003, and RAI Clarification G (RAI B.4.1-1, RAI 3.1.2.1-4, RAI 3.1.2.1-5) in RNP letter RNP-RA/03-0074, dated June 13, 2003.

In the response to RAI Clarification G, RNP amended part (3) of the commitment associated with the Nickel-Alloy Nozzles and Penetration Program to the following:

> *"(3) RNP will maintain its involvement in industry initiatives and will implement any actions, unless impracticable, that are agreed upon between the NRC and the nuclear power industry to monitor for, detect, evaluate, and correct cracking in the VHP nozzles, specifically as the actions relate to ensuring the integrity of VHP nozzles in the RNP upper reactor vessel head during the extended period of operation."*

RNP will add items detailed in Table 3.1-2, AMR Item 9, to the program if required by the results of the commitment stated above. Also, note that in the response to RAI Clarification G, RNP also agreed to submit, for review and approval, its inspection plan for the Nickel-Alloy Nozzles and Penetrations Program, as it will be implemented from participation in industry initiatives prior to July 31, 2009.

Confirmatory Item 3.3.2.3.3-1 :

(confirmation that the diesel and motor driven fire pumps are overhauled on a 10-year cycle and this overhaul includes inspection of the bowls)

During the AMR inspection (June 9–13, 2003), the staff reviewed the applicant's replacement frequency for fire pump casings for the Fire Protection Program, see LRA Table 3.3-2, Item 30. The audit noted that there is an error in the application and the fire pumps do not have casings, rather the vertical shaft pumps used at RNP use bowls for the pressure boundary function. Furthermore, the inspection indicated that these bowls are not replaced on a 10-year cycle, rather the pumps are overhauled on a 10-year cycle. Overhaul does not specifically require replacement of the bowls. The applicant explained during a phone call on June 12, 2003, that the frequency of the overhaul of the fire pumps is consistent with OE and that the current Preventive Maintenance Program is effective at ensuring the pumps remain operable during a 10-year service between overhauls. A Confirmatory Item 3.3.2.3.3-1 will be included for the applicant to confirm that the diesel and motor driven fire pumps are overhauled on a 10-year cycle and this overhaul includes inspection of the bowls (i.e., the pressure retaining portion of the pump), and the bowls may or may not be replaced based upon their condition.

Confirmatory Item 3.3.2.3.3-1 Response:

The diesel and motor driven fire pumps are overhauled on a 10-year cycle and this overhaul includes inspection of the bowls.

Clarification of the aging management program is shown in the following updated LRA table:

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
30. Diesel and Motor Driven Fire Pumps	Carbon Steel	Raw Water	Loss of Material from General Corrosion	Preventive Maintenance Program	Based on RNP operating experience, the Diesel and Motor Driven Fire Pumps are refurbished every 10 years in accordance with the Preventive Maintenance Program. Included in this activity is inspection of the pump bowls. This activity is used to manage degradation caused by corrosion of the external surface of the pumps in the "splash zone."

<u>Confirmatory Item 3.1.2.4.5.5-1:</u>

(Nickel-based alloy incore flux thimbles tubes)

The staff seeks confirmation that the scope of AMR 16 of LRA Table 3.1-2 is for nickel-based alloy incore flux thimbles tubes and not the retractable incore flux thimbles. An inspection-based program should be used in conjunction with the Water Chemistry Program to manage SCC in these components and therefore the staff also seeks confirmation that the applicant will credit both the PWR Vessel Internals Program and the Water Chemistry Program to manage SCC (including PWSCC and/or IASCC) in the nickel-based alloy incore flux thimble tubes. This is Confirmatory Item 3.1.2.4.5.5-1.

<u>Confirmatory Item 3.1.2.4.5.5-1 Response:</u>

This response revises the response to RAI 3.1.2.4.5-2, provided by RNP letter RNP-RA/03-0031, dated April 28, 2003.

The incore flux thimble tubes are described in the RNP LRA on page 2.3-6. It states that:

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

This component is evaluated in LRA Table 3.1-1, Item 1; Table 3.1-1, Item 28; Table 3.1-2, Item 2; and, Table 3.1-2, Item 16.

The aging management programs credited in the RNP LRA only provide inspections for loss of material due to wear (Table 3.1-1, Item 28, credits the Flux Thimble Eddy Current Inspection Program).

The AMR for Table 3.1-2, Item 16, is revised such that cracking due to SCC will be managed by the Water Chemistry Program and the PWR Vessel Internals Program.

Confirmatory Item 3.6.2.3.1.2-1:

(non-EQ insulated cables and connections program)

In LRA Section B.4.6, "Non-EQ Insulated Cables and Connections Program," the applicant described its AMP to manage aging in Non-EQ insulated cables and connections. The LRA stated that this AMP is consistent with GALL AMPs XI.E1, "Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements," with no deviations. In response to the staff's concern (RAI B.4.6-2) about excluding non-PVC cables inside and outside containment in adverse localized environment from the sample, the applicant in a letter dated June 13, 2003, stated that the scope of this program includes plant cables and connections of various insulation material types (not just PVC) that may be located in an adverse, localized environment. On the basis of its review, the staff finds that its concern is not resolved. In subsequent discussions with the NRC staff to resolve this issue, the applicant stated that the statement in LRA Section B.4.6 regarding "The sample locations will consider the location of PVC cables inside and outside containment as well as any known adverse localized environments, (PVC was determined to be the limiting insulation material)" will be modified by "The sample locations will consider the location of cables and connections inside and outside containment as well as any known adverse localized environments." The staff finds that the applicant's resolution of this issue as acceptable because the sample will consider all insulation material types used inside and outside containment as well as any known adverse localized environments. However, the applicant needs to submit its resolution under oath and affirmation; therefore, this is Confirmatory Item 3.6.2.3.1.2-1

Confirmatory Item 3.6.2.3.1.2-1 Response:

The LRA Section B.4.6 is revised to read as follows:

> *"The Non-EQ Insulated Cables and Connections Program is credited for aging management of cables and connections not included in the RNP EQ Program.*
>
> *The aging effects/mechanisms of concern are as follows:*
>
> - *Reduced Insulation Resistance*
> - *Electrical Failure*
>
> *The technical basis for selecting a sample of cables to be inspected will be defined prior to the period of extended operation. The sample locations will consider the location of cables and connections inside and outside containment, as well as any known adverse localized environments."*

Commitment 36 is revised to delete "polyvinyl chloride-insulated."

Confirmatory Item 3.6.2.3.2.2-1:

(AMP for non-EQ electrical cables used in instrumentation circuits (B.4.7))

Operating experience shows that changes in instrument calibration data can be caused by degradation of the circuit cable and are a possible indication of potential cable degradation. The staff finds that the applicant did not address the operating experience in the formal response. In subsequent discussions with the NRC staff to resolve this issue, the applicant stated that this element will be revised to address the operating experience as follows: Industry operating experience indicates that changes in instrument calibration data can be caused by degradation of the circuit cable and are a possible indication of potential cable degradation. This program is for the non-EQ portions of the high range radiation monitoring cabling systems. These cabling systems are located in non-harsh environments and none have experienced age related degradation. The staff finds that the applicant's resolution of the open item is acceptable because the applicant adequately addressed the operating experience. However, the applicant needs to submit its resolution under oath and affirmation; therefore, this is Confirmatory Item 3.6.2.3.2.2-1.

Confirmatory Item 3.6.2.3.2.2-1 Response:

The Operating Experience section of the aging management program for Non-EQ Electrical Cables Used in Instrumentation Circuits described in Attachment 2 to the response to RAI 3.6.1-2 submitted in RNP letter RNP-RA/03-0031, dated April 28, 2003, is modified to read as follows:

"Operating Experience
Industry operating experience indicates that changes in instrument calibration data can be caused by degradation of the circuit cable and are a possible indication of potential cable degradation. This program is for the non-EQ portions of the high range radiation monitoring cabling systems. These cabling systems are located in non-harsh environments and none have experienced age related degradation."

Confirmatory Item 3.6.2.3.2.2-2:

(AMP for neutron flux instrumentation (B.4.8))

To detect aging effects, the cables used in neutron flux instrumentation circuits will be tested at least once every 10 years. Testing may include insulation resistance tests, TDR tests, I/V testing, or other testing judged to be effective in determining cable insulation condition. Following issuance of a renewed operating license for RNP, the initial test will be completed before the end of the initial 40-year license term for Unit 2 (July 31, 2010). The staff finds that this testing is acceptable because the testing will determine cable insulation resistance (potential degradation); however, the staff is concerned about the 10-year testing frequency. This last concern remained as open issue. In subsequent discussions with the NRC staff to resolve this issue, the applicant stated that a review of site operating experience found no age related failures for neutron monitoring cables or connectors. The only industry operating experience identified for these cables was Westinghouse Technical Bulletin 86-01. This Bulletin identified industry concerns with cables used for the source range detector regarding cable degradation due to high operating voltage, radiation, heat, and moisture. Both the source range and intermediate range detector cables inside containment were replaced in 1991 as a result of that bulletin. These cables had operated for 20 years without failure prior to being replaced. The replacement cables were manufactured to Class 1E standards and have remained functional during the last twelve years. The power range cables are the original installed cables and are the same cable type (Amphenol/Essex 21-529) that was originally used in the source range and intermediate range circuits. They have operated for over 32 years without failure, which demonstrates their ability to operate over long periods without a loss of intended function.

In addition, the licensee stated that initial testing of all in-scope neutron monitoring cables will be performed prior to the end of the current license term. This testing will provide a positive means of detecting any significant aging that has occurred since the cables were installed, which in the case of the power range cables will be after 33-40 years of operation. Given the operating experience of these cables and the gradual nature of cable insulation aging, the 10 year testing frequency subsequent to the initial testing provides reasonable assurance that the cables will continue to perform their intended function. The staff finds that the applicant's resolution of the issue is acceptable because the cable insulation degradation is a slow process and RNP operating experience did not identify any cable insulation degradation. Additionally, this 10 year frequency is consistent with NUREG-1801 cable aging management programs frequency. However, the applicant needs to submit its resolution under oath and affirmation; therefore, this is Confirmatory Item 3.6.2.3.2.2-2.

Confirmatory Item 3.6.2.3.2.2-2 Response:

A review of site operating experience found no age related failures for neutron monitoring cables or connectors. The only industry operating experience identified for these cables was Westinghouse Technical Bulletin 86-01. This bulletin identified industry concerns with cables used for the source range detector regarding cable degradation due to high operating voltage, radiation, heat, and moisture. Both the source range and intermediate range detector cables inside the containment were replaced in 1991 as a result of that technical bulletin. These cables had operated for 20 years without failure prior to being replaced. The replacement cables were manufactured to Class 1E standards and have remained functional during the last twelve years.

The power range cables are the original installed cables and are the same cable type (Amphenol / Essex 21-529) that was originally used in the source range and intermediate range circuits. They have operated for over 32 years without failure, which demonstrates their ability to operate over long periods without a loss of intended function.

Initial testing of all in-scope neutron monitoring cables will be performed prior to the end of the current license term. This testing will provide a positive means of detecting any significant aging that has occurred since the cables were installed, which, in the case of the power range cables, will be after 33 to 40 years of operation. Given the operating experience of these cables and the gradual nature of cable insulation aging, the 10 year testing frequency subsequent to the initial testing provides reasonable assurance that the cables will continue to perform their intended function. This conclusion is consistent with the 10 year frequency established for NUREG-1801 cable aging management programs XI.E1, Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements, and XI.E3, 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.

The Operating Experience section of the aging management program for neutron monitoring circuitry described in Attachment 2 to the response to RAI 3.6.1-2 submitted in RNP letter RNP-RA/03-0031, dated April 28, 2003, is modified to read as follows:

"Operating Experience
Exposure of electrical cables and connectors to adverse localized environments caused by heat, radiation, or moisture can result in reduced IR. Industry operating experience has shown that the vast majority of failures have occurred near the reactor vessel. This program is for non-EQ neutron monitoring cabling systems. A review of site operating experience found no age related failures for neutron monitoring cables or connectors. However, Westinghouse Technical Bulletin 86-01 did identify concerns with cables used for the source range detector regarding cable degradation due to high operating voltage, radiation, heat, and moisture. Both the source range and intermediate range detector cables inside the containment were replaced in 1991 as a result of that technical bulletin. The replacement cables have remained functional during the last twelve years. The power range cables are the original installed cables and are the same cable type (Amphenol / Essex 21-529) that was originally used in the source range and intermediate range circuits. The operating history for these cables demonstrates their reliability and provides reasonable assurance that they will continue to perform their intended function

throughout the period of extended operation."

Confirmatory Item 4.2.3-1:

(update of UFSAR Supplement in accordance with the RT_{PTS} and USE values listed in WCAP-15828)

The staff requests confirmation that, at the next update of the UFSAR Supplement for RNP, the applicant will update Sections A.4.2.1 and A.4.2.2 of Appendix A in the LRA to reference the applicability of PTS and USE analyses in WCAP-15828, Revision 0, to the 60-year PTS and USE assessments for the RNP RV beltline materials and will update the corresponding UFSAR Supplement summary descriptions to reference the RT_{PTS} and USE values listed in the report for the limiting PTS and USE materials in the beltline of the reactor vessel.

Confirmatory Item 4.2.3-1 Response:

Based on information provided in responses to RAIs 4.2.1-1 and 4.2.2-1 in RNP letter RNP-RA/03-0031, dated April 28, 2003, the UFSAR Supplement information contained in Section A.3.2.1 (Confirmatory Item should refer to A.3.2.1 and A.3.2.2) now reads:

"A.3.2.1 Reactor Vessel Neutron Embrittlement

A.3.2.1.1 Pressurized Thermal Shock

10 CFR 50.61 requires the reference temperature (RT_{PTS}) for reactor vessel beltline materials be less than the "PTS screening criteria" at the expiration date of the operating license unless otherwise approved by the NRC. The screening criteria limit the amount that the material reference temperature, RT_{PTS}, may increase following neutron irradiation.

WCAP-15828, Revision 0, provides an evaluation of PTS for RNP that incorporates the results of the surveillance Capsule X evaluation. The calculated RT_{PTS} temperatures for reactor vessel beltline materials, including plates, forgings, axial welds, inlet nozzles, outlet nozzles, and nozzle welds have been demonstrated to remain below the 270°F PTS screening criterion throughout the 60-year period of extended operation. The limiting location is Circumferential Weld Seam 10-273, which has an RT_{PTS} temperature of 297°F.

Therefore the TLAA for Pressurized Thermal Shock has been projected to the end of the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

A.3.2.1.1 Upper Shelf Energy

10 CFR Part 50, Appendix G, paragraph IV.A.1, requires that reactor vessel beltline materials have a Charpy upper-shelf energy (USE) of no less than 50 ft-lb (68 J) throughout the life of the reactor vessel unless otherwise approved by the NRC. WCAP-15828, Revision 0, Appendix A, provides an evaluation of USE for the RNP incorporating the results of the surveillance Capsule X evaluation. WCAP-15828, Appendix A, Table A-3, provides predicted end-of-extended-license (50 EFPY) USE values for the beltline region materials. The limiting value is for Upper Shell Plate W 10201-3, which has a predicted 60-year USE of 48.4 ft-lbs. This exceeds the applicable 42 ft-lbs minimum requirement from the Equivalent Margins Analysis provided in WCAP-13587, Revision 1, for this material.

Based on the foregoing discussion, the TLAA for reactor pressure vessel USE has been projected to the end of the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii)."

Confirmatory Item 4.3.2-1:

(auxiliary feedwater fatigue analysis)

In RAI 4.3-7, the staff requested the applicant to provide (1) calculated cumulative utilization factors (CUFs) of the six replacement branch connections, (2) confirmation that no other nonstandard components were used or provide justification of the acceptability for use in safety systems at RNP, and (3) description of the AMPs that will be used to provide assurance that the CUFs for these connections will not exceed the limit of 1.0 for the period of extended operation. In its response by a letter dated June 13, 2003, the applicant stated that there are three 4" to 16" auxiliary feedwater-to-feedwater connections downstream of the motor-driven and the steam-driven AFW pump. The three connections downstream from the steam-driven pump could not be qualified for the full 40-year design transient set, so a reduced number of design transients was postulated. This resulted in a CUF value of 0.99 for 40-year life. Based upon projections of actual transients to date, the qualified number of transients is not expected to be reached until approximately year 50. The applicant indicated that the number of transients used in the analysis will be tracked by the Fatigue Monitoring Program. The applicant further indicated that the components will be either reanalyzed or replaced prior to exceeding the number of transients tracked by the Fatigue Monitoring Program. The staff finds that the applicant's proposed options provide acceptable plant-specific approaches to address fatigue of the connections between the auxiliary and main feedwater lines for the period of extended operation in accordance with 10 CFR 54.21(c)(1). However, in accordance with 10 CFR 54.21(d), these options need to be included in the UFSAR Supplement (Confirmatory Item 4.3.2-1).

Confirmatory Item 4.3.2-1 Response:

UFSAR Supplement Section A.3.2.2.1 is modified to include the following:

"The three connections downstream from the steam-driven pump could not be qualified for the full 40-year design transient set, so a reduced number of design transients were postulated. The number of transients used in the analysis will be tracked by the Fatigue Monitoring Program (FMP). The components will be either reanalyzed or replaced prior to exceeding the transient limits tracked by the FMP."

Confirmatory Item 4.3.2-2:

(aging management of surge line for period of extended operation)

In RAI 4.3-10, the staff requested that the applicant provide additional clarification regarding aging management of the surge line during the period of extended operation. The applicant's June 13, 2003, response indicated that fatigue of the surge line will be managed using one or more options. Options include further refinement of the fatigue analyses to maintain the environmentally assisted fatigue (EAF)-adjusted CUF below 1.0, or repair of the affected locations, or replacement of the affected locations, or management of the effects of fatigue through the use of an augmented ISI program reviewed and approved by the NRC.

The applicant commits to provide the NRC with the details of the inspection program prior to the period of extended operation if the last option is selected. As indicated by the applicant, the use of an inspection program to manage fatigue will require prior staff review and approval. The applicant indicated that LRA Section A.3.2.2.2 would be revised to include the applicant's proposed options for managing the surge line fatigue. The staff finds the applicant's proposed options provide acceptable plant-specific approaches to address EAF of the RNP pressurizer surge line for the period of extended operation in accordance with 10 CFR 54.21(c)(1). Revision of the UFSAR Supplement is Confirmatory Item 4.3.2-2.

Confirmatory Item 4.3.2-2 Response:

UFSAR Supplement Section A.3.2.2.1 is modified to include the following:

"Fatigue of the surge line will be managed using one or more of the following options: further refinement of the fatigue analyses to maintain the EAF-adjusted CUF below 1.0, or repair of the affected locations, or replacement of the affected locations, or manage the effects of fatigue through the use of an augmented in-service inspection program that has been reviewed and approved by the NRC.

RNP will provide the NRC with the details of the inspection program prior to the period of extended operation if the last option is selected; the use of an inspection program to manage fatigue will require prior staff review and approval."

Confirmatory Item B.4.3-1:

(issued with regard to the staff's assessment of LRA Section B.4.3, PWR Vessel Internals Program, as evaluated in Section 3.1.2.3.8 of the SER)

The staff will confirm that the applicant has incorporated the commitment regarding the Nickel-Alloy Nozzles and Penetrations Program into the UFSAR Supplement summary description of Section A.3.1.30 of Appendix A to the LRA when the applicant revises its UFSAR Supplement for this AMP. This is Confirmatory Item B.4.3-1.

Confirmatory Item B.4.3-1 Response:

The program description is revised by adding a sentence at the end of the first paragraph of UFSAR Supplement Section A.3.1.28 as follows:

> *"The program includes (a) primary water stress corrosion cracking (PWSCC) susceptibility assessment to identify susceptible components, (b) monitoring and control of reactor coolant water chemistry to mitigate PWSCC, and (c) inservice inspection of reactor vessel head penetrations to monitor PWSCC and its effect on the intended function of the component. For susceptible penetrations and locations, the program includes an industry-wide, integrated, long-term inspection program based on the industry responses to NRC Generic Letter (GL) 97-01. This program includes the augmented requirements in NRC Order No. EA-03-009 for the RNP reactor vessel head and VHP nozzles."*