3.5 Containments, Structures, and Component Supports

This section addresses the aging management of the structures and structural components. The structures that make up this group are described in the following SER sections.

- Containment (2.4.1)
  —Containment Structures (2.4.1.1)
  —Containment Internal Structural Components (2.4.1.2)
  —Containment External Structural Components (2.4.1.3)

- Other Structures (2.4.2)
  —Reactor Auxiliary Building (2.4.2.1)
  —Fuel-Handling Building (2.4.2.2)
  —Turbine Building (2.4.2.3)
  —Dedicated Shutdown Diesel Generator Building (2.4.2.4)
  —Radwaste Building (2.4.2.5)
  —Intake Structures (2.4.2.6)
  —North Service Water Header Enclosure (2.4.2.7)
  —Emergency Operations Facility/Technical Support Center Security Diesel Generator Building (2.4.2.8)
  —Discharge Structures (2.4.2.9)
  —Lake Robinson Dam (2.4.2.10)
  —Pipe Restraint Tower (2.4.2.11)
  —Yard Structures and Foundations (2.4.2.12)
  —Refueling System (2.4.2.13)

As discussed in Section 3.0.1 of this SER, the structures and structural components are included in one of two LRA tables. LRA Table 3.5-1 consists of structural components that are evaluated in the GALL Report, and LRA Table 3.5-2 consists of structural components not addressed in the GALL Report.

3.5.1 Summary of Technical Information in the Application

In LRA Section 3.5, the applicant described its AMR for structural components within the containment, other Class 1 structures, and component supports at RNP. The passive, long-lived components in these structures that are subject to an AMR are identified in LRA Tables 2.4-1 through 2.4-12.

The applicant's AMRs included an evaluation of plant-specific and industry operating experience. The plant-specific evaluation included reviews of condition reports and discussions with appropriate site personnel to identify aging effects that require management. These reviews concluded that the aging effects requiring management based on RNP operating experience were consistent with aging effects identified in GALL. The applicant's review of industry operating experience included a review of operating experience through 2001. The results of this review concluded that aging effects requiring management based on industry operating experience were consistent with aging effects identified in GALL. The applicant's ongoing review of plant-specific and industry-wide operating experience is conducted in accordance with the RNP's Corrective Action and Operating Experience Programs.

3.5.2 Staff Evaluation

In Section 3.5 of the LRA, the applicant describes its AMR for structural components at RNP. The staff reviewed LRA Section 3.5 to determine whether the applicant had provided sufficient information to demonstrate that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB throughout the period of extended operation, in accordance with the requirements of 10 CFR 54.21(a)(3), for structural components that are determined to be within the scope of license renewal and subject to an AMR.

The applicant referenced the GALL Report in its AMR. The staff has previously evaluated the adequacy of the aging management of structural components for license renewal as documented in the GALL Report. Thus, the staff did not repeat its review of the items described in the GALL Report, except to ensure that the material presented in the LRA was applicable, and to verify that the applicant had identified the appropriate programs as described and evaluated in the GALL Report.

The staff evaluated those aging management issues recommended for further evaluation in the GALL Report, as well as the applicant's AMR for structural components not addressed in the GALL Report. In addition, the staff evaluated the AMPs used by the applicant to manage the aging of structural components. Finally, the staff reviewed the structural components listed in LRA Section 2.4 to determine whether the applicant properly identified the applicable aging effects and AMPs needed to adequately manage the aging effects.

Table 3.5-1 below provides a summary of the staff's evaluation of components, aging effects/mechanisms, and AMPs listed in LRA Section 3.5 that are addressed in GALL.

Table 3.5-1

Staff Evaluation for RNP Structures and Structural Components Described in the GALL Report

Common Components of All Types of PWR and BWR Containment

Component Group	Aging Effect/Mechanism	AMP in GALL Report	AMP in LRA	Staff Evaluation
Penetration sleeves penetration bellows, and dissimilar metal welds	Cumulative fatigue damage (CLB fatigue analysis exists)	TLAA evaluated in accordance with 10 CFR 54.21(c)	TLAA (4.3)	Consistent with GALL. GALL recommends further evaluation (See Section 3.5.2.2.1.6 below).

Penetration sleeves, penetrations bellows, and dissimilar metal welds	Cracking due to cyclic loading, or crack initiation and growth due to SCC	Containment ISI and Containment leak rate test	Containment ISI (B.3.13); Containment leak rate test (B.2.7); Water Chemistry Program (B.2.2) and Boric Acid Corrosion Program (B3.2)	Consistent with GALL. GALL recommends further evaluation (See Section 3.5.2.2.1.7 below).
Penetration sleeves, penetration bellows, and dissimilar metal welds	Loss of material due to corrosion	Containment ISI and containment leak rate test	Containment ISI (B.3.13); Containment leak rate test (B.2.7)	Consistent with GALL. (See Section 3.5.2.1 below).
Personnel airlock and equipment hatch	Loss of material due to corrosion	Containment ISI and containment leak rate test	Containment ISI (B.3.13); Containment leak rate test (B.2.7)	Consistent with GALL. (See Section 3.5.2.1 below).
Personnel airlock and equipment hatch	Loss of leak tightness in closed position due to mechanical wear of locks, hinges, and closure mechanism	Containment leak rate test and plant technical specifications	Containment ISI (B.3.13); Containment leak rate test (B.2.7)	Consistent with GALL. (See Section 3.5.2.1 below).
Seals, gaskets, and moisture barriers	Loss of sealant and leakage through containment due to deterioration of joint seals, gaskets, and moisture barriers	Containment ISI and containment leak rate test	Containment ISI (B.3.13); Containment leak rate test (B.2.7)	Consistent with GALL. (See Section 3.5.2.1 below).

PWR Concrete (Reinforced and Prestressed) and Steel Containment
BWR Concrete (Mark II and III) and Steel (Mark I, II, and III) Containment

Component Group	Aging Effect/Mechanism	AMP in GALL Report	AMP in LRA	Staff Evaluation
Concrete elements: foundation, walls, dome	Aging of accessible and inaccessible concrete areas due to leaching of calcium hydroxide, aggressive chemical attack, and corrosion of embedded steel	Containment ISI	Containment ISI (B.3.14)	Consistent with GALL. GALL recommends further evaluation (See Section 3.5.2.2.1.1 below).
Concrete elements: foundation	Cracks, distortion, and increases in component stress level due to settlement	Structures Monitoring	Containment ISI (B.3.14)	Consistent with GALL. (See Section 3.5.2.2.2.2 below).
Concrete elements: foundation	Reduction in foundation strength due to erosion of porous concrete subfoundation	Structures Monitoring	None	Consistent with GALL. (See Section 3.5.2.2.1.2. below).
Concrete elements: foundation, dome, and wall	Reduction of strength and modulus due to elevated temperature	Plant-specific	None	Consistent with GALL. GALL recommends further evaluation (See Section3.5.2.2.1 .3 below).
Prestressed containment: tendons and anchorage components	Loss of prestress due to relaxation, shrinkage, creep, and elevated temperature	TLAA evaluated in accordance with 10 CFR 54.21(c)	TLAA (4.5)	Consistent with GALL. GALL recommends further evaluation (See Section 3.5.2.2.1.5 below).

Steel elements: liner plate, containment shell	Loss of material due to corrosion in accessible and inaccessible areas	Containment ISI and Containment leak rate test	Containment ISI (B.3.13); Containment leak rate test (B.2.7)	Consistent with GALL. GALL recommends further evaluation (See Section 3.5.2.2.1.4 below).
Steel elements: vent header, drywell head, torus, downcomers, pool shell	Cumulative fatigue damage (CLB fatigue analysis exists)	TLAA evaluated in accordance with 10 CFR 54.21(c)	None	BWR
Steel elements: protected by coating	Loss of material due to corrosion in accessible areas only	Protective coating monitoring and maintenance	None	Not applicable to RNP
Prestressed containment: tendons and anchorage components	Loss of material due to corrosion of prestressing tendons and anchorage components	Containment ISI	None	Not applicable to RNP
Concrete elements: foundation, dome, and wall	Scaling, cracking, and spalling due to freeze-thaw; expansion and cracking due to reaction with aggregate	Containment ISI	Containment ISI (B.3.14)	Consistent with GALL. (See Section 3.5.2.1 below).
Steel elements: vent line bellows, vent headers, downcomers	Cracking due to cyclic loads or crack initiation and growth due to SCC	Containment ISI and Containment leak rate test	None	BWR
Steel elements: Suppression chamber liner	Crack initiation and growth due to SCC	Containment ISI and Containment leak rate test	None	BWR

Steel elements: drywell head and downcomer pipes	Fretting and lock up due to wear	Containment ISI	None	BWR

Class I Structures

Component Group	Aging Effect/Mechanism	AMP in GALL Report	AMP in LRA	Staff Evaluation
All Groups except Group 6: accessible interior/exterior concrete & steel components	All types of aging effects	Structures Monitoring	Structures Monitoring Program (B.3.15)	Consistent with GALL. (See Section 3.5.2.2.2.1 below).
Groups 1-3, 5, 7-9: inaccessible concrete components, such as exterior walls below grade and foundation	Aging of inaccessible concrete areas due to aggressive chemical attack and corrosion of embedded steel	Plant-specific	Structures Monitoring Program (B.3.15)	Consistent with GALL. GALL recommends further evaluation (See Section 3.5.2.2.2.2 below).
Group 6: all accessible/inaccessible concrete, steel, and earthen components	All types of aging effects, including loss of material due to abrasion, cavitation, and corrosion	Inspection of water-control structures or FERC/US Army Corps of Engineers dam inspections and maintenance	Dam Inspection Program	Consistent with GALL. (See Section 3.5.2.1 below).
Group 5: liners	Crack initiation and growth from SCC and loss of material due to crevice corrosion	Water Chemistry and monitoring of spent fuel pool water level	Water Chemistry Program and Monitoring of spent fuel pool water level per RNP Technical Specifications	Consistent with GALL. (See Section 3.5.2.1 below).

Group 1-3, 5, 6: all masonry block walls	Crack due to restraint, shrinkage, creep, and aggressive environment	Masonry Wall	Structures Monitoring Program (B.3.15)	Consistent with GALL. (See Section 3.5.2.1 below).
Group 1-3, 5, 7-9: foundation	Cracks, distortion, and increases in component stress level due to settlement	Structures Monitoring	Structures Monitoring Program (B.3.15)	Consistent with GALL. (See Section 3.5.2.2.1.2 below).
Group 1-3, 5-9: foundation	Reduction in foundation strength due to erosion of porous concrete subfoundation	Structures Monitoring	None	Consistent with GALL. (See Section 3.5.2.2.1.2 below).
Group 1-5: concrete	Reduction of strength and modulus due to elevated temperature	Plant-specific	None	Consistent with GALL. GALL recommends further evaluation. (See Sections 3.5.2.2.1.3 and 3.5.2.4.2.2 below.)
Groups 7, 8: liners	Crack initiation and growth due to SCC and loss of material due to crevice corrosion	Plant-specific	None	Not applicable to RNP

Component Supports

Component Group	Aging Effect/Mechanism	AMP in GALL Report	AMP in LRA	Staff Evaluation
All Groups: support members: anchor bolts, concrete surrounding anchor bolts, welds, grout pad, bolted connections, etc.	Aging of component support	Structures Monitoring	Structures Monitoring Program (B.3.15)	Consistent with GALL. (See Section 3.5.2.2.3.1 below).
Groups B1.1, B1.2, and B1.3: support members: anchor bolts, welds	Cumulative fatigue damage (CLB fatigue analysis exists)	TLAA evaluated in accordance with 10 CFR 54.21(c)	None	Consistent with GALL. GALL recommends further evaluation (See Section 3.5.2.2.3.2 below).
All Groups: support members: anchor bolts, welds	Loss of material due to boric acid corrosion	Boric Acid Corrosion	Boric Acid Corrosion Program (B.3.2)	Consistent with GALL. (See Section 3.5.2.1 below).
Groups B1.1, B1.2, and B1.3: support members: anchor bolts, welds, spring hangers, guides, stops, and vibration isolators	Loss of material due to environmental corrosion and loss of mechanical function due to corrosion, distortion, dirt, overload, etc.	ISI	ASME Section XI, Subsection IWF Program (B.2.6)	Consistent with GALL. (See Section 3.5.2.1 below).
Group B1.1: high strength low-alloy bolts	Crack initiation and growth due to SCC	Bolting Integrity	None	Consistent with GALL. (See Section 3.5.2.1 below).

The staff's review of the structural components for the RNP LRA is contained within four sections of this SER. Section 3.5.2.1 is the staff's review of structures and structural components that the applicant indicated are consistent with GALL and do not require further evaluation. Section 3.5.2.2 is the staff's review of structures and structural components that the applicant indicates are consistent with GALL, and for which GALL recommends further evaluation. Section 3.5.2.3 is the staff evaluation of the AMPs that are specific to the aging management of structural components. Section 3.5.2.4 contains an evaluation of the adequacy of aging management for components in each structure and includes an evaluation of structures and structural components that the applicant indicates are not in GALL.

3.5.2.1 Aging Management Evaluations in the GALL Report That Are Relied On for License Renewal, Which Do Not Require Further Evaluation

For component groups evaluated in the GALL Report for which the applicant has claimed consistency with GALL, and for which GALL does not recommend further evaluation, the staff sampled components in these groups to determine whether the plant-specific components contained in these GALL component groups were bounded by the GALL evaluation. The staff also sampled component groups to determine whether the applicant had properly identified those component groups in the GALL Report that were not applicable to its plant.

On the basis of this review, the staff has determined that the applicant's basis of managing aging effects associated with structures and structural components is consistent with GALL.

3.5.2.2 Aging Management Evaluations in the GALL Report That Are Relied On for License Renewal, For Which GALL Recommends Further Evaluation

For component groups evaluated in GALL for which the applicant has claimed consistency with GALL, and for which GALL recommends further evaluation, the staff reviewed the applicant's evaluation to determine whether it adequately addressed the issues for which GALL recommended further evaluation. In addition, the staff sampled components in these groups to determine whether the plant-specific components contained in these GALL component groups were bounded by the GALL evaluation.

The GALL Report indicates that further evaluation should be performed for the component groups described in the following sections.

3.5.2.2.1 Containments

3.5.2.2.1.1 Aging of Inaccessible Concrete Areas

As stated in the SRP-LR, the GALL Report recommends further evaluation to manage the aging effects for containment concrete components located in inaccessible areas, if certain aging mechanisms, including (1) leaching of calcium hydroxide, (2) aggressive chemical attack, or (3) corrosion of embedded steel, are significant. Possible aging effects for containment concrete structural components due to these three aging mechanisms are cracking, change in material properties, and loss of material.

The AMP recommended by the GALL Report for managing the above aging effects for containment concrete components in accessible portions of the containment structures is the ASME Section XI, Subsection IWL (XI.S2) Program. The staff's evaluation of the applicant's ASME Section XI, Subsection IWL Program is found in Section B.3.14 of this SER.

Subsection IWL exempts from examination those portions of the concrete containment that are inaccessible (e.g., foundation, below-grade exterior walls, concrete covered by liner). For inaccessible portions of the containment structure, 10 CFR 50.55a(b)(2)(ix) requires that the licensee evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to inaccessible areas.

The applicant addressed the specific criteria defined in the GALL Report regarding the need for further evaluation to manage the potential aging of containment concrete structural components in inaccessible areas in LRA Table 3.5-1. The GALL Report recommends further evaluation for containment concrete in inaccessible areas if certain aging mechanisms, including (1) leaching of calcium hydroxide, (2) aggressive chemical attack, or (3) corrosion of embedded steel are significant.

Regarding the aging mechanism, leaching of calcium hydroxide, the applicant stated the following in LRA Table 3.5-1.

> RNP concrete is not exposed to flowing water, is dense, well cured, has low permeability, and was constructed in accordance with ACI recommendations at the time of construction. Thus, leaching of calcium hydroxide is not applicable to RNP concrete structures.

Regarding the aging mechanisms, aggressive chemical attack and corrosion of embedded steel, the applicant stated the following in LRA Table 3.5-1.

> RNP ground water values for chlorides and sulfates are much less than the threshold values necessary for aggressive chemical attack. However, the aging mechanisms associated with aggressive chemical attack and corrosion of embedded steel are potentially applicable to below-grade concrete structures owing to slightly acidic ground water (average pH of 4.4). The ASME Section XI, Subsection IWL Program is applicable to the containment structure. However, RNP will enhance the inspection requirements to apply a special inspection provision for monitoring aging effects potentially caused by aggressive chemical attack and corrosion of embedded steel.

Since the below-grade reinforced concrete at RNP is exposed to an aggressive environment (low pH), the staff requested, in RAI 3.5.1-3, that the applicant provide available RNP ground water chemistry test results including chlorides, sulphate, and pH values, and discuss the proposed AMP, as well as past inspection results of below-grade concrete at RNP. RAI 3.5.1-9 stated that the staff is unclear as to how the inspection for below-grade containment concrete will be performed by the ASME Section XI, Subsection IWL Program and requested that additional information, such as the locations, depth, and frequency of soil excavation, related to the AMR of below-grade containment concrete be provided. In response to RAI 3.5.1-3, the applicant stated the following.

Based on a long-term environmental monitoring report, from 1975 to 1995, the following environmental parameters have been identified for lake water at the intake structure:

average chloride concentration 3.14 ppm
average sulfate concentration 3.67 ppm
average pH 5.46

Based on semi-annual ground water monitoring reports, required by the State of South Carolina, the following environmental parameters have been identified from Well #4.

chloride concentration no data available
sulfate concentration 21.0 ppm
ground water pH 4.41

In response to RAI 3.5.1-9, the applicant stated the following.

> Based on the relatively low pH value for both ground water and lake water, an aggressive environment was assumed for the determination of aging effects associated with below-grade concrete.

The intended scope for the inspection of below-grade concrete, related to Item 7 of LRA Table 3.5-1, includes the concrete foundation and below-grade walls for the containment structure. The referenced AMP for this item is the Containment ISI Program for IWL, which is implemented through two plant procedures, the IWL inspection procedure and the site excavation backfill procedure. The inspection of inaccessible, below-grade concrete will be performed using the inspection criteria of ASME Section XI, Subsection IWL, for the subject item.

The site excavation procedure requires the user to notify design engineering of proposed excavations and requires an inspection prior to backfilling against exposed concrete surfaces. Excavations will not be performed with the sole purpose of concrete inspection. However, below-grade examinations of concrete have been performed at certain locations with satisfactory results. These include a below-grade section of the RAB, internal surfaces of electrical manholes exposed to ground water, submerged portions of the intake structure, and the dam spillway exposed to lake water. The lake water environment for the intake structure and dam spillway is essentially the same as that of aggressive ground water (pH values are both below 5.5, and chloride and sulfate levels are well below the trigger levels). As such, inspection results of the submerged portions should envelop aging effects encountered by below-grade concrete of other structures.

Having reviewed the applicant's response above, as well as its response to RAI B.3.14-1, the staff requested the applicant to provide a summary of the results of inspections performed in the below-grade sections of the RAB, the submerged portions of the intake structure, and the dam spillway that would support a conclusion that the below-grade structures have not been degraded, and the scope of the enhanced inspection is adequate to detect any significant degradation of the below-grade structures during the extended period of operation. The applicant provided the following response.

A summary of the results of inspections performed in the (1) below-grade sections of the RAB, (2) submerged portions of the intake structure, (3) dam spillway, and (4) other below-grade concrete are provided below:

(1) below-grade sections of the RAB

A visual inspection of the below-grade portion of the RAB foundation approximately three feet deep was performed in July 1999 while the east foundation was exposed during excavation for construction of the north service water header support slab. This general visual inspection monitored for spalling, scaling, erosion, swelling, bulging, signs of corrosion, cracking, settlement, and exposed rebar. In addition, the interior of manholes 35 and 36, which about the

RAB, were inspected on September 30, 2002. The interior, which had been exposed to ground water since initial construction, had no signs of spalling or other concrete degradation.

(2) submerged portion of the Intake Structure

An inspection of the inaccessible areas was performed during RFO-19 from September 28, 1999, to October 2, 1999, using divers and video equipment. The results of the inspection are as follows. The concrete surface had very little marine growth. There was little or no sediment on the bottom slab. The concrete located at the water line showed signs of erosion from the constant wave action. The top coat of mortar has eroded away leaving the aggregate exposed. The average loss of cover is approximately 1/16 inch to 1/8 inch. The concrete surface was cleaned of marine growth in a number of locations with a wire brush. The top coat came off with minor effort, thereby exposing the aggregate. Sound material was observed at all cleaned locations. Several repairs were observed to have been made in various locations. One repair had flaked off and rebar was observed (one end cut). The repair material thickness was approximately 2 inches and the repair area was about 1 square foot. This area was determined by the RNP Engineering Section to have no impact on the structural integrity of the concrete.

(3) dam spillway

An underwater inspection was performed on June 20, 2000, by divers. The spillway inspection examined the condition of concrete, especially at the tainter gates. A spalled portion of concrete (6" by 8" by 4" deep) was identified. This area is scheduled to be reinspected and repaired prior to the period of extended operation. The Dam Inspection Program will monitor the condition of the normally inaccessible submerged spillway concrete surfaces at a frequency not to exceed 10 years. No other underwater concrete degradation was identified.

(4) other

The interiors of eight security manholes were visually examined in August 2002. The interior concrete has been partially submerged from ground water and provides a similar environment as below-grade concrete (exposure to slightly acidic ground water). No cracking, loss of material, or change in material properties was observed in the concrete surface.

In a conference call with the applicant which occurred on June 16, 2003, the staff pointed out that the applicant did not specify appropriate remedial measures to be followed if the results of RNP's periodic, submerged inspection of the intake structure concrete show significant concrete degradation. Subsequent to this conference call, the applicant, through an email communication, has agreed to the following in order to ensure adequate aging management of below-grade structural concrete that is within the scope of the AMR:

- Degradation to submerged concrete observed during periodic under water inspections at the intake structure and RNP dam spillway will be used as a leading indicator for potential degradation to below-grade concrete structures in the scope of license renewal. Below-grade concrete will be evaluated and/or examined for potential degradation and corrective actions taken as determined by Engineering. This applies to below-grade concrete examined by the Structures Monitoring Program (SMP) and the ASME Section XI, Subsection IWL Program. Applicable SMP and IWL Program procedures will be enhanced to incorporate these changes.

- Ground water and lake water monitoring results (pH, chlorides, sulfates) will be reviewed by Engineering and trended. Increasing aggressiveness of the ground water and lake water will also be used as a leading indicator for potential degradation to below-grade concrete structures in the scope of license renewal as described above.

- Below-grade concrete, when exposed during excavation, already requires notification of Engineering for inspection. However, degradation to below-grade concrete due to aggressive ground water, when exposed during excavation, will also be used as a leading indicator for potential degradation to other below-grade concrete structures in the scope of license renewal as described above.

The staff finds the above commitments adequate to address its concerns regarding the aging management of below-grade, in-scope concrete structural components at RNP. The applicant also committed to provide appropriate documentation of the above agreement. This item was designated as Confirmatory Item 3.5-1.

By letter dated August 14, 2003 (RNP Serial RNP-RA/03-0094), the applicant responded to a number of Confirmatory Items identified by the staff (via an email dated July 14, 2003, from Mr. S.K. Mitra, NRC, to Mr. Roger Stewart, RNP). The staff reviewed the revised contents of Items 25, 26, and 27 of Attachment II (Revised License Renewal Commitments) to the applicant's August 14, 2003, letter. The staff also reviewed the specific response to Confirmatory Item 3.5-1 provided in Attachment III (Response to License Renewal Confirmatory Items) to the same letter. Based on these reviews, the staff finds that the applicant has provided adequate information Confirmatory Item 3.5-1 is closed.

Because of the slightly acidic RNP ground water environment, the applicant conservatively assumed existence of an aggressive chemical environment and proposed the above described plant-specific AMPs (an enhanced ASME Section XI, Subsection IWL Program for containment and an enhanced Structures Monitoring Program for other Category 1 structures) to manage the aging effects of below-grade concrete. As such, the staff finds RAIs 3.5.1-3 and 3.5.1-9 to be fully resolved.

On the basis of its review, the staff finds that the applicant has adequately evaluated the management of aging of inaccessible concrete areas for containment, as recommended in the GALL Report.

3.5.2.2.1.2 Cracking, Distortion, and Increase in Component Stress Level Due to Settlement; Reduction of Foundation Strength due to Erosion of Porous Concrete Subfoundations, If Not Covered by Structures Monitoring Program

As stated in the SRP-LR, the GALL Report recommends, for the containment foundation, further evaluation of certain aging effects, including (1) cracking due to settlement, and (2) change in material properties as manifested by a reduction of foundation strength due to erosion of the porous concrete subfoundation, if these two effects are not covered by a structures monitoring AMP. In addition, the GALL Report recommends verification of the continued functionality of a dewatering system during the license renewal period, if relied on by the applicant to lower the site ground water level.

The applicant addressed the above criteria defined in the GALL Report regarding the need for further evaluation to manage the potential aging of the containment foundation in LRA Table 3.5-1. In row entries 8 and 9 of LRA Table 3.5-1, the applicant stated that the aging effect were not applicable. However, based on the applicant's response to Interim Staff Guidance on Concrete Aging (letter to NRC Serial: RNP-RA/02-0159), the applicant stated RNP would examine accessible concrete using the SMP or the IWL Program. For the containment structure, the applicant is using the IWL Program for managing the aging effects of cracking, change in material properties, and loss of material. The staff's evaluation of the applicant's IWL Program is found in Section B.3.14 of this SER.

Regarding the aging effect, cracking due to settlement, the applicant stated the following in row 8 of the LRA Table 3.5-1.

The RNP AMR determined that cracking due to settlement is not applicable. Monitoring for settlement was performed during construction of the plant. Based on the results of the monitoring program and 30 years of operating experience, settlement is not an applicable aging mechanism and no dewatering system was used at RNP. Refer to Table 3.5-1 of this SER.

Regarding the reduction in strength due to erosion of porous concrete subfoundation, the applicant stated the following in row 9 of the LRA Table 3.5-1.

The RNP AMR for concrete determined that RNP concrete foundations are not constructed of porous concrete and, therefore, are not susceptible to this aging mechanism. Refer to Table 3.5-1 of this SER. Table 3.5-1 lists "none" for the AMP for this effect because porous concrete does not exist at RNP.

Because the applicant is managing cracking and change in material properties for the containment foundation as recommended by the GALL Report, the staff finds that the applicant has adequately addressed this further evaluation criteria.

On the basis of its review, the staff finds that the applicant has adequately evaluated the management of cracking, distortion, and increase in component stress level due to settlement and the reduction of foundation strength due to erosion of porous concrete subfoundations for containment components, as recommended in the GALL Report.

3.5.2.2.1.3 Reduction of Strength and Modulus of Concrete Structures Due to Elevated Temperature

As stated in the SRP-LR, the GALL Report recommends, for the containment structure, further evaluation to manage the aging effect change in material properties as manifested by a reduction in strength and modulus, if any portion of the containment concrete exceeds the temperature limit of 150 °F. The GALL Report notes that the implementation of Subsection IWL examinations and 10 CFR 50.55a would not be able to detect the reduction of concrete strength and modulus due to elevated temperature and also notes that no mandated aging management exists for managing this aging effect.

The GALL Report recommends that a plant-specific evaluation be performed if any portion of the concrete containment components exceeds specified temperature limits, (i.e., general temperature 66 °C (150 °F) and local area temperature 93 °C (200 °F)). The staff verified that the applicant's discussion in the renewal application indicated that the affected PWR containment components are not exposed to temperatures that exceed the above temperature limits. For concrete containment components that operate above these temperature limits, the staff reviewed the applicant's proposed programs to ensure that the effects of elevated temperature will be managed during the period of extended operation.

The applicant addressed the above criterion defined in the GALL Report regarding the need for further evaluation in LRA Table 3.5-1. In row 10 of LRA Table 3.5-1, the applicant stated the following regarding temperatures within the containment structure.

Generally, RNP concrete elements do not experience temperatures that exceed the temperature limits associated with aging degradation due to elevated temperature. During an accident, uninsulated concrete may experience a temperature greater than 200 °F for less than 10 seconds, but this was considered to have minimal effects. Therefore, this aging effect is not applicable. However, a TLAA was evaluated to demonstrate the continuing capability of one containment penetration when subject to temperature cycles that exceed 200 °F in adjacent concrete.

RNP subsequently determined the concrete temperature surrounding the subject containment penetration did not exceed 200 °F. The TLAA for this was therefore eliminated. The applicant asserted that RNP concrete elements do not experience temperatures that exceed the temperature limits associated with aging degradation due to elevated temperature. Therefore, this aging effect is not applicable.

In RAI 3.5.1-12, the staff requested that the applicant provide further information regarding the highest temperatures of in-scope concrete elements at RNP, with respect to general high temperature areas and localized hot spots, and compare them to the ACI 349 Code temperature limits. In response to RAI 3.5.1-12, the applicant stated the following.

No concrete elements at RNP exceed the ACI 349 Code temperature limits. The maximum ambient atmospheric air temperatures are as follows for the various RNP in-scope structures:

Outdoor 95 °F
Indoor Air Conditioned 85 °F
Indoor Not Air Conditioned 104 °F (excluding containment)
Containment 120 °F (bulk average temperature)

Based on initial conditions used in the design basis analyses, the containment bulk average temperature is maintained below 120°F and verified through Technical Specifications surveillance on a 24 hour frequency. As such, containment bulk average temperature is below the ACI normal operation value for general areas (i.e., 150 °F).

The temperature of concrete in the vicinity of the reactor vessel is kept within acceptable limits by the reactor vessel insulation casing, air spacing between the insulation and primary shield wall, and supplemental cooling. Concrete in this area is managed by the Structures Monitoring Program and no degradation has been identified.

Localized hot spots within containment can be characterized as the pressurizer cubicle and the concrete surrounding hot piping penetrations. Documented temperatures for the pressurizer cubicle are as follows:

175 °F (9 percent of the time)
165 °F (25 percent of the time)
155 °F (66 percent of the time)

These values are below the ACI 349 normal operation value for local areas (200 °F).

There are no concrete areas around containment penetrations where sustained temperatures exceed 200 °F.

Based on the RNP operational data reported above, the staff determined that (1) the monitoring and management of the concrete temperature for the RNP containment concrete is based on periodic temperature measurements at key containment locations, some of which are verified through Technical Specifications surveillance on a 24 hour frequency, (2) containment bulk average temperature is below the ACI normal operation value for general areas (i.e., 150 °F), and (3) there are no localized concrete hot spots within containment or around containment penetrations where sustained temperatures exceed the 200 °F acceptance limit set by ACI 349 Code. These RNP specific operational data provide an acceptable basis for the staff to conclude that the applicant has implemented reasonable and adequate procedures for managing elevated temperature induced containment concrete degradation. As such, the applicant's response to RAI 3.5.1-12 is acceptable.

On the basis of its review, the staff finds that the applicant has adequately evaluated the management of the reduction of strength and modulus of concrete structures due to elevated temperatures for structures and structural components, as recommended in the GALL Report.

3.5.2.2.1.4 Loss of Material Due to Corrosion in Inaccessible Areas of Steel Containment Shell or Liner Plate

As stated in the SRP-LR, the GALL Report recommends further evaluation to manage the aging effect, loss of material due to corrosion for the embedded containment liner, if corrosion of the embedded liner is significant. The AMP recommended by the GALL Report for managing loss of material for accessible steel elements within the containment structure is the ASME Section XI, Subsection IWE (XI.S1) Program. The staff's evaluation of the applicant's ASME Section XI, Subsection IWE Program is found in Section 3.0.3.3 of this SER.

Subsection IWE exempts from examination portions of the containments that are inaccessible, such as embedded or inaccessible portions of steel liners and steel containment shells, piping, and valves penetrating or attaching to the containment. To cover inaccessible areas, 10 CFR 50.55a(b)(2)(ix) requires that the licensee evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to inaccessible areas.

The applicant addressed the above criterion defined in the GALL Report regarding the need for further evaluation to manage the potential aging of the embedded containment liner in LRA Table 3.5-1. In row entry 12 of LRA Table 3.5-1, the applicant stated the following regarding the potential for significant corrosion of the RNP steel containment liner.

Certain inaccessible areas in the Containment were identified which are required to be evaluated because conditions exist in accessible areas that could indicate the presence of or result in degradation to inaccessible areas. These areas include the containment liner plate at elevation 228 feet and the containment liner plate beneath the concrete floor below 228 feet. As noted in the 90-day ISI Summary Report submitted by letter RNP-RA/ 01- 0125, dated 8/10/2001, these areas have been evaluated to be acceptable until 2005. A One-Time Inspection Program action has been identified to verify the results of the evaluation and to manage any aging effects at these locations. At that time, the GALL-recommended AMPs will continue to manage the aging effects. This is consistent with the GALL Report. In addition, if the corrosion is caused by leakage of borated water onto carbon steel components, the Boric Acid Corrosion Program in addition to the ISI Program would be applied to manage the localized degradation caused by aggressive chemical attack.

Therefore, the ASME Section XI, Subsection IWE, the 10 CFR 50, Appendix J, the Boric Acid Corrosion, and One-Time Inspection Programs are used to manage corrosion in accessible and inaccessible areas. Aging management for this component/commodity group is consistent with the GALL Report.

In RAI 3.5.1-7, the staff raised a concern regarding the potential for loss of material associated with inaccessible containment vessel liners located below the concrete and requested the applicant to explain how the portions of inaccessible containment vessel liners that are located below the concrete were evaluated. The staff also requested that the applicant briefly summarize the basis for concluding that the other "inaccessible" areas below the concrete are acceptable for continued service until 2005. The applicant stated the following in its response to RAI 3.5.1-7.

A section of the liner was examined (approximately 1 foot deep by 4 feet long in a pre-existing void) below the concrete floor at the 228 foot elevation. A visual examination determined there were tightly adhered corrosion products on the liner surface. A UT examination for actual liner plate thickness determined there was no loss of material thickness. Water samples located in this void area were alkaline, stagnant, low re-oxygenation, low chloride concentration, and low boron concentration. The vertical liner below the concrete floor was in better condition and less pitted than the liner surface immediately above the concrete floor. The liner surface immediately above the concrete floor had pitting corrosion up to 0.1875 inch which was the worst case. This corrosion rate was estimated based on the worst-case degradation occurring from the containment flooding event in 1975 to the liner inspections in 1998 (0.1875 inch/ 23 years). The corrosion rate was then applied to the difference between the actual thickness examined for the liner and minimum design thickness. The worst-case corrosion area above the concrete was determined to conservatively meet the liner design thickness until year 2005. The liner plate thickness below the concrete, which had no degradation, was determined to be acceptable (exceeding the minimum wall thickness) for continued service until 2005. By 2005, either further evaluation or inspection will be required for the inaccessible portion of the liner below the concrete.

In RAI 3.5.1-19, the staff requested that the applicant provide a basis for concluding that (1) the existing conditions of the containment liner (behind the moisture barrier) and the moisture barrier are acceptable, and (2) the inspection to be performed under a One-Time Inspection Program will be sufficient to monitor the condition of the containment liner behind the insulation and the moisture barrier during the extended period of operation. In response to RAI 3.5.1-19, the applicant provided the following response.

The existing condition of the containment liner (behind the moisture barrier) and the moisture barrier was determined to be acceptable based on visual examinations. These visual examinations of the containment liner, behind the removed moisture barrier, determined that the corrosion observed did not impact the structural integrity or leak tightness of the containment.

The inspection to be performed under the One-Time Inspection Program was determined to be sufficient to monitor the condition of the containment liner behind the insulation and the moisture barrier during the extended period of operation. Liner plate areas (behind the moisture barrier) with identified corrosion will be prepared, re-coated, and a new moisture barrier installed. No additional examinations are planned beyond those required by the IWE Program. In accordance with LRA Table 3.5-1, Items 6 and 12, the existing IWE Program is committed to for the extended period of operation, and the one-time inspection will be completed before the end of 2005.

Because the existing condition of the containment liner (behind the moisture barrier) and the moisture barrier itself was determined to be acceptable based on visual examinations, and because the applicant has committed to perform a One-Time Inspection Program to reconfirm the acceptability of the condition of the containment liner behind the insulation, the staff finds that the applicant has provided a reasonable basis for concluding that the aging of the containment liner behind the insulation and the moisture barrier will be adequately managed consistent with its CLB during the extended period of operation. In addition, the ASME XI, Subsection IWE Program manages the aging of the accessible portions of the liner with the stipulation that the applicant evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to inaccessible areas. As such, the staff considers that RAIs 3.5.1-7 and 3.5.1-19 are closed.

On the basis of its review, the staff finds that the applicant has adequately evaluated the management of the loss of material due to corrosion in inaccessible areas of the steel containment shell or liner plate for structures and structural components, as recommended in the GALL Report. Due to the corrosion of the liner plate, the applicant proposed to implement a one-time inspection and to take necessary remedial actions that might be required as a result of the one-time inspection to ensure the integrity of the containment liner during the extended period of operation, thus, adequately fulfilling the further evaluation provision recommended by the GALL Report.

3.5.2.2.1.5 Loss of Prestress Due to Relaxation, Shrinkage, Creep, and Elevated Temperature

As stated in the SRP-LR, the GALL Report identifies loss of prestress due to relaxation, shrinkage, creep, and elevated temperature for prestressed containment tendons and anchorage components as a TLAA to be performed for the period of extended operation. The applicant covered this TLAA in Section 4.5 of the application and the staff evaluation of this TLAA is addressed in Section 4.5 of this SER.

Because the prestressing tendons of RNP containment are protected from corrosion by means of specially formulated grout, the requirements of Subsection IWL are not applicable to the

RNP prestressing tendons.

In addition to loss of prestress, the staff also evaluated loss of material as a potential aging effect for the containment tendons and anchorage components. LRA Section 3.5 states that the tendons and their anchorage components are embedded and cannot be accessed for inspection. In addition, the applicant had performed inspections of sample surveillance blocks at 5-year and 25-year intervals. Based on the results of the inspection of these surveillance blocks, the applicant concluded that grouting has proven to be an effective means of preventing corrosion of the tendons and anchorage components.

To get an understanding of the surveillance block tendons and their role in preventing corrosion of the containment tendons, the staff issued RAI 3.5.1-20. Following is the applicant's response.

a) The surveillance tendons consist of six 1-3/8 inch diameter bars grouted in a six inch pipe sheath with anchor plates and prestressing hardware, which is identical to the service tendon except for the length. They are embedded in a section of concrete approximating the same environment as that of the service tendons. The surveillance blocks were placed next to the containment to subject them to a similar unsheltered outdoor environment.

b) The surveillance tendons are 1-3/8 inches in diameter which is the same size as the tendons used in the containment structure.

c) There are no records that would indicate the surveillance block tendons were prestressed. However, inspection results from the surveillance note a snap-back of the tendons into the casing as each rod was severed. The test lab suggested that the snap-back indicated a level of stress had been maintained in the rods by the grout.

d) The surveillance blocks were not instrumented for time-dependent stress/strain measurements.

e) The conclusions for both the 5- and 25-year surveillance blocks indicate there is no significant corrosion, and mechanical testing of the tendon bars also show no significant change in properties. While no specific inspection criteria were provided for the grout, it was noted that the grout cracked as the pipe was cut and stress relieved from the bars. Also in some areas, separated grout had a reddish-brown stain at the contact surface with the bars that was suspected to be an oxide that formed during construction.

The applicant also provided the detailed reports with photographs to the staff.

As indicated at the beginning of this section, the applicant addressed the loss of prestress due to relaxation, shrinkage, creep, and elevated temperature aging effects as part of RNP's TLAA in Section 4.5 of the LRA. The staff evaluation of this TLAA, including the above RNP response to RAI 3.5.1-20, is presented in Section 4.5 of this SER.

Having reviewed the information provided in Section 4.5 and Appendix A of the LRA and the applicant's responses to RAIs 3.5.1-20, 4.5-1, 4.5-2, and 4.5-3, including a commitment to perform structural integrity testing (SIT) and making the necessary observations during the tests, the staff finds the applicant's RAI responses and its commitment to perform SIT reasonable and acceptable because it would assess the integrity of the prestressing tendons and the RNP containment during the extended period of operation. RAI 3.5.1-20 is considered closed and closure of RAIs 4.5-1 through 4.5-3 is provided in Section 4.5 of this SER. On the basis of the above findings, the staff concludes that there is reasonable assurance that the structures and structural components subject to loss of prestress aging effects will be adequately managed during the period of extended operation.

3.5.2.2.1.6 Cumulative Fatigue Damage

As stated in the SRP-LR, the GALL Report identifies cumulative fatigue damage as a TLAA for penetration sleeves, penetration bellows, and dissimilar metal welds to be performed for the period of extended operation. The applicant covered this TLAA in Section 4.6of the application, and the staff evaluation of this TLAA is addressed in Section 4.6 of this SER.

On the basis of the staff's review of LRA Section 4.3.5 , the staff concludes that the containment penetration bellows subject to fatigue will be adequately managed during the period of extended operation.

3.5.2.2.1.7 Cracking Due to Cyclic Loading and SCC

As stated in the SRP-LR, the GALL Report recommends further evaluation of the AMPs to manage cracking of containment penetrations (including penetration sleeves, penetration bellows, and dissimilar metal welds) due to cyclic loading or SCC for all types of PWR containments. Containment ISI and leak rate testing may not be sufficient to detect cracks. The staff evaluated the applicant's proposed programs to verify that adequate inspection methods will be implemented to ensure that cracking of containment penetrations is detected.

Items 2 and 3 of Table 3.5-1 of the LRA discuss the plant-specific operating experience related to cracking due to SCC and/or cyclic loading, as well as loss of material of the penetration sleeves and bellows. In addition to its Containment Inservice Inspection and Containment Leak Rate Testing AMPs, the applicant uses its Water Chemistry Program to identify degradation of SS components which are subjected to borated, treated water. The applicant uses its Boric Acid Corrosion Program if the corrosion is caused by leakage of borated water on carbon steel components. To better understand the plant-specific operating experience related to the degradation of penetration bellows, the staff requested additional information in RAIs 3.5.1-16 and 3.5.1-17.

RAI 3.5.1-16 requested the applicant to provide further information regarding the leak rate testing of the containment bellows. In response to RAI 3.5.1-16, the applicant provided the following information.

(a) Bellows (inside and outside containment) are testable by Appendix J, Type B testing.

(b) Administrative leakage limits are not established for individual penetrations that have bellows. However, administrative limits are established for groups of mechanical penetrations. If any group of mechanical penetrations exceeds its administrative limit, individual penetration(s) can be isolated for evaluation and repair. This allows detection of degradation of individual bellows on the penetrations during Type B testing. The overall leakage limit is specified in the Technical Specifications section 5.5.16.

(c) Type B tests are conducted on a refueling outage interval, not to exceed a maximum interval of two years. This frequency of testing will continue to be used for the extended period of operation. In addition, the following information is provided:

A review of plant OE determined many of the original bellows have been replaced. Replacements were generally made due to excessive leakage from damaged bellows. The following OE provides assurance the 10 CFR 50 Appendix J Program has been successful at detection of leakage at penetration bellows and implementing actions to replace bellows as necessary. Before 1992, several bellows were replaced with like-for-like bellows when leakage was identified. This was determined by monitoring the PPS which was used at that time to continuously provide design pressure to the containment penetrations. This system is now only used for testing. No aging mechanisms were determined for these replacement bellows. On July 20, 1995, a potential breach of containment integrity was discovered when the PPS indicated leakage greater than the limits established in the Technical Specifications. A Steam Generator Blowdown (SGB) bellows failed due to a crack caused by TGSCC. Condensation of water from the PPS supplied air inside the penetration wetted the pipe insulation and transported the chlorides contained in the insulation materials to the penetration bellows. The presence of the chlorides on the SS material of the bellows caused the bellows to fail. Additional thermal stresses due to isolation of service water to the penetration coolers contributed to the event. The penetration bellows and end plates were removed on all the SGB bellows per a plant modification. The insulation was replaced with chloride free insulation. Pipe caps replaced the inside end plates. Based on a new design without bellows, the aging mechanism no longer exists for the SGB line penetrations. This was also documented in a Licensee Event Report (LER 95-005-00). On October 7, 1996, a leak was found on the bellows inside the containment on penetration 63, sleeve 5. This was discovered during pressure testing of a new bellows installed on penetration 51, which is also on sleeve 5. It was found that the bellows convolutions had been compressed and damaged due to work performed on other bellows in the area during a previous outage. The penetration bellows were replaced in Refueling Outage-18. There were no aging mechanisms identified.

RAI 3.5.1-17 requested that the applicant provide further information regarding the accessibility of the outside plate/bellows and the possible existence of a penetration pressurization system (PPS) at RNP, which continuously monitors the leakage from the penetrations. The applicant provided the following response to RAI 3.5.1-17.

(a) Outside plate/bellows are accessible for inspection. However, these plates/bellows are not part of the containment pressure boundary and are only used during testing.

(b) The penetration pressurization system (PPS) installed at RNP does not continuously monitor the leakage from the penetrations. The PPS is used during power operation to test the personnel airlock and during outages to test containment penetrations (local leak rate tests). The PPS was originally installed as a continuous monitoring system but the system was modified in 1995 to change to an intermittent monitoring system, and PPS was isolated to the containment penetrations.

Based on the fact that (1) the Type B leak rate testing, performed as part of the 10 CFR Part 50, Appendix J testing, has been successful at detecting leakage at penetration bellows, (2) the applicant has replaced degraded bellows as necessary, and (3) the appropriate AMPs, as discussed above, are credited to manage the aging of the identified components, the staff finds that the applicant has adequately evaluated the management of cracking of containment penetrations (including penetration sleeves, penetration bellows, and dissimilar metal welds) due to cyclic loading and SCC, as recommended in the GALL Report. On the basis of this finding, the staff concludes that there is reasonable assurance that this aging effect will be adequately managed during the period of extended operation, as required by 10 CFR 54.21(a)(3). As such, RAIs 3.5.1-16 and 3.5.1-17 are considered closed.

3.5.2.2.1.8 Conclusions

The staff has reviewed the applicant's evaluation of the issues for which GALL recommends further evaluation for the containment structural components in Sections 3.5.2.2.1.1 through 3.5.2.2.1.7. On the basis of its review, the staff finds that the applicant has provided sufficient information to demonstrate that the issues for which GALL recommends further evaluation have been adequately addressed and that the subject aging effects will be adequately managed for the period of extended operation, as required by 10 CFR 54.21(a)(3).

The staff also reviewed the UFSAR Supplements for the AMPs and concludes that they provide adequate summary descriptions of the programs and activities credited for managing the effects of aging for containment components for which the applicant claimed consistency with GALL to satisfy 10 CFR 54.21(d).

3.5.2.2.2 Class I Structures

3.5.2.2.2.1 Aging of Structures Not Covered by Structures Monitoring Program

As stated in the SRP-LR, the GALL Report recommends further evaluation for certain structure/aging effect combinations, if they are not covered by the applicant's Structures Monitoring Program. These include (1) scaling, cracking, and spalling due to repeated freeze-thaw for Groups 1-3, 5, and 7-9 structures, (2) scaling, cracking, spalling, and increase in porosity and permeability due to leaching of calcium hydroxide and aggressive chemical attack for Groups 1-5 and 7-9 structures, (3) expansion and cracking due to reaction with aggregates for Groups 1-5 and 7-9 structures, (4) cracking, spalling, loss of bond, and loss of material due to corrosion of embedded steel for Groups 1-5 and 7-9 structures, (5) cracks, distortion, and increase in component stress level due to settlement for Groups 1-3, 5, and 7-9 structures, (6) reduction of foundation strength due to erosion of porous concrete subfoundation for Groups 1-3 and 5-9 structures, (7) loss of material due to corrosion of structural steel components for Groups 1-5, 7, and 8 structures, (8) loss of strength and modulus of concrete structures due to elevated temperatures for Groups 1-5 structures, and (9) crack initiation and growth due to SCC and loss of material due to crevice corrosion of SS liner for Groups 7 and 8 structures. Further evaluation is necessary only for structure/aging effect combinations that are not covered by the applicant's Structures Monitoring Program.

The applicant addressed the above criterion defined in the GALL Report regarding the need for further evaluation to manage the potential aging of concrete and steel structural components, in LRA Table 3.5-1. In row entry 16 of LRA Table 3.5-1, the applicant stated the following.

The Structures Monitoring Program is applied to components/commodities in this group that have aging effects. For concrete, the RNP AMR methodology concluded that above-grade concrete/grout structures have no aging effects; for steel, in addition to the Structures

Monitoring Program, the Boric Acid Corrosion Program is applicable for corrosion caused by leakage of borated water onto carbon steel components of this component/commodity group; protective coatings are not credited for aging management of steel components; Lubrite Reactor Pressure Vessel Supports use bearing plates of high strength, hard tool steel instead of Lubrite and owing to the wear-resistant material used, the low frequency of movement, and the slow movement between sliding surfaces, mechanical wear was determined not to be an aging mechanism, and similarly, lock-up due to wear is not considered to be an aging effect at RNP.

The above statements by the applicant raised a question as to whether the applicant will use its Structures Monitoring Program to manage the aging effects identified above, as recommended in the GALL Report. The staff issued RAI 3.5.1-8 to clarify this concern. In response to RAI 3.5.1-8, the applicant stated the following.

The letter from J. Moyer (CP&L) to NRC, Serial: RNP-RA/02-0159: "Supplement to Application for Renewal of Operating License," dated October 23, 2002, addresses aging management of concrete components. RNP committed to an AMP for monitoring accessible concrete based on Interim Staff Guidance, and agreed to credit the Structures Monitoring Program and the Dam Inspection Program for examination of accessible concrete. The Component/Commodity Group of "Reinforced Concrete" or "Concrete Tank Foundation" includes grout. Masonry block walls were not specifically identified in the October 23, 2002, letter. However, the Structures Monitoring Program is credited for monitoring the masonry block walls. LRA Table 3.5.1, Item 16, should state that based on Interim Staff Guidance, the Structures Monitoring Program will be used to monitor accessible concrete. LRA Table 3.5-2, Item 10, should be deleted. LRA Table 3.5.1, Item 20, should state that based on Interim Staff Guidance, the Structures Monitoring Program will be used to monitor accessible masonry walls. Based on GALL XI.S5, the Structures Monitoring Program can be used for the aging management of masonry walls.

The above response resolved the staff's concern regarding the concrete components listed in Item 16 of the LRA Table 3.5-1; however, the applicant did not commit to use the Structures Monitoring Program to manage the aging effects of the carbon steel components listed in Item 16. On May 22, 2003, the staff had a telephone conference to inform the applicant that full resolution of the RAI requires the aging management for all of the steel components listed in Item 16 of LRA Table 3.5-1. The applicant proposed to append with the following sentence.

In addition, the Structures Monitoring Program will be used for aging management of the steel components listed in LRA Table 3.5.1, Item 16.

Because the applicant is managing the aging effects for both the concrete and steel structural items covered by Item 16 of LRA Table 3.5-1, as recommended by the GALL Report, the staff finds that the applicant has adequately addressed this further evaluation criterion and RAI 3.5.1-8 is considered closed. The staff's evaluation of the applicant's Structures Monitoring Program is found in Section 3.5.2.3.5 of this SER.

On the basis of its review, the staff finds that the applicant has adequately evaluated the management of aging of structures not covered by the Structures Monitoring Program, as recommended in the GALL Report.

3.5.2.2.2.2 Aging Management of Inaccessible Areas

As stated in the SRP-LR, the GALL Report recommends further evaluation for aging of inaccessible concrete areas, such as below-grade foundation and exterior walls exposed to ground water, due to aggressive chemical attack, if an aggressive below-grade environment exists. An aggressive below-grade environment could result in either cracking or loss of material for concrete components subjected to such an environment. The GALL Report recommends that a plant-specific AMP be developed by the applicant, if an aggressive below-grade environment exists.

The applicant addressed the above criterion defined in the GALL Report, regarding the potential aging of below-grade concrete exposed to an aggressive environment, in LRA Table 3.5-1. In item 17 of LRA Table 3.5-1, the applicant stated the following.

The aging mechanisms associated with aggressive chemical attack and corrosion of embedded steel are applicable only to below-grade concrete/grout structures owing to the slightly acidic pH of ground water. The Structures Monitoring Program is applicable to these structures. RNP will apply a special, plant-specific inspection provision to monitor aging effects caused by aggressive chemical attack and corrosion of embedded steel for below-grade concrete in this component/commodity group. This will include inspection of below-grade concrete and grout that is exposed during excavation. These aging management activities are consistent with the GALL Report.

In RAI 3.5.1-10, the staff asked the applicant to explain how the inspection for below-grade Class I structural concrete will be performed by an RNP plant-specific AMP, as recommended in the GALL Report. The staff also requested the applicant to provide additional information, such as the locations, depth, and frequency of soil excavation. The applicant provided the following response to RAI 3.5.1-10.

Inspection of inaccessible, below-grade concrete will be performed using the concrete inspection criteria of the Structures Monitoring Program for the subject item., e.g., planned construction, corrective maintenance, etc. Inaccessible, below-grade, concrete will be inspected when it is exposed during plant excavations for other activities. The site excavation procedure requires notification of Engineering for proposed excavations, and requires an inspection prior to backfilling. Such below-grade examinations of concrete have been performed at certain locations with satisfactory results. These include a below-grade section of the RAB, internal surfaces of electrical manholes exposed to ground water, submerged portions of the intake structure, and the dam spillway exposed to lake water. The lake water environment for the intake structure and dam spillway is essentially the same as that of aggressive ground water (pH values are both below 5.5, and chloride and sulfate levels are well below the trigger levels). Therefore, inspection results of the submerged portions should envelope aging effects encountered by below-grade concrete of other structures. For additional information regarding inspection of inaccessible, below-grade, concrete associated with the containment pressure boundary, please refer to the RNP Response to RAI 3.5.1-3.

As stated previously in Section 3.5.2.2.1.1 of this SER, the staff found that RNP's approach of inspecting below-grade concrete only when it happens to be exposed during plant excavations done for other activities to be insufficient. As such, the staff requested further measures be taken to ensure the adequate aging management of below-grade concrete at RNP. In response to the staff's concerns, the applicant proposed to use its periodic inspections of the submerged portions of the intake structure and dam spillway as indicators for the condition of below-grade concrete at RNP. Because the ground water and lake chemistry are similar, degradation to submerged concrete will be used as a leading indicator for the potential degradation to below-grade concrete structures. This commitment was designated as Confirmatory Item 3.5-1.

Based on the discussion related to closure of Confirmatory Item 3.5-1 provided in Section 3.5.2.2, "Aging Management Evaluations in the GALL Report That Are Relied On for License Renewal, For Which GALL Recommends Further Evaluation," of this SER, Confirmatory Item 3.5-1 is closed.

The staff finds that the applicant has adequately evaluated the aging management of inaccessible concrete areas for Category 1 structures, as recommended in the GALL Report.

3.5.2.2.2.3 Conclusions

The staff has reviewed the applicant's evaluation of the issues for which GALL recommends further evaluation for Class I structures in sections 3.5.2.2.2.1 and 3.5.2.2.2.2. On the basis of its review, the staff finds that the applicant has provided sufficient information to demonstrate that the issues for which GALL recommends further evaluation have been adequately addressed and that the subject aging effects will be adequately managed for the period of extended operation, as required by 10 CFR 54.21(a)(3).

The staff also reviewed the UFSAR Supplement for the AMPs and concludes that they provide adequate summary descriptions of the programs and activities credited for managing the effects of aging for Class I structures for which the applicant claimed consistency with GALL to satisfy 10 CFR 54.21(d).

3.5.2.2.3 Component Supports

3.5.2.2.3.1 Aging of Supports Not Covered by Structures Monitoring Program

As stated in the SRP-LR, the GALL Report recommends further evaluation of certain component support/aging effect combinations if they are not covered by the Structures Monitoring Program. This includes (1) reduction in concrete anchor capacity due to degradation of the surrounding concrete for Groups B1-B5 supports, (2) loss of material due to environmental corrosion for Groups B2-B5 supports, and (3) reduction/loss of isolation function due to degradation of vibration isolation elements, for Group B4 supports. Further evaluation is necessary only for the structure/aging effect combinations listed above that are not covered by the applicant's Structures Monitoring Program.

The applicant addressed the above criterion defined in the GALL Report, regarding the need for further evaluation to manage the potential aging of component supports, in LRA Table 3.5-1. In item 25 of LRA Table 3.5-1, the applicant stated that it will use its Structures Monitoring Program to manage the aging effects identified in the preceding paragraph. The applicant further stated that RNP's Structures Monitoring Program will be enhanced to assure that additional concrete structures, which provide support to component support members, are included in the required monitoring. Carbon steel parts of slide bearing plates used for non-ASME components are also included in this Item 25 group.

Since the applicant is managing the aging effects for the component supports covered by Item 25 of LRA Table 3.5-1, as recommended by the GALL Report, the staff finds that the applicant has adequately addressed this further evaluation criterion. The staff's evaluation of the applicant's Structures Monitoring Program is found in Section 3.5.2.3.5 of this SER.

3.5.2.2.3.2 Cumulative Fatigue Damage Due to Cyclic Loading

As stated in the SRP-LR, the GALL Report identifies cumulative fatigue damage as a TLAA for support members, anchor bolts, and welds for Groups B1.1, B1.2, and B1.3 component supports, if a CLB fatigue analysis exists. Since a CLB fatigue analysis does not exist at RNP, cumulative fatigue damage for component supports is not addressed by the applicant.

3.5.2.2.3.3 Conclusions

The staff has reviewed the applicant's evaluation of the issues for which GALL recommends further evaluation for component supports. On the basis of its review, the staff finds that the applicant has provided sufficient information to demonstrate that the issues for which the GALL recommends further evaluation have been adequately addressed and that the subject aging effects will be adequately managed for the period of extended operation, as required by 10 CFR 54.21(a)(3).

The staff also reviewed the UFSAR Supplement for the AMP and concludes that it provides an adequate summary description of the programs and activities credited for managing the effects of aging for component supports for which the applicant claimed consistency with GALL to satisfy 10 CFR 54.21(d).

3.5.2.3 Aging Management Programs for Containment, Structures, and Component Supports

In SER Section 3.5.2.1, the staff evaluated the applicant's conformance with the aging management recommended by GALL for containment, other Class I structures, and component support component groupings. In SER Section 3.5.2.2, the staff reviewed the applicant's evaluation of the issues for which GALL recommends further evaluation. In this SER section, the staff presents its evaluation of the programs used by the applicant to manage the aging of the component groups within the containment, other Class I structures, and component supports.

RNP credits 13 AMPs to manage the aging effects associated with the containment, other Class 1 structures, and components supports. Four of the AMPs are credited to manage aging for components in other system groups (common AMPs), six AMPs are credited with managing aging only for structural components, and three are evaluated as mechanical systems. The staff's evaluation of the common AMPs credited with managing in structures is provided in Section 3.0.3. The AMPs evaluated as machanical systems include:

- Fire Water System Program (SER Section 3.3.2.3.3)
- Fire Protection Program (SER Section 3.3.2.3.2)
- Inspection of Overhead Heavy-Load and Light-Load Handling Systems Program (SER Section 3.3.2.3.1)

The common AMPs include the following programs:

- Water Chemistry Program (SER Section 3.0.3.3)
- Boric Acid Corrossion Program (SER Section 3.0.3.4)
- One-Time Inspection Program (SER Section 3.0.3.9)
- Preventive Maintenance Program (SER Section 3.0.3.12)

The staff's evaluation of the six structure-specific AMPs are provided in the sections below.

3.5.2.3.1 ASME Section XI, Subsection IWE Program

3.5.2.3.1.1 Summary of Technical Information in the Application

The applicant described its ASME Section XI, Subsection IWE Program in Section B.3.13 of the LRA. The LRA states that this program is consistent with GALL Program XI.S1, "ASME Section XI, Subsection IWE," with the following exceptions, (1) RNP will use the One-Time Inspection Program for inspecting inaccessible portions of the containment liner and the moisture barrier inside the containment at the liner plate/floor concrete interface, (2) RNP identified additional aging mechanisms not identified in the GALL Report (e.g., aggressive chemical attack for the containment liner plate and galvanic and general corrosion for penetration bellows), and (3) RNP did not identify SCC for the penetration sleeve and bellows because the environmental stressors required to initiate cracking from SCC are not present at RNP.

The applicant credits the ASME Section XI, Subsection IWE Program for aging management of selected components of the reactor containment building at RNP. The applicant identified the following aging effects/mechanisms of concern, (1) loss of material due to general corrosion, (2) loss of material due to galvanic corrosion, (3) loss of material due to aggressive chemical attack, (3) loss of material due to crevice corrosion, (4) loss of material due to pitting corrosion, (5) change in material properties due to elevated temperature, (6) cracking due to elevated temperature, and (7) cracking due to thermal fatigue.

The applicant further stated that, as a result of the license renewal review, administrative controls associated with program element *Confirmation Process* for the program will be enhanced to (1) specify the requirements for conducting reexaminations, and (2) document that repairs meet the specified acceptance standards.

Under the program element "Operating Experience," the applicant states that the program is implemented and maintained in accordance with the general requirements of engineering programs, and asserts that the programs (in general) are effectively implemented through the use of qualified personnel and adequate resources, and are managed in accordance with plant administrative controls. Moreover, the applicant makes a point that generic operating experience includes NUREG-1522, "Assessment of Inservice Conditions of Safety Related Nuclear Plant," June 1995, and that RNP was one of the six plants that was inspected in support of this document.

In the plant-specific operating experience, the applicant identifies degradation of containment as (1) corrosion of the cylinder wall at the bottom of the equipment hatch, (2) degradation of protective insulation sheathing, (3) cracking due to transgranular stress-corrosion cracking

(TGSCC) of a SG blowdown penetration bellows, (4) localized bulging of the containment liner, (5) numerous instances of corrosion of liner, and (5) potential for boric acid leakage penetrating the epoxy construction seal in the vicinity of the emergency core cooling system (ECCS) sump. For these occurrences, the applicant states that it has taken appropriate corrective actions. The applicant further states that this AMP is continually upgraded based on the industry experience and research, and that the Corrective Action Program has been effective in ensuring that the program is continually improving.

3.5.2.3.1.2 Staff Evaluation

In LRA Section B.3.13, "ASME Section XI, Subsection IWE Program," the applicant described its program to manage aging of the containment building at RNP. The LRA states that this program is consistent with GALL Program XI.S1, "ASME Section XI, Subsection IWE," with the following exceptions, (1) RNP will use the One-Time Inspection Program for inspecting inaccessible portions of the containment liner and the moisture barrier inside the containment at the liner plate/floor concrete interface, (2) RNP identified additional aging mechanisms not identified in the GALL Report (e.g., aggressive chemical attack for the containment liner plate and galvanic and general corrosion for penetration bellows), and (3) RNP did not identify SCC for the penetration sleeve and bellows because the environmental stressors required to initiate cracking from SCC are not present at RNP. The staff confirmed the applicant's claim of consistency during the AMR inspection. Furthermore, the staff reviewed the deviations and their justification to determine whether the AMP, with the deviations, remains adequate to manage the aging effects for which it is credited. The staff also reviewed the UFSAR Supplement to determine whether it provides an adequate description of the revised program. In addition, the staff determined whether the applicant properly applied the GALL program to its facility.

The staff conceptually considers the Appendix J Program as a program to ensure the leak-tight integrity of the containment (as described in GALL Section XI.4), and the Containment Inservice Inspection Program (Subsection IWE program) as the AMP for detecting the aging degradation of containment pressure boundary components. These programs complement each other and are required to assure that the containment continues to perform its intended functions as described in Table 2.4-1 of the LRA. The LRA appropriately describes the purpose of the program; however, the staff requested clarification of some of the program elements and exceptions (GALL Section XI.S1) associated with the ASME XI Section, Subsection IWE Program.

In addressing the program element *Confirmation Process*, the applicant stated that the program will be enhanced to require reexaminations and document that repairs meet the specified acceptance standards. The requirements for supplemental examinations, additional examinations, and documentation of acceptance criteria are parts of Subsection IWE of the ASME Code, as modified by 10 CFR 50.55a, and endorsed in GALL Section XI.S1. The staff asked the applicant, in RAI B.3.13-1, to provide information regarding what the enhancements consist of which are not currently required. By letter dated April 28, 2003, the applicant provided the following response.

> The site procedure for the IWE Program meets the requirements of IWA-4000, IWA-2200, and Table IWE-3410-1 for repairs and reexaminations, except as allowed by 10 CFR 50.55a(b)(2)(ix)(B) and approved requests for relief. However,

> an improvement was recommended to add the following statement to the IWE Program procedure: "Reexaminations are conducted in accordance with the requirements of IWA-2200, and the recorded results are to demonstrate that the repair meets the acceptance standards set forth in Table IWE-3410-1." This was recommended to clearly summarize the requirements in one location.

The staff considers the applicant's action of incorporating all the acceptance criteria in one location prudent in implementing the requirements of Subsection IWE of Section XI of the ASME Code and finds it acceptable.

Based on the database on degradation of the moisture barrier between the concrete floor and the cylinder liner, Subsection IWE of Section XI of the ASME Code (as referenced in GALL Section XI.S1) requires 100 percent examination of the moisture barrier once every inspection interval. During the IWE examinations, a number of licensees have discovered degradation of moisture barriers and significant corrosion of liner plates below the concrete floor levels. The staff asked the applicant, in RAI B.3.13-2, to provide a technical justification for the exception taken (i.e., one-time inspection of this area). By letter dated April 28, 2003, the applicant provided the following response.

> RNP has received NRC approval for relief from Subsection IWE of ASME Section XI. This is documented in a letter from Herbert N. Berkow (NRC) to D.E. Young (CP&L) dated July 26, 1999 titled, "Evaluation of Relief Requests IWE/IWL-1 through IWE/IWL-9: Implementation of Subsections IWE and IWL of ASME Section XI For Containment Inspection for Carolina Power and Light Company's H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2) (TAC No. MA4637)." Relief Request IWE/IWL-01 has been approved to provide a VT-3 examination on those portions of the insulated moisture barriers and liner plate that are exposed when a maintenance activity requires removal of the insulation. Although Relief Requests IWE/IWL-01 and IWE/IWL-02 do not require examination of these "inaccessible" areas, 10 CFR 50.55a(b)(2)(ix)(a) does require the evaluation of these inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. These areas of the moisture barrier and containment liner were made accessible by removing the liner insulation and performing an examination. These areas were analyzed as stated in RNP Response to RAI 3.5.1-19 and determined not to impact the structural integrity or leak-tightness of containment. Some areas of the moisture barrier and liner plate are behind permanent structures, or due to ALARA concerns some could not be inspected. These inaccessible areas were analyzed and determined not to impact the structural integrity or leak-tightness of containment and determined to be acceptable for continued service until 2005, based on using worst case corrosion rates as discussed in the RNP Responses to RAI 3.5.1-7 and RAI 3.5.1-19. A one-time inspection was assigned for completing these inspections by year 2005. If additional inspections are required, they will be determined and scheduled at that time.

The staff reviewed this response in conjunction with the applicable relief request and the responses provided to RAIs 3.5.1-7 and 3.5.1-19. Based on these reviews, the staff determined that (1) by the 2005 outage, the applicant will perform a focused inspection of the liner plate behind the moisture barrier and the insulation at the junction of the wall and the concrete at elevation 228 ft., (2) the applicant will perform the periodic examination of these areas as required by 10 CFR 50.55a and Subsection IWE, and (3) as a result of the inspection performed in 2005 , if additional inspections are required, the applicant will determine the time and schedule of the additional examinations. Based on this determination, the staff finds the mechanism used by the applicant to monitor these areas acceptable.

The applicant summarized its implementation process, and the operating experience related to the degradation of the liner, protective insulation sheathing, penetration bellows, bulging of the liner plate, and corrosion of the external vertical liner plate of the ECCS sump. The applicant stated that it has evaluated all these degradations, taken corrective actions where warranted, and ensured itself that the requirements of containment structure are met. The staff asked the applicant, in RAI B.3.13-3, to provide acceptance criteria for bulging of the liner plate. By letter dated April 28, 2003, the applicant provided the following response.

> The bulge in the containment liner was analyzed in the "HB Robinson Unit No. 2 Containment Liner Stress Analysis Report," dated June 21, 1974. A finite element approach was used for the liner and stud stress analysis. Broken adjacent stud anchors were postulated. Neither the stud load nor liner stress exceeded the allowable criteria of the materials used. The bulged liner and remaining anchor studs were determined to be effective to meet their functional requirements during a LOCA and during normal plant operating conditions. The bulge is believed to have been present since initial construction. A strain monitoring program was initiated for one cycle which indicated no gross movement or growth of the liner. A letter from E. Utley (CP&L) to Robert W. Reid (NRC), Serial NG-76-443, dated March 25, 1976, summarized the findings and provided a summary of the analysis used to demonstrate the integrity of the bulged liner. Two additional bulged liner areas were discovered in 1992. These areas are also believed to have existed since initial construction. These bulges were determined to be enveloped by the evaluation performed for the bulge discovered in 1974. These bulges were monitored in 1993 with negligible movement and were considered stable and acceptable, with no further monitoring required.

A review of the summary of the bulged liner plate analysis in the applicant's March 1976 letter and the recent examinations indicate that the bulges are stable and the maximum liner strain associated with the bulged liner is 0.0013, which is less than 40 percent of the strain permissible by Table CC-3720-1 of Division 2 of Section III of the ASME Code. Based on the observations made by the applicant during subsequent pressure tests and inspections, the staff concludes that such bulging will not be detrimental to the containment function during the period of extended operation. However, the staff recommends monitoring of such liner plate bulges during subsequent inspections performed under this program.

The staff's review of the applicant's program implementation process and the method of evaluating containment degradation indicates that the applicant is effectively implementing the AMP and, therefore, the staff finds these actions acceptable.

Section A.3.1.21 of the UFSAR Supplement briefly summarizes the program and makes a note that prior to the start of the extended period of operation, the program will be enhanced to (1) specify the requirements for conducting reexaminations, and (2) document that repairs meet the specified acceptance standards. Neither the LRA nor the UFSAR Supplement states the edition and addenda of the ASME Code being implemented. As amendment of UFSAR is a continuing process, the staff believes it would be appropriate to state the edition and addenda of the ASME Code being used in the UFSAR Supplement. The relief requests granted from the specific edition and addenda of the Code should also be listed in the UFSAR Supplement (and subsequent addenda). The applicant was asked in RAI B.3.13-4 to provide information pertinent to the implementation of the program. By letter dated April 28, 2003, the applicant provided the following response.

> The current code of record for the IWE/IWL Containment Examination Program is the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 1992 edition through 1992 Addenda, subject to the limitations and modifications of 10 CFR 50.55a(b)(2). The current program comprises the first containment inspection interval and is effective from September 9, 1998 to September 8, 2008. The relief requests are listed in a letter from Herbert N. Berkow (NRC) to D. E. Young (CP&L), titled: "Evaluation of Relief Requests IWE/IWL-1 through IWE/IWL-9: Implementation of Subsections IWE and IWL of ASME Section XI For Containment Inspection for Carolina Power and Light Company's H.B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2) (TAC No. MA4637)," dated July 26, 1999. The first Containment Examination Program Interval (2008) ends prior to the extended period of operation (2010). During the extended period of operation, RNP will continue to meet the requirements of the Code incorporated by reference in 10 CFR 50.55a. Therefore, please note that the Code of record and relief requests will change prior to the extended period of operation. In consideration of the above, the information in the first paragraph of LRA Subsection A.3.1.21, ASME Section XI, Subsection IWE Program, is modified to read:
>
> The ASME Section XI, Subsection IWE, Program consists of periodic visual, surface, and volumetric inspection of steel containment components for signs of degradation, assessment of damage, and corrective actions. This program is in accordance with ASME Section XI, Subsection IWE, and in accordance with 10 CFR 50.55a(g), with modifications and approved relief requests.

The applicant provided the requested information about the implementation of Subsection IWE of Section XI of the ASME Code. With the modification noted in the above paragraph, the applicant has properly characterized the scope of the IWE program, and the staff finds the modified paragraph in LRA Subsection A.3.1.21 acceptable.

3.5.2.3.1.3 Conclusions

On the basis of its review and audit of the applicant's program, the staff finds that those portions of the program for which the applicant claims consistency with the GALL program are consistent with the GALL program. In addition, the staff has reviewed the exceptions to the GALL program and finds that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the UFSAR Supplement for this AMP and finds that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

Therefore, the staff concludes that actions have been identified and have been or will be taken to manage the effects of aging during the period of extended operation on the functionality of SCs subject to an AMR such that there is reasonable assurance that the activities authorized by a renewed license will continue to be conducted in accordance with the CLB, as required by 10 CFR 54.29(a).

3.5.2.3.2 ASME Section XI, Subsection IWL Program

3.5.2.3.2.1 Summary of Technical Information in the Application

The applicant described its ASME Section XI, Subsection IWL Program in Section B.3.14 of the LRA. The LRA states that this program is consistent with GALL Program XI.S2, "ASME Section XI, Subsection IWL," with the following exceptions, (1) RNP did not identify the aging effects of cracking and loss of bond due to corrosion of embedded steel, but did identify loss of material due to the aging mechanism of corrosion of embedded steel and applies the ASME Section XI, Subsection IWL Program, (2) the requirements of ASME Section XI, Subsection IWL, do not apply to the RNP prestressing system because the plant design includes a grouted tendon system, which is outside the scope of Subsection IWL, (3) RNP aging effects/mechanisms include cracking of concrete and change in material properties of concrete due to fatigue at penetration anchors, while these are not addressed in the GALL, (4) erosion of porous concrete subfoundation is not an applicable aging mechanism since porous concrete was not used at RNP under the containment building, and (5) GALL identifies "Increase in porosity, permeability" as aging effects for concrete in Section II.A1, while RNP considers this effect a part of "change in material properties." The applicant credits the ASME Section XI, Subsection IWL Program for aging management of selected components of the reactor containment building at RNP.

The applicant identified the aging effects/mechanisms of concern as (1) change in material properties due to aggressive chemical attack, (2) loss of material due to aggressive chemical attack, (3) loss of material due to corrosion of embedded steel, (4) change in material properties due to fatigue, and (5) cracking due to fatigue.

The applicant further stated that as a result of the license renewal review, administrative controls associated with program element *Scope of Program* will be enhanced to notify Civil/Structural Design Engineering of the location and extent of proposed excavations and to require Civil/Structural Design Engineering to examine representative samples of below-grade concrete when excavated for any reason.

Under the program element "Operating Experience," the applicant states that the program is implemented and maintained in accordance with the general requirements of engineering programs, and asserts that the programs (in general) are effectively implemented through the use of qualified personnel and adequate resources and are managed in accordance with plant administrative controls. Moreover, the applicant makes a point that generic operating experience includes NUREG-1522, "Assessment of Inservice Conditions of Safety Related Nuclear Plant," June 1995, and that RNP was one of the six plants that was inspected in support of this document.

In the plant-specific operating experience, the applicant identified degradation of containment concrete as (1) concrete surface staining, cracking, and spalling at the north and south cable vault rooms, (2) degraded radial construction joint at the base of the crane wall in the area of the ECCS sump, (3) degraded concrete between elevations 226 and 232 ft. on the southwest side of the containment between the equipment hatch and the CV access area (1992), and (4) degradation of grout covering in the dome (1984). For these occurrences, the applicant briefly described corrective actions taken.

The applicant further stated that this AMP is continually upgraded based on the industry experience and research, and that the Corrective Action Program has been effective in ensuring that the ASME Section XI, Subsection IWL Program is continually improving.

3.5.2.3.2.2 Staff Evaluation

In LRA Section B.3.14 , "ASME Section XI, Subsection IWL Program," the applicant described its program to manage aging of containment building components at RNP. The LRA states that this program is consistent with GALL Program XI.S2, "ASME Section XI, Subsection IWL," with the following exceptions, (1) RNP did not identify the aging effects of cracking and loss of bond due to corrosion of embedded steel, but did identify loss of material due to the aging mechanism of corrosion of embedded steel and applies the ASME Section XI, Subsection IWL Program, (2) the requirements of ASME Section XI, Subsection IWL, do not apply to the RNP prestressing system because the plant design includes a grouted tendon system, which is outside the scope of Subsection IWL, (3) RNP aging effects/mechanisms include cracking of concrete and change in material properties of concrete due to fatigue at penetration anchors, while these are not addressed in the GALL, (4) erosion of porous concrete subfoundation is not an applicable aging mechanism since porous concrete was not used at RNP under the containment building, and (5) GALL identifies "increase in porosity, permeability" as aging effects for concrete in Section II.A1, while RNP considers this effect to be part of "change in material properties." The staff confirmed the applicant's claim of consistency during the AMR inspection. Furthermore, the staff reviewed the deviation and its justification to determine whether the AMP, with the deviation, remains adequate to manage the aging effects for which it is credited. The staff also reviewed the UFSAR Supplement to determine whether it provides an adequate description of the revised program. In addition, the staff determined whether the applicant properly applied the GALL program to its facility.

The applicant has appropriately described the purpose of the program and the aging effects/mechanisms that will be managed through the implementation of the program. Moreover, the applicant states that administrative controls associated with the program element *Scope of Program* will be enhanced to notify Civil/Structural Design Engineering of the location and extent of proposed excavations and to require Civil/Structural Design Engineering to examine representative samples of below-grade concrete when excavated for any reason. Because of the high acidity of the soil at the plant site, the staff considers the enhancement appropriate.

The staff asked the applicant to provide information regarding the present condition of the below-grade concrete basemat based on the inspections performed during certain maintenance activities. By letter dated April 28, 2003, the applicant provided the following response.

> The soil at Robinson Nuclear Plant is considered aggressive because of the ground water pH being slightly less than 5.5. This is considered to be slightly acidic, rather than highly acidic. Below-grade examinations of concrete have been performed at certain locations with satisfactory results. These include a below-grade section of

the RAB, internal surfaces of electrical manholes exposed to ground water, submerged portions of the intake structure, and the dam spillway exposed to lake water. The lake water environment for the intake structure and dam spillway is essentially the same as that of aggressive ground water (pH values are both below 5.5); as such, inspection results in these areas should envelope aging effects encountered by below-grade concrete of other structures, such as the containment basemat. In addition, an enhancement has already been made to a plant procedure, which requires an examination of any exposed concrete surfaces by engineering prior to backfilling. Please refer to the RNP Response to RAI 3.5.1-3 for more detailed discussion of lake water and ground water chemistry.

Having reviewed the RNPs response to RAI B.3.14-1, the staff requested the applicant to provide a summary of the results of inspections performed (1) in the below-grade sections of the RAB, (2) the submerged portions of the intake structure, and (3) the dam spillway, that would support a conclusion that the below-grade structures have not been degraded, and that the scope of the enhanced inspection is adequate to detect any significant degradation of the below-grade structures during the extended period of operation. The applicant provided the following summary of the results of inspections performed in the (1) below-grade sections of the RAB, (2) submerged portions of the intake structure, (3) dam spillway, and (4) other below-grade concrete.

(1) Below-grade sections of the RAB

A visual inspection of the below-grade portion of the RAB foundation approximately three feet deep was performed in July 1999 while the east foundation was exposed during excavation for construction of the north service water header support slab. This general visual inspection monitored for spalling, scaling, erosion, swelling, bulging, signs of corrosion, cracking, settlement, and exposed rebar. In addition, the interior of Manholes 35 and 36, which abut the RAB, were inspected on September 30, 2002. The interior, which had been exposed to ground water since initial construction, had no signs of spalling or other concrete degradation.

(2) Submerged portion of the Intake Structure

An inspection of the inaccessible areas was performed during Refueling Outage 19 from September 28, 1999, to October 2, 1999, using divers and video equipment. The results of the inspection are as follows. The concrete surface had very little marine growth. There was little or no sediment on the bottom slab. The concrete located at the water line showed signs of erosion from the constant wave action. The top coat of mortar has eroded away leaving the aggregate exposed. The average loss of cover is approximately 1/16 inch to 1/8 inch. The concrete surface was cleaned of marine growth in a number of locations with a wire brush. The top coat came off with minor effort, thereby exposing the aggregate. Sound material was observed at all cleaned locations. Several repairs were observed to have been made in various locations. One repair had flaked off and rebar was observed (one end cut). The repair material thickness was approximately 2 inches and the repair area was about one square foot. This area was determined by the Robinson Engineering Section to have no impact on the structural integrity of the concrete.

(3) Dam Spillway

An underwater inspection was performed June 20, 2000, by divers. The spillway inspection examined the condition of concrete, especially at the tainter gates. A spalled portion of concrete (6" by 8" by 4" deep) was identified. This area is scheduled to be reinspected and repaired prior to the period of extended operation. The Dam Inspection Program will monitor the condition of the normally inaccessible submerged spillway concrete surfaces at a frequency not to exceed 10 years. No other underwater concrete degradation was identified.

(4) Other

The interior of eight security manholes were visually examined in August 2002. The interior concrete has been partially submerged from ground water and provides a similar environment as below-grade concrete (exposure to slightly acidic ground water). No cracking, loss of material, or change in material properties was observed in the concrete surface.

In a conference call with the applicant which occurred on June 16, 2003, the staff pointed out that the applicant did not specify appropriate remedial measures to be followed if the results of RNP's periodic, submerged inspection of the intake structure concrete show significant concrete degradation. Subsequent to this conference call, the applicant, through an e-mail communication, has agreed to the following in order to ensure adequate aging management of below-grade structural concrete that is within the scope of the AMR.

Degradation to submerged concrete observed during periodic under water inspections at the Intake Structure and RNP Dam Spillway will be used as a leading indicator for potential degradation to below-grade concrete structures in the scope of License Renewal. Below-grade concrete will be evaluated and/or examined for potential degradation and corrective actions taken as determined by Robinson Engineering Support Section. This applies to below-grade concrete examined by the Structures Monitoring Program (SMP) and the ASME Section XI, Subsection IWL Program. Applicable SMP and IWL Program procedures will be enhanced to incorporate these changes.

Ground water and lake water monitoring results (pH, chlorides, sulfates) will be reviewed by Engineering and trended. Increasing aggressiveness of the ground water and lake water will also be used as a leading indicator for potential degradation to below-grade concrete structures in the scope of License Renewal as described above.

Below-grade concrete, when exposed during excavation, already requires notification of Robinson Engineering Support Section for inspection. However, degradation to below-grade concrete due to aggressive ground water, when exposed during excavation, will also be used as a leading indicator for potential degradation to other below-grade concrete structures in the scope of License Renewal as described above.

The staff finds the above commitments adequate to address its concerns regarding the aging management of below-grade, in-scope concrete structural components at RNP. The applicant also committed to provide appropriate documentation of the above agreement. This item was designated as Confirmatory Item 3.5-1. Based on the discussion related to closure of Confirmatory Item 3.5-1 provided in Section 3.5.2.2, "Aging Management Evaluations in the GALL Report That Are Relied On for License Renewal, For Which GALL Recommends Further Evaluation," of this SER, Confirmatory Item 3.5-1 is closed.

Because of the slightly acidic RNP ground water environment, the applicant conservatively assumed existence of an aggressive chemical environment and proposed the above described plant-specific AMPs (an enhanced ASME, Section XI, Subsection IWL Program for containment and an enhanced Structures Monitoring Program for other Category 1 structures) to manage the aging effects of below-grade concrete. As such, the staff finds RAI B.3.14-1 to be fully resolved.

The applicant also described the operating experience related to the degradation of containment concrete, and the evaluation and corrective actions taken. The operating experience related to the containment concrete degradation states, "An evaluation concluded that not providing cooling to the penetrations with hot piping does not degrade the concrete. Degradation has not occurred and does not require augmented examinations." The staff notes that most of the high-temperature-related degradation would be in the concrete around the liner plate (or insert plate). Any degradation occurring in the area cannot be seen by visual examination. Therefore, the staff asked the applicant, in RAI B.3.14-2, to provide information on (1) the sustained temperature in the concrete/liner interface around the hot penetrations, and (2) the use of other NDE examination to ensure that the concrete on the back of the liner is not degraded. By letter dated April 28, 2003, the applicant provided the following information.

> The maximum pipe temperature is 380 °F, and the temperature of the sleeve and concrete was calculated as 208.5 °F. This is conservative, since the calculation assumed 130 °F ambient air over a period of 200 hours. The RHR system is in operation above 200 °F during cooldown for 10 hours, and for 22 hours during the heatup transient. These values are based on plant experience, rather than the 40 hours conservatively assumed in the plant calculation. After 22 hours, the temperature of the sleeve and concrete is at 162.3 °F.
>
> No other examinations have been completed or are planned for the affected concrete, other than those required in accordance with the ASME Section XI, Subsection IWL Program. A concrete surface examination of the area around the applicable RHR penetration (S-15) performed in May 2001 in accordance with the ASME Section XI, Subsection IWL Program identified some notches which had been cut out for small piping routed to the penetration. The inspection found no evidence of in-service degradation, and the inspection results were acceptable.

Additionally, the applicant asserts that the concrete at the RHR penetration meets the design requirements as discussed in the RNP response to RAI 4.6.3-2. The staff reviewed the above in conjunction with the applicant's response to RAI 4.6.3-2. The Code requirements pertinent to the temperatures in concrete are those contained in Subparagraph CC-3440 of Section III, Division 2 of the ASME Code. The requirements permit sustained temperatures up to 200 °F for the concrete around penetrations. The discussion in the applicant's responses indicate that the maximum temperatures around RHR penetration will be 208 °F, for 10 hours during the cooldowns, and 22 hours during heatup transients. Under this type of temperature conditions, the staff believes that the applicant's evaluation related to the concrete compressive strength provided in response to RAI 4.6.3-2 is conservative. The surface inspections performed of the concrete around the penetration did not indicate evidences of inservice degradation. As the applicant will be performing IWL inspections during the extended period of operation, the staff considers the applicant's evaluation of the concrete around the RHR penetration acceptable.

The staff reviewed the exceptions to the GALL Program XI.S2 and concludes that all the plant specific exceptions are reasonable and appropriate.

The staff's review of the applicant's program implementation process and the method of evaluating containment degradation indicate that the applicant is effectively implementing the AMP and the staff finds these actions to be acceptable.

Section A.3.1.22 of the UFSAR Supplement briefly summarizes the program and makes a note that prior to the start of the extended period of operation, the program will be enhanced to (1) specify the requirements for conducting reexaminations, and (2) document that repairs meet the specified acceptance standards. Neither the LRA nor the UFSAR Supplement states the edition and addenda of the ASME Code being implemented. RAI B.3.13-3 pertained to this subject. In its response dated April 28, 2003, the applicant proposed to change the information in the first paragraph of LRA Subsection A.3.1.22, "ASME Section XI, Subsection IWL Program," to include the following.

> The ASME Section XI, Subsection IWL Program consists of periodic visual inspection of concrete surfaces of reinforced and prestressed concrete containments for signs of degradation, assessment of damage, and corrective actions. This program is in accordance with the ASME Code Section XI, Subsection IWL, and addenda in accordance with 10 CFR 50.55a(g), with modifications and approved relief requests. The RNP prestressing tendons are grouted in place. Therefore, ASME Section XI, Subsection IWL rules regarding unbonded post-tensioning systems are not applicable.

The proposed change adequately describes the process to be used for performing inspections in accordance with the ASME Section XI, Subsection IWL Program during the period of extended operation and is acceptable.

3.5.2.3.2.3 Conclusions

On the basis of its review and audit of the applicant's program, the staff finds that those portions of the program for which the applicant claims consistency with the GALL program are consistent with the GALL program. In addition, the staff has reviewed the exceptions to the GALL program and finds that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the UFSAR Supplement for this AMP and finds that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

Therefore, the staff concludes that actions have been identified and have been or will be taken to manage the effects of aging during the period of extended operation on the functionality of SCs subject to an AMR such that there is reasonable assurance that the activities authorized by a renewed license will continue to be conducted in accordance with the CLB, as required by 10 CFR 54.29(a).

3.5.2.3.3 ASME Section XI, Subsection IWF Program

3.5.2.3.3.1 Summary of Technical Information in the Application

The applicant described its ASME Section XI, Subsection IWF Program in Section B.2.6 of Appendix B of the LRA. The LRA states that this program is consistent with GALL Program XI.S3, "ASME Section XI, Subsection IWF." The applicant stated that the program is credited for aging management of Class 1, 2, and 3 component supports (including piping supports) for loss of material due to general corrosion.

The program is a condition monitoring program that provides for the implementation of ASME Code Section XI, Subsection IWF, in accordance with the provisions of 10 CFR 50.55a. The 10-year examination plan provides a systematic guide for performing NDE of passive components in the scope of license renewal.

Under the program element "Operating Experience," the LRA states that discrepancies found during the visual examination of supports have been transmitted to engineering personnel for evaluation. The LRA also states that the processes at RNP are continually upgraded based upon industry experience, research, and ongoing self-assessments.

3.5.2.3.3.2 Staff Evaluation

In LRA Section B.2.6, "ASME Section XI, Subsection IWF Program," the applicant described its program to manage aging of Class 1, 2, and 3 component supports at RNP. The applicable aging effect is loss of material. The LRA states that this program is consistent with GALL Program XI.S3, "ASME Section XI, Subsection IWF." The staff confirmed the applicant's claim of consistency during the AMR inspection. In addition, the staff determined whether the applicant properly applied the GALL program to its facility. Furthermore, the staff reviewed the UFSAR Supplement to determine whether it provides an adequate description of the revised program.

In Section B.2.6 of the LRA, the applicant identified loss of material due to general corrosion as the only aging effect/mechanism of concern. The program would examine hangers for loss of mechanical function; however, loss of mechanical function was not identified as an age-related degradation in the RNP AMR. In RAI B.2.6-2, the staff asked the applicant to elaborate on the extent to which the component supports are examined for loss of mechanical function and explain why loss of mechanical function for supports was not identified as an age-related degradation in its AMR. The staff also asked the applicant to discuss how its visual examination would be consistent with the GALL IWF program in monitoring or inspecting component supports for corrosion, deformation, misalignment, improper clearances, improper spring settings, damage to close tolerance machined or sliding surfaces, and missing, detached, and/or loosened support items. By letter dated April 28, 2003, the applicant stated that the RNP AMR for the IWF program component supports concluded that the only aging effect/mechanism of concern was loss of material due to general corrosion. The applicant stated that the concerns for loss of mechanical function were addressed in the AMR but their occurrence could not be specifically attributed to aging. The applicant stated that a review of the potential loss of component support intended functions, and the RNP plant reports for component support deficiencies, determined that they could be design related or due to an unplanned plant operational occurrence, but not due to aging. However, the RNP IWF program for component supports currently requires supports to undergo periodic inspections, and the program does examine supports for loss of material due to general corrosion and loss of mechanical function. Although not a requirement for the LRA, the applicant stated that the program examines supports for loss of mechanical function in accordance with Table IWF-2500-1 of Subsection XI (1989 Edition) in the following manner.

- (F1.10) mechanical connections to pressure-retaining components and building structure
- (F1.20) weld connections to building structure
- (F1.30) weld and mechanical connections at intermediate joints in multi-connected integral and nonintegral supports
- (F1.40) clearances of guides and stops, alignment of supports, and assembly of support items
- (F1.50) spring supports and constant load supports
- (F1.60) sliding surfaces
- (F1.70) hot and cold position of spring supports and constant load supports

Because the applicant has committed to manage loss of mechanical function and the information provided above by the applicant resolves the staff's concern regarding the extent of the support examination, the staff finds it acceptable.

The applicant stated that the program provides for VT-3 visual examination for ASME Class 1, 2, and 3 component supports, consistent with GALL requirements. The applicant stated that the operating experience review determined that documentation exists which demonstrates that discrepancies found during the visual examination of supports are transmitted to engineering personnel for evaluation. The visual examinations of ASME Class 1, 2, and 3 component supports look for deformations or structural degradations, corrosion, and other conditions, as stated above, that could affect the intended function of the support. The staff believes that fairly large cracks would be identified for the component supports that are inspected and finds the applicant's VT-3 visual examination to be consistent with GALL and, therefore, acceptable.

The applicant confirmed that this program will be implemented consistently with the requirements of 10 CFR 50.55a throughout the period of extended operation to satisfy the requirements for the aging management of ASME Class 1, 2, and 3 component supports. The LRA states that the program is subject to ongoing self-assessments and, when weaknesses are noted, the Corrective Action Program is used to initiate program improvements. The staff finds that the operating experience supports the applicant's conclusion that the ASME Section XI, Subsection IWF Program is effectively managing aging and is, therefore, acceptable.

The staff reviewed the UFSAR Supplement summary description of the ASME Section XI, Subsection IWF program in Appendix A .3.1.6 of the LRA. The staff finds that the information in the UFSAR Supplement provides an adequate summary of the program activities..

3.5.2.3.3.3 Conclusions

On the basis of its review and audit of the applicant's program, the staff finds that those portions of the program for which the applicant claims consistency with the GALL program are consistent with the GALL program. In addition, the staff has reviewed the exceptions to the GALL program and finds that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the UFSAR Supplement for this AMP and finds that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

Therefore, the staff concludes that actions have been identified and have been or will be taken to manage the effects of aging during the period of extended operation on the functionality of SCs subject to an AMR such that there is reasonable assurance that the activities authorized by a renewed license will continue to be conducted in accordance with the CLB, as required by 10 CFR 54.29(a).

3.5.2.3.4 10 CFR Part 50, Appendix J Program

3.5.2.3.4.1 Summary of Technical Information in the Application

The applicant's 10 CFR Part 50, Appendix J Program is discussed in Section B.2.7 of the LRA. The LRA states that the program is consistent with GALL Program XI.S4, "10 CFR Part 50, Appendix J." The applicant credits the program for aging management of selected components of the reactor containment building at RNP. The LRA identifies the aging effects/mechanisms of concern as (1) cracking due to elevated temperature, (2) cracking due to thermal fatigue, (3) change in material properties due to elevated temperature, (4) loss of material due to general corrosion, wear, aggressive chemical, crevice corrosion, galvanic corrosion, and pitting.

Under the program element "Operating Experience," the LRA states that the program is implemented in accordance with the general requirements of engineering programs, and that the programs (in general) are effectively implemented through the use of qualified personnel and adequate resources, and are managed in accordance with plant administrative controls. Moreover, the applicant stated that the program is continually upgraded based on industry experience and research. This AMP has provided an effective means of ensuring the structural integrity and leak tightness of the RNP containment. The LRA also states that, in addition to industry experience, plant-operating experiences are shared between CP&L and Florida Power Corporation (FPC) sites through regular peer group meetings.

The applicant provided the following broad statement regarding its operating experience.

> Based on a review of condition reports and inspection results, the corrective action program (CAP) has been effective in ensuring that the Appendix J program is continually improving. Several Condition Reports have been generated as a result of as-found conditions or as a result of assessments (site and corporate). When

weaknesses are noted, actions are taken under the CAP to initiate program improvements. Program improvements were also made as a result of NRC Inspections.

3.5.2.3.4.2 Staff Evaluation

In LRA Section B.2.7, "10 CFR Part 50, Appendix J Program," the applicant described its AMP to manage various components in the reactor containment building. The LRA stated that this program is consistent with GALL Program XI.S4, "10 CFR Part 50, Appendix J," with no deviations. The staff confirmed the applicant's claim of consistency during the AMR inspection. In addition, the staff determined whether the applicant properly applied the GALL program to its facility. The staff also reviewed the UFSAR Supplement to determine whether it provides an adequate description of the program.

The staff conceptually considers the Appendix J program as a program to ensure the leak-tight integrity of the containment (as described in GALL Section XI.S4), and the containment ISI program (ASME, Section XI, Subsection IWE Program) as the AMP for detecting the aging degradation of containment pressure boundary components. These programs complement each other and are required to assure that the containment continues to perform its intended functions, as described in Table 2.4-1 of the LRA.

The staff noted that the LRA description of the purpose of the program is not consistent with the program description stated in GALL Program XI.S4. The LRA identified aging effects/mechanisms of concern that cannot be readily detected by performing leakage rate tests as described in GALL Program XI.S4. In RAI B.2.7-1, the staff asked the applicant to provide either a clear description of the purpose of the program that would be consistent with GALL Program XI.S4, or to develop a 10 element program that is consistent with the intended use of the program and an explanation of how the leak-tight integrity of the containment will be maintained during the extended period of operation. By letter dated April 28, 2003, the applicant explained that the implementation of this program detects degradation of the pressure retaining components in conjunction with the implementation of Subsection IWE of Section XI of the ASME Code, and reiterated that the program is consistent with Section XI.S4 of the GALL Report. The staff finds this interpretation of the purpose of the program acceptable.

In RAI B.2.7-2, the staff asked the applicant to clarify which of the options will be used during the extended period of operation, since in the element *Scope of Program* of GALL Section XI.S4, the program provides an option for leakage testing of containment isolation valves either (1) under Appendix J, Type C test, or (2) along with the tests of the systems containing isolation valves. By letter dated April 28, 2003, the applicant provided the following response.

> RNP currently performs Appendix J, Type C tests on containment isolation valves at intervals prescribed by and in accordance with the requirements of 10 CFR 50, Appendix J. While there are no plans to change the method of testing in the near future, the RNP Appendix J Program is continually upgraded based on industry experience and research. Additionally, improved technology or techniques may result in the adoption of different leakage testing techniques during the extended period of operation. Any such changes are expected to involve a license amendment request, or will otherwise be controlled in accordance with 10 CFR 50.59 and/or applicable plant procedures.

The staff recognizes the potential for changes in performing leakage rate testing of containment isolation valves based on the improved technology or techniques, and finds the stated processes that will be utilized for making those changes adequate and acceptable.

The LRA, under *Operating Experience*, states, "Several Condition Reports have been generated as a result of as-found conditions or as a result of assessments (site and corporate)." In RAI B.2.7-3, the staff asked the applicant to provide a summary of condition reports where significant as-found leakages (Type A, Type B, and Type C tests) were found (e.g., more than twice the acceptance criteria), including the corrective action taken. By letter dated April 28, 2003, the applicant stated that a review of the Corrective Action Program database identified no specific conditions where as-found leakages were greater than twice the acceptance criteria. The applicant stated that the as-found conditions cited in the LRA involve generic issues, such as using instruments with the wrong calibrated range, assessment findings of more desirable valve line-ups, or more desirable testing configurations. The applicant also stated that two instances involved findings that containment purge isolation valve V12-8 had exceeded its leakage acceptance criterion by a small margin; however, the condition was resolved by establishing that the original acceptance criterion was overly restrictive. The staff considers the applicant's process for corrective action adequate and acceptable.

The staff reviewed the UFSAR Supplement summary description of the 10 CFR Part 50, Appendix J program in Appendix A .3.1.7 of the LRA. The staff finds that the information in the UFSAR Supplement provides an adequate summary of the program activities.

3.5.2.3.4.3 Conclusions

On the basis of its review and audit of the applicant's program, the staff finds that those portions of the program for which the applicant claims consistency with the GALL program are consistent with the GALL program. In addition, the staff has reviewed the exceptions to the GALL program and finds that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the UFSAR Supplement for this AMP and finds that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

Therefore, the staff concludes that actions have been identified and have been or will be taken to manage the effects of aging during the period of extended operation on the functionality of SCs subject to an AMR such that there is reasonable assurance that the activities authorized by a renewed license will continue to be conducted in accordance with the CLB, as required by 10 CFR 54.29(a).

3.5.2.3.5 Structures Monitoring Program

3.5.2.3.5.1 Summary of Technical Information in the Application

The applicant described its Structures Monitoring Program in Section B.3.15 of the LRA. The LRA states that this program is consistent with GALL Program XI.S6, "Structures Monitoring Program." The applicant credits this program with aging management of civil SCs within the scope of license renewal. The LRA states that the aging effects and mechanisms of concern include the following.

Steel aging effects and mechanisms

- loss of material due to general corrosion
- loss of material due to crevice corrosion
- loss of material due to pitting corrosion

Concrete (below-grade) aging effects/mechanisms

- loss of material due to aggressive chemical attack
- loss of material due to corrosion of embedded steel
- change in material properties due to aggressive chemical attack

Elastomer aging effects/mechanisms:

- change in material properties due to elevated temperature
- cracking due to elevated temperature

The LRA also identifies a number of enhancements that the applicant will make to its current program (developed for the Maintenance Rule) for the condition monitoring of structures including the following.

- Include buildings and structures, and associated acceptance criteria, in scope for license renewal, but outside the scope of the Maintenance Rule. (Structures addressed in the Maintenance Rule already are in the Program.)

- Identify interfaces between structures monitoring inspections of concrete surfaces and the Fire Protection Program requirements for barriers.

- State clearly the boundary definition between systems and structures. The physical structure is inspected as part of the structure/building walkdown and includes the concrete structure and all structural steel (such as main building—structural steel, platform support steel, stairways, etc.).

- Revise administrative controls to provide inspection criteria for portions of systems covered by structures monitoring. Provide acceptance categories similar to those used for structures monitoring, and require that a condition report be initiated for all inspection attributes found to be unacceptable.

- Expand system walkdown inspection criteria to include observation of selected, adjacent components.

- Revise personnel responsibilities to include responsibilities to (1) provide assistance in evaluating structural deficiencies when requested by the Responsible Engineer, (2) inspect excavated concrete, and (3) notify Civil/Structural Design Engineering of location and extent of proposed excavations.

Under *Operating Experience*, the LRA states that the Structures Monitoring Program is a combination of the existing corporate procedure for condition monitoring of structures and the existing plant procedure for system walkdown, both of which were developed to support implementation of the Maintenance Rule, with the addition of the enhancements described above. The LRA states that the subject administrative controls have been proven effective for implementing the Maintenance Rule and are supported by the excellent operating experience for systems, SCs. The applicant stated that a review of condition reports and inspections performed has concluded that administrative controls are in effect and effective in identifying age-related degradation, implementing appropriate corrective actions, and continually upgrading the administrative controls used for structural monitoring.

3.5.2.3.5.2 Staff Evaluation

In LRA Section B.3.15 , "Structures Monitoring Program," the applicant described its program to manage the aging of civil SCs within the scope of license renewal. The LRA states that this program is consistent with GALL Program XI.S6, "Structures Monitoring Program." The staff confirmed the applicant's claim of consistency during the AMR inspection. In addition, the staff determined whether the applicant properly applied the GALL program to its facility. The staff also reviewed the UFSAR Supplement to determine whether it provides an adequate description of the program.

For the aging management of below-grade concrete structural components, the GALL Report recommends that additional measures be taken if an aggressive soil/ground water environment is present. Because RNP has acknowledged an aggressive soil/ground water environment due to a low pH value (< 5.5), the additional measure proposed for the aging management of below-grade concrete is to inspect these components when exposed during plant excavations done for other activities.

As stated in Section 3.5.2.2.1.1 of this SER, the staff found that RNP's approach of inspecting below-grade concrete only when it happens to be exposed during plant excavations done for other activities to be insufficient. As such, the staff requested further measures be taken to ensure the adequate aging management of below-grade concrete at RNP. In response to the staff's concerns, the applicant proposed to use its periodic inspections of the submerged portions of the intake structure and dam spillway as indicators for the condition of below-grade concrete at RNP. Because the ground water and lake chemistry are similar, degradation to submerged concrete will be used as a leading indicator for the potential degradation to below-grade concrete structures. In addition, the applicant committed to modify the Structures Monitoring Program to add this enhancement. This commitment was designated as Confirmatory Item 3.5-1. Based on the discussion related to closure of Confirmatory Item 3.5-1 provided in Section 3.5.2.2, "Aging Management Evaluations in the GALL Report That Are Relied On for License Renewal, For Which GALL Recommends Further Evaluation," of this SER, Confirmatory Item 3.5-1 is closed.

For concrete SCs outside of containment, the applicant stated that it will use the Structures Monitoring Program to manage loss of material and change in material properties. However, the applicant did not indicate that it would manage cracking as specified in the GALL Report. In addition, for several of the table entries in LRA Table 3.5-1, the applicant stated that the aging effect/mechanism combinations identified in the GALL Report are not applicable to RNP. The staff requested, in RAIs 3.5.1-3, 3.5.1-8 , and 3.5.1-11, that the applicant clarify its intent to manage the aging effect/mechanism combinations as recommended by the GALL Report. In response, the applicant stated that although it does not consider these aging effects to be applicable, it will manage the aging of concrete structures at RNP as recommended by the

GALL Report. As the applicant has committed to manage the aging of accessible concrete structural components at RNP, including cracking, the staff considers the response to the RAIs adequate.

The staff requested additional information (RAI B.3.15-2) regarding the aging management of elastomers. By letter dated April 28, 2003, the applicant stated, "The [Structures Monitoring Program] manages aging of the seismic joint filler commodity by visual inspection to note any indication of movement or distress, as well as a determination that the gaps meet design requirements and are free of debris. The [Structures Monitoring Program] manages aging of roof material by a visual inspection for degradation, damage, and/or leakage." The staff finds that this consistent is with GALL and acceptable.

The staff reviewed the UFSAR Supplement summary description of the Structures Monitoring program in Appendix A .3.1.23 of the LRA. The staff finds that the information in the UFSAR Supplement provides an adequate summary of the program activities.

3.5.2.3.5.3 Conclusions

On the basis of its review and audit of the applicant's program, the staff finds that those portions of the program for which the applicant claims consistency with the GALL program are consistent with the GALL program. In addition, the staff has reviewed the exceptions to the GALL program and finds that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the UFSAR Supplement for this AMP and finds that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

Therefore, the staff concludes that actions have been identified and have been or will be taken to manage the effects of aging during the period of extended operation on the functionality of SCs subject to an AMR such that there is reasonable assurance that the activities authorized by a renewed license will continue to be conducted in accordance with the CLB, as required by 10 CFR 54.29(a).

3.5.2.3.6 Dam Inspection Program

3.5.2.3.6.1 Summary of Technical Information in the Application

The applicant described its Dam Inspection Program in Section B.3.16 of the LRA. The applicant credits this program for aging management of selected components for Lake Robinson Dam within the scope of license renewal.

The LRA states that the aging effects and mechanisms of concern include the following.

Steel structures aging effects and mechanisms

- loss of material due to general corrosion
- loss of material due to crevice corrosion
- loss of material due to pitting corrosion
- loss of material due to microbiologically induced corrosion

Concrete structures aging effects and mechanisms

- loss of material due to aggressive chemical attack
- loss of material due to corrosion of embedded steel
- change in material properties due to aggressive chemical attack

Earthen structures aging effects and mechanisms:

- loss of form due to settlement

The applicant's program uses the Federal Energy Regulatory Commission FERC/US Army Corps of Engineers program, "Recommended Guidelines for Safety Inspection of Dams," which is one of the acceptable alternatives for managing the aging effects for water control structures documented in GALL, Section III.A6. This is a plant-specific program (e.g., not based on a GALL program), so the applicant described the program using the 10 elements from Appendix A of the SRP-LR.

3.5.2.3.6.2 Staff Evaluation

In LRA Section B.3.16, "Dam Inspection Program," the applicant described its program to manage aging of the Lake Robinson Dam. The program is not based on a GALL program; therefore, the staff reviewed the program using the guidance in Branch Technical Position RLSB-1 in Appendix A of the SRP-LR. The staff's evaluation focused on management of aging effects through incorporation of the following 10 elements from RLSB-1—program scope, preventive actions, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls, and operating experience. The applicant indicated that the corrective actions, confirmation process, and administrative controls for license renewal are in accordance with the site-controlled Quality Assurance Program. The staff's evaluation of the applicant's Quality Assurance Program is provided separately in Section 3.0.4 of this SER and the evaluation of the remaining seven elements is provided below. The staff also reviewed the UFSAR Supplement to determine whether it provides an adequate description of the program.

Program Scope: The LRA indicates that the program covers components of the Lake Robinson Dam and associated concrete structures consistent with the FERC/US Army Corps of Engineers program, "Recommended Guidelines for Safety Inspection of Dams." The staff has accepted the FERC program as a comprehensive program for managing the aging effects of dams. Therefore, the staff finds this acceptable.

Preventive Actions: The LRA states that the Dam Inspection Program is a condition monitoring program; therefore, preventive actions are not required. The staff agrees that the dam inspections are condition monitoring, and the staff had not identified the need for additional preventive actions; therefore, the staff finds this acceptable.

Parameters Monitored or Inspected: The LRA states that the parameters monitored are addressed in detail under Appendix II of "Recommended Guidelines for Safety Inspection of Dams." They include inspection of concrete structures, embankments, spillways, outlet works (gates, channels, sluices, etc.). The staff finds that this is consistent with the FERC program and, therefore, acceptable.

Detection of Aging Effects: The LRA states that the method of identifying aging effects is based on an independent inspection using the "Recommended Guidelines for Safety Inspection of Dams." The detection of aging effects uses a combination of visual field inspection and office review of available data, including records and operating history, to identify any actual or potential deficiencies, whether in the condition of the project works, the quality and adequacy of project maintenance, surveillance, or in the methods of operation. The dam inspections are conducted at five year intervals. The staff finds that this is consistent with the FERC program and, therefore, acceptable.

Monitoring and Trending: The LRA states that the dam inspections are conducted at five year intervals. The LRA further states that the "Recommended Guidelines for Safety Inspection of Dams," Phase I, Appendix 1, investigation report instructs the user to review the "history of previous failures or deficiencies and pending remedial measures for correcting known deficiencies and the schedule for accomplishing remedial measures should be indicated...," and recommends a review of inspection history, including the results of the last safety inspection. The staff finds that the overall monitoring and trending techniques proposed by the applicant are acceptable because inspections and review of inspection history, including the results of the last safety inspection activities, will effectively manage the applicable aging effects.

Acceptance Criteria: The LRA states that the acceptance criteria for the inspection and monitoring of Lake Robinson Dam are in accordance with the requirements of the "Recommended Guidelines for Safety Inspection of Dams," and, as such, will ensure the structure or component intended functions are maintained. The staff finds that this is consistent with the FERC program and, therefore, acceptable.

Operating Experience: The LRA states that five dam inspection reports (five-year intervals starting in 1980) were reviewed, along with a sample of Unit 1 visual inspection reports, yearly South Carolina dam inspections, and a year 2000 underwater visual inspection report for the spillway. Recommendations were made in each report and photographs were taken of typical areas and areas of concern. The LRA states that no significant issues were identified, and that recommended maintenance activities have been performed, as evidenced by succeeding inspection reports. The staff finds that the operating experience supports the applicant's conclusion that the Dam Inspection Program will effectively manage aging of the Lake Robinson Dam.

The staff reviewed the UFSAR Supplement summary description of the dam inspection program in Appendix A .3.1.24 of the LRA. The staff finds that the information in the UFSAR Supplement provides an adequate summary of the program activities.

3.5.2.3.6.3 Conclusions

On the basis of its review of the applicant's program, the staff finds that the program adequately addresses the 10 program elements defined in Branch Technical Position (BTS) RLSB-1 in Appendix A.1 of the SRP-LR, and that the program will adequately manage the aging effects for which it is credited so that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 50.21(a)(3). The staff also reviewed the UFSAR Supplement for this AMP and finds that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

Therefore, the staff concludes that actions have been identified and have been or will be taken to manage the effects of aging during the period of extended operation on the functionality of SCs subject to an AMR such that there is reasonable assurance that the activities authorized by a renewed license will continue to be conducted in accordance with the CLB, as required by 10 CFR 54.29(a).

3.5.2.4 Aging Management Review of Plant-Specific Structures and Structural Components

In this section of the SER, the staff presents its review of the applicant's AMR for specific structural components. To perform its evaluation, the staff reviewed the components listed in LRA Table 2.4-1 through 2.4-12 to determine whether the applicant properly identified the applicable aging effects and AMPs needed to adequately manage these aging effects. This portion of the staff's review involved identification of the aging effects for each component, ensuring that each component was evaluated in the appropriate LRA AMR Table in Section 3, and that management of the aging effect was captured in the appropriate AMP. The results of the staff's review are provided below.

3.5.2.4.1 Containment

3.5.2.4.1.1 Summary of Technical Information in the Application

The AMR results for the containment are presented in Tables 3.5-1 and 3.5-2 of the LRA. The applicant used the GALL Report format to present its AMR of containment components in LRA Table 3.5-1. In LRA Table 3.5-2, the applicant identified the component group designation along with its (1) material, (2) environment, (3) aging effect/mechanism, and (4) AMP(s).

As described in Section 2.4.1.1 of the LRA, the containment structure is a steel-lined concrete shell in the form of a vertical right circular cylinder with a hemispherical dome and a flat base. The containment includes the protective concrete structure outside the containment around the personnel and equipment hatch areas. The containment encloses the reactor and major components of the RCS and other important systems that interface with the RCS. Also, the containment houses and supports components required for reactor refueling. These include the polar crane, refueling cavity, and portions of the fuel handling system, which are included with components on the interior of the containment structure.

The materials of construction for the containment structure, as discussed in Section 2.4.1 of the LRA, are concrete, steel, and miscellaneous materials such as containment liner insulation and elastomers. These materials are exposed to containment air, outdoor air, borated water, and a buried environment.

<u>Aging Effects</u>

The LRA identifies the following aging effects for the containment structure.

- cracking, loss of material, and change in material properties for concrete components
- cracking and loss of material for SS penetration sleeves, bellows, and other SS components

- cumulative fatigue for penetration bellows (TLAA)
- loss of material for carbon steel components
- loss of prestress for containment tendons (TLAA)
- change in material properties and cracking for elastomers (results in loss of seal)

Aging Management Programs

The LRA credits the following AMPs with managing the identified aging effects for the containment structure.

- ASME Section XI, Subsection IWL Program
- ASME Section XI, Subsection IWE Program
- ASME Section XI, Subsection IWF Program
- 10 CFR Part 50, Appendix J Program

- Water Chemistry Program
- Structures Monitoring Program
- Boric Acid Corrosion Program
- One-Time Inspection Program
- Inspection of Overhead Heavy Load Handling Systems Program and Light Load Handling Systems
- Fire Water System

A description of these AMPs is provided in Appendix B of the LRA.

3.5.2.4.1.2 Staff Evaluation

In addition to Section 3.5 of the LRA, the staff reviewed the pertinent information provided in Section 2.4, "Scoping and Screening Results—Structures," and the applicable AMP descriptions provided in Appendix B of the LRA, to determine whether the aging effects for the containment components have been properly identified and will be adequately managed during the period of extended operation, as required by 10 CFR 54.21(a)(3).

This section of the SER provides the staff's evaluation of the applicant's AMR for the aging effects and the appropriateness of the programs credited for the aging management of the containment structural components at RNP. The staff's evaluation included a review of the aging effects considered and the basis for the applicant's elimination of certain aging effects. In addition, the staff has evaluated the appropriateness of the AMPs that are credited for managing the identified aging effects for the containment components.

Aging Effects

Concrete: For containment concrete components, the applicant's AMR is consistent with the recommendations in the GALL Report. As such, the applicant has committed to manage cracking, change in material properties, and loss of material for containment concrete components that are accessible. However, for several of the table entries in LRA Table 3.5-1, the applicant stated that the aging effect/mechanism combinations identified in the GALL Report are not applicable to RNP. In RAIs 3.5.1-8, 3.5.1-11, and 3.5.1-14, the staff requested that the applicant clarify its intentions to manage the aging effect/mechanism combinations for concrete SCs as recommended by the GALL Report. In its response to these RAIs, the applicant stated that it has "...committed to an AMP for monitoring accessible concrete based on Interim Staff Guidance." The staff position concerning the aging management of concrete SCs, which is discussed in an Interim Staff Guidance paper for concrete, is that concrete SCs need to be periodically inspected in order to adequately monitor their performance or condition in a manner that allows for the timely identification and correction of degraded conditions. In addition, in response to RAI 3.5.1-8, the applicant stated that Item 10 in LRA Table 3.5-2 will be deleted. Item 10 states that concrete and grout would experience no aging effects. This item includes accessible concrete/grout components located in the containment . Because the applicant has committed to monitor accessible containment concrete/grout components for cracking, loss of material, and change in material properties using the appropriate AMPs, the staff considers the applicant's response to be adequate. As such, the staff considers RAIs 3.5.1-8, 3.5.1-11, and 3.5.1-14 closed.

In RAI 3.5.1-8, the staff requested further information regarding the aging management of the masonry walls in the containment. Item 20 in LRA Table 3.5-1 states that "...the RNP AMR determined that no aging effects are applicable, based on the locations and design of the Masonry Walls at RNP." In its response to RAI 3.5.1-8, the applicant stated that Item 20 in LRA Table 3.5-1 will be changed based on Interim Staff Guidance for concrete aging and that the Structures Monitoring Program will be used to monitor accessible masonry walls for cracking.

For below-grade containment concrete components, the GALL Report recommends aging management only for an aggressive below-grade soil/ground water environment. Since ASME Section XI, Subsection IWL exempts from examination those portions of the concrete containment that are inaccessible, the GALL Report recommends that a plant-specific AMP be developed for concrete that may be exposed to an aggressive below-grade soil/ground water environment. As stated previously in SER Sections 3.5.2.2.1.1 and 3.5.2.2.2.2, the low pH value (< 5.5) for the ground water at RNP suggests a potentially aggressive environment for below-grade concrete. Therefore, a plant-specific AMP, or special provisions to an existing AMP for below-grade concrete components, is warranted. As described previously in Section 3.5.2.2.1.1 of this SER, the applicant has committed to use its periodic underwater inspections at the Intake Structure and RNP Dam Spillway as a leading indicator for potential degradation to below-grade concrete structures. Both these structures are exposed to lake water, which has similar pH, chloride, and sulfate values as the ground water at RNP. In the event that significant degradation to the submerged portions of the Intake Structure or Dam Spillway is observed or ground water and lake water trending results indicate increasing aggressiveness, the applicant will evaluate and examine below-grade concrete through both the ASME Section XI, Subsection IWL (for containment) and Structures Monitoring Program (for other Class I structures) AMPs.

By letter dated August 14, 2003 (RNP Serial RNP-RA/03-0094), the applicant responded to a number of confirmatory items identified by the staff. The staff reviewed the revised contents of Items 25, 26, and 27 of Attachment II (Revised License Renewal Commitments). The staff also reviewed the specific response to Confirmatory Item 3.5-1 provided in Attachment III (Response to License Renewal Confirmatory Items) in the same letter. Based on these reviews, the staff finds that the applicant has provided adequate information, and Confirmatory Item 3.5-1 is closed.

The staff finds the applicant's approach for evaluating the applicable aging effects for concrete components in containment to be reasonable and acceptable. The staff concludes that the applicant has properly identified the aging effects for concrete components in containment.

Steel: Consistent with the GALL Report recommendations, the applicant identified loss of material for containment carbon steel structural components, cracking, and loss of material as applicable aging effects for steel containment penetrations. In addition, loss of leak tightness in the closed position is identified as an aging effect for the containment equipment hatch and the personnel airlock. The applicant identifies this as loss of material due to wear. Loss of prestress for containment tendons is also identified as an applicable aging effect by the applicant.

Loss of material due to corrosion of the embedded containment liner and cracking of containment penetrations due to cyclic loading are identified by the GALL Report as aging effects requiring further evaluation and are covered in detail in Sections 3.5.2.2.1.4 and 3.5.2.2.1.7, respectively, of this SER. Loss of prestress for containment tendons is evaluated as a TLAA and reviewed by the staff in Section 4.5 of this SER. Fatigue damage is evaluated as a TLAA in Section 4.3.5 of this SER.

For carbon steel components that are completely encased in concrete (i.e., penetration sleeves, liner plate, airlock and hatch penetrations, anchorages/embedments, floor drains, and grouted tendons), the applicant did not identify loss of material as an applicable aging effect. In RAI 3.5.1-2, the staff requested that the applicant justify its conclusion regarding the aging management of the above components. In response to RAI 3.5.1-2, the applicant stated the following.

> The basis for determining that carbon steel components completely encased in RNP concrete would experience no loss of material aging effect includes consideration of the concrete design, in combination with the highly alkaline environment of concrete, and no plant operating experience identifying corrosion of embedded steel as an issue. Section 3.8.1.6.1.2 of the UFSAR states: "All reinforcing steel and frames which form an extension of the reinforcing steel are encased completely within the highly alkaline environment of the concrete wall and dome and are, therefore, protected from corrosion." Section 3.8.1.6.1.3 of the UFSAR states: "Concrete has been used successfully for many years as a protective covering for steel." As specified in NUREG-1557, and referenced in the GALL, the attributes of a concrete design for which corrosion is not significant are the same as specified for the RNP concrete design, specifically the concrete design is per ACI 318-63 with a low water-to-cement ratio and adequate air entrainment. Plant operating experience supporting this position is found in the corrosion inspection reports for the grouted surveillance tendons, which notes in the conclusions: "Based upon the results of the investigations documented in this report, it is concluded that there is no significant corrosion in the Robinson Nuclear Power Plant 25-year containment surveillance block provided for investigation." Additionally, the absence of any deficiencies identified in the Corrective Action Program, associated with the loss of material from embedded components, provides further evidence that the aging effect is not credible for the subject components. A combination of all the attributes listed in the above discussion provides reasonable assurance that carbon steel components completely encased in RNP concrete would experience no loss of material aging effect.

Based on the RNP concrete design, which is ACI Code compliant, RNP's plant-specific operating experience, and the highly alkaline environment of the concrete that encases the carbon steel components, the staff finds the potential for significant loss of material is not likely. As such, the staff considers RAI 3.5.1-2 to be closed.

The staff finds the applicant's approach for evaluating the applicable aging effects for steel components in containment to be reasonable and acceptable. The staff concludes that the applicant has properly identified the aging effects for steel components in containment.

Elastomers (moisture barriers, seals): Consistent with the GALL Report recommendations, the applicant identified loss of seal as an applicable aging effect for the containment moisture barrier and seals/gaskets. The aging effects identified by the applicant are change in material properties and cracking of elastomers. These aging effects are considered to result in loss of seal.

Item 6 of LRA Table 3.5-1, states that the leak tightness of seals and gaskets of containment penetrations is ensured by means of an Appendix J program. Performance based Option B of Appendix J (of 10 CFR 50) provides flexibility to the users of the option to perform Type B tests at an interval as long as 10 years (except for the air locks). Considering that some leakage is allowed during the type B tests (i.e., minor degradation is permissible), RAI 3.5.1-18 requested that the applicant discuss how it will manage the degradation of penetration seals and gaskets between the test intervals during the extended period of operation. In response to RAI 3.5.1-18, the applicant stated the following.

> RNP uses Option A of 10 CFR 50, Appendix J, for Type B testing (for gaskets and seals). Type B tests are conducted on a refueling outage interval, not to exceed a maximum interval of two years with the following exceptions: 1. The containment air lock is tested at six-month intervals. 2. If the air lock is opened during periods when containment integrity is not required, it is tested at the end of such periods prior to restoring the reactor to an operating mode that requires containment integrity. 3. If the air lock is opened during periods when containment integrity is required, the door seals are tested within 3 days after being opened. This current frequency of testing was evaluated to be adequate for the extended period of operation. Due to the short testing intervals, credit was not taken for additional inspections made as part of preventative maintenance. The Appendix J Program at RNP is consistent with GALL Section XI.S4, as discussed in LRA Appendix B, Item B.2.7.

Since the applicant is using Option A of 10 CFR 50, Appendix J, for Type B testing for managing the degradation of penetration seals and gaskets, which requires more frequent testing than Option B, the staff finds the proposed aging management adequate and reasonable and considers RAI 3.5.1-18 closed.

The staff finds the applicant's approach for evaluating the applicable aging effect for elastomers in containment to be reasonable and acceptable. The staff concludes that the applicant has properly identified the aging effect for elastomers in containment.

Miscellaneous Materials (copper alloy, bronze/graphite, insulation): For the bronze sliding bearing plates and threaded fasteners, copper alloy components, and insulation materials located in containment, the applicant did not identify any aging effects. In RAI 3.5.1-6, the staff requested that the applicant justify the above conclusion for each of these materials. In response to RAI 3.5.1-6, the applicant stated the following.

> The slide bearing plates identified in Item 13 of LRA Table 3.5-2 are fabricated from copper alloys (bronze material) impregnated with a graphitic lubricant with the trade name Lubrite or Lubron. Item 13 was used to categorize the copper alloy component or bronze material. Item 14 of LRA Table 3.5-2 was used to categorize the miscellaneous component or the graphite based lubricant. ASM Handbook, Volume 13, Corrosion – page 617, describes the corrosive ratings for various copper alloys in boric acid as "Excellent: resists corrosion under almost all conditions of service." Additionally, past ISI inspection reports for the reactor coolant pump supports and steam generator supports have identified no recordable degradation of the slide bearing plates. Based on the above, there is reasonable assurance that the subject item will experience no credible aging effects requiring an AMP.
>
> The containment liner insulation is fabricated from a PVC or polyamide foam. The subject insulation is used for thermal insulation of the containment liner, and is in direct contact with the external surface of the liner on one side, and is covered with a stainless steel sheathing (sheet metal) on the other side. There have not been specific inspections performed for the insulation panels, but, inspection reports for liners have not identified age related degradation of the insulation, and no condition reports have been identified that are associated with liner insulation degradation. Therefore, based on an absence of age related degradation operational experience, there is reasonable assurance that the containment liner insulation will experience no credible aging effects requiring an AMP.
>
> The containment penetration insulation commodities are identified as high density penetration insulation (BTU-BLOCK Flexible by Manville) and fiberglass blankets for the main steam lines, and ceramic fiber insulation for the steam generator blowdown lines. The subject insulation is located in the containment air environment not subject to boric acid leaks. No aging effects have been identified based on review of RNP operating experience, and based on the protective location of the subject insulation (inside penetrations), no mechanical degradation is expected. Therefore, no aging effects are identified that require management and an AMP is not required.

Since the applicant's previous operating experience with the materials identified above demonstrates that there are no applicable aging effects, the staff finds the applicant's response to RAI 3.5.1-6 adequate.

The staff finds the applicant's approach for evaluating the applicable aging effect for miscellaneous materials in containment to be reasonable and acceptable. The staff concludes that the applicant has properly evaluated the potential aging of miscellaneous materials in containment.

On the basis of its review, the staff finds the applicant has identified the appropriate aging effects for the materials and environments associated with the containment.

Aging Management Programs

Tables 3.5-1 and 3.5-2 of the LRA credit the following AMPs with managing the identified aging effects for the components in the containment.

- ASME Section XI, Subsection IWL Program
- ASME Section XI, Subsection IWE Program
- ASME Section XI, Subsection IWF Program

- Structures Monitoring Program
- Boric Acid Corrosion Program
- One-Time Inspection Program
- 10 CFR Part 50, Appendix J Program
- Water Chemistry Program

The Boric Acid Corrosion Program, Water Chemistry Program, and One-Time Inspection Program are credited with managing the aging of several components in several different structures and systems and are, therefore, considered common AMPs. The staff's review of these common AMPs can be found in Section 3.0.3 of this SER. The staff's evaluation of the noncommon, or structure-specific, AMPs, listed above, is presented in Section 3.5.2.3 of this SER. Two additional AMPs manage aging effects for containment components, but are not identified in Table 3.5-1 or Table 3.5-2. The Inspection of Overhead Heavy Load and Light Load Haandling Systems Program is reviewed in Section 3.3.2.3.1 of this SER. The Fire Water System Program is reviewed in Section 3.3.2.3.3 of this SER.

After evaluating the applicant's AMR for each of the components in the containment, the staff evaluated the AMPs listed above to determine if they are appropriate for managing the identified aging effects. For those components identified in Table 3.5-1 of the LRA, the staff verified that the applicant credited the AMPs recommended by the GALL Report. For the components identified in LRA Table 3.5-2, the staff verified that the applicant credited an AMP that is appropriate for the identified aging effect(s).

On the basis of its review, the staff finds the applicant has credited the appropriate AMPs to manage the aging effects for the materials and environments associated with containment. In addition, the staff found the associated program descriptions in the UFSAR Supplement to be acceptable.

The staff has reviewed the information in Sections 2.4 and 3.5 of the LRA, the applicant's responses to the staff's RAIs, and the applicable AMP descriptions in Appendix B of the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the containment components will be adequately managed so that these components will perform their intended functions in accordance with the CLB during the period of extended operation.

3.5.2.4.2 Other Structures

3.5.2.4.2.1 Summary of Technical Information in the Application

The AMR results for other structures are presented in Tables 3.5-1 and 3.5-2 of the LRA. The applicant used the GALL Report format to present its AMR of structural components in LRA Table 3.5-1. In LRA Table 3.5-2, the applicant identified the component group designation along with its (1) material, (2) environment, (3) aging effect(s), and (4) AMP(s). The structural components listed in Tables 3.5-1 and 3.5-2 of the LRA are in the following structures.

- Reactor Auxiliary Building
- Fuel Handling Building
- Turbine Building
- Dedicated Shutdown Diesel Generator Building
- Radwaste Building
- Intake Structure
- North Service Water Header Enclosure
- Emergency Operations Facility/Technical Support Center Security Emergency Diesel Generator Building
- Lake Robinson Dam
- Pipe Restraint Tower
- Yard Structures and Foundations

A brief description of each of the above structures is provided in Section 2.4.2, "Other Structures," of the LRA. The materials of construction identified in the LRA for each of the above structures are (1) steel, (2) concrete, (3) aluminum, (4) elastomers, and (5) miscellaneous material, such as soil and ceiling and floor tiles. These materials are exposed to outdoor, buried, indoor air-conditioned, indoor not-air-conditioned, borated water, and raw water environments.

The spent fuel storage racks, neutron absorbing sheets in spent fuel storage racks and cranes including bridge and trolleys and rail system in load handling systems are scoped under structures. The AMR results of these structural components are presented in Tables 3.3-1 and 3.3-2 of the LRA.

Aging Effects

Tables 3.5-1 and 3.5-2 of the LRA identify the following applicable aging effects for components in structures outside the containment.

- loss of material
- change in material properties and cracking of elastomers
- cracking
- loss of mechanical function
- loss of form
- corrosion of embedded steel
- reduction in concrete anchor capacity
- cracking of masonry walls

Aging Management Programs

Tables 3.5-1 and 3.5-2 of the LRA credit the following AMPs with managing the identified aging effects for the components in structures outside the containment.

- ASME Section XI, Subsection IWF Program
- Boric Acid Corrosion Program
- Dam Inspection Program
- Structures Monitoring Program
- Water Chemistry Program

The applicant credited the above listed Water Chemistry Program to manage the loss of material aging effect of the spent fuel storage racks. The applicant credits Inspection of Overhead Heavy Load and Light Load Handling System Program to manage aging effects of RNP cranes and their related components.

A description of these AMPs is provided in Appendix B of the LRA.

3.5.2.4.2.2 Staff Evaluation

In addition to Section 3.5 of the LRA, the staff reviewed the pertinent information provided in Section 2.4, "Scoping and Screening Results—Structures," and the applicable AMP descriptions provided in Appendix B of the LRA to determine whether the aging effects for components in structures outside the containment have been properly identified and will be adequately managed during the period of extended operation, as required by 10 CFR 54.21(a)(3).

This section of the SER provides the staff's evaluation of the applicant's AMR for the aging effects and the appropriateness of the programs credited for the aging management of structures outside the containment at RNP. The staff's evaluation includes a review of the aging effects considered and the basis for the applicant's elimination of certain aging effects. In addition, the staff has evaluated the appropriateness of the AMPs that are credited for managing the identified aging effects for the components in structures outside the containment.

Aging Effects

Concrete: For concrete components in structures outside the containment, the applicant's AMR is consistent with the recommendations in the GALL Report. As such, the applicant has committed to manage cracking, change in material properties, and loss of material for concrete structural components that are accessible. As stated previously in Section 3.5.2.4.1.2 of this SER, several of the table entries in LRA Table 3.5-1 stated that the aging effect/mechanism combinations identified in the GALL Report are not applicable to RNP. The staff requested, in RAIs 3.5.1-8 and 3.5.1-10 , that the applicant clarify its intent to manage the aging effect/mechanism combinations as recommended by the GALL Report. In response, the applicant stated that although it does not consider these aging effects to be applicable, it will manage the aging of concrete structures at RNP as recommended by the GALL Report. As the applicant committed to manage the aging of accessible concrete structural components at RNP, the staff considers the response to the RAIs adequate.

For below-grade concrete structural components, the GALL Report recommends aging management only for an aggressive below-grade soil/ground water environment. Item 17 of LRA Table 3.5-1 states the following.

> The aging mechanisms associated with aggressive chemical attack and corrosion

of embedded steel are applicable only to below-grade concrete/grout structures owing to the slightly acidic pH of ground water. The Structures Monitoring Program is applicable to these structures. RNP will apply a special, plant-specific inspection provision to monitor aging effects caused by aggressive chemical attack and corrosion of embedded steel for below-grade concrete in this component/commodity group. This will include inspection of below-grade concrete and grout that is exposed during excavation. These aging management activities are consistent with the GALL Report.

As stated previously in SER Sections 3.5.2.2.1.1 and 3.5.2.2.2.2, the low pH value (< 5.5) for the ground water at RNP suggests a potentially aggressive environment for below-grade concrete. Therefore, a plant-specific AMP, or special provisions to an existing AMP for below-grade concrete components is warranted. The provision proposed above by the applicant is to include inspection of below-grade concrete and grout that is exposed during excavation as part of the Structures Monitoring Program. As stated previously in Section 3.5.2.2.1.1 of this SER, the staff found the RNP's approach of inspecting below-grade concrete only when it happens to be exposed during plant excavations done for other activities to be insufficient. As such, the staff requested that further measures be taken to ensure the adequate aging management of below-grade concrete at RNP. As described previously in Section 3.5.2.2.1.1 of this SER, the applicant has committed to use its periodic underwater inspections at the intake structure and RNP dam spillway as a leading indicator for potential degradation to below-grade concrete structures. Both these structures are exposed to lake water, which has similar pH, chloride, and sulfate values as the ground water at RNP. In the event that significant degradation to the submerged portions of the intake structure or dam spillway is observed, the applicant will evaluate and examine below-grade concrete through both the ASME Section XI, Subsection IWL (for containment) and Structures Monitoring Program (for other Class I structures) AMPs. The applicant's commitment to provide appropriate documentation of the above agreement was designated as Confirmatory Item 3.5-1. Based on the discussion related to closure of Confirmatory Item 3.5-1 provided in Section 3.5.2.2, "Aging Management Evaluations in the GALL Report That Are Relied On for License Renewal, For Which GALL Recommends Further Evaluation," of this SER, Confirmatory Item 3.5-1 is closed.

The staff finds that the applicant's approach for evaluating the applicable aging effects for concrete components in structures outside the containment to be reasonable and acceptable. The staff concludes that the applicant has properly identified the aging effects for concrete components in structures outside the containment.

Steel: Consistent with the recommendations of the GALL Report, the applicant identified loss of material as an applicable aging effect for carbon steel components in structures outside the containment. This includes all Class I structures identified in the GALL Report.

For some of the carbon steel structural components listed in Section 2.4, "Scoping and Screening Results—Structures," the staff was unable to verify that the aging effect(s) identified for these components in Table 3.5-1 of the LRA will be managed by an appropriate AMP. In RAI 3.5.1-13, the staff requested the applicant to provide clarification regarding the AMR conclusions for carbon steel structural components inside containment, as well as for structures outside containment.

In response to RAI 3.5.1-13, the applicant stated the following.

> Loss of material is an applicable aging effect for carbon steel components inside or outside containment and is managed by one of the following programs for the structural components listed in Section 2.4.
>
> - Structures Monitoring Program
> - Boric Acid Corrosion Program
> - IWF Program
> - IWE Program
> - Appendix J Program
> - One-Time Inspection Program
> - Dam Inspection Program
>
> These AMPs are considered to be appropriate for managing the aging effects for carbon steel components that were identified in the AMR.

As the applicant has clarified its intention to manage loss of material for carbon steel structural components, as recommended by the GALL Report, the staff finds the applicant's response to RAI 3.5.1-13 adequate.

For below-grade carbon steel foundation pilings, the applicant identified corrosion of the piles as a TLAA and performed an evaluation for a 40-year corrosion loss. The staff's evaluation of this TLAA is found in Section 4.6.2 of this SER.

For SS components, the applicant identified loss of material as an applicable aging effect for (1) liners in the fuel storage facility and refueling canal, (2) the fuel transfer tube and associated bellows, and (3) detector and manway cover, spent fuel racks, and reactor cavity seal ring plate. In Table 3.5-1 of the LRA, the applicant indicated that stress- corrosion cracking is not applicable for the SS reactor cavity or spent fuel pool liners. The applicant stated that cracking due to SCC requires both high temperatures (> 140 °F) and exposure to an aggressive environment to be applicable. Because the normal temperatures in the fuel pool and reactor cavity do not exceed 140 °F, the applicant concluded that SCC is not applicable. As the applicant's position is consistent with the GALL Report, the staff concurs with this position.

The AMR results of Neutron absorbing sheets in spent fuel storage racks are provided on LRA Table 3.3-1. Section 4.6.4.2 of this SER discusses staff evaluation for the Boraflex degradation and the related Confirmatory Item 4.6.4-1. By letter dated December 22, 2003, License Amendment No. 198, the staff approved the applicant's request to eliminate the need to credit the Boraflex neutron absorbing material for reactivity control in the spent fuel storage pool. In place of the Boraflex material (i.e., panels), the staff approved the applicant's request to take credit for a combination of soluble boron and controlled fuel loading patterns in the spent fuel pool to maintain the required subcriticality margins in the spent fuel storage pool. On the basis of the final issuance of License Amendment No. 198, the staff finds that Confirmatory Item 4.6.4-1 is closed.

With regard to the AMR of spent fuel storage racks, the applicant concluded that stress corrosion cracking was not applicable to RNP spent fuel storage racks, because the temperature of the fluid is normally less than 140 degree F. However, the applicant determined that loss of material due to crevice and pitting corrosion is an applicable aging effect for spent fuel storage racks. The applicant credits Water Chemistry Program to manage this aging effect. The staff finds the above RNP determination adequate and acceptable.

The applicant identified general corrosion, but not wear, as an aging mechanism for crane rails. The applicant also stated that regardless of the aging mechanism, Inspection of Overhead Heavy Load and Light Load Handling Systems Program is credited to manage loss of material aging effect for in-scope cranes including bridge and trolleys and rail system in load handling systems. The staff finds this RNP position adequate and acceptable.

The staff finds the applicant's approach for evaluating the applicable aging effects for steel components in structures outside the containment to be reasonable and acceptable. The staff concludes that the applicant has properly identified the aging effects for steel components in these structures.

Elastomers: For the structures outside containment, the applicant identified change in material properties and cracking from elevated temperature as applicable aging effects in Table 3.5-2 of the LRA. The applicant credited the Structures Monitoring Program to manage these two aging effects of elastomeric material.

The staff finds that the applicant's approach for evaluating the applicable aging effects for elastomers in structures outside the containment to be reasonable and acceptable. The staff concludes that the applicant has properly identified the aging effects for elastomers in these structures.

Miscellaneous materials: The in-scope miscellaneous materials identified by the applicant in structures outside the containment are soil for the Lake Robinson earthen dam, and ceiling and floor tiles for the control room.

For the Lake Robinson earthen dam, the applicant identified loss of form due to settlement as an applicable aging effect and proposed to use its Dam Inspection Program. The identification of loss of form as an applicable aging effect for earthen embankments or dams is consistent with the GALL Report. In addition, the applicant's Dam Inspection Program is a FERC/US Army Corps of Engineers program, which is also consistent with the GALL Report.

No aging effects were identified by the applicant for the floor and ceiling tiles in the control room. In RAI 3.5.1-6, the staff requested further information regarding the previous operating experience for these components. In response, the applicant provided the following information.

> For the control room, ceiling the acoustical ceiling tiles are mineral fiberboard, manufactured by Armstrong. The suspended grid system for the acoustical tile is a heavy duty exposed tee system by Armstrong. The control room ceiling is supported by a combination of structural steel, threaded rod, and unistrut attached to the building by welding or expansion bolts. The material is either coated steel or galvanized steel. The control room raised floor access floor system is constructed of epoxy painted carbon steel pedestals, stringers, and floor panels furnished by Tate Access Floors, Inc. Fasteners are either carbon steel or galvanized steel. The cable spread room raised floor access floor system is constructed of epoxy painted

carbon steel pedestals, stringers, and perforated floor panels furnished by Tate Access Floors, Inc. Fasteners are either carbon steel or galvanized steel. The control room and cable spreading room are indoor-air-conditioned environments. Therefore, the carbon steel structural supports for the control room and cable spreading room raised floors do not require aging management. Additionally, based on RNP operating experience, no aging effects requiring management for the control room ceiling material or raised floors have been identified. Therefore, no AMP is required.

Because the applicant has not identified any previous aging of the floor and ceiling tiles and because these tiles are in an air-conditioned indoor environment, the staff concurs with the applicant 's conclusion that there are no applicable aging effects. As such, RAI 3.5.1-6 is closed.

On the basis of its review, the staff finds the applicant has identified the appropriate aging effects for the materials and environments associated with the structures outside the containment.

Aging Management Programs

Tables 3.5-1and 3.5-2 of the LRA credit the following AMPs with managing the identified aging effects for the components in structures outside the containment.

- ASME Section XI, Subsection IWF Program
- Boric Acid Corrosion Program

- Dam Inspection Program
- Structures Monitoring Program
- Water Chemistry Program

The applicant credits the above listed AMPs to manage the aging effects associated with structures and structural components outside the containment. Two AMPs (i.e., Water Chemistry Program and Boric Acid Corrosion Program) are common AMPs, while the remaining three AMPs are credited with managing aging only for structures and structural components outside the containment. The staff's evaluation of the common AMPs credited with managing aging in structures and structural components outside the containment is provided in Section 3.0.3 of this SER.

Table 3.3-1 of the LRA credits Inspection of Overhead Heavy Load and Light Load Handling Systems Program with managing of the identified aging effects for cranes including bride and trolleys and rail system in load handling systems. The staff evaluation of this crane inspection program is provided in SER Section 3.3.2.3.1.2.

Other structural components are managed by additional AMPs. These AMPs and locationwhere the staff evaluated these AMPs are listed below:

- Fire Water System Program (SER Section 3.3.2.3.3)
- Fire Protection System Program (SER Section 3.3.2.3.2)
- Preventive Maintenance Program (SER Section 3.0.3.12)

- Inspection of Overhead Heavy Load and Light Load Handling System Program (SER Section 3.3.2.3.1)

Additional staff evaluation of the structural components outside the containment can be found in the applicable technical evaluations provided in Section 3.5.2.2.2 of this SER.

After evaluating the applicant's AMR for each of the components in structures outside the containment, the staff evaluated the AMPs listed above to determine if they are appropriate for managing the identified aging effects. For those components identified in Table 3.5-1 of the LRA, the staff verified that the applicant credited the AMP recommended by the GALL Report. For the components identified in LRA Table 3.5-2, the staff verified that the applicant credited an AMP that is appropriate for the identified aging effect(s).

The staff also reviewed the applicable UFSAR Supplement program descriptions and concludes that the UFSAR Supplements provide adequate program descriptions of the AMPs credited for managing aging in structures and structural components outside the containment.

The staff has reviewed the information in Sections 2.4 and 3.5 of the LRA, the applicant's responses to the staff's RAIs, and the applicable AMP descriptions in Appendix B of the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the components in structures outside the containment will be adequately managed so that these components will perform their intended functions in accordance with the CLB during the period of extended operation.

3.5.2.4.3 Component Supports

3.5.2.4.3.1 Summary of Technical Information in the Application

The AMR results for the component supports are presented in Tables 3.5-1 and 3.5-2 of the LRA. The applicant used the GALL Report format to present its AMR of the components in LRA Table 3.5-1. In LRA Table 3.5-2, the applicant identified the component group designation along with its (1) material, (2) environment, (3) aging effect(s), and (4) AMP(s).

Component supports are those components that provide support or enclosure for mechanical and electrical equipment. The component supports identified in LRA Section 2.4 include (1) anchorages/embedments, (2) electrical component supports, (3) expansion anchors, (4) instrument line supports, (5) instrument racks and frames, (6) pipe supports, (7) pressurizer surge line supports, (8) SG supports, (9) vibration isolators, (10) battery racks, (11) HVAC duct supports, (12) tube track supports, and (13) several other supports .

The materials of construction for the component supports, which are subject to an AMR, are steel, and copper alloy. These materials are exposed to internal, external, borated water leaks, and embedded environments.

<u>Aging Effects</u>

Tables 3.5-1 and 3.5-2 of the LRA identify the following applicable aging effects for the component supports.

- loss of material
- cracking
- loss of mechanical function

Aging Management Programs

Tables 3.5-1 and 3.5-2 of the LRA credit the following AMPs with managing the identified aging effects for the component supports.

- Boric Acid Corrosion Program
- Structures Monitoring Program
- ASME Section XI, Subsection IWF Program

A description of these AMPs is provided in Appendix B of the LRA.

3.5.2.4.3.2 Staff Evaluation

In addition to Section 3.5 of the LRA, the staff reviewed the pertinent information provided in Section 2.4, "Scoping and Screening Results—Structures," and the applicable AMP descriptions provided in Appendix B of the LRA to determine whether the aging effects for the component supports have been properly identified and will be adequately managed during the period of extended operation, as required by 10 CFR 54.21(a)(3).

This section of the SER provides the staff's evaluation of the applicant's AMR for the aging effects and the appropriateness of the programs credited for the aging management of the component supports at RNP. The staff's evaluation includes a review of the aging effects considered and the basis for the applicant's elimination of certain aging effects. In addition, the staff has evaluated the appropriateness of the AMPs that are credited for managing the identified aging effects for the component supports.

Aging Effects

Steel: Consistent with the recommendations of the GALL Report, the applicant identified loss of material as an applicable aging effect for the carbon steel component supports in non-air-conditioned environments (internal and external). For SS component supports, either in an outdoor or borated water environment, the applicant identified loss of material as an applicable aging effect. In addition, for galvanized steel component supports in an outdoor environment, the applicant also identified loss of material as an applicable aging effect.

However, for galvanized structural steel in indoor, containment air, or exposed to borated water leaks, Items 2 and 11 of LRA Table 3.5-2 state that there are no applicable aging effects. In RAI 3.5.1-5, the staff requested that the applicant discuss past incidents of borated water leakage including ponding of leaked borated water at RNP. Additionally, Item 12 of LRA Table 3.5-2 states that there are no applicable aging effects for SS threaded fasteners (among other SS components). As part of RAI 3.5.1-5, the staff also requested that the applicant confirm that there are no SS threaded fasteners used in a wetted or highly moist air environment. In response to RAI 3.5.1-5, the applicant stated the following.

For galvanized steel, no operating experience examples were identified regarding borated water leaks causing aging to the galvanized steel components identified in LRA Table 3.5-2, items 2 and 11. As a conservative measure, RNP has decided to include loss of material due to corrosion for galvanized steel in a borated water leakage environment as an aging effect/mechanism. As such, borated water leakage environment should be deleted as an applicable environment in LRA Table 3.5-2, Item 2. In addition, galvanized steel should be deleted as a material and from the discussion column of LRA Table 3.5-2, Item 11. In LRA Table 3.5 -1, Item 16, the discussion column for steel should include galvanized steel.

For stainless steel, no operating experience examples were identified regarding borated water leaks causing aging to the stainless steel components identified in LRA Table 3.5 -2, Items 2 and 11. At RNP, LR did not identify occurrences of stainless steel threaded fasteners in a wetted or highly moist environment.

Because the applicant has committed to manage loss of material, due to corrosion, for galvanized steel components in a borated/water leakage environment and because the applicant did not identify any occurrences of SS threaded fasteners in a wetted environment, the staff finds the applicant's response to RAI 3.5.1-5 adequate.

For the high strength carbon steel threaded fasteners, the applicant did not identify cracking due to SCC as an applicable aging effect. Item 29 of LRA Table 3.5-1 states the following.

The RNP AMR, which included operating experience, determined that SCC is not an applicable aging mechanism for RNP bolting. In general, high strength structural bolting, i.e., bolting with specified yield strength > 150 ksi, is not being used; and, for the one case where high strength bolts have been installed, the environment experienced by the bolts is considered benign with respect to SCC, i.e., the bolts are located in a dry environment high up on the steam generator above any source of leakage and, therefore, not exposed to an aggressive or aqueous environment. Based on these results, no AMP is required to manage cracking due to SCC.

Conditions that may contribute to the occurrence of SCC for high strength carbon steel threaded fasteners are elevated temperatures, an aggressive environment (e.g., borated water leaks), and wetted air with an oxygen concentration. For the one case where high strength bolting is used at RNP, the applicant stated that none of these conditions are prevalent. As such, the staff concurs with the applicant that SCC is not an applicable aging effect for high strength carbon steel threaded fasteners.

Item 28, Table 3.5-1, of the LRA states that RV nozzle supports are inaccessible and not currently inspected under the RNP ASME Section XI, Subsection IWF Program and that RNP plans to implement an inspection under the One-Time Inspection Program to verify effective management of potential corrosion of the supports. RAI 3.5.1-1 requested that the applicant discuss the specific steps to be adopted in performing the one-time inspection of the inaccessible nozzle supports and provide the basis for concluding that a one-time inspection would suffice to ensure effective aging management of these inaccessible supports. The applicant provided the following response to RAI 3.5.1-1.

RNP has elected to remove the RV nozzle supports from the One-Time Inspection Program and will include them within the ASME Section XI, Subsection IWF Program. Therefore, a RV nozzle support will be inspected by the IWF Program during the Fourth Ten-Year ISI Interval prior to the end of the current 40-year Operating License. Due to the limited accessibility of the supports, a limited visual inspection will be made using remote visual technology. The RV nozzle supports will continue to be inspected by the ASME Section XI, Subsection IWF Program during the period of extended operation. A review of operating experience (OE) indicated a condition report was identified in April 2001 (during Refueling Outage-21). This OE information was a consideration in the decision to include the RV nozzle supports in the ASME Section XI, Subsection IWF Program.

Because the applicant has committed to periodic inspections of the RV nozzle supports through the ASME Section XI, Subsection IWF Program, rather than a single inspection under the One-Time Inspection Program, the staff finds the above response adequate and RAI 3.5.1-1 closed.

Copper Alloy: For the copper alloy slide bearing plate inside containment, the applicant did not identify any applicable aging effects. The staff's review of these slide bearing plates is provided in Section 3.5.2.4.1.2 of this SER.

On the basis of its review, the staff finds that the applicant has identified the appropriate aging effects for the materials and environments associated with component supports.

Aging Management Programs

Tables 3.5-1 and 3.5-2 of the LRA credit the following AMPs with managing the identified aging effects for the component supports.

- Boric Acid Corrosion Program
- Structures Monitoring Program
- ASME Section XI, Subsection IWF Program

The Boric Acid Corrosion Program, Bolting Integrity Program, and One-Time Inspection Program are credited with managing the aging of several components in several different structures and systems and are, therefore, considered common AMPs. The staff's review of these common AMPs can be found in Section 3.0.3 of this SER. The staff's evaluation of the noncommon or structure-specific AMPs, listed above, is provided in Section 3.5.2.3 of this SER.

After evaluating the applicant's AMR for each of the components in the containment, the staff evaluated the AMPs listed above to determine if they are appropriate for managing the identified aging effects. For those components identified in Table 3.5-1 of the LRA, the staff verified that the applicant credited the AMPs recommended by the GALL Report. For the components identified in LRA Table 3.5-2, the staff verified that the applicant credited an AMP that is appropriate for the identified aging effect(s).

On the basis of its review, the staff finds the applicant has credited the appropriate AMPs to manage the aging effects for the materials and environments associated with the component supports. In addition, the staff found the associated program descriptions in the UFSAR Supplement to be acceptable.

3.5.2.4.3.3 Conclusions

On the basis of its review, the staff concludes that, the applicant has adequately identified the aging effects, and the AMPs credited for managing the aging effects, of the containment, structures, and component supports plant specific components in Sections 3.5.2.4.1 through 3.4.2.4.3, such that the component intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the applicable UFSAR Supplement program description and concludes that it provides an adequate program description of the AMPs credited for managing aging of the containment, structures, and component supports plant specific components, as required by 10 CFR 54.21(d).

3.5.3 Evaluation Findings

The staff has reviewed the information in Section 3.5 of the LRA. On the basis of its review, the staff concludes that the applicant has adequately identified the aging effects, and the AMPs credited for managing the aging effects, for the containments, structures, and component supports, such that there is reasonable assurance that the component intended functions will be maintained consistent with the CLB for the period of extended operation. The staff also reviewed the applicable UFSAR Supplement program descriptions and concludes that the UFSAR Supplement provides an adequate program description of the AMPs credited for managing aging effects, as required by 10 CFR 54.21(d).

3.6 Electrical and Instrumentation and Controls

This section addresses the aging management of the components of the electrical and instrumentation and control (I&C) systems group. The systems that make up this group are described in the following LRA sections:

- Bus Duct (2.5.3.1)
- Insulated Cables and Connections (2.5.3.2)
- Electrical/Instrumentation and Control Penetration Assemblies (2.5.3.3)

As discussed in Section 3.0.1 of this SER, the electrical and instrumentation and controls are included in one LRA table. LRA Table 3.6-1 consists of electrical and I&C components that are evaluated in the GALL Report.

3.6.1 Summary of Technical Information in the Application

In LRA Section 3.6, the applicant described its AMRs for the electrical and I&C systems group at RNP.

The applicant stated that the methodology used for AMR of this system group employs the "plant spaces" approach in which the plant is segregated into areas (or spaces) where common bounding environmental parameters can be assigned. Each bounding environmental parameter is evaluated against the most limiting (worst-case) material in the area to determine if the components will be able to maintain their intended functions through the period of extended operation.

The Department of Energy (DOE), "Aging Management Guideline for Commercial Nuclear Power Plants—Electrical Cable and Terminations," (the Cable AMG) was used to identify aging effects for all electrical commodity groups within the scope of this review. The applicant determined that the potential aging effects are based upon materials of construction and their exposure to environmental stressors, such as heat, radiation, and moisture.

The AMR identifies one or more AMPs to be used to demonstrate that the effects of aging will be managed to assure that the intended functions will be maintained consistent with the CLB for the period of extended operation. The programs to be used for managing the effects of aging were compared to those listed in the GALL Report and evaluated for consistency with GALL Report programs that are relied on for license renewal. The results are documented and discussed in Subsection 3.6.2 using the format suggested by the SRP-LR. AMPs are described in Appendix B.

Based on a review of potential aging effects using the Cable AMG, the following stressors and aging effects were identified.

Applicable Stressor	Voltage Category [1]	Applicability	Potential Aging Effects
Heat, oxygen	Low & Medium	All insulation materials	Reduced IR; electrical failure
Radiation, oxygen	Low & Medium	All insulation materials	Reduced IR electrical failure
Moisture and voltage stress	Medium	All insulation materials exposed to standing water	Electrical failure (caused by a breakdown of the insulation)

Notes: 1. Low-voltage (≤1000 volts alternating current (Vac) or ≤250 volts direct current (Vdc)) and medium-voltage (2 kVac—15 kVac)

The applicant's AMRs included an evaluation of site-specific and industry operating experience. The site-specific evaluation included reviews of (1) the Corrective Action Program, (2) licensee event reports, (3) the Maintenance Rule database, and (4) interviews with systems engineers. These reviews concluded that the aging effects requiring management based on RNP operating experience were consistent with aging effects identified in GALL.

The applicant's review of industry operating experience included a review of operating experience published since the effective date of the GALL Report. The results of this review concluded that aging effects requiring management based on industry operating experience were consistent with aging effects identified in GALL.

The applicant's ongoing review of plant-specific and industry-wide operating experience is conducted in accordance with the RNP Corrective Action and Operating Experience Programs.

3.6.2 Staff Evaluation

In Section 3.6 of the LRA, the applicant described its AMR for electrical and I&C systems at RNP. The staff reviewed LRA Section 3.6 to determine whether the applicant has provided sufficient information to demonstrate that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB throughout the period of extended operation, in accordance with the requirements of 10 CFR 54.21(a)(3), for electrical and I&C system components that are determined to be within the scope of license renewal and are subject to an AMR.

The applicant referenced the GALL Report in its AMR. The staff has previously evaluated the adequacy of the aging management of electrical and I&C system components for license renewal as documented in the GALL Report. Thus, the staff did not repeat its review of the matters described in the GALL Report, except to ensure that the material presented in the LRA was applicable, and to verify that the applicant had identified the appropriate programs as described and evaluated in the GALL Report. The staff evaluated those aging management issues recommended for further evaluation in the GALL Report. The staff also reviewed aging management information submitted by the applicant that was different from that in the GALL Report or was not addressed in the GALL Report. Finally, the staff reviewed the UFSAR Supplement to ensure that it provided an adequate description of the programs credited with managing aging for the electrical and I&C system components.

In LRA Section 2.5, the applicant provided a brief description of the electrical and I&C systems and summarized the results of its AMR of the electrical and I&C system components at RNP in LRA Section 3.6.

Table 3.6-1 below provides a summary of the staff's evaluation of components, aging effects/mechanisms, and AMPs listed in LRA Section 3.6 that are addressed in the GALL Report.

Table 3.6-1

Staff Evaluation Table for RNP Electrical Components Evaluated in the GALL Report

Component Group	Aging Effect/Mechanism	AMP in GALL Report	AMP in LRA	Staff Evaluation
Electrical equipment subject to 10 CFR 50.49 EQ requirements	Degradation due to various aging mechanisms	Environmental qualification of electrical components	B.2.9 (This AMP was not in the original LRA). See RAI 4.4-2.	See Section 4.4

Component Group	Aging Effect/Mechanism	AMP in GALL Report	AMP in LRA	Staff Evaluation
Electrical cables and connections not subject to 10 CFR 50.49 EQ requirements	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced IR; electrical failure caused by thermal/thermoxidative degradation of organics; radiolysis and photolysis (ultraviolet (UV) sensitive materials only) of organics; radiation-induced oxidation; moisture intrusion	AMP for electrical cables and connections not subject to 10 CFR 50.49 EQ requirements	B.4.6	Consistent with GALL. (See Section 3.6.2.1 below)
Electrical cables used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements that are sensitive to reduction in conductor IR (high-range radiation monitoring instrumentation circuits)	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced IR; electrical failure caused by thermal/thermoxidative degradation of organics; radiation-induced oxidation; moisture intrusion	AMP for electrical cables used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements	B.4.7 (This AMP was not in the original LRA). See RAI 3.6.1-2.	Consistent with GALL (see Section 3.6.2.3.2 below)
Electrical cables used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements that are sensitive to reduction in conductor IR (neutron flux instrumentation circuits)	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced IR; electrical failure caused by thermal/thermoxidative degradation of organics; radiation-induced oxidation; moisture intrusion	AMP for electrical cables used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements	B.4.8 (This AMP was not in the original LRA). See RAI 3.6.1-2.	Non-GALL Program (see Section 3.6.2.3.2 below)

Component Group	Aging Effect/Mechanism	AMP in GALL Report	AMP in LRA	Staff Evaluation
Inaccessible medium-voltage (2 kV to 15 kV) cables (e.g., installed in conduit or direct buried) not subject to 10 CFR 50.49 EQ requirements	Formation of water trees, localized damage leading to electrical failure (breakdown of insulation); water trees caused by moisture intrusion	AMP for inaccessible medium-voltage cables not subject to 10 CFR 50.49 EQ requirements	No AMP Required	(see Section 3.6.2.3.3 below)
Electrical connectors not subject to 10 CFR 50.49 EQ requirements that are exposed to borated water leakage	Corrosion of connector contact surfaces caused by intrusion of borated water	AMP for boric acid corrosion	B.3.2	Consistent with GALL (see Section 3.6.2.3 below)

3.6.2.1 Aging Management Evaluations in the GALL Report That Are Relied On for License Renewal, Which Do Not Require Further Evaluation

For component groups evaluated in GALL for which the applicant has claimed consistency with GALL, and for which GALL does not recommend further evaluation, the staff sampled components in these groups to determine whether the plant-specific components contained in these GALL component groups were bounded by the GALL evaluation. The staff also sampled component groups to determine whether the applicant had properly identified those component groups in GALL that were not applicable to its plant.

On the basis of this review, the staff has determined that the applicant's basis of managing aging effects associated with electrical and I&C system components is consistent with GALL.

3.6.2.2 Electrical Equipment Subject to Environmental Qualification

Environmental qualification is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The staff reviewed the evaluation of this TLAA separately in Section 4.4 of this SER, following the guidance in Section 4.4 of the SRP-LR.

3.6.2.3 Aging Management Programs for Electrical and Instrumentation and Controls Components

In SER Sections 3.6.2.1, the staff determined that the applicant's AMRs and associated AMPs will adequately manage component aging in electrical and I&C systems. The staff then reviewed specific electrical and I&C system components to ensure that they were properly evaluated in the applicant's AMR.

To perform its review, the staff reviewed the components listed in LRA Table 2.5-1 to determine whether the applicant had properly identified the applicable AMRs and AMPs needed to adequately manage the aging effects of the components. This portion of the staff's review involved identifying the aging effects for each component, ensuring that each aging effect was evaluated using the appropriate AMR in Section 3, and ensuring that management of the aging effect was captured in the appropriate AMP. The results of the staff's review are provided below.

The staff also reviewed the UFSAR Supplements for the AMPs credited with managing aging in electrical and I&C system components to determine whether the program descriptions adequately describe the programs.

The applicant credits five AMPs to manage the aging effects associated with electrical and I&C components. One of the AMPs is credited to manage aging for components in other system groups (common AMP) while the other four AMPs are credited with managing aging only for electrical and I&C components. The staff's evaluation of the common AMP (Boric Acid Corrosion Program), credited with managing aging in electrical and I&C components, is provided in Section 3.0.3.4 of this SER.

The staff's evaluation of the other electrical and I&C components system AMP is provided below.

3.6.2.3.1 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

3.6.2.3.1.1 Summary of Technical Information in the Application

The Non-EQ Insulated Cables and Connections Program is credited for aging management of cables and connections not included in the RNP Environmental Qualification Program. The aging effects/mechanisms of concern are as follows.

- reduced insulation resistance
- electrical failure

The technical basis for selecting a sample of cables to be inspected will be defined prior to the period of extended operation. The sample locations will consider the location of PVC cables inside and outside containment, as well as any known adverse localized environments. (PVC was determined to be the limiting insulation material.)

The Non-EQ Insulated Cables and Connections Program is a new program with no operating experience history. However, as noted in the GALL Report, industry operating experience has shown that adverse localized environments caused by heat or radiation for electrical cables and connections have been shown to exist and have been found to produce degradation of insulating materials that is visually observable.

Upon defining the technical basis for the sample of cables to be inspected under the Non-EQ Insulated Cables and Connections Program, the program will be consistent with GALL XI.E1, "Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements."

The scope of the Non-EQ Insulated Cables and Connections Program will also be applied to instrument cable insulation, as addressed in Section XI.E2 of the GALL Report; however, the calibration of instrument circuits for the purpose of detecting insulation degradation, as called for in GALL XI.E2, is not part of the RNP program. This is acceptable because the visible effects of localized adverse environments caused by heat or radiation would be manifest on all electrical cables, including instrument cables, prior to significant IR degradation.

3.6.2.3.1.2 Staff Evaluation

In Table 3.6-1, the applicant identifies embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced IR, electrical failure caused by thermal/thermoxidative degradation of organics, radiolysis and photolysis (UV sensitive materials only) of organics, radiation-induced oxidation, and moisture intrusion as the aging effects of cables and connections due to heat or radiation. The staff concurs with the aging effects identified by the applicant. These aging effects are consistent with the aging effects identified by the staff in the GALL Report.

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 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 from the sample, non-PVC cables inside and outside containment in an adverse, localized environment, 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, "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 the requested information is 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. In response to the Confirmatory Item, the applicant, in a letter dated September 16, 2003, revised LRA Section B.4.6 to read, "The sample locations will consider the location of cables and connections inside and outside containment as well as any known adverse localized environments." This is acceptable. On this basis, Confirmatory Item 3.6.2.3.1.2-1 is closed.

Aging Management Program for Non-EQ Insulated Cables and Connections (B.4.6)

As a result of the AMP audit conducted at RNP on May 28 and 29, 2003, the applicant revised AMP B.4.6, "Non-EQ Insulated Cables and Connections," on June 13, 2003. The applicant stated that this is a condition monitoring program designed to provide reasonable assurance that age-related degradation will not inhibit the intended function of insulated cables and connectors within the scope of license renewal during the period of extended operation. The non-EQ insulated cables and connections managed by this program include those used in power, instrumentation, control, and communication applications. The aging effects managed include embrittlement, discoloration, cracking, swelling, or surface contamination leading to reduced IR or electrical failure.

The evaluation of the applicant's AMP for non-EQ insulated cables and connections focused on program elements. To determine whether the applicant's AMP is adequate to manage the effects of aging so that the intended functions will be consistent with the CLB for the period of extended operation, the staff evaluated the following seven elements—(1) scope of program, (2) preventive actions, (3) parameters monitored or inspected, (4) detection of aging effects, (5) monitoring and trending, (6) acceptance criteria, and (7) operating experience. The staff's evaluation of the applicant's corrective action, confirmation process, and administrative controls is provided separately in Section 3.0.4 of the staff's safety evaluation.

Scope of Program: The Non-EQ Insulated Cables and Connections Program includes accessible (i.e., able to be approached and easily viewed) insulated cables and connections installed in structures (i.e., areas) within the scope of license renewal. This program includes cables and connections installed in an adverse, localized environment caused by heat or radiation, as well as other plant areas. An adverse, localized environment is defined as a condition in a limited plant area that is significantly more severe than the specified service condition for the cable or connection. Except for the low level signal instrumentation circuits discussed in Section 3.6.2.3.2, the staff concludes that the scope of the program is acceptable because it includes all accessible non-EQ cables and connections that are subject to a potentially adverse, localized environment of heat and radiation that could cause applicable aging effects in these cables and connections.

Preventive Actions: No actions are taken as part of this program to prevent or mitigate aging degradation. This is acceptable because the staff finds no need for such actions.

Parameters Monitored or Inspected: A representative sample of accessible electrical cables and connections installed in adverse, localized environments are visually inspected for cable and connection jacket surface anomalies, such as embrittlement, discoloration, cracking, swelling, or surface contamination. Cable and connection jacket surface anomalies are precursor indications of conductor insulation aging degradation from heat or radiation in the presence of oxygen, and may indicate the existence of an adverse, localized environment. The staff finds the visual technique to be acceptable because it provides indications that can be visually implemented to preclude aging effects of accessible cables and connections.

Detection of Aging Effects: Accessible insulated cables and connections installed in areas within the scope of license renewal will be inspected at least once every 10 years. Following issuance of a renewed operating license for RNP, the initial inspection will be completed before the end of the initial 40-year license term for RNP (July 31, 2010). The staff finds that a 10-year inspection frequency is an adequate period to preclude failure of the conductor insulation because aging degradation is a slow process.

Monitoring and Trending: Trending of discrepancies will be performed as required in accordance with the RNP Corrective Action Program. Corrective action, as described in Chapter 17 of the RNP UFSAR, is implemented by the RNP Quality Assurance Program in accordance with 10 CFR 50, Appendix B. The staff finds the absence of trending to be acceptable because the ability to trend inspection results is limited and the staff did not see a need for such activities. The staff also finds the trending of discrepancies in accordance with the RNP Corrective Action Program to be acceptable.

Acceptance Criteria: The acceptance criterion is no unacceptable, visual indications of jacket surface anomalies which would suggest that conductor insulation applicable aging effects may exist, as determined by engineering evaluation. An unacceptable indication is defined as a noted condition or situation that, if left unmanaged, could lead to a loss of the license renewal intended function. The staff finds the acceptance criterion to be acceptable because it ensures that the cables and connections intended functions are maintained under all CLB design conditions for the period of extended operation.

Operating Experience: This is a new program; there is no existing operating experience to validate the effectiveness of this program. The GALL Report is based on industry operating experience through April 2001. Subsequent RNP operating experience will be captured through the operating experience review process. The operating experience review process is fully implemented at RNP and used to improve plant procedures and operating practices. This process will continue throughout the period of extended operation. The staff finds that the applicant has adequately addressed operating experience.

<u>Aging Management Program for Fuse Holders (B.4.9)</u>

In response to the staff's concern about the fuse holder (RAI 2.5.2-1), the applicant stated, in a letter dated April 28, 2003, that fuse holders are typically constructed of blocks of rigid insulating material, such as phenolic resins. Metallic clamps are attached to the blocks to hold each end of the fuse. The clamps can be spring-loaded clips that allow the fuse ferrules or blades to slip in, or they can be bolt lugs to which the fuse ends are bolted. The clamps are typically made of either copper or aluminum. The program focuses on the metallic clamp (clip) portion of the fuse holder. By letter dated June 13, 2003, the applicant clarified that the insulating material for the fuse holders will be managed by the Non-EQ Insulated Cables and Connections Program.

The applicant identified oxidation, corrosion, thermal fatigue from ohmic heating and electrical transients, mechanical fatigue from frequent removal and replacement, or vibration as the principal aging effects for the fuse holder. The staff concurs with the aging effects identified by the applicant. These aging effects are consistent with the aging effects identified by the staff in ISG-05.

RNP has elected to implement an AMP for fuse holders to ensure that they will continue to perform their intended function for the extended period of operation. The program applies to susceptible fuse holders outside of active devices. The program focuses on the metallic clamp (or clip) portion of the fuse holder. The parameters monitored include oxidation, corrosion, chemical contamination, thermal fatigue in the form of high resistance caused by ohmic heating, thermal cycling or electrical transients, and mechanical fatigue caused by frequent manipulation of the fuse itself or vibration. The evaluation of the applicant's AMP for fuse holders focused on program elements. To determine whether the applicant's AMP is adequate to manage the effects of aging so that the intended function will be consistent with the CLB for the period of extended operation, the staff evaluated the following seven elements—(1) scope of program, (2) preventive actions, (3) parameters monitored or inspected, (4) detection of aging effects, (5) monitoring and trending, (6) acceptance criteria, and (7) operating experience. The staff's evaluation of the applicant's corrective action, confirmation process, and administrative controls is provided separately in Section 3.0.4 of the staff's safety evaluation report.

Scope of Program: This program applies to fuse holders located outside of active devices that have been identified as being susceptible to aging effects. Fuse holders inside an active device are not within the scope of this program. The staff considers the scope of the program acceptable.

Preventive Actions: No actions are taken as part of this program to prevent or mitigate aging degradation. This is acceptable because the staff finds no need for such actions.

Parameters Monitored or Inspected: This program will focus on the metallic clamp (or clip) portion of the fuse holder. The parameters monitored include thermal fatigue in the form of high resistance caused by ohmic heating, thermal cycling or electrical transients, mechanical fatigue caused by frequent manipulation of the fuse itself or vibration, chemical contamination, corrosion, and oxidation. The staff finds this acceptable because it provides a means for monitoring the applicable aging effects on the metallic clamp portion of the fuse holder.

Detection of Aging Effects: Identified fuse holders within the scope of license renewal that are located outside of an active device will be tested at least once every 10 years. Testing may include thermography, contact resistance testing, or other appropriate methods to be determined prior to testing. Following issuance of a renewed operating license for RNP, the first test will be completed before the end of the initial 40-year license term for Unit 2 (July 31, 2010). The staff finds the above testing acceptable because these tests will locate hot spots (potential degradation). The staff also finds a 10-year testing frequency is an adequate period to preclude failure of the fuse holders because aging degradation is a slow process.

Monitoring and Trending: Trending of discrepancies will be performed as required in accordance with the Corrective Action Program. Corrective action, as described in Chapter 17 of the Unit 2 UFSAR, is part of the RNP Quality Assurance Program. The staff finds this process to be acceptable because the trending of discrepancies will be performed in accordance with Corrective Action Program.

Acceptance Criteria: The acceptance criteria will be determined based on the test selected for this inspection program. The staff finds this to be acceptable because the acceptance criteria is dependent on the test selected.

Operating Experience: Site-specific and industry-wide operating experience has shown that the loosening of fuse holders is an aging mechanism that, if left unmanaged, has led to a loss of electrical continuity function. The staff finds that the applicant has adequately addressed operating experience.

The staff also reviewed the UFSAR Supplement of the AMPs and finds that it provides an adequate summary description of the program

3.6.2.3.1.3 Conclusions

On the basis of its review of the applicant's Non-EQ Insulated Cables and Connections and fuse holders programs, the staff finds that the programs adequately address the 10 program elements defined in Branch Technical Position (BTS) RLSB-1 in Appendix A.1 of the SRP-LR, and that the programs will adequately manage the aging effects for which they are credited so that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 50.21(a)(3). The staff also reviewed the UFSAR Supplement for these AMPs and finds that they provide an adequate summary description of the program, as required by 10 CFR 54.21(d).

Therefore, the staff concludes that actions have been identified and have been or will be taken to manage the effects of aging during the period of extended operation on the functionality of SCs subject to an AMR such that there is reasonable assurance that the activities authorized by a renewed license will continue to be conducted in accordance with the CLB, as required by 10 CFR 54.29(a).

3.6.2.3.2 Electrical Cables Used in Instrumentation Circuits Not Subject to 10 CFR 50.49 EQ Requirements That Are Sensitive to Reduction in Conductor Insulation Resistance

3.6.2.3.2.1 Summary of Technical Information in the Application

The applicant stated that the scope of the Non-EQ Insulated Cables and Connections Program will also be applied to instrument cable insulation, as addressed in Section XI.E2 of the GALL Report; however, the calibration of instrument circuits for the purpose of detecting insulation degradation, as called for in GALL XI.E2, is not part of the RNP program. The applicant determined that this is acceptable because the visible effects of localized, adverse environments caused by heat or radiation would be manifest on all electrical cables, including instrument cables, prior to significant IR degradation.

3.6.2.3.2.2 Staff Evaluation

The applicant stated that the GALL Report contains an AMP specifically for cables with sensitive, low-level signals. However, RNP applies the Non-EQ Insulated Cables and Connections Program to this area. The applicant claimed that the inspection required by this program would be effective in identifying visual indications of insulation deterioration caused by environmental conditions (e.g., embrittlement, cracking, melting, discoloration, and swelling). This approach is considered by the applicant to be a preferred alternative to the AMP identified in the GALL Report.

The aging management activity (Table 3.6-1, Item 3, and Table 3.6-2, Item 2 of LRA) submitted by the applicant does not utilize the calibration approach for non-EQ electrical cables used in circuits with sensitive, low-level signals. Instead, these cables are simply combined with all other non-EQ cables under the visual inspection activity. The staff believes, however, that visual inspection alone would not necessarily detect reduced IR levels in cable insulation before the intended function is lost. Exposure of electrical cables to localized environments caused by heat, radiation, or moisture can result in reduced IR. Reduced IR causes an increase in leakage currents between conductors and from individual conductors to ground. A reduction in IR is a concern for circuits with sensitive, low-level signals, such as radiation monitoring and nuclear instrumentation, because it may contribute to inaccuracies in the instrument loop.

The staff is not convinced that aging of these cables will initially occur on the outer jacket resulting in sufficient damage to enable visual inspection to be effective in detecting the degradation before IR losses lead to a loss of its intended function, particularly if the cables are also subject to moisture. Therefore, the staff requested the applicant to provide a technical justification that will demonstrate that visual inspection will be effective in detecting damage before current leakage can affect instrument loop accuracy, or propose an alternate aging management activity (RAI 3.6.1-2). In response to the staff's above concern, the applicant, in a letter dated April 28, 2003, stated that RNP will implement AMPs to manage the aging effects of high-range radiation and neutron flux instrumentation circuits. These are two separate, but related programs. The AMP for the high-range radiation monitoring instrumentation circuits is consistent with the Non-EQ Electrical Cables Used in Instrumentation Circuits Program presented in the GALL Report, Volume 2, Section XI.E2. As this cable monitoring program is modeled after the GALL Report, the staff concluded that the requirements of 10 CFR 54.21(a)(3) have been met.

The applicant further stated that neutron flux monitoring instrumentation cables that may experience a reduction in IR require a different program other than the one presented in the GALL Report, Volume 2, Section XI.E2, because these cables are disconnected from their circuits during calibration. The applicant provided the details of the AMP for neutron flux instrumentation circuits. The scope of the program includes those cables associated with the source range, intermediate range, power range, and gamma-metrics circuits of the excore nuclear instrumentation system.

Aging Effects

In Table 3.6-1, the applicant identifies embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced IR, electrical failure caused by thermal/thermoxidative degradation of organics, radiation-induced oxidation, and moisture intrusion as aging effects of cables and connections due to heat or radiation. The staff concurs with the aging effects identified by the applicant. These aging effects are consistent with the aging effects identified by the staff in the GALL Report.

Aging Management Program

RNP will implement an AMP for high-range radiation monitoring instrumentation circuits. The scope of the program is limited to the cables associated with the containment vessel (CV) high range monitors. The High-Range Radiation Monitoring Instrumentation Circuits Program is consistent with the GALL XI.E2 Program. In this AMP, calibration results or findings of surveillance testing programs are used to identify the potential existence of aging degradation.

Additionally, RNP will implement an AMP for neutron flux instrumentation circuits. The scope of the program is limited to the cables associated with the source range, intermediate range, power range, and gamma-metrics circuits of the excore nuclear instrumentation system. This is a non-GALL program. In this AMP, an appropriate test, such as IR tests, time domain reflectometry (TDR) tests, or I/V testing will be used to identify the potential existence of a reduction in cable IR.

The evaluation of the applicant's AMP focused on program elements. To determine whether the applicant's AMP is adequate to manage the effects of aging so that the intended function will be consistent with the CLB for the period of extended operation, the staff evaluated the following seven elements—(1) scope of program, (2) preventive actions, (3) parameters monitored or inspected, (4) detection of aging effects, (5) monitoring and trending, (6) acceptance criteria, and (7) operating experience. The staff's evaluation of the applicant's corrective action, confirmation process, and administrative controls is provided separately in

Section 3.0.4 of the staff's safety evaluation.

<u>Aging Management Program for Non-EQ Electrical Cables Used in Instrumentation Circuits (B.4.7)</u>

Scope of Program: This program applies to the non-EQ cables used in CV high-range radiation monitoring instrumentation circuits. The staff finds that the scope of the program is acceptable because these cables are part of the calibration program. Cables associated with neutron flux instrumentation circuits are not included in this program because the calibration program does not include these cables.

Preventive Actions: No actions are taken as part of this program to prevent or mitigate aging degradation. This is acceptable because the staff finds no need for such actions.

Parameters Monitored or Inspected: The parameters monitored are determined from the specific calibrations or surveillances performed and are based on the specific instrumentation circuit under surveillance or being calibrated, as documented in plant surveillance calibration or surveillance procedures. The staff finds this approach to be acceptable because it provides a means for monitoring the aging effects of non-EQ electrical cables used in instrumentation circuits.

Detection of Aging Effects: Review of calibration results or findings of surveillance programs can provide an indication of aging effects by monitoring key parameters and providing data based on acceptance criteria related to instrumentation circuit performance. Reviews of results obtained during normal calibrations or surveillances may detect severe aging degradation prior to loss of cable intended function. The first reviews will be completed before the end of the initial 40-year license term for Unit 2 (July 31, 2010) and every 10 years thereafter. All calibrations or surveillances that fail to meet the acceptance criteria will be reviewed at that time. The staff finds this action to be acceptable because the review of calibrations or surveillances that fail to meet the acceptance criteria will provide reasonable assurance that age-related degradation of the cables will be detected prior to loss of cable intended function.

Monitoring and Trending: Trending actions are not included as part of this program because the ability to trend test results is dependent on the specific type of test chosen. Trending of discrepancies will be performed as required in accordance with the RNP Corrective Action Program. Corrective action, as described in Chapter 17 of the Unit 2 UFSAR, is part of the RNP Quality Assurance Program. The staff finds this process to be acceptable because trending of discrepancies will be performed in accordance with the Corrective Action Program.

Acceptance Criteria: Calibration results or findings of surveillances are to be within the acceptance criteria, as set out in the calibration or surveillance procedure. The staff finds this to be acceptable because surveillance or calibration activity ensures that cable intended functions used in instrumentation circuits are maintained under all CLB design conditions during the period of extended operation.

Operating Experience: 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 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 requested information 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. In response to Confirmatory Item 3.6.2.3.2.2-1, the applicant, in a letter dated September 16, 2003, revised the operating experience to include the following statement:

> "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 found this statement to be acceptable. On this basis, Confirmatory Item 3.6.2.3.2.2-1 is closed. The staff also reviewed the UFSAR Supplement of the AMPs and finds that it provides an adequate summary description of the program.

<u>Aging Management Program for Neutron Flux Instrumentation (B.4.8)</u>

Scope of Program: This program applies to the non-EQ cables used in the source range, intermediate range, power range, and gamma-metrics instrumentation circuits of the excore nuclear instrumentation system. The staff finds the scope of the program to be acceptable because these cables are not part of the calibration program.

Preventive Actions: No actions are taken as part of this program to prevent or mitigate aging degradation. This is acceptable because the staff finds no need for such actions.

Parameters Monitored or Inspected: The parameters monitored include a loss of dielectric strength caused by thermal/thermoxidative degradation of organics or radiation-induced oxidation (radiolysis) of organics. The staff finds this to be acceptable because loss of dielectric strength will lead to reduced IR.

Detection of Aging Effects: The cables used in neutron flux instrumentation circuits will be tested at least once every 10 years. Testing may include IR 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 first test will be completed before the end of the initial 40-year license term for Unit 2 (July 31, 2010). The staff finds the above testing acceptable because such testing will determine cable IR (potential degradation). However, the staff is concerned about the 10-year testing frequency. 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 12 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. In response, the applicant, in a letter dated September 16, 2003, stated the following:

> 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. In addition, the applicant modified the *Operating Experience* element as described below. This is acceptable. On this basis, Confirmatory Item 3.6.2.3.2.2-2 is closed.

Monitoring and Trending: Trending of discrepancies will be performed as required in accordance with the RNP Corrective Action Program. Corrective action, as described in Chapter 17 of the Unit 2 UFSAR, is part of the RNP Quality Assurance Program. The staff finds this to be acceptable because trending of discrepancies will be performed in accordance with the Corrective Action Program.

Acceptance Criteria: The acceptance criteria will be determined based on the test selected for this program. The staff finds this to be acceptable because the acceptance criteria is dependent on the test selected.

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 detectors 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.

The staff finds that the applicant has adequately addressed operating experience.

The staff also reviewed the UFSAR Supplement of the AMPs and finds that it provides an adequate summary description of the program.

3.6.2.3.2.3 Conclusions

On the basis of its review and audit of the applicant's program, the staff finds that the AMP for high-range radiation monitoring instrumentation is consistent with the GALL XI.E2 program and this program provides adequate management of the aging effects of the cables used in high-range radiation monitoring instrumentation. The staff also reviewed the UFSAR Supplement for this AMP and finds that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

The staff concludes that the applicant has demonstrated that the AMP for high-range radiation monitoring instrumentation circuits will effectively manage the aging effects of cables used in high-range radiation monitoring instrumentation circuits and that these circuits will perform its intended function in accordance with the CLB, as required by 10 CFR 54.29(a).

On the basis of its review, the staff finds that the AMP for neutron flux instrumentation is a non-GALL program and that this program provides adequate management of the aging effects of the cables used in neutron flux instrumentation. The staff also reviewed the UFSAR Supplement for this AMP and finds that it provides an adequate summary description of the program as required by 10 CFR 54.21(d).

The staff concludes that the applicant has demonstrated that the AMP for neutron flux instrumentation circuits will effectively manage the aging effects of cables used in neutron flux instrumentation circuits, and these circuits will perform its intended function in accordance with the CLB, as required by 10 CFR 54.29(a).

3.6.2.3.3 Inaccessible Medium-Voltage Cable Not Subject to 10 CFR 50.49 EQ Requirements

3.6.2.3.3.1 Summary of Technical Information in the Application

The applicant stated that no medium-voltage cables that are potentially susceptible to wetting provide any license renewal intended function. Therefore, no aging management activities are required.

3.6.2.3.3.2 Staff Evaluation

The applicant states that no AMP is required for 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 applicant determined that no medium-voltage cables, that are potentially susceptible to wetting, provide any license renewal intended function. The staff believes that some circuits (e.g., service water pumps) will be susceptible to wetting and hence an AMP is necessary. The staff requested the applicant, in RAI 3.6.1-4, to identify cables that are installed in conduits or direct buried and explain how the aging due to wetting will be managed. In response to the staff's request, the applicant, in a letter dated April 28, 2003, stated that energized medium-voltage cables are subject to a phenomenon known as water treeing which can ultimately result in failure of the cable insulation. For the purposes of license renewal, medium-voltage is defined as 2 kV to 15 kV. According to the DOE/Sandia Aging Management Guideline (SAND 96-0344), the incidence of cable failure due to water treeing has been found to be more prevalent as voltage level increases. The RNP evaluated all medium-voltage circuits to determine which inscope components were fed by cables that were direct buried, in underground conduits, or in duct banks. This review found that there were no in-scope energized and wetted medium-voltage cables at RNP. This aging mechanism has not been observed in low-voltage cables, which are defined as cables rated at less than 2 kV.

The staff finds that the applicant provided adequate justification for not having an AMP for inaccessible medium-voltage cables not subject to 10 CFR 50.49 EQ requirements.

3.6.2.3.3.3 Conclusions

On the basis of its review, the staff concludes that no AMP is needed to manage the aging of inaccessible medium-voltage cables susceptible to wetting.

3.6.2.4 Aging Management of Plant-Specific Components

The applicant credits one AMP to manage the aging effect associated with plant-specific electrical and I&C components. The following sections provide the results of the staff's evaluation of the adequacy of aging management for plant-specific electrical and I&C components.

3.6.2.4.1 Bus Duct

3.6.2.4.1.1 Summary of Technical Information in the Application

The applicant stated that a bus duct provides a means of connecting electrical power between equipment utilizing a preassembled, metal-enclosed raceway with conductors installed on insulated supports. Bus ducts were not evaluated in the GALL Report. Based on the RNP AMR, no applicable aging effects were identified for the bus duct. Therefore, it is concluded that no aging management activities are required for the extended period of operation.

3.6.2.4.1.2 Staff Evaluation

In the LRA Section 2.5.2, the applicant determined whether bus ducts meet the screening criteria of 10 CFR 54.21(a)(1)(i) and evaluated these components against 10 CFR 54.21(a)(1)(ii). However, in Table 3.6-2, the applicant stated that, "Based on the RNP AMR, no applicable aging effects were identified for the bus duct. Therefore, it is concluded that no aging management activities are required for the extended period of operation." The staff requested the applicant to explain why the connections (two end devices and intermediate points) will not require any aging management (RAI 2.5.2-2). These circuits may be exposed to appreciable ohmic or ambient heating during operation and may experience loosening related to the repeated cycling of connected loads or the ambient temperature environment (described in SAND 96-0344).

In response to the staff's above concern, the applicant, by letter dated April 28, 2003, stated that although the loosening of bolted connections is not a credible aging effect for RNP bus ducts, RNP has conservatively elected to implement an AMP (B.4.10) to identify and manage potential aging degradation.

The applicant stated that the bus ducts utilize preassembled raceway (enclosure) design with internal conductors installed on electrically insulated supports. Bus ducts are constructed of various metals, porcelain, PVC, and silicon caulk. Bus ducts at RNP include (1) generator isolated phase bus ducts, and (2) nonsegregated 4.16 kV and 480 V bus ducts. Bus ducts electrically connect specified sections of an electrical circuit to deliver voltage or current to various equipment and components throughout the plant. In LRA Section 2.5.3.1, the applicant stated that there are no bus ducts within the scope of license renewal that are included in the 10 CFR 50.49 program.

<u>Aging Effects</u>

The applicant identified oxidation, loosening of bolted connections due to thermal cycling, and corrosion due to moisture as the aging effects/mechanism for the bus ducts. The staff concurs with the aging effects identified by the applicant. The staff finds cracks, foreign debris, excessive dust buildup, and evidence of water intrusion as additional aging effects which are addressed in the AMP.

<u>Aging Management Programs (B.4.10)</u>

The applicant stated that although the loosening of bolted connections is not a credible aging effect for RNP bus ducts, RNP has conservatively elected to implement an AMP to identify and manage potential aging degradation. This is a non-GALL program and will provide reasonable assurance that the bus ducts will continue to perform their intended function consistent with the CLB through the period of extended operation.

The evaluation of the applicant's AMP focused on program elements. To determine whether the applicant's AMP is adequate to manage the effects of aging so that the intended function will continue to be performed consistent with CLB for the period of extended operation, the staff evaluated the following seven elements—(1) scope of program, (2) preventive actions, (3) parameters monitored or inspected, (4) detection of aging effects, (5) monitoring and trending, (6) acceptance criteria, and (7) operating experience. The staff's evaluation of the applicant's corrective action, confirmation process, and administrative controls is provided separately in Section 3.0.4 of the staff's safety evaluation.

Scope of Program: This program applies to the iso-phase bus duct, as well as the non-segregated 4.16 kV and 480 V bus ducts within the scope of license renewal. This is acceptable to the staff because the program will include all bus ducts within the scope of license renewal.

Preventive Actions: No actions are taken as part of this program to prevent or mitigate aging degradation. This is acceptable because the staff finds no need for such actions.

Parameters Monitored or Inspected: A sample of accessible bolted connections will be checked for proper torque. This program will also inspect the bus duct for cracks, corrosion, foreign debris, excessive dust buildup, and evidence of water intrusion. The bus itself will be inspected for signs of cracks, corrosion, or discoloration, which may indicate overheating. The internal bus supports will be inspected for structural integrity and signs of cracks. The staff finds that the visual inspection of bus ducts, bus bar, and internal bus supports will provide an indication of aging effects. Additionally, checking of sample bolted connections for proper torque will provide assurance that bus ducts are not exposed to excessive ohmic or ambient heating.

Detection of Aging Effects: This program will be completed before the end of the initial 40-year license term for Unit 2 (July 31, 2010) and every 10 years thereafter. The staff finds that the 10-year inspection frequency is an adequate period to preclude failure of bus ducts because industry experience has shown that the aging degradation is a slow process.

Monitoring and Trending: Trending actions are not included as part of this program. Trending will be performed in accordance with the Corrective Action Program. Corrective action, as described in Chapter 17 of the UFSAR, is part of the RNP Quality Assurance Program. The staff finds this to be acceptable because trending will be performed in accordance with the Corrective Action Program.

Acceptance Criteria: Bolted connections must meet the minimum torque specifications. Additional acceptance criteria include no unacceptable indications of cracks, corrosion, foreign debris, excessive dust buildup, or discoloration, which may indicate overheating or evidence of water intrusion. An "unacceptable indication" is defined as a noted condition or situation that, if left unmanaged, could lead to a loss of license renewal intended function. The staff finds the acceptance criteria to be acceptable because the bolted connections must meet the minimum torque requirement specified by the manufacturer.

Operating Experience: Industry experience has shown that bus ducts exposed to appreciable ohmic or ambient heating during operation may experience loosening of bolted connection related to the repeated cycling of connected loads or the ambient temperature environment. This phenomenon can occur in heavily loaded circuits (i.e., those exposed to appreciable ohmic heating or ambient heating) that are routinely cycled. The staff finds that the proposed program will provide assurance that bus ducts are not exposed to excessive ohmic or ambient heating.

The staff also reviewed the UFSAR Supplement of the AMPs and finds that it provides an adequate summary description of the program.

3.6.2.4.1.3 Conclusions

On the basis of its review of the applicant's program, the staff finds that the program adequately addresses the 10 program elements defined in Branch Technical Position (BTS) RLSB-1 in Appendix A.1 of the SRP-LR, and that the program will adequately manage the aging effects for which it is credited so that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 50.21(a)(3). The staff also reviewed the UFSAR Supplement for this AMP and finds that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

Therefore, the staff concludes that actions have been identified and have been or will be taken to manage the effects of aging during the period of extended operation on the functionality of SCs subject to an AMR such that there is reasonable assurance that the activities authorized by a renewed license will continue to be conducted in accordance with the CLB, as required by 10 CFR 54.29(a).

3.6.2.4.2 Non-EQ Electrical Penetration Assemblies

3.6.2.4.2.1 Summary of Technical Information in the Application

The applicant stated that the components of non-EQ electrical penetration assemblies subject to AMR are the organic insulating materials associated with electrical conductors and connections. Therefore, the non-EQ electrical penetration assemblies are included with the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualifications Requirements Program. Considering cable systems to include penetration assemblies is consistent with GALL XI.E1, "Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements in the GALL Report."

3.6.2.4.2.2 Staff Evaluation

In the LRA Section 3.6.2.1, the applicant states that the components of non-EQ electrical penetration assemblies subject to AMR are the organic materials associated with electrical conductors and connections. It is not clear to the staff why the epoxy seal and other insulating material associated with the electrical penetration assemblies do not require an AMR.

In response to the above concern, documented in RAI 3.6.1-1, the applicant, by letter dated April 28, 2003, stated that electrical penetration assemblies are used to pass electrical circuits through the containment wall while maintaining containment integrity. They provide electrical continuity for the circuit, as well as a pressure boundary for the containment. The pressure boundary function of electrical penetration assemblies is addressed in LRA Table 2.4-1. The intent of the electrical AMR of electrical penetration assemblies is to preserve the assemblies' electrical continuity function. The focus of this review is the interaction between the assemblies' organic insulating materials and their operating environment. The organic insulating materials comprise the penetration assemblies' primary insulation system.

In addition to organic insulating materials, there are other materials (metals and inorganic materials) used in the construction of the penetration assembly. These include cable fillers, epoxies, potting compounds, connector pins, plugs, and facial grommets. Consistent with the DOE/Sandia Aging Management Guideline (i.e., SAND 96-0344) these items have no significant effect on the normal aging process of the primary insulation system and do not adversely affect the electrical continuity function. Accordingly, they are not included in the AMR of electrical penetration assemblies. The staff concurred that the components subject to aging in the electrical penetration assemblies are the materials used for the electrical cables and connections.

By letter dated June 13, 2003, the applicant clarified that the electrical penetrations used for high-range radiation monitoring circuits and neutron flux instrumentation circuits are in the EQ Program and, therefore, are not credited to manage the aging effects of non-EQ electrical penetration assemblies. The staff agrees with the applicant that the non-EQ electrical penetration assemblies are included with the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program. Section 3.6.2.3.1 provides more detail on this program.

3.6.2.4.2.3 Conclusions

On the basis of its review, the staff concludes that the applicant has adequately identified the aging effect and has an adequate AMP in place for managing the aging effects for containment electrical penetrations, such that the intended functions for the component will be maintained consistent with the CLB for the period of extended operation as required by 10 CFR 54.21(a)(3).

The staff also reviewed the applicable USAR Supplement program descriptions and concludes that the USAR Supplement provides an adequate program description of the AMPs credited for managing aging in containment electrical penetrations to satisfy 10 CFR 54.21(d)..

3.6.2.4.3 High-Voltage Electrical Switchyard Bus

3.6.2.4.3.1 Summary of Technical Information in the Application

The switchyard bus electrically connects specified sections of an electrical circuit to deliver voltage or current to various equipment and components throughout the plant. The switchyard bus is used in switchyards to connect two or more elements of an electrical power circuit, such as active disconnect switches and passive transmission conductors. The material used for the switchyard bus is aluminum and iron.

<u>Aging Effects</u>

The applicant identified connection surface oxidation and vibration as the aging effects/mechanism for the switchyard bus.

<u>Aging Management Program</u>

The applicant states that connection surface oxidation is an applicable aging effect. All switchyard bus connections have welded and/or compression connections. For the service conditions encountered at RNP, no aging effects have been identified that could cause a loss of intended function. Vibration is not an applicable aging mechanism because the switchyard bus has no connections to moving or vibrating equipment. Switchyard buses are connected to flexible conductors that do not normally vibrate and are supported by insulators mounted to static, structural components, such as cement footings and structural steel. This configuration provides reasonable assurance that the switchyard bus will perform its intended function for the period of extended operation. No AMP for switchyard bus is required.

3.6.2.4.3.2 Staff Evaluation

In Table 1, "AMR Results for the Offsite Power System Electrical Components," of the RAI 2.5.1-1 response, the applicant identified connection surface oxidation and vibration as the aging effects/mechanism for the switchyard bus. The staff concurs with the aging effects identified by the applicant. The staff also finds that the applicant adequately addressed the reasons that these aging effects are not applicable aging effects at RNP. The staff agrees that there is reasonable assurance that the switchyard bus will perform its intended function for the period of extended operation.

3.6.2.4.3.3 Conclusions

On the basis of the staff's review of the information presented in the RAI 2.5.1-1 response, the staff concludes that the switchyard bus has no aging effects that require management.

3.6.2.4.4 High-Voltage Transmission Conductors

3.6.2.4.4.1 Summary of Technical Information in the Application

Transmission conductors are uninsulated, stranded electrical cables used in switchyards, switching stations, and transmission lines to connect two or more elements of an electrical power circuit, such as active disconnect switches, power circuit breakers, and transformers to a passive switchyard bus. Transmission conductors are made of aluminum core steel reinforced (ACSR).

Aging Effects

The licensee identified loss of conductor strength and vibration as the aging effects/mechanism for the transmission conductors.

Aging Management Program

The applicant stated that loss of conductor strength due to corrosion of aluminum core steel reinforced transmission conductors is a very slow process. This process is even slower for rural areas with generally less suspended particles and sulfur dioxide concentrations in the air than urban areas. RNP is located in a rural area where airborne particle concentrations are comparatively low. Consequently, this is not considered a significant contributor to the aging of RNP transmission conductors. Transmission conductor vibration would be caused by wind loading. Wind loading is considered in the initial design and field installation of transmission conductors and high-voltage insulators throughout the CP&L transmission and distribution network. Loss of material (wear) and fatigue that could be caused by transmission conductor vibration or sway are not considered applicable aging effects that warrant aging management.

3.6.2.4.4.2 Staff Evaluation

In Table 1, "Aging Management Review Results for the Offsite Power System Electrical Components," of its RAI 2.5.1-1 response, the applicant identified loss of conductor strength and vibration as the aging effects/mechanism for transmission conductors. The staff concurs with the aging effects identified by the applicant. The staff also finds that the applicant adequately addressed the reasons these aging effects are not applicable at RNP. Additionally, the staff is aware of tests performed by Ontario Hydroelectric which showed a 30 percent loss of composite conductor strength of an 80-year-old ACSR conductor due to corrosion. The National Electric Safety Code requires that tension on installed conductors be a maximum of 60 percent of the ultimate conductor strength. Therefore, the staff concludes that there is reasonable assurance that the transmission conductors will perform their intended function for the period of extended operation.

3.6.2.4.4.3 Conclusions

On the basis of the staff's review of the information presented in the RAI 2.5.1-1 response, the staff concludes that transmission conductors have no aging effects that require management.

3.6.2.4.5 High-Voltage Insulators

3.6.2.4.5.1 Summary of Technical Information in the Application

High-voltage insulators typically used on transmission towers are insulating materials in a form designed to (1) support a conductor physically, and (2) separate the conductor electrically from another conductor or object. High-voltage insulators serve as an intermediate support between a supporting structure (such as a transmission tower or support pedestal) and switchyard bus or transmission conductor. Materials used for the high-voltage insulators are porcelain and metal.

Aging Effects

The applicant identified surface contamination, cracking, and loss of material due to wear as the aging effects/mechanism for the switchyard bus.

Aging Management Program

The applicant stated that surface contamination is not an applicable aging mechanism. The buildup of surface contamination is typically a slow, gradual process. The RNP is located in a rural area where airborne particle concentrations are comparatively low. Consequently, the rate of contamination buildup on the insulators is not significant. Any such contamination accumulation is washed away naturally by rainwater. The glazed surface on high-voltage insulators at RNP aids in the removal of this contamination. Therefore, there are no applicable aging effects that require management. Cracking is not an applicable aging mechanism. Cracking or breaking of porcelain insulators is typically caused by physical damage which is event driven, rather than an age-related mechanism. Mechanical wear is an aging effect for strain and suspension insulators if they are subject to significant movement. RNP transmission conductors do not normally swing, and when they do, because of strong winds, they dampen quickly once the wind has subsided. Loss of material due to wear has not been identified during routine inspections at RNP. No AMP is required.

3.6.2.4.5.2 Staff Evaluation

In Table 1, "Aging Management Review Results for the Offsite Power System Electrical Components," of its RAI 2.5.1-1 response, the applicant identified surface contamination, cracking, and loss of material due to wear as the aging effects/mechanism for high-voltage insulators. The staff concurs with the aging effects identified by the applicant. The staff also finds that the applicant adequately addressed the reasons these aging effects are not applicable at RNP. The staff agrees that there is reasonable assurance that the high-voltage insulators will perform their intended function for the period of extended operation.

3.6.2.4.5.3 Conclusion

On the basis of the staff's review of the information presented as in the RAI 2.5.1-1 response, the staff concludes that high-voltage insulators have no aging effects that require management.

3.6.3 Evaluation Findings

The staff has reviewed the information in Section 3.6 of the LRA and the RAI responses dated April 28, 2003, and June 13, 2003. On the basis of its review, the staff concludes that the applicant has adequately identified the aging effects, and the AMPs credited for managing the aging effects, for the electrical instrumentation and controls, such that there is reasonable assurance that the component intended functions will be maintained consistent with the CLB for the period of extended operation. The staff also reviewed the applicable UFSAR Supplement program descriptions and concludes that the UFSAR Supplement provides an adequate program description of the AMPs credited for managing aging effects, as required by 10 CFR 54.21(d).

4 TIME-LIMITED AGING ANALYSES

4.1 Identification of Time-Limited Aging Analyses

This section addresses the identification of time-limited aging analyses (TLAAs). The applicant discusses the TLAAs in license renewal application (LRA) Sections 4.2 through 4.6. The staff's review of the TLAAs can be found in Sections 4.2 through 4.6 of this safety evaluation report (SER).

The TLAAs include certain plant-specific safety analyses that are based on an explicitly assumed 40-year plant life. Pursuant to Title 10 of the *Code of Federal Regulations* (CFR), Part 54.21(c)(1), the applicant for license renewal provides a list of TLAAs, as defined in 10 CFR 54.3.

In addition, pursuant to 10 CFR 54.21(c)(2), an applicant must provide a list of plant-specific exemptions granted under 10 CFR 50.12 that are based on TLAAs. For any such exemptions, the applicant must provide an evaluation that justifies the continuation of the exemptions for the period of extended operation.

4.1.1 Summary of Technical Information in the Application

The applicant evaluated calculations for Robinson Nuclear Plant (RNP) against the six criteria specified in 10 CFR 54.3 to identify the TLAAs. The applicant indicated that calculations that meet the six criteria were identified by searching current licensing basis documents, including technical specifications, the updated final safety analysis report (UFSAR), environmental reports, docketed licensing correspondence, and industry documents such as NUREG-1800, Westinghouse Owner's Group Topical Reports, NUREG-1800, and Nuclear Energy Institute (NEI) 95-10. The applicant listed the following TLAAs in Table 4.1-1 of the LRA:

- reactor vessel neutron embrittlement, including analyses for upper shelf energy, pressurized thermal shock
- metal fatigue, including reactor vessel underclad cracking, reactor internals holddown springs and alignment pins, pressurizer insurge/outsurge, steam generators, pressurizer surge line thermal stratifications, and auxiliary feedwater lines
- environmental equipment qualification
- containment tendon stress relaxation
- containment penetration bellows fatigue
- reactor coolant pump fatigue and Code Case N-481 fracture mechanics analyses
- primary loop leak-before-break analysis
- crane mechanical fatigue

- Boraflex depletion allowance
- containment pile corrosion
- containment concrete temperature cycles

Pursuant to 10 CFR 54.21(c)(2), the applicant stated that no exemptions granted under 10 CFR 50.12 that were based on a TLAA, as defined in 10 CFR 54.3, were identified.

4.1.2 Staff Evaluation

In LRA Section 4.1, the applicant identified the TLAAs applicable to RNP and discussed exemptions based on TLAAs. The staff reviewed the information to determine whether the applicant provided information adequate to meet the requirements of 10 CFR 54.21(c)(1) and 10 CFR 54.21(c)(2).

As indicated by the applicant, TLAAs are defined in 10 CFR 54.3 as calculations and analyses that meet the following six criteria.

(1) involve systems, structures, and components within the scope of license renewal, as delineated in section 54.4(a)

(2) consider the effects of aging

(3) involve time-limited assumptions defined by the current operating term, for example, 40 years

(4) were determined to be relevant by the applicant in making a safety determination

(5) involve conclusions, or provide the basis for conclusions, related to the capability of the system, structure, and component to perform its intended functions, as delineated in Section 54.4(b)

(6) are contained or incorporated by reference in the current licensing basis

The applicant listed the TLAAs applicable to RNP in Table 4.1-1 of the LRA. Tables 4.1-2 and 4.1-3 in NUREG-1800 identify potential TLAAs determined from the review of other license renewal applications. In RAI 4.1-1 the staff requested that the applicant discuss two other issues:

(1) whether there are any calculations or analyses at RNP that address the topics listed in Tables 4.1-2 and 4.1-3 of NUREG-1800 and were not included in Table 4.1-1 of the LRA

(2) if they do exist, how these calculations or analyses were evaluated against the TLAA definition provided in 10 CFR 54.3

In its response dated April 28, 2003, to the request for additional information (RAI), the applicant indicated the following topics listed in NUREG-1800 are applicable to pressurized water reactor (PWR) facilities and were not included in Table 4.1-1 of the LRA.

(1) inservice flaw growth analysis of structure stability
(2) metal containment corrosion allowance
(3) high-energy line break analysis based on cumulative usage factor
(4) reactor vessel low temperature overpressure protection analysis
(5) main steam supply lines to the auxiliary feedwater pump
(6) reactor coolant pump flywheel fatigue analysis
(7) reactor vessel internals transient analysis
(8) reactor vessel internals fracture toughness ductility reduction
(9) containment liner plate fatigue analysis

On the basis of a search for RNP-specific TLAAs, the applicant identified calculations or analyses applicable to the reactor vessel (RV) for low temperature overpressure protection (LTOP) analysis (item 4), the main steam supply lines to auxiliary feedwater (AFW) pump (item 5), and the reactor coolant pump (RCP) flywheel fatigue analysis (item 6).

The analysis of the main steam supply lines to the AFW pump (item 5) is addressed in LRA Section 4.3.2. No explicit fatigue analysis of the main steam supply lines to the steam-driven AFW pump has been identified for RNP. Items 4 and 6 were determined not to meet the criterion from 10 CFR 54.3 that the analysis involves time-limited assumptions defined by the current operating term. The RNP LTOP analyses (item 4) have been performed for periods less than the current operating term and are periodically updated. Further discussion on this matter is provided in the applicant's response to RAI 4.2.3-1, Part 2. The RCP flywheel fatigue analysis (item 6) has been performed using an operating life of 60 years.

The supplemental RAI response, submitted by letter June 13, 2003, confirmed that, of the nine potential TLAA categories, only categories 4, 5, and 6 are applicable to RNP. On the basis of the discussion above, the staff finds acceptable the applicant's identification of the TLAAs applicable to RNP.

4.1.3 Conclusions

On the basis of its review, the staff concludes that the applicant has provided an acceptable list of TLAAs as defined in 10 CFR 54.3, as required by 10 CFR 54.21(c)(1), and has confirmed that no 10 CFR 50.12 exemptions have been granted on the basis of a TLAA, as required by 10 CFR 54.21(c)(2).

4.2 Reactor Vessel Neutron Embrittlement

During plant service, neutron irradiation reduces the fracture toughness of ferritic steel in the reactor vessel beltline region of light-water nuclear power reactors. Areas of review to ensure that the reactor vessel has adequate fracture toughness to prevent brittle failure during normal and off-normal operating conditions are (1) upper-shelf energy, (2) pressurized thermal shock for PWRs, (3) heatup and cooldown (P-T limits) curves and LTOP setpoints. The staff has evaluated the adequacy of these TLAAs for the items for the period of extended operation.

4.2.1 Summary of Technical Information in the Application

4.2.1.1 Pressurized Thermal Shock

In Section 4.2.1 of the LRA, the applicant summarized the applicable requirements in 10 CFR 50.61 for determining whether the RNP RV beltline materials will have adequate protection against PTS. The applicant stated that the calculated RT_{PTS} temperatures for RV beltline materials, including axial welds, circumferential welds, and plates, have been demonstrated to remain below the applicable PTS screening criteria throughout the 60-year license renewal period. The applicant stated that the limiting location is circumferential weld 10-273, which has a 60-year RT_{PTS} reference temperature more than 25 °F below the screening criterion (i.e., 60-year RT_{PTS} = 275 °F vs the 300 °F screening criterion for circumferential welds). The applicant stated that the RT_{PTS} values were calculated using the methodology found in 10 CFR 50.61.

The applicant also stated that conservative 60-year RT_{PTS} reference temperatures were also calculated for the RV inlet and outlet nozzles and welds, and that the highest 60-year RT_{PTS} reference temperature for the nozzles was 35 °F below the screening criterion (i.e., 60-year RT_{PTS} = 235 °F vs the 270 °F screening criterion for plates, forgings, and axial welds). The applicant stated that the nozzles and nozzle welds have been shown to meet the PTS criteria for 60 years and have been shown not to be the limiting components, since the beltline materials were closer to the limit. The applicant therefore stated that the inlet and outlet nozzles and welds need not be added to the RV Surveillance Program.

The applicant stated that the analysis associated with PTS has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

4.2.1.2 Reactor Vessel Upper-Shelf Energy

In Section 4.2.2 of the LRA, the applicant summarizes the applicable requirements for upper-shelf energies (USE) of RV beltline materials, as stated in Section IV.A.1 of 10 CFR Part 50, Appendix G. The applicant stated that the USE values for the RNP RV beltline materials were calculated for a 60-year operating period using methodology from 10 CFR Part 50, Appendix G, and RG 1.99, Revision 2, and the 60-year fluence projections.

The applicant stated that for welds and forgings exposed to end of life (EOL) fluence, the USE screening criterion is 50 ft-lbs minimum. The applicant stated that the projected 60-year USE values for reactor beltline axial and circumferential welds were shown to be above the minimum USE screening criteria. The limiting location is weld 2-273A, with a 60-year USE value of 56 ft-lbs, which is acceptable.

The applicant stated that for RV plate materials, a 42 ft-lbs minimum USE acceptance criterion has been established, based upon WCAP-13587, Revision 1, which demonstrated equivalent margins of safety for RNP vessel plates with USE as low as 42 ft-lbs. The applicant also stated that the 60-year USE values were calculated for RNP vessel plates and that the limiting plate location is plate W 10201-4, with a 60-year USE value of 45 ft-lbs, which is acceptable.

The applicant stated that the nozzle forgings have a 60-year USE value of 53 ft-lbs and that the nozzle welds have a 60-year USE value of 52 ft-lbs, compared with the 50 ft-lbs minimum criterion for welds and forgings from 10 CFR Part 50, Appendix G, which is acceptable.

The applicant stated that the analysis associated with 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).

4.2.1.3 Plant Heatup/Cooldown (Pressure/Temperature) Curves/Low-Temperature Overpressure Protection Power-Operated Relief Valve Setpoints

In Section 4.2.3 of the LRA, the applicant considered other analyses impacted by neutron embrittlement, specifically those for establishing the heatup/cooldown curves and LTOP setpoints for the RNP RV. These were determined not to be TLAAs because they are not based upon end-of-license fluence projections. The applicant stated that these analyses are periodically updated as required by regulations based upon fluence projections that bound the current period of operation, but that this period is not necessarily associated with the end of license. The applicant also stated that these analyses are also updated whenever new information is available that would significantly affect the projections, either from the Reactor Vessel Surveillance Program or from other industry sources, and that these analyses do not require updating as a part of the license renewal process since they will be updated when required in accordance with applicable regulations.

4.2.2 Staff Evaluation

Pursuant to 10 CFR 54.21(c), the applicant is required to provide a list of TLAAs as part of the application for the renewal of a license. The applicant stated that the group of TLAAs in Section 4.2 of the LRA deals with the cumulative effect of neutron irradiation on the materials that were used to fabricate the beltline region of the RV and whether neutron irradiation could lead to unacceptable embrittlement (i.e., loss of fracture toughness) in these materials before the end of the extended period of operation for RNP. These TLAAs therefore have direct relation to the structural integrity of the RV during the extended period of operation for RNP. For PWR light-water reactors, including RNP, the staff assesses the impacts of neutron irradiation on the following three parameters related to structural integrity for the RV materials:

(1) the reference temperatures for embrittlement (i.e., RT_{PTS} value) to ensure that the RV beltline materials will be adequately protected against postulated PTS events through the end of the extended period of operation for RNP

(2) the Charpy-V notch USE values for the RV beltline materials to ensure that the materials will have adequate ductility through the end of the extended period of operation for RNP

(3) the P-T limits and LTOP setpoints for the reactor vessel to protect the RNP RV during normal, transient, and pressure-test operating conditions through the end of the extended period of operation for RNP

The staff reviewed the TLAAs identified by the applicant and described in Sections 4.2.1, 4.2.2, and 4.2.3 of the LRA to ensure that the RV beltline materials would have sufficient remaining margins of safety for these parameters, as assessed in compliance with the safety margin/screening criteria requirements for these parameters defined in 10 CFR 50.61, Section

IV.A.1 of 10 CFR Part 50, Appendix G, and Section IV.A.2 of 10 CFR Part 50, Appendix G, respectively. The staff also reviewed these TLAAs to determine if the applicant had demonstrated that the TLAAs for parameters related to structural integrity had been adequately projected to the end of the period of extended operation for RNP, as required by 10 CFR 54.21(c)(1)(ii). The staff evaluates these TLAAs for PTS, USE, and P-T/LTOP limits in Sections 4.2.2.1, 4.2.2.2, and 4.2.2.3 of this SER, respectively.

4.2.2.1 Pressurized Thermal Shock

The requirements for demonstrating that RVs in U.S. PWR light-water reactor facilities will have adequate protection against PTS events are specified in 10 CFR 50.61. The rule establishes PTS screening criteria[1] for RV beltline forging, plate, and weld materials, and requires applicants to calculate a PTS reference temperature (i.e., the RT_{PTS} value) for each beltline material in the reactor vessel. The applicant must also demonstrate that the RT_{PTS} values for the materials will remain below the PTS screening criteria until the end of the license for the facility. The rule also contains the requirements for calculating the RT_{PTS} values for the beltline materials, which are based on the calculation methods contained in RG 1.99, Revision 2 (May 1988). The applicant did not include its end-of-extended-operating-period RT_{PTS} value calculations for the RNP beltline RV materials in its TLAA for PTS; instead, it only summarized the RT_{PTS} values for the limiting shell and nozzle materials in the RNP RV beltline through the expiration of the extended period of operation. The applicant stated that the limiting beltline material in the RNP RV was circumferential Weld 10-273 and that the RT_{PTS} value for this material at the expiration of the extended period of operation is 275 °F, which provides a 25 °F margin of safety when compared to the screening criterion for RV circumferential weld materials (300 °F). The applicant stated that for the RV nozzle materials within the RV beltline region, the RT_{PTS} value for the limiting nozzle material at the expiration of the extended period of operation is 235 °F, which is 35 °F less than the screening criterion for RV base metal and axial weld materials (270 °F).

Pursuant to 10 CFR 54.21(c)(1), the TLAA for PTS must demonstrate that RT_{PTS} values for the beltline materials will remain below the PTS screening criteria until the end of the period of extended operation for RNP. In order to demonstrate compliance with the requirements of both 10 CFR 54.21(c)(1) and 10 CFR 50.61, the staff requested, in RAI 4.2.1-1, that the applicant provide the inputs and results for the end-of-extended-operating-period RT_{PTS} calculations for all RNP beltline shell and nozzle materials and their associated weldments. The applicant provided its response to RAI 4.2.1-1 by letter dated May 15, 2003. In this letter, the applicant attached nonproprietary Class 3 topical report WCAP-15828, Revision 0 (March 2003), which provides the updated PTS assessments for the RNP RV through both the current and extended period of operation.

The staff reviewed the data and information in WCAP-15828, Revision 0, as it relates to the PTS assessment for RNP through the expiration of the extended period of operation for the unit (i.e., 60 years total of licensed life, 50 effective full power years (EFPYs). The staff performed an independent assessment of the PTS data in WCAP-15828, Revision 0, to assess the validity of the 50-EFPY RT_{PTS} calculations for the beltline plate, nozzle forgings, and weld materials in

[1]The PTS screening criteria in 10 CFR 50.61 are 270 °F for RV beltline forgings, plates, and longitudinal (axial) welds and 300 °F for RV beltline circumferential welds.

the RNP reactor vessel. The staff applied the 50 EFPY neutron fluence values cited in the report for the respective beltline materials in the RNP RV. These fluences are based on the material test data from the latest capsule withdrawal for the RNP Reactor Vessel Material Surveillance Program (i.e., Capsule X, as reported in WCAP-15805 March 2002).

The staff's independent calculation of the RT_{PTS} values for the RNP reactor vessel beltline materials through 50 EFPYs of operation confirms that all of the materials will have sufficient protection and margin of safety against PTS events through the expiration of the extended period of operation for the unit. The staff based its RT_{PTS} calculations on the 50-EFPY neutron fluences reported in WCAP-15828 for the RNP beltline materials. For the RNP RV, the limiting beltline material for PTS is upper shell-to-lower shell circumferential weld 10-273 (Weld Heat No. W5214). The staff calculated two RT_{PTS} values for this material—the first RT_{PTS} value as calculated if the chemistry factor (CF) for the material is obtained from the material copper and nickel alloying contents and determined from Table 1 in 10 CFR 50.61, and the second RT_{PTS} value as calculated if the CF is determined from applicable RV material surveillance capsules for this heat of material (i.e., from Capsules T, V, and X data as applicable to Weld Heat No. W5214). A full safety margin is applied to the calculations. The staff calculated the RT_{PTS} values for these materials to be 282 °F if Table 1 in 10 CFR 50.61 is used to calculate the CF, and 295 °F if the surveillance data are used to determine the CF, respectively. The corresponding RT_{PTS} values reported by the applicant in WCAP-15828 were 289 °F and 297 °F, respectively, and are slightly more conservative than those calculated by the staff.

The applicant and the staff calculations were in reasonable agreement with each other, and all values calculated by the applicant and the staff are below the corresponding PTS screening criterion for circumferential welds stated in 10 CFR 50.61. The staff therefore concludes that the applicant has sufficiently resolved the data requested in RAI 4.2.1-1. The staff also concludes that, based on the RT_{PTS} values for the RNP beltline materials, as calculated by both the applicant and the staff, the RNP RV beltline materials will have sufficient protection against PTS through the expiration of the period of extended operation for RNP. Based on this assessment, the staff concludes that the applicant's TLAA for PTS meets the acceptance criterion stated in 10 CFR 54.21(c)(1)(ii) and is acceptable.

4.2.2.2 Reactor Vessel Upper-Shelf Energy

Section IV.A.1 to 10 CFR Part 50, Appendix G, provides the Commission's requirements for demonstrating that reactor vessels in U.S. PWR light-water reactor facilities will have ductility throughout their service lives. The rule requires that the RV beltline materials have USE values in the transverse direction for the base metal and along the weld for the weld material of no less than 75 ft-lb initially, and must maintain USE values throughout the life of the vessel of no less than 50 ft-lb. However, USE values below these criteria may be acceptable if it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that the lower values of USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," provides an expanded discussion regarding the calculations of USE values and describes two methods for determining USE values for RV beltline materials, depending on whether or not a given RV beltline material is represented in the plant's Reactor Vessel Material Surveillance Program.

The applicant did not include its end-of-extended-operating-period USE value calculations for the RNP beltline RV materials in its TLAA for USE; instead, it summarized the end-of-extended-operating-period USE values only for the shell, weld, and nozzle forging materials in the RNP RV beltline through the expiration of the extended period of operation. The applicant stated that intermediate shell welds 2-273 A, B, and C will have the lowest USE values for all RNP beltline weld materials at the end of the extended operating period and that the USE values for these welds at the expiration of the extended period of operation are 56 ft-lbs. The applicant also stated that RNP RV nozzle forging materials within the RV beltline region have a USE value of 53 ft-lb at the end of the extended period of operation and that the RNP RV nozzle weld materials have a USE value of 52 ft-lb at the end of the extended period of operation. All of these USE values are above the end-of-life USE value screening criterion of 50 ft-lb and therefore meet the applicable USE requirements of 10 CFR Part 50, Appendix G.

The applicant also indicated that the limiting RV beltline materials for USE are beltline plates which have been evaluated using an equivalent margins analysis (EMA) that demonstrates that the plate materials would have equivalent safety margins for USE down to 42 ft-lb, when compared to the safety margin requirements required by Section XI of the ASME Boiler and Pressure Vessel Code. The applicant indicated that this EMA, as applicable through the end of the extended period of operation for RNP, is provided in topical report WCAP-13587, Revision 1.

For LRAs, pursuant to 10 CFR 54.21(c)(1), the TLAA for USE must demonstrate either that USE values for all RNP beltline materials will remain above the 50 ft-lb screening criterion of Section IV.A.1 of 10 CFR Part 50, Appendix G, through to the expiration of the period of extended operation for RNP, or that the beltline materials will have an acceptable margin of safety against ductile failure equivalent to that if the margin of safety is calculated in accordance with Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code. Therefore in RAI 4.2.2-1, Part 1, in order to demonstrate that the EMA in WCAP-13587, Revision 1, would still be bounding and in compliance with both 10 CFR 54.21(c)(1)(i) and Section IV.A.1 of 10 CFR Part 50, Appendix G, the staff requested that the applicant provide its inputs and results for the USE evaluations for all RNP beltline shell and nozzle materials and their associated weldments through the expiration of the extended period of operation for RNP. In RAI 4.2.2-1, Part 2, the staff requested confirmation that the EMA in WCAP-13587, Revision 1, has been submitted for review and approval by the staff.

The applicant provided its response to RAI 4.2.2-1, Parts 1 and 2, by letter dated May 15, 2003. In its response to RAI 4.2.2-1, Part 1, the applicant submitted nonproprietary Class 3 topical report WCAP-15828, Revision 0 (March 2003), which provides the updated USE assessments for RNP reactor vessel through both the current license period and extended period of operation for RNP. In its response to RAI 4.2.2-1, Part 2, the applicant stated that the assessment in WCAP-13587, Revision 1, provided a bounding EMA for Westinghouse Owners Group plants, and confirmed that the generic EMA in WCAP-13587, Revision 1, was reviewed and approved by the staff.

The RNP is a three-loop Westinghouse light-water reactor design. The NRC safety assessment of April 21, 1994, to the Nuclear Management and Resource Council (NUMARC, which is now the NEI) provides the staff's assessment of Westinghouse Electric Company's generic EMAs for two-loop, three-loop, and four-loop Westinghouse light-water reactor designs. In this safety assessment, the staff summarized the results of its independent elastic-plastic fracture mechanics evaluations (i.e., EMAs) for two-loop, three-loop, and four-loop Westinghouse light-water reactor designs. The staff concluded that three-loop Westinghouse light-water reactor designs will have acceptable safety margins against fracture (i.e., on USE) down to a minimum value of 42 ft-lb.

Appendix A of WCAP-15828 provides the applicant's USE analyses for the beltline plate, weld, and nozzle forging materials in the RNP reactor vessel through the expiration of the extended period of operation for the unit. The staff reviewed the USE data and information in Appendix A of WCAP-15828, Revision 0, as it relates to the USE assessment for RNP through the expiration of the extended period of operation for the unit (i.e., 60 years total of licensed life, 50 EFPYs). The staff also performed an independent assessment of the USE data in WCAP-15828, Revision 0, to assess the validity of the 50 EFPY USE calculations for the beltline plate, nozzle forging, and weld materials in the RNP reactor vessel.

The staff's independent calculation of the USE values for the RNP RV beltline materials through 50 EFPYs of operation confirmed that all of the materials will have a sufficient margin of safety against fracture equivalent to that required by Section XI of the ASME Code through the expiration of the extended period of operation for the unit. The staff applied the 50-EFPY neutron fluence values for the beltline materials at the 1/4T location of the vessel, as cited in WCAP-15828, Revision 0. The 1/4T fluences for the beltline materials at EOLE (i.e., through 50 EFPYs) are based on the latest capsule withdrawal from the RNP Reactor Vessel Material Surveillance Program (i.e., on test data from Capsule X, as reported in WCAP-15805 (March 2002)). For the RNP reactor vessel, the limiting beltline material for USE is upper-shell plate W10201-3 (Plate Heat No. B1255-1). The staff calculated the USE value for this material to be 48.6 ft-lb through 50 EFPY of operation. The corresponding USE value reported by the applicant in WCAP-15828, Revision 0, was 48.4 ft-lb, which is in good agreement with the value calculated by the staff. This value is higher than the minimum allowable value (42 ft-lb) cited in the April 21, 1994, safety assessment for three-loop Westinghouse plants and is therefore acceptable. Based on the information provided by the applicant in its responses to RAI 4.2.2-1, Parts 1 and 2, the staff concludes that the applicant has sufficiently addressed the information and data requested by the staff, and RAI 4.2.2-1, Parts 1 and 2, is resolved. The staff also concludes that, based on the 50-EFPY USE values for the RNP beltline materials, as calculated by both the applicant and the staff, the RNP RV beltline materials will have adequate ductility (i.e., sufficient levels of USE) through the expiration of the period of extended operation for RNP. Based on this assessment, the staff concludes that the applicant's TLAA for USE meets the safety margin requirements of 10 CFR Part 50, Appendix G, and the acceptance criterion stated in 10 CFR 54.21(c)(1)(ii) is acceptable.

4.2.2.3 Plant Heatup and Cooldown (Pressure/Temperature) Curves/Low-Temperature Overpressure Protection Power-Operated Relief Valves Setpoints

The P-T limits and LTOP limits for operating reactors are provided to protect the reactor vessels against fracture during transients that can significantly affect the pressure or temperature of the reactor. The P-T and LTOP limits are established by calculations that utilize the materials and fluence data obtained through the unit-specific Reactor Surveillance Capsule Program. Normally, the P-T limits are calculated for several years into the future and remain valid for an established period of time not to exceed the expiration date for the current operating license. For RNP, the current P-T limit curves are valid through 24 EFPYs.

The P-T limit curve requirements and LTOP limit requirements for RNP are currently included within the scope of the limiting conditions for operation for the plant. Pursuant to 10 CFR 50.90, the applicant is required to submit any proposed changes to the P-T limit requirements or LTOP limit requirements to the staff for review pursuant to the license amendment process of 10 CFR 50.90. The applicant used the licensing protocol to conclude that it does not consider the P-T and LTOP limits for RNP to be TLAAs. In RAI 4.2.2.3-1, the staff informed the applicant that, in all previous applications, the P-T limits and LTOP limits for operating light-water reactors have been identified as TLAAs that fall within the scope of 10 CFR 54.3(a). The staff asked the applicant to confirm whether the P-T limits and LTOP limits for RNP are within scope of the definition for TLAAs, as defined in 10 CFR 54.3(a).

The applicant responded to RAI 4.2.2.3-1 by letter dated May 15, 2003. In its response to RAI 4.2.2.3-1, the applicant indicated that it does not consider the P-T limit and LTOP limits for RNP to be TLAAs for the facility because the current curves, which have been approved through 24 EFPY, are not based on time-limited assumptions for the current operating period (40 years of licensed life, 29 EFPYs). Based on this discussion, the staff concludes that the P-T limits and LTOP limits do not fall within the scope of the definition of TLAAs, as given in 10 CFR 54.3(a), because the current P-T limits and LTOP limits are not based on the end of the licensed life for the facility. However, since the current P-T limits and LTOP limits for RNP are included within the scope of the limiting conditions for operations for RNP, the applicant is required to submit new P-T limits and LTOP limits for the facility for staff review and approval prior to expiration of the P-T limit curves and LTOP limits currently approved in the technical specifications. Pursuant to 10 CFR 54.35, this review process will carry over into the period of extended operation for RNP and ensures that the P-T limit curves and LTOP limits for the extended period of operation will be reviewed by the staff for approval, pursuant to the license amendment process. The staff's review of the P-T limit curves and LTOP limits for the period of extended operation, when submitted, will ensure that the operations of the RNP reactor will be done in a manner that ensures the integrity of the reactor coolant system (RCS) during the extended period of operation. Based on this assessment, the staff concludes that the P-T limits and LTOP limits for RNP do not have to be included within the scope of the TLAAs defined under 10 CFR 54.3(a), and RAI 4.2.2.3-1 is resolved.

4.2.3 UFSAR Supplement

Section 54.21(d) of Title 10 of the *Code of Federal Regulations* requires, in part, applicants to provide a summary description of TLAAs for the periods of extended operation for their facilities. Section A.3.2.1 of the LRA provides the applicant's UFSAR Supplement descriptions for the TLAAs for neutron irradiation embrittlement. The applicant provides its UFSAR Supplement descriptions for the TLAAs on PTS and USE in Sections A.3.2.1.1 and A.3.2.1.2 of the LRA, respectively. The staff reviewed the UFSAR Supplement descriptions for the TLAAs on PTS and USE, as given in Sections A.3.2.1.1 and A.3.2.1.2 of the LRA. In RAI 4.2.3-1, Part 1, the staff requested that the applicant amend the UFSAR supplement descriptions for PTS and USE to provide the technical bases why the TLAAs have been demonstrated to be in compliance with the requirements of 10 CFR 54.21(c)(1)(ii). The applicant provided its response to RAI 4.2.3-1, Part 1, by letter dated April 28, 2003. In this response, the applicant stated that the responses to RAIs 4.2.1-1 and 4.2.2-1, Part 1, describe how the TLAAs for PTS and USE are acceptable for the period of extended operation, respectively, and that the analyses for PTS and USE were identified as TLAAs and were described and evaluated in Section A.3.2.1 of the UFSAR Supplement. The applicant clarified that Section A.3.2.1 of the

UFSAR Supplement provides the technical basis for compliance with the requirements of 10 CFR 54.21(c)(1).

The applicant's responses to RAIs 4.2.1-1 and 4.2.2-1, Part 1, which reference WCAP-15828, Revision 0, provide the TLAAs for PTS and USE. In Section 4.2.2.1 of this SER, the staff concluded that the PTS assessment in WCAP-15828, Revision 0, was acceptable and demonstrates that the RV beltline materials would be in compliance with the PTS screening criteria of 10 CFR 50.61 through the expiration of the extended period of operation for RNP.

In Sections 4.2.2.1 and 4.2.2.2 of this SER, the staff concluded that the PTS and USE assessments in WCAP-15828, Revision 0, were acceptable and demonstrates that the RV beltline materials would be in compliance with the PTS screening criteria of 10 CFR 50.61 and the USE acceptance criteria of 10 CFR Part 50, Appendix G, through the expiration of the extended period of operation for RNP. However, the RT_{PTS} and USE values listed for the limiting PTS and USE materials in the RNP reactor vessel are not current with the limiting values for these materials listed in WCAP-15828, Revision 0. The staff requests confirmation that, at the next update of the UFSAR Supplement for RNP, the applicant will update Sections A.3.2.1 and A.3.2.2 of Appendix A to 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. This is Confirmatory Item 4.2.3-1.

In its response to Confirmatory Item 4.2.3-1 dated September 16, 2003, the applicant stated that it would amend the FSAR Supplement summary descriptions for the TLAAs on PTS and USE, as given in *Sections A.3.2.1 and A.3.2.2, respectively, to read as follows:*

> A.3.2.1 Reactor Vessel Neutron Embrittlement
>
> A.3.2.1.1 Pressurized Thermal Shock
>
> 10 CFR S5.61 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 ac
>
> 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 50ft-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).

The applicant's amended UFSAR Supplement summary descriptions for the TLAAs on PTS and USE (1) provide a sound basis as to why the TLAA for PTS and USE, as given in Sections A.3.2.1 and A.3.2.2 of the LRA, comply with the requirements in 10 CFR 50.61 for PTS and in 10 CFR Part 50, Appendix G, for USE through the expiration of the extended period of operation for RNP, and (2) provide a reference to the extended period of operation licensing basis documents containing the TLAAs for PTS and USE. Since the UFSAR Supplement summary descriptions demonstrate why the TLAAs are acceptable and reference the applicable licensing basis documents, the staff therefore concludes that the applicant's UFSAR Supplement summary descriptions for the TLAAs on PTS and USE, as given in Sections A.3.2.1 and A.3.2.2 of the LRA, and amended by the applicant's response to Confirmatory Item 4.2.3-1, are acceptable. Confirmatory Item 4.2.3-1 is resolved.

In Section 4.2.2.3, the staff assessed whether P-T limits and LTOP limits for RNP were within the scope of the staff's definition for TLAAs, as given in 10 CFR 54.3(a). In RAI 4.2.3-1, Part 2, the staff requested that the applicant provide its UFSAR Supplement description for the RNP P-T limits and LTOP limits. The staff's issuance of RAI 4.2.3-1 was based on the assumption that the P-T limits and LTOP limits for RNP would fall within the scope of the definition for TLAAs, as promulgated in 10 CFR 54.3(a). In its response to RAI 4.2.3-1, Part 2, the applicant stated that the Robinson LRA did not have to include a UFSAR Supplement summary description for the RNP P-T limits and LTOP limits because they are not within the scope of 10 CFR 54.3(a) for TLAAs. In Section 4.2.2.3 of this SER, the staff provided its basis for concluding that the P-T limits and LTOP limits for RNP were not considered to be within the scope of the staff's definition of TLAAs, as given in 10 CFR 54.3(a). Since the P-T limits and LTOP limits for RNP are not within the scope of the definition for TLAAs, as required in 10 CFR 54.3(a), the staff concludes that the LRA does not need to include a UFSAR Supplement summary description for the plant's P-T limits and LTOP limits, as would otherwise be mandated by the provisions of 10 CFR 54.21(d).

4.2.4 Conclusions

On the basis of its review, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that, for the RV neutron embrittlement TLAA, the analyses have been projected to the end of the period of extended operation. The staff also concludes that the UFSAR Supplement contains an appropriate summary description of the RV neutron embrittlement TLAA evaluation for the period of extended operation, as required by 10 CFR 54.21(d). Therefore, the staff concludes that the safety margins established and maintained during the current operating term will be maintained during the period of extended operation, as required by 10 CFR 54.21(c)(1).

4.3 Metal Fatigue

A metal component subjected to cyclic loading at loads less than the static design load may fail due to fatigue. Metal fatigue of components may have been evaluated based on an assumed number of transients or cycles for the current operating term. The validity of such metal fatigue analysis is reviewed for the period of extended operation.

4.3.1 Summary of Technical Information in the Application

The applicant discussed the explicit fatigue design requirements for RNP components in Section 4.3.1 of the LRA. Explicit fatigue analyses, in accordance with ASME Boiler and Pressure Vessel (B&PV) Code Section III, Class A (now Class 1) requirements, were performed during the design process for the Class 1 RCS primary system components. Components were subjected to all transients intended to envelop all foreseeable thermal and pressure cycles within a 40-year operating life. Originally, the methodology was applied to the RV, steam generators (SGs), RCPs, and pressurizer. Additional explicit fatigue analyses were performed to address new fatigue issues such as thermal stratification, insurge/outsurge flow in the pressurizer and surge line, RV internals, and thermal cycling of AFW to main feedwater connections.

The applicant tracks the number of design transients with its Fatigue Monitoring Program. The Fatigue Monitoring Program is discussed in Section B.3.19 of the LRA. The applicant indicated that, based on review of the frequency and severity of actual operating transients, it projects that the original 40-year transient set will remain bounding for 60 years of plant operation. Therefore, the applicant concluded that the fatigue analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Section 4.3.1.1 of the LRA describes the applicant's evaluation of the pressurizer surge line. The pressurizer surge line, originally designed to American National Standards Institute (ANSI) B31.1 rules, was reanalyzed by the explicit fatigue method to account for the impact of thermal stratification issues raised in NRC Bulletin 88-11. The hot-leg nozzle was identified as the limiting fatigue location. The applicant indicated that the number of design transients bounds the number of transients expected for 60 years of plant operation. Therefore, the applicant concluded that the surge line stratification analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Section 4.3.1.2 of the LRA describes the applicant's evaluation of pressurizer insurge and outsurge transients. Additional plant-specific analyses were performed to account for insurges and outsurges in the pressurizer and to account for actual plant operation. The plant-specific analyses were performed because the temperature monitoring data indicated that the temperature profile assumed in previous analyses did not bound the observed data. The plant-specific analyses found the limiting location in the pressurizer to be the surge line nozzle. The applicant indicated that the number of design transients bounds the number of transients expected for 60 years of plant operation. Therefore, the applicant concluded that the analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Section 4.3.1.3 of the LRA describes the applicant's evaluation of RV internals. Explicit fatigue analyses were presented in a Westinghouse topical report, WCAP-10322, Revision 1, October 1984, for the reactor internals holddown spring and alignment pins. Since WCAP-10322, Revision 1, has been incorporated by reference, the fatigue analyses for the reactor internals holddown spring and alignment pins were identified as TLAAs. The calculated cumulative utilization factors (CUFs) were 0.073 and 0.008 for the holddown spring and alignment pin, respectively. The applicant indicated that the number of design transients bounds the number of transients expected for 60 years of plant operation. Therefore, the applicant concluded that the analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Section 4.3.1.4 of the LRA describes the applicant's evaluation of the AFW line. The applicant reported a 1972 leakage, attributed to thermal fatigue cracking, at the 4"x16" connection between the auxiliary and main feedwater (AFW to FW) upstream of the B steam generator. The AFW connections were replaced with thermal-sleeved tees designed to ASME Code Section III, Subsection NB requirements (although this piping was designed originally using United States of America Standards (USAS) B31.1 Code). A fatigue analysis performed for the feedwater branch connection reinforcement plate resulted in an acceptable CUF value of less than 1.0 for the 40-year operating life and for the period of license renewal extended operation. The applicant indicated that assuming successful limitation of transient cycles for the 60-year operational period, the fatigue analyses will remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Section 4.3.2 of the LRA describes the applicant's evaluation of components with implicit fatigue design. The applicant stated that most RNP piping, including RCS piping, has been designed to USAS B31.1, "Power Piping Code." The code requires the application of reduction factors to allowable stresses to account for specified cyclic loadings. No explicit fatigue analyses were required. The applicant indicated that the 40-year design transient set has been demonstrated to be conservative for 60 years of operation for the RCS and, consequently, the number of thermal cycles imposed upon the RCS piping systems is not expected to exceed the original design assumptions. Therefore, the applicant concluded that the current design and licensing basis will be maintained throughout the license renewal period.

Auxiliary heat exchangers at RNP were designed in accordance with Westinghouse specifications and ASME Section III, Class C, or ASME Section VIII requirements. Each of the heat exchangers was designed for a specified number and magnitude of transients required by the specification complying with the rules of implicit fatigue design defined in the applicable codes, including ASME Section III, Class C, which are essentially identical to the B31.1 stress range reduction factors. The applicant indicated that any reductions in allowable stress needed for the components to safely withstand the specified thermal transients would have occurred during the original design of these heat exchangers in order to meet the code design requirements. The applicant indicated that the number of pressure and temperature cycles projected for the 60-year license renewal period does not exceed the number of pressure and temperature cycles originally specified and analyzed for 40 years. Therefore, the applicant concluded that the current designs for the specified heat exchangers, including fatigue considerations, remain valid for the 60-year license renewal period.

Section 4.3.3 of the LRA describes the applicant's evaluation of environmentally assisted fatigue (EAF). The applicant indicated that plant-specific environmental fatigue calculations were performed for the high-fatigue locations identified in NUREG/CR-6260 for older vintage Westinghouse plants. For RNP, four of these locations have ASME Section III explicit fatigue analyses, and the remaining three have USAS B31.1 implicit fatigue analyses. EAF relationships developed in NUREG/CR-6583 for carbon and low-alloy steels, and NUREG/CR-5704 for stainless steels, were used. The calculations use the environmental fatigue multiplier (F_{en}) approach. For the locations with an implicit fatigue evaluation, a comparison with the fatigue analyses in NUREG/CR-6260 was performed by comparing RNP plant-specific design attributes with those used in the NUREG/CR-6260 analyses. The F_{en} was computed for each case and was applied to the CUF values obtained from the NUREG/CR-6260 fatigue analysis. All EAF-adjusted CUFs were less than 1.0. For the locations with an ASME Section III fatigue analyses, EAF factors were calculated and applied to the CUFs from the fatigue analyses. The results showed that of the four locations, only the pressurizer surge line was not shown to have an EAF-adjusted CUF value below 1.0.

As part of the EAF-adjusted CUF analysis, the number of load/unload transients was reduced from 29,000 to 19,000 cycles. Since RNP does not operate in daily load-following mode, the number of load/unload transients experienced to date is less than 300, and the 60-year projection is approximately 600. The applicant indicated that a revision will be made to the RNP design transient set in the UFSAR prior to the license renewal period to limit these transients to a maximum of 19,000 cycles.

In addition to the locations specified in NUREG/CR-6260, the applicant performed environmental fatigue calculations for seven RNP pressurizer locations using 19,000 load/unload transients. The results of the analyses indicated that all locations have an EAF-adjusted CUF value of less than 1.0, except for the pressurizer surge nozzle safe end. Therefore, the applicant concluded that both the welds joining the surge line to the RCS hot leg and to the pressurizer surge nozzle are the limiting locations.

The applicant committed to manage the fatigue of surge line components by performing periodic volumetric examinations in accordance with ASME Section XI, Subsections IWB, IWC, and IWD. The frequency of these inspections, at least once every 10-year interval, is specified within the program documents. These inspections are considered adequate to detect the initiation of fatigue cracking prior to propagation into an unstable flaw. If unacceptable indications are identified, further evaluation, repair, or replacement will be performed as required by ASME Section XI. The applicant indicated that this program is adequate to manage thermal fatigue of the surge line and adjacent components during the license renewal period.

4.3.2 Staff Evaluation

4.3.2.1 Explicit Fatigue Analysis (ASME Section III, Class A)

The applicant performed explicit fatigue analyses, in accordance with ASME B&PV Code, Section III, Class 1, requirements, for the RCS primary system components subjected to transients intended to envelop foreseeable thermal and pressure cycles within a 40-year operating life. Originally, this methodology was applied to the RV, SGs, RCPs, and pressurizer. Additional explicit fatigue analyses were performed to address new fatigue issues such as thermal stratification, insurge/outsurge flow on pressurizer and surge lines, RV internals, and thermal cycling of AFW to main FW connections. The staff reviewed the applicant's evaluation of these components for compliance with the provisions of
10 CFR 54.21(c)(1).

The specific design criterion for fatigue analysis of RCS components involves calculating the CUF. The fatigue damage in the component caused by each transient depends on the magnitude of the resulting stresses. The CUF sums the fatigue damage resulting from each transient pair. The design criterion requires that the CUF not exceed 1.0. The applicant indicated that review of the RNP plant operating histories shows that the number of cycles and severity of the transients assumed in the design of these components envelop the expected transients during the period of extended operation.

The applicant used the terms "design transients," "postulated transients," and "selected transients" interchangeably in LRA Section 4.3.1. In RAI 4.3-1, the staff requested clarification as to the differences and specific designation of the category of transients that was used in the design of the RCS components. In its RAI response dated April 28, 2003, the applicant indicated that during the design process, thermal transient and postulated cycles that were used as the design basis for the 40-year life have been referred to as both "design transients" and "postulated transients" and these terms may be used interchangeably. "Selected transients" are those monitored directly in the Fatigue Monitoring Program, and represent design cycles that bound the actual cycles anticipated during the period of extended plant operation. The staff finds the applicant's clarification acceptable.

Section 4.3.1 of the LRA also discusses the adjustments to "cumulative cycle counts." While partial cycle of design transients is defined and used in the ASME B&PV Code, Section III (the Code), the staff requested that the applicant provide additional clarification of this procedure. In RAI 4.3-2, Part 1, the staff requested that the applicant provide the number of design cycles, current operating cycles, and a description of the transients, and for partial cycle transients, the method used to determine the fraction of a full cycle. In its response dated April 28, 2003, the applicant identified the applicable design codes for RNP components and transient descriptions with design and operating cycles in two tables, including applicable notes. For partial cycle transients, the methodology provided in Section 102.3.2 of USAS B31.1, "Power Piping Code," 1967 edition, was used to determine the fraction of a full cycle. The heatup transient was presented as an example to demonstrate how the equivalent full-temperature range cycles were calculated. The staff finds this method acceptable.

In RAI 4.3-2, Part 2, the staff requested that the applicant provide the number of full-range operating cycles estimated for past operation, the method used to estimate the number of cycles for the remaining and extended life, and the basis of developing assumed cycle data on past and present operations. In its response dated April 28, 2003, the applicant stated that, for transients except plant heatups, cooldowns, and reactor trips, cycles are conservatively extrapolated to 60 years based on the actual average number of transients per year to date (through April 2003). For heatup, cooldown, and trip transients, the extrapolation was based on "learning curve effects" and system shakedowns which occurred early in plant life. For these transients, the rate of accumulation was very high during the first 20 years of plant life (3.8 per year for plant heatups and cooldowns and 9.1 per year for reactor trips) but has diminished dramatically down to 1.1 transients per year for each transient in the last 10 years. This reduced rate of accumulation is believed to represent the best estimate of future operation. The staff finds the applicant's method of transient extrapolation for the remaining and extended life reasonable and conservative, and, therefore, acceptable.

In RAI 4.3-2, Part 3 , the staff requested that the applicant describe the proposed mechanism to adjust and track transients included in the LRA for the remaining and extended life of the plant if operational procedures for future operation are modified. The applicant responded by letters dated April 28 and June 13, 2003, that if operating procedures are changed to the extent that the associated fatigue usage could increase beyond that of the most recent fatigue analysis, the affected fatigue analyses would be revised to account for the more severe thermal transients. If the number of allowable cycles to maintain CUF less than 1.0 remains unchanged, then no change would be required to the Fatigue Monitoring Program limits. If the number of allowable cycles had to be reduced to obtain a CUF value less than 1.0, this reduced number of cycles would become the new Fatigue Monitoring Program cycle limit. The reduction of load/unload transient limit from 29,000 to 19,000 cycles to qualify the pressurizer spray nozzle safe end CUF was used as an example of this process applied to the environmental fatigue calculations performed for license renewal. The staff finds the description of transient adjustment and tracking to keep the Fatigue Monitoring Program allowable cycle limits, using the pressurizer spray nozzle as an example, reasonable and acceptable.

In RAI 4.3-2, Part 4, the staff requested that the applicant provide a quantitative comparison of the cycles and severity of the design transients listed in the LRA with the transients monitored by the Fatigue Monitoring Program described in Section B.3.19 of the LRA and identification of any transients listed in the LRA that are not monitored by the Fatigue Monitoring Program and an explanation of why it is not necessary to monitor these transients. In its RAI response dated April 28, 2003, the applicant stated that the transients that are counted are those most severe and likely to result in fatigue cracking of one or more components. Those that are less likely to result in fatigue, due to low contribution to fatigue usage, would not be useful fatigue indicators and need not be counted. They are denoted by "N/C" in the transient description table attached to the response to RAI 4.3-2. For a given component, the influence of any particular transient on the CUF and the magnitude of total CUF determine whether or not that particular event should be counted and tracked. Based on these factors, a review was performed to identify the design cycles from those in the table that have a significant impact on the component fatigue analyses for RNP. First, component locations with individual CUF values of 0.1 or more were identified. Then, the individual transients that contribute to 50 percent or more of the fatigue usage for these locations with a CUF value of 0.1 or more were identified. These are required to be tracked. The loss of load transient and partial loss of flow transient had not been included in the Fatigue Monitoring Program prior to the evaluation but were added to the program because they meet the criteria specified above. Records were reviewed to determine past occurrences, and the counts were updated as required to assure that they are not approaching their design limits. Using these methods, RNP was able to demonstrate that the original 40-year transient set is conservative and bounding for the 60-year operation of the plant. The staff finds the described method of transient monitoring reasonable and acceptable.

4.3.2.1.1 Pressurizer Surge Line Thermal Stratification

The applicant indicated that plant-specific analyses were performed for pressurizer surge line stratification because the temperature monitoring data indicated that the temperature profile assumed in the Westinghouse generic analyses did not bound the observed plant-specific data. In RAI 4.3-3, the staff requested that the applicant (1) provide data or references to justify that the number of transients projected for 60 years of operation is significantly less than that of transients originally postulated for 40 years, (2) justify the projected RNP transient cycles in view of past and future heatup and cooldown methods, and (3) discuss how the TLAA reanalysis will be performed, if the operations during the extended period are different from those assumed in the design assumptions.

The responses to requests 1 and 2 are detailed in the replies to RAI 4.3-2, Part 2, and RAI 4.3-4, respectively. The applicant's response to RAI 4.3.2-2 was discussed in the previous section of the SER. The applicant's response to RAI 4.3-4 is discussed in the next section. Previous transients that exceeded the specified pressurizer heatup and cooldown limits were evaluated, along with several extra cycles to allow for any unanticipated future transients above these limits. RNP has modified the methods for plant heatup and cooldown to mitigate the pressurizer insurge/outsurge transients, and to assure that the existing heatup rate limit of 100 °F/hr and cooldown rate limit of 200 °F/hr are maintained as required by the technical specifications. The method for performing plant heatup and cooldown during the extended operating period will continue to conform to the specified pressurizer heatup and cooldown limits. If a change in operational method were contemplated that might result in exceeding the specified heatup or cooldown rates, the fatigue analyses for the pressurizer and surge line would be evaluated and, if necessary, revised to account for the increased fatigue usage. However, no such change is anticipated. The staff finds the responses provided to RAIs 4.3-2, 4.3-3, and 4.3-4 adequately address transient cycles for 60-year operation and are acceptable.

4.3.2.1.2 Pressurizer Insurge/Outsurge

Pressurizer cooldown limits may be exceeded if a significant temperature difference exists between the pressurizer and the RCS hot leg. The applicant indicated that the cooldown limit had been exceeded in February 1994 and that a detailed evaluation of the transient was performed. RAI 4.3-4 requested the applicant to provide this information and the RNP-specific temperature difference limit during heatup and cooldown.

In its response, the applicant identified technical specification limits of 100 °F/hr for heatups and 200 °F/hr for cooldowns. If a transient exceeds these limits, actions must be taken to evaluate and determine the effects of the out-of-limit condition on the structural integrity of components. The detailed evaluation of the February 1994 out-of-limit transient also included previous occurrences of transients exceeding the technical specifications limits identified through review of plant operating history. The evaluation included identification of past out-of-limit pressurizer transients, development of enveloping transients, determination of stresses in critical locations, and evaluation of these stresses on the structural integrity of the pressurizer. Pressurizer structural integrity was evaluated with respect to nonductile fracture and fatigue requirements. Fracture analysis showed stress intensity factors calculated for a range of assumed flaw depths to remain below the material fracture toughness. The ASME Code fatigue analysis showed that the increase in fatigue usage from these transient events was small.

The analysis of the February 1994 pressurizer out-of-limit transient included other past out-of-limit transients, totaled 16 cooldown and 8 heatup excursions, and included two new enveloping models that were used to bound the fatigue usage. The analysis conservatively calculated the fatigue usage that would result from 40 occurrences of each of the two new transients. The pressurizer surge line was instrumented for one operating cycle to validate the assumptions used in the analysis and to provide detailed transient data for a more accurate analysis. These data determined that moment ranges were larger than previously analyzed. The measured data were used in a structural reanalysis and revised fatigue analysis. The limiting location at the RCS hot-leg nozzle was determined to have a CUF value of 0.96.

In its response to RAI 4.3-6, the applicant confirmed that none of the pressurizer components which have an explicit fatigue analysis has a 40-year or 60-year CUF value that exceeds 1.0

without consideration of environmental effects. Analyzed components include the pressurizer lower head, heater well, spray nozzle, spray nozzle safe end, surge nozzle, surge nozzle safe end, and instrument nozzles. On the basis of the applicant' s responses to the RAIs, the staff finds that the applicant has adequately addressed insurge/outsurge transients.

When environmental fatigue effects were considered, the only component in the pressurizer that was determined to have an EAF-adjusted fatigue value that exceeds 1.0 is the pressurizer surge nozzle safe end (stainless steel) weld to the pressurizer surge line. Fatigue of this component will be managed in the same manner as the adjacent stainless steel pressurizer surge line components, including the surge line piping and RCS hot-leg nozzle. Section 4.3.2.3 of this SER discusses the management of fatigue for the surge line components with EAF-adjusted CUF values over 1.0.

4.3.2.1.3 Reactor Internals Holddown Spring and Alignment Pins

The applicant reported in Section 4.3.1.3 of the LRA that explicit fatigue analyses for the reactor internals holddown spring and alignment pins were presented in a Westinghouse report. The calculated CUFs were 0.073 and 0.008 for the holddown spring and alignment pin, respectively. The Westinghouse report is the stress report on 312 standard reactor core structures. In RAI 4.3-5, the staff requested that the applicant provide justification of the direct applicability of this stress report to the RNP reactor internals holddown spring and alignment pins.

In its April 28, 2003, response, the applicant confirmed that the Westinghouse report is not directly applicable to RNP. The RNP performed an engineering evaluation of materials used for replacement control rod guide tube support pins. This evaluation included references to two Westinghouse documents, which in turn referenced the Westinghouse report in question. Direct reference to the fatigue evaluation in the Westinghouse report was not part of the engineering evaluation, and RNP was not required to establish a TLAA for the RV internals. However, RNP conservatively incorporated the indirect reference to the fatigue evaluation for these components as being within the scope of license renewal. The staff finds the applicant's clarification acceptable. The applicant has also indicated that the number of transients assumed for 40-year design life bounds the number expected for 60 years of operation. On the basis that the number of design transients bounds the number expected for 60 years of plant operation, the staff finds that fatigue of the reactor internals holddown spring and alignment pins has been adequately evaluated for the period of extended operation.

4.3.2.1.4 Auxiliary Feedwater Line Fatigue Analysis

The applicant reported a 1972 leakage, attributed to thermal fatigue cracking, at the 4"x16" connection between the AFW and main FW lines upstream of the B steam generator. Although the piping was originally designed to USAS B31.1 Code, the AFW to main FW connections were replaced with thermal-sleeved tees designed to ASME Code Section III, Subsection NB, requirements. A fatigue analysis, considered to be a TLAA, was performed for the branch connection reinforcement plate. The RNP reported a CUF value of less than 1.0 for the 40-year life and for the period of extended 60-year operation. These connections are considered as nonstandard (ASME) components for which stress intensification factors may not be defined. In RAI 4.3-7, the staff requested the applicant to provide (1) calculated CUF of the six replacement branch connections, (2) confirmation that no other nonstandard components were used or justification of the acceptability for use in safety systems at RNP, and (3) description of the aging management programs (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 letter dated June 13, 2003 (RAI 4.3-7), the applicant stated that there are three 4" to 16" AFW to main FW connections downstream of the motor-driven and the steam-driven AFW pumps. These connections were designed in accordance with USAS B31.1 requirements. Due to detected leakage, the three connections downstream from the motor-driven pumps were replaced with a better design employing a thermal sleeve, also designed to B31.1 requirements.

The three connections downstream from the steam-driven pumps, two of the pad plate reinforcing plate design and one with the saddle reinforcing plate design, were not replaced. In the early 1990s, more rigorous fatigue analyses were performed for each of these two configurations using methodology from ASME Section III, Class 1, rules. The analyses showed that the saddle plate design was inferior to the pad plate design, and a modification was performed to replace the saddle reinforcement plate with a pad-type reinforcing plate. In conjunction with that modification, an ASME Section III fatigue analysis was performed for the pad plate design for the three connections, and this analysis was determined to be a TLAA for license renewal. However, during the license renewal review of this fatigue analysis, an error was discovered in the analysis, and the analysis was revised in 2002 to correct the error. The three connections downstream from the steam-driven pumps 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. This was identified as Confirmatory Item 4.3.2-1.

By letter dated September 16, 2003, the applicant provided a modification to UFSAR Supplement Section A.3.2.2.1 which includes the proposed options to address fatigue of the connections between the auxiliary and main feedwater lines for the period of extended operation. The staff finds the modification to UFSAR Supplement Section A.3.2.2.1 acceptable. Confirmatory Item 4.3.2-1 is closed.

In response to Part 3 of the RAI, the applicant performed reviews during the RNP integrated plant assessment (IPA) and found no nonstandard components used in safety systems, based on USAS B31.1 as the design code. This includes each type of AFW/FW connection. ASME Code, Section III, is not the applicable design code, even though portions of it were used as a basis for preparing the fatigue analyses.

Based on the above review of the LRA and the applicant's responses to the RAI provided in the June 13, 2003, letter, the staff finds that the applicant has provided adequate justification to assure the proper fatigue management of the FW/AFW connections for the extended period of operation.

4.3.2.2 Implicit Fatigue Design (ASME Section III, Class C, ANSI B31.1)

ANSI B31.1 requires that a reduction factor be applied to the allowable bending stress range if the number of full range thermal cycles exceeds 7000. The applicant indicated that the number of design transient cycles was found to bound the number of transient cycles expected for 60 years of plant operation. Therefore, the applicant concluded that the analyses of these piping components remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

In RAI 4.3-8, the staff requested that the applicant provide justification of that expectation and assumption that the USAS B31.1 limit of 7000 equivalent full range cycles will not be exceeded during the period of extended operation for the B31.1 piping systems.

In its April 28, 2003, response, the applicant indicated that the 60-year transient projection results apply to both the explicit Class A fatigue analyses and the implicit Class C (and USAS B31.1) analyses. Fatigue Monitoring Program transient data were evaluated to show that the number of transients expected in 60 years is less than the number postulated for 40 years in the original design. In its June 13, 2003, response, the applicant indicated that the primary sampling piping is no longer used for sampling and was not accumulating additional thermal cycles. Therefore, the applicant concluded that the number of thermal cycles for the primary sampling system would not exceed the USAS B31.1 limits during the period of extended operation. The staff finds the applicant's assessment reasonable and acceptable.

The applicant indicated that auxiliary heat exchangers at RNP were designed in accordance with Westinghouse specifications and ASME Section III, Class C, or ASME Section VIII requirements for a specified set of transients required by the specification complying with the rules of implicit fatigue design method defined in the design code using the stress reduction factors described above. The applicant concluded that no further reductions are needed because, as described previously, the number of pressure and temperature cycles projected for the 60-year license renewal period does not exceed the number of cycles originally specified and analyzed for the 40-year life. Therefore, the current designs for the specified heat exchangers, including fatigue considerations, remain valid. In RAI 4.3-8, the staff also requested that the applicant provide the fatigue design method for this case.

The applicant's April 28, 2003, response indicated that there is no requirement to reduce the allowable stress based on cyclic loadings. ASME Section VIII requires that loads not induce a combined maximum primary membrane plus primary bending stress across the thickness exceeding 1.5 times the maximum allowable stress. It is recognized that high localized discontinuity stresses may exist in accordance with these rules. Insofar as practical, design rules have been written to limit such stresses to a safe level consistent with experience. The staff finds this is consistent with the Code and, therefore, acceptable.

4.3.2.3 Environmentally Assisted Fatigue Evaluation

Generic Safety Issue (GSI)-166, "Adequacy of the Fatigue Life of Metal Components," raised concerns regarding the conservatism of the fatigue curves used in the design of the RCS components. Although GSI-166 was resolved for the current 40-year design life of operating components, the staff identified GSI-190, "Fatigue Evaluation of Metal Components for 60-year Plant Life," to address license renewal. The NRC closed GSI-190 in December 1999 with the following conclusions:

> The results of the probabilistic analyses, along with the sensitivity studies performed, the iterations with industry (NEI and EPRI), and the different approaches available to the licensees to manage the effects of aging, lead to the conclusion that no generic regulatory action is required, and that GSI-190 is closed. This conclusion is based primarily on the negligible calculated increases in core damage frequency in going from 40- to 60-year lives. However, the calculations supporting resolution of this issue, which included consideration of environmental effects, and the nature of age-related degradation indicate the potential for an increase in the frequency of pipe leaks as plants continue to operate. Thus, the staff concludes that, consistent with existing requirements in 10 CFR 54.21, licensees should address the effects of coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.

The LRA indicates that the EAF relationship developed later in NUREG/CR-6583 and NUREG/CR-5704 was used in the calculation of the environmental fatigue multiplier (F_{en}). The LRA indicated that the EAF usage factors were less than 1.0 except for the pressurizer surge line. In RAI 4.3-9, the staff requested that the applicant provide the results of the F_{en} and EAF-adjusted CUF calculation for each of the seven component locations listed in NUREG/CR-6260.

The applicant's April 28, 2003, response provided a table which included the F_{en} values and the EAF-adjusted CUFs for the seven component locations listed in NUREG/CR-6260 that are applicable to an older vintage Westinghouse plant. The staff compared the results presented by the applicant with the results presented in NUREG/CR-6260. On the basis of this comparison, the staff finds the applicant's evaluations are reasonable.

The applicant indicated that the EAF-adjusted usage factor for the surge line would exceed 1.0 during the period of extended operation. The applicant further indicated that it would use an AMP to address surge line fatigue during the period of extended operation. The AMP would rely on ASME Section XI inspections. The staff has not endorsed a procedure on a generic basis which allows for ASME Section XI inspections in lieu of meeting the fatigue usage criteria. 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 of the following options:

- further refinement of the fatigue analyses to maintain the EAF-adjusted CUF below 1.0
- repair of the affected locations
- replacement of the affected locations
- management of the effects of fatigue through the use of an augmented inservice inspection program that has been 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). The staff identified revision of the UFSAR Supplement as Confirmatory Item 4.3.2-2.

By letter dated September 16, 2003, the applicant provided a modification to UFSAR Supplement Section A.3.2.2.1 which includes the proposed options to address fatigue of the surge line for the period of extended operation. The staff finds the modification to UFSAR Supplement Section A.3.2.2.1 acceptable. Confirmatory Item 4.3.2-2 is closed.

4.3.3 Conclusions

On the basis of its review, the staff concludes that the applicant has provided an acceptable demonstration, that, for the metal fatigue TLAA, the effects of aging on the intended functions will be adequately managed for the period of extended operation. Therefore, the staff has concluded that the safety margins established and maintained during the current operating term will be maintained during the period of extended operation, as required by 10 CFR 54.21(c)(1).

4.3.4 Reactor Vessel Underclad Cracking

In Section 4.3.4 of the LRA, the applicant provides the TLAA for assuring that postulated underclad cracks in the RNP RV would remain acceptable for service through the expiration of the extended period of operation for RNP, as evaluated in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

4.3.4.1 Summary of Technical Information in the Application

In the TLAA evaluation of RV underclad cracks, the applicant considers the effect that additional operation cycles during the period of extended operation would have on postulated underclad cracks in the RNP RV. The applicant cites as a reference a fracture mechanics analysis that was completed in 1971 and which concluded that fatigue growth of potential underclad flaws in RV base metal was insignificant over a 40-year operating life.

The applicant states that the underclad cracking analysis has been updated by a Westinghouse topical report, WCAP-15338, which is applicable to the evaluation of underclad cracks in the RNP RV through the end of the extended period of operation. The applicant states that this report has been approved by the staff in a generic safety evaluation for the Westinghouse Electric Company and that this report demonstrates that postulated underclad cracks in the RNP RV will be acceptable through the expiration of the extended period of operation.

4.3.4.2 Staff Evaluation

WCAP-15338 provides Westinghouse Electric's generic evaluation for underclad cracks in Westinghouse-designed RVs. In order to justify operation of Westinghouse-designed light-water reactors through 60 years of operation, the report evaluates the effect of additional operating cycles during the period of extended operation on fatigue-induced growth of detected underclad cracks in the RVs. The report evaluates the effects that the additional operational cycles would have on a bounding 0.295-inch semi-elliptical surface flaw, which is assumed to grow under the influence of transient cycles for a period of 60 years. In a safety evaluation (SE) dated July 15, 2002, the staff concluded that the flaw depths for detected RV underclad cracks, as evaluated in WCAP-15338, would be acceptable for service without repair over 60 years of licensed operation for two-loop, three-loop, and four-loop Westinghouse-designed light-water reactors. In the SE of July 15, 2002, the staff states that applicants for license renewal may reference that WCAP-15338 satisfies the TLAA requirement of 10 CFR 54.21(c)(1), as it relates to the demonstration that RV underclad cracks are acceptable for service over 60 years of operating life for a licensed Westinghouse-design PWR. However, in order to take credit for the evaluation in WCAP-15338, the staff informed applicants for license renewal that they would need to complete the following two action items:

(1) The applicant is to verify that its plant is bounded by the WCAP-15338 report. Specifically, the renewal applicant is to indicate whether the number of design cycles and transients assumed in the WCAP-15338 analysis bounds the number of cycles for 60 years of operation of the applicant's RV.

(2) To satisfy the requirements of 10 CFR 54.21(d), the renewal applicant referencing WCAP-15338 would need to ensure that the UFSAR description for the TLAA appropriately summarizes the TLAA for RV underclad cracks, including a reference to WCAP-15338 as being bounding and applicable to the evaluation of RV underclad cracks at the applicant's Westinghouse-design light-water reactor facility.

In Section 4.3.4 of the LRA, the applicant indicated that it has verified that WCAP-15338 is applicable to the evaluation of RV underclad cracks at RNP. The applicant also indicated that it has verified that (1) the number of design cycles and transients assumed in the WCAP-15338 analysis bounds the number of cycles for 60 years of operation of the RNP RV, and (2) a summary description of the WCAP-15338 analysis has been included in the RNP UFSAR Supplement. The applicant's TLAA for the RNP RV underclad cracks has been performed in accordance with the staff's evaluation and action items on WCAP-15388, which provided the criteria for ensuring that underclad cracks will be adequately managed to meet the requirements of 10 CFR 54.21(c)(1)(i). The staff therefore concludes that the applicant's TLAA for RV underclad cracking is acceptable.

4.3.4.3 Updated Final Safety Analysis Report Supplement

The applicant provides its UFSAR Supplement description for the TLAA on RV underclad cracking in Section A.3.2.2.3 of the LRA. The staff has reviewed the UFSAR Supplement description for the TLAA on RV underclad cracking and has confirmed that the applicant has provided a sufficient summary of this TLAA in Section A.3.2.2.3 of the LRA. The staff confirmed that the applicant appropriately referenced WCAP-15338 as being applicable to the evaluation of underclad cracks at RNP and that the flaw evaluation for RV underclad cracks in WCAP-15338 bounds the evaluation of underclad cracks at RNP. The staff therefore concludes that the UFSAR Supplement description for the applicant's TLAA on RV underclad cracking is acceptable.

4.3.4.4 Conclusions

On the basis of its review, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the analysis for the RV underclad cracking remains valid until the end of the period of extended operation. The staff also concludes that the UFSAR Supplement contains an appropriate summary description of the TLAA for RV underclad cracking for the period of extended operation, as required by 10 CFR 54.21(d). Therefore, the staff has concluded that the safety margins established and maintained during the current operating term will be maintained throughout the period of extended operation as required by 10 CFR 54.21(c)(1).

4.3.5 Containment Penetration Bellows Fatigue

4.3.5.1 Summary of Technical Information in the Application

The applicant stated that the fatigue of containment components was reviewed to identify potential TLAAs. Fatigue TLAAs were identified for three replacement bellows assemblies used for hot piping penetrations. The fatigue analysis of the three replacement bellows shows that they are designed to withstand 4000 cycles without cracking. The applicant also stated that the original bellows do not have analyses that fit the definitions of TLAAs.

The significant thermal transients that result in flexure of the hot pipe penetration bellows are those that involve a full-range temperature change in the piping system. This includes the plant heatup and cool downcycles. The original 40-year design basis of the plant specifies 200 heatup and cooldown cycles. The applicant indicated, in Section 4.3.1 of the LRA, that the 40-year transient counts remain conservative for 60 years of operation.

The applicant stated that the number of cycles for which the three containment bellows were qualified in the fatigue calculations exceeds the 200 heatup/cooldown cycles applicable to 60 years of operation. These calculations therefore remain valid for the period of extended operation. The applicant concludes that the analyses associated with containment bellows fatigue remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.3.5.2 Staff Evaluation

In RAI 4.3-11, the staff requested that the applicant identify the design code to which the containment penetrations are designed and provide a description of the methodology on which the fatigue analysis of the hot penetrations is based. The applicant was also asked to support its conclusion that the bellows can withstand 4000 cycles of operation without fatigue cracking. In response, the applicant stated that the fatigue evaluation of the hot penetrations is limited to the bellows only. According to the design specifications for the bellows, they are designed in accordance with ASME Code Section III, Subsection NC, and bellows performance equations as listed in Section C of the "Standards of the Expansion Joint Manufacturers Association," 5th Edition, 1980, including the 1985 Addenda.

The other components of the containment penetrations at RNP are described in Section 3.8.1.1.6 of the UFSAR. The applicable codes and standards for the design of hot containment penetrations are described in Section 3.8.1.2. This section states that penetrations conform to the applicable sections of USAS N6.2-1965, "Safety Standard for the Design, Fabrication, and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors."

In RAI 4.3-12, the staff inquired if the containment penetration bellows are included within the scope of the RNP Fatigue Monitoring Program. The applicant stated that at RNP, the plant heatup and cooldown transients that involve full-temperature changes in the piping systems are controlled and monitored by the RNP Fatigue Monitoring Program. The UFSAR limits these to 200 heatup and cooldown cycles, based on the 40-year design basis of the plant. These are also the cycles that contribute to the fatigue of the containment penetration bellows. The containment penetration bellows are therefore implicitly included within the scope of the RNP Fatigue Monitoring Program. For license renewal, the number of heatup and cooldown cycles to date were analyzed and projected to 60-year plant operation. The projection demonstrated that the present limit of 200 heatup and cooldown cycles is conservative for 60-year operation. Since the bellows were analyzed for 4000 cycles, the bellows will not exceed their design limits during the period of extended operation. The staff finds the applicant's evaluation acceptable.

4.3.5.3 Conclusions

On the basis of its review, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1), that, for the hot containment penetrations bellows fatigue TLAAs, the analyses remain valid for the period of extended operation. The staff also concludes that Section A.3.2.2.4 of the UFSAR Supplement contains an appropriate summary description of the containment penetrations bellows fatigue TLAA evaluation for the period of extended operation as required by 10 CFR 54.21(d). Therefore the staff has concluded that in accordance with current industry practice, the safety margins established and maintained during the current operating term will be maintained during the period of extended operation, as required by 10 CFR 54.21(c)(1).

4.3.6 Crane Cycle Load Limits

4.3.6.1 Summary of Technical Information in The Application

The applicant states that the load cycle limits for cranes were identified as a potential TLAA and that two following RNP cranes in the scope of license renewal have a TLAA, which requires evaluation for 60 years. These two cranes are the containment polar crane and the spent fuel cask crane.

<u>Containment Polar Crane</u>

The applicant states that the RNP containment polar crane was designed in accordance with "Electric Overhead Crane Institute (EOCI) Specification for Electric Overhead Traveling Cranes," 1961 (EOCI-61), and American Institute of Steel Construction (AISC), "Manual of Steel Construction," 6th Edition. According to the applicant, EOCI-61 did not require a reduction in allowable stresses for fatigue. However, the AISC 6th Edition permitted up to 10,000 complete stress reversals at maximum stress to occur for the life of the structure.

The applicant has provided an analysis to project the current RNP containment polar crane fatigue analysis for 60 years of plant operation. This analysis is summarized below:

> The total number of lift cycles for the Containment Polar Crane is directly dependent on the number of Refueling Outages. The total number of Refueling Outages for 60 years of operation has been established as 40. The total number of upper and mid-range lifts is 110 per outage for a

> total of 40 outages, which equates to a 60-year projection of 4,400 lift cycles. This is less than the 10,000 permissible lift cycles and is therefore acceptable.

Spent Fuel Cask Crane

The applicant has provided a similar assessment to demonstrate that the current RNP spent fuel cask crane fatigue analysis is valid for 60 years of plant operation. This analysis is summarized below:

> The number of lift cycles originally projected for 40 years was 2,500. This can be multiplied by a factor of 1.5 to determine the number of cycles for 60-year life. Therefore, number of load cycles projected for 60 years is 3,750. This is less than the 20,000 permissible cycles and is therefore acceptable.

Based on the above information, the applicant concludes that the analyses associated with fatigue of the containment polar crane and the spent fuel cask crane have been projected to the end of the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

4.3.6.2 Staff Evaluation

The method of review applicable to the crane cyclic load limit TLAA involves (1) reviewing the existing 40-year design basis to determine the number of load cycles considered in the design of each of the cranes in the scope of license renewal and (2) developing 60-year projections for load cycles for each of the cranes in the scope of license renewal and comparing them with the number of design cycles for 40 years.

Section 4.3.6 of the LRA states that the basic allowable stress calculation of the spent fuel cask crane includes dead weight, live load, and impact allowance. In RAI 4.3-13, the staff requested the applicant to discuss the specific requirements on which the impact allowance was based and indicate its magnitude. In its response dated April 28, 2003, and additional clarification provided during a meeting on May 20, 2003, the applicant made the following statement:

> The spent fuel cask handling crane underwent a load rating capacity upgrade during the 1974/75 time frame. The structural upgrade was performed in accordance with CMAA-70. The CMAA-70 specific requirement for impact allowance of the rated capacity is taken as 1/2% of the load per foot per minute of hoisting speed, but not less than 15%, nor more than 50%, of rated load. The spent fuel cask handling crane support structure modifications utilized an impact allowable of 15% of the lift load.

The staff finds the applicant's response reasonable and acceptable because it clarifies the specific requirements on which the impact allowance is based and it meets the Crane Manufacturers Association of America (CMAA)-70 requirements.

Section 4.3.6 of the LRA states that the spent fuel crane is designed for 20,000 to 100,000 load cycles. In RAI 4.3-14, the staff requested the applicant to provide the basis for the upper and lower limits. In its response dated April 28, 2003, the applicant stated the following:

> The load cycle design requirement for the RNP spent fuel crane was based on less than 2500 load cycles over a 40-year period. This equates to a design requirement of less than 3750 load cycles for the 60-year license renewal period. The CMAA-70 crane classification for the RNP spent fuel

> crane is Class A1. Due to its low usage, the spent fuel crane was designed for the lowest range of cycles (20,000 to 100,000).

The applicant further stated that "Class A1 cranes, which are standby Class A cranes, are used for standby service, with infrequent maintenance and long idle periods, i.e., 'low usage.' Additionally, crane specification CMAA-70 code provides an allowable stress range for structural design dependant on its usage (i.e., number of loading cycles)." Based on the above discussion, the staff finds that the applicant has provided an adequate explanation for the upper and lower limits of the load cycles used in the spent fuel crane design.

The applicant also contends that a review of the operational history of the RNP spent fuel crane indicates that the original design requirement was conservative and will not be exceeded for the 40-year period. Therefore, by extrapolation, the requirement for the 60-year period will not be exceeded. The staff concurs with this assessment.

The minimum factor of safety for the spent fuel crane, as discussed in Section 4.3.6 of the LRA, is based on a maximum tensile strength of 58,000 psi for American Society for Testing and Materials (ASTM)-A36 material. In RAI 4.3-15, the staff asked the applicant to verify that no members of the crane have a lower tensile strength and also identify the members with the minimum factors of safety.

In its response dated April 28, 2003, the applicant stated the following:

> The structural load-bearing members for the RNP spent fuel crane have been fabricated in accordance with CMAA-70 from ASTM A-36 steel (tensile strength of 58,000 psi). A minimum factor of safety was provided for structural load bearing members based on a maximum allowable stress. The maximum basic allowable stress for any member under tension or compression is 17,600 psi. The 17,600 psi allowable is the not to be exceeded allowable stress as stated in the CMAA-70 crane specification for members subjected to repeated loading. The factor of safety reported in the LRA was given based on the tensile strength for ASTM A-36.

Based on its review of the applicant's response, as discussed above, the staff finds that the applicant has satisfactorily addressed the concerns related to the minimum factor of safety.

4.3.6.3 Conclusions

On the basis of its review, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that, for the crane cycle load limit TLAA, the analyses have been projected to the end of the period of extended operation. The staff also concludes that the UFSAR Supplement contains an appropriate summary description of the crane cycle limit TLAA evaluation for the period of extended operation, as reflected in the license condition as required by 10 CFR 54.21(d). Therefore, the staff has concluded that the safety margins established and maintained during the current operating term will be maintained during the period of extended operation, as required by 10 CFR 54.21(c)(1)(ii).

4.4 Environmental Qualification of Electrical Equipment

The 10 CFR 50.49 Environmental Qualification Program has been identified as a TLAA for the purposes of license renewal. The TLAA of environmental qualification (EQ) components includes all long-lived, passive and active electrical and I&C components and commodities that are located in a harsh environment and are important to safety, including safety-related and Q-list equipment, non-safety-related equipment whose failure could prevent satisfactory accomplishment of any safety-related function, and the necessary post-accident monitoring equipment.

The staff has reviewed Section 4.4, "Environmental Qualification," of the RNP LRA to determine whether the applicant submitted information adequate to meet the requirements of 10 CFR 54.21(c)(1) for evaluating the EQ TLAA. The staff also reviewed Section 4.4.2, "GSI-168, Environmental Qualification of Electrical Components," of the LRA.

On the basis of this review, the staff requested additional information in a letter to the applicant dated February 11, 2003, with a supplement dated February 21, 2003. The applicant responded to this RAI in letters to the staff dated April 28, 2003, and June 13, 2003.

4.4.1 Electrical and I&C Component Environmental Qualification Analyses

4.4.1.1 Summary of Technical Information in the Application

In the LRA, Section 4.4, the applicant describes the TLAA evaluation methodology and how the results from these evaluations were used to demonstrate that (1) the analyses remain valid for the period of extended operation, (2) the analyses have been projected to the end of the period of extended operation, or (3) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The following is a summary of the methodology used by the applicant to evaluate the EQ TLAAs and the results from this evaluation.

The Environmental Qualification Program at RNP is a centralized plant support program administered by Design Engineering in order to maintain compliance with 10 CFR 50.49. The scope of the Environmental Qualification Program includes the following categories of electrical equipment located in a harsh environment:

- safety-related equipment
- non-safety-related equipment whose failure could adversely affect safety-related equipment
- the necessary post-accident monitoring equipment

The identification of EQ equipment is specified by procedural controls, and a component database is utilized to maintain an EQ equipment master list.

The Environmental Qualification Program includes three main elements—identifying applicable equipment and environmental requirements, establishing the qualification, and maintaining (or preserving) qualification.

Components included in the RNP Environmental Qualification Program have been evaluated to determine if existing environmental qualification aging analyses remain valid for the period of extended operation. Qualification for the license renewal period will be treated the same as for components currently qualified at RNP for 40 years or less. The Environmental Qualification

Program manages component thermal, radiation, and wear cycle aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, environmentally qualified components must be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation. Aging evaluation for environmentally qualified components that specify a qualification of at least 40 years are considered TLAAs for license renewal.

Age-related service conditions that are applicable to environmentally qualified components (i.e., 60 years of exposure versus 40 years) were evaluated for the period of extended operation to verify that the current EQ analyses are bounding. Temperature and radiation values assumed for service conditions in the EQ analyses are either design operating values or measured values for RNP. The following paragraphs describe the thermal, radiation, and wear cycle aging effects that were evaluated.

Thermal Considerations

The component qualification temperatures were calculated for 60 years using Arrhenius method, as described in Electric Power Research Institute (EPRI) NP-1558, "A Review of Equipment Aging Theory and Technology." If the component qualification temperature bounded the service temperatures throughout the period of extended operation, then no additional evaluation was required.

Radiation Considerations

The RNP Environmental Qualification Program has established bounding radiation dose qualification values for all environmentally qualified components. Typically, these bounding radiation dose values were determined by component vendors through testing. To verify that the bounding radiation values are acceptable for the period of extended operation, integrated dose values were determined and then compared to the bounding values. The total integrated dose (TID) through the period of extended operation is determined by adding the established accident dose to the normal operating dose for the component.

Wear Cycle Aging Considerations

Wear cycle aging is a factor for some equipment within the Environmental Qualification Program. In cases for which wear cycle aging was considered a credible aging mechanism, wear cycles were evaluated through the end of the new license term.

4.4.1.2 Staff Evaluation

The staff reviewed Section 4.4 of the RNP LRA to determine whether the applicant submitted adequate information to meet the requirements of 10 CFR 54.21(c)(1). In addition, the staff met with the applicant to obtain clarifications and to review specific EQ calculations and reviewed the applicant's response to the staff's RAIs.

In response to the staff's concern about the use of measured values in the EQ analyses (RAI 4.4.1-1), the applicant, by letter dated April 28, 2003, stated that the temperature and radiation values used for service conditions in the EQ analyses discussed in LRA Section 4.4.1 are either the design values or are based on measured values. Design values are based on plant design documentation that supports the CLB including the UFSAR, design calculations, and Environmental Qualification Program evaluations. Measured values are actual measured values taken over a period of 1 year or more.

The pressurizer cubicle is the only area in the containment that uses actual measured temperatures, since temperatures in this area routinely exceed the bulk average containment temperature. Components located in the pressurizer cubicle that were found to be qualified for 60 years had sufficient margin to absorb the increases in normal operating temperatures in the pressurizer cubicle. These components included Rockbestos Firewall III cable and Raychem splice material.

Outside containment, the qualified life calculations are based on either the design temperature of 104 °F or actual measured temperatures. Measured temperatures are based on temperature readings taken each shift by operations personnel. There are no defined harsh temperature areas in the Environmental Qualification Program outside of containment. In the one case where measured temperatures are used for EQ, a qualified life of over 60 years resulted. Aging in this case was based on aging performed for PVC insulated cables that were then subjected to a loss-of-coolant-accident (LOCA). For these cables located outside containment, survival of a LOCA is not a requirement, which results in additional conservatism.

Area radiation levels are monitored continuously in various locations in the containment and reactor auxiliary building (RAB). UFSAR Section 11.5 describes the process and effluent radiation monitoring system. Radiation levels in these areas are indicated, recorded, and alarmed in the control room.

Daily operator rounds, radiation monitoring by health physics personnel (surveys of areas in the RAB at least monthly, and in some cases daily or weekly), and maintenance and engineering personnel provide feedback to engineering through the Corrective Action Program when changes to the plant environment or EQ equipment are encountered. Changes in temperature or radiation levels that could adversely affect qualification would be readily identified. RNP plant procedures govern the frequency of surveillances, radiation surveys, and plant walkdowns. The frequencies range from each shift to each outage.

Containment temperature and radiation are logged at least daily, and other EQ areas are subject to operator rounds at least daily while the plant is operating. The temperature and radiation data obtained are representative of the service conditions of EQ equipment, and any change in temperature or radiation that could adversely affect qualification would be readily identified.

Based upon the above information, the staff finds that the applicant has adequately addressed the subject of concern in RAI 4.4.1-1.

In response to the staff's concern regarding the controls used to monitor changes in plant environmental conditions to periodically validate the environmental data used in analyses (RAI 4.4.1-2), the applicant, by letter dated June 13, 2003, provided the following response:

(a) RNP completed a new containment accident analysis in 1999 that resulted in revision of the temperature versus time profile used as a basis for environmental qualification. Also, RNP completed an Appendix K power uprate in 2002 that resulted in an approximate 1.7% increase in power level.

The Appendix K power uprate resulted in no change to temperature values and a minor change to radiation values. Radiation dose was increased by 1.02 times the current value. When this multiplier was applied to the current dose rates in the containment for the remaining period through the end of the new license term, it was found that the change in dose was minimal and well within the 10% margin typically added to environmentally qualified equipment. Environmental qualification packages are undergoing revision at this time and will be updated prior to the end of the current license term (Commitment Number 41).

(b) The qualification basis for the equipment impacted by the aforementioned changes had sufficient conservatism to maintain existing qualification.

(c) Containment temperature and radiation are logged at least daily, and other EQ areas are subject to operator rounds at least daily while the plant is operating. The temperature and radiation data obtained is representative of the service conditions of EQ equipment, and any change in temperature or radiation that could adversely affect qualification would be readily identified.

UFSAR Section 11.5 describes the Process and Effluent Radiation Monitoring System. Radiation levels in these areas are indicated, recorded and alarmed in the control room.

Operator daily rounds, radiation monitoring by Health Physics personnel (surveys of areas in the RAB at least monthly, and in some cases daily or weekly), and Maintenance and Engineering personnel provide feedback to Engineering through the Corrective Action program when changes to the plant environment or EQ equipment are encountered. Changes in temperature or radiation levels that could adversely affect qualification would be readily identified. RNP plant procedures govern the frequency of surveillances, radiation surveys, and plant walkdowns. The frequencies range from each shift to each outage.

Based upon the above information, the staff finds that the applicant has adequately addressed the subject of concern.

In response to the staff's concern regarding TID through the period of extended operation from the 40-year values (RAI 4.4.1-5), the applicant stated by letter dated April 28, 2003, that the RNP EQ Program has established bounding radiation dose qualification values for environmentally qualified components. Typically, these bounding radiation dose values were determined by component vendors through testing. To verify that the bounding radiation values are acceptable for the period of extended operation, integrated dose values were determined and then compared to the bounding values. The TID through the period of extended operation is determined by adding the established accident dose to the normal operating dose for the component. The normal 60-year operating dose was determined by multiplying the normal 40-year dose by 1.5. Based on this information, the staff finds that the applicant has adequately addressed the subject of concern.

On October 23, 2002, representatives of RNP met with the NRC staff to review a sample of EQ calculations. The staff reviewed the following calculations:

- EQDP-1.0, Revision 9, ASCO Solenoid Valves—AQR Report (4.4.1.2)

- EQDP-1.1, Revision 2, ASCO Solenoid Valves
- EQDP-2.0, Revision 6, Limitorque Model SB-3 and SBM-00 MOV Actuators—Inside Containment (4.4.1.4)
- EQDP-2.1, Revision 5, Limitorque MOV Actuators
- EQDP-3.0, Revision 13, Rockbestos Cable—Firewall III (4.4.1.5)
- EQDP-8.1, Revision 6, Westinghouse Motors—Frame 506 UPZ, 509US, and SBDP-RHR, SI Pumps, HVA 6A, 8A, and 8B (4.4.1.11)
- EQDP-9.0, Revision 4, Crouse-Hinds Electrical Penetration Assemblies (4.4.1.13)
- EQDP-15.1, Revision 6, Kerite FR2/FR3 Insulated Multiconductor Cable (4.4.1.27)
- EQDP-18.1, Revision 2, Westinghouse CET/CCM—Reference Junction Boxes and Potting Adaptors (4.4.1.32)
- EQDP-19.1, Revision 4, Gamma—Metrics Excore Neutron Detectors (4.4.1.34)
- EQDP-31.0, Revision 6, Cable—PVC and XLPE Outside Containment (4.4.1.43)
- EQDP-33.0, Revision 4, Grease—Motors and MOVs (4.4.1.44)
- EQDP-12.1, Revision 2, Raychem Splices—NPKV Stub Kits (4.4.1.19)
- EQDP-34.0, Revision 6, Target Rock Solenoid Valves (4.4.1.45)

The staff verified that the applicant is using standard, approved EQ methodologies and acceptance criteria applicable to EQ as defined by NRC Bulletin 79-01B (the Division of Operating Reactors guidelines), including Supplements 1, 2, and 3; NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Revision 1; 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants"; RG 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," Revision 1; various NRC generic letters and information notices; and NRC safety evaluation reports on EQ.

The staff found that all EQ calculations were done using design temperature or measured temperature. The measured temperatures at pressurizer cubicles are higher than the design temperature. These higher temperature values are used for equipment in that area. The staff found that activation energies have not been changed and ohmic heating for power cables was properly considered. A 32 °C rise due to ohmic heating over 40 °C ambient was used for power cables. Wear cycle aging for motors, limit switches, solenoid valves, and multipin connectors was not addressed. By letter dated April 28, 2003, the applicant provided a response to the staff's concerns (RAI 4.4.1-3). On the basis of its review, the staff concludes that the applicant has adequately addressed these concerns.

<u>TLAA Demonstration for Option 10 CFR 54.21(c)(1)(ii)</u>

For the following list of electrical equipment identified in Section 4.4.1 of the LRA, the applicant cites 10 CFR 54.21(c)(1)(ii) in its TLAA evaluation to demonstrate that the analyses have been projected to the end of the period of extended operation:

- 4.4.1.1 ASCO NP8316 and NP8321 Series Solenoid Valves
- 4.4.1.2 ASCO Solenoid Valves—AQR Report
- 4.4.1.4 Limitorque Model SB-3 and SBM-00 Motor-Operated Valve (MOV) Actuators—Inside Containment
- 4.4.1.5 Rockbestos Cable—Firewall III
- 4.4.1.6 Rockbestos RSS-6-104/LE Series Coaxial Cable
- 4.4.1.7 Rockbestos Cable—Firezone R
- 4.4.1.8 GEMS Liquid Level Transmitters—Model XM-54853 and XM-54854
- 4.4.1.9 B&W Valve Monitoring System
- 4.4.1.10 Westinghouse Reactor Containment Fan Cooler (RCFC) Motors
- 4.4.1.11 Westinghouse Motors—Frame 506UPZ, 506US, and SBDP-RHR, SI Pumps, HVA 6A, 6B, 8A, and 8B
- 4.4.1.12 Westinghouse Motors—Model S068C20085—Containment Spray Pumps
- 4.4.1.13 Crouse-Hinds Electrical Penetration Assemblies
- 4.4.1.14 Continental Shielded Instrument Cable—CC2115
- 4.4.1.15 Continental/Anaconda Cable—Instrumentation
- 4.4.1.16 Samuel Moore Dekoron Instrumentation Cables (EPDM and XLPO Insulations)
- 4.4.1.17 Eaton Corporation Dekoron Cable 16 AWG
- 4.4.1.18 Raychem WCSF-N Splices
- 4.4.1.19 Raychem Splices—NPKV Stub Kits
- 4.4.1.20 Raychem Splices—NPK Connection Kits
- 4.4.1.21 Raychem Splices—NMCK Connection Kits
- 4.4.1.22 Raychem Splices—NESK End Seal Kits

- 4.4.1.23 AMP Butt Splices
- 4.4.1.24 AMP PIDG Terminals
- 4.4.1.25 CM-303 Tape Splices Assemblies—Scotch 27 and Scotch 70
- 4.4.1.26 Kerite HTK Power Cable
- 4.4.1.27 Kerite FR2/FR3 Insulated Multiconductor Cable
- 4.4.1.28 Thomas and Betts STA-KON Terminal
- 4.4.1.29 Conax Electrical Conductor Seal Assemblies—ECSA
- 4.4.1.30 Conax Electrical Penetration Assemblies
- 4.4.1.31 Westinghouse CET/CCM—Incore T/C Connectors and MI Cable Assemblies
- 4.4.1.32 Westinghouse CET/CCM—Reference Junction Boxes and Potting Adaptors
- 4.4.1.33 Westinghouse CET/CCM—Intermediate Disconnect Box Connectors
- 4.4.1.34 Gamma—Metrics Excore Neutron Detectors
- 4.4.1.35 Pyco Resistance Temperature Detectors
- 4.4.1.36 Buchanan Terminal Blocks
- 4.4.1.37 Barton Pressure Switches—Model 580A
- 4.4.1.38 NAMCO Receptacle and Connector/Cable Assemblies—Model EC210
- 4.4.1.39 Victoreen High Range Radiation Detectors
- 4.4.1.40 Brand Rex Cable—Instrumentation
- 4.4.1.41 Brand Rex Cable—Control
- 4.4.1.42 Raychem Cable—Flamtrol
- 4.4.1.43 Cable—PVC and XLPE Outside Containment
- 4.4.1.44 Greases—Motors and MOVs
- 4.4.1.45 Target Rock Solenoid Valves
- 4.4.1.46 Boston Insulated Wire—Cable
- 4.4.1.47 Honeywell Model V4-21 Microswitch Assembly

• 4.4.1.48 RAM-Q Connectors

TLAA Demonstration for Option 10 CFR 54.21(c)(1)(iii)

For the list of electrical equipment identified in Section 4.4.1 of the LRA, the applicant cites 10 CFR 54.21(c)(1)(iii) in its TLAA evaluation to demonstrate that the aging effects of the EQ equipment identified in this TLAA will be managed during the extended period of operation by the Environmental Qualification Program activities described in Section B.4.1 of the LRA.

4.4.1.3 Limitorque SBM Motor-Operated Valve Actuators—Outside Containment

In LRA Section 4.4, the applicant stated that the Environmental Qualification Program manages component thermal, radiation, and wear cycle aging through the use of aging evaluation based on 10 CFR 50.49(f) qualification methods. Appendix B, "Aging Management Programs," did not include the Environmental Qualification Program as one of the existing programs. This program will be credited to manage the aging of EQ components. In response to this staff concern (RAI 4.4-2), the applicant, by letter dated April 28, 2003, stated that new Section B.2.9, "Environmental Qualification (EQ) of Electrical Components," should be added to Appendix B. The applicant provided the details of the program.

The staff reviewed the EQ Program to determine whether it will assure that the electrical/I&C components covered under this program will continue to perform their intended function consistent with the CLB for the period of extended operation. The staff's evaluation of the component qualification focused on how the program manages the aging effect through effective incorporation of seven elements—scope of program, preventive action, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance criteria, and operating experience. The staff's evaluation of the applicant's corrective actions, confirmation process, and administrative controls is provided separately in Section 3.0.4 of the SER.

Scope of Program—The RNP Environmental Qualification Program includes certain electrical components that are important to safety and could be exposed to harsh environment accident conditions, as defined in 10 CFR 50.49. The staff considers the scope of the program to be acceptable.

Preventive Actions—Actions that prevent aging effects are not requried by 10 CFR 50.49. The RNP Environmental Qualification Program actions that could be viewed as preventive actions include (1) establishing the component service condition tolerance and aging limits (for example, qualified life or condition limit), (2) refurbishment, replacement, or requalification of installed equipment prior to reaching these aging limits, and (3) where applicable, requiring specific installation, inspection, monitoring, or periodic maintenance actions to maintain equipment aging effects within the qualification. The staff considers these actions acceptable because 10 CFR 50.49 does not require actions that prevent aging effects.

Parameter Monitored or Inspected—EQ component aging limits are not typically based on condition or performance monitoring. However, per RG 1.89 Revision 1, such a monitoring program is an acceptable basis to modify aging limits. Monitoring or inspection of certain environmental, condition, or equipment parameters may be used to ensure that the equipment is within its qualification or as a means to modify qualification. The staff considers this monitoring appropriate because the program objective is to ensure that the qualified life of devices established is not exceeded.

Detection of Aging Effects—The detection of aging effects for inservice components is not required by 10 CFR 50.49. Monitoring of aging effects may be used as a means to modify component aging limits. The staff considers the applicant's program to use the monitoring of aging effects as a means to modify component aging limits acceptable.

Monitoring and Trending—Monitoring and trending of component condition or performance parameters of inservice components to manage the effects of aging are not required by 10 CFR 50.49. Environmental Qualification Program actions that could be viewed as monitoring include monitoring how long qualified components have been installed. Monitoring or inspection of certain environmental, condition, or component parameters may be used to ensure that a component is within its qualification or as a means to modify the qualification. The staff considers this acceptable since 10 CFR 50.49 does not require monitoring and trending of component condition or performance parameters of inservice components to manage the effects of aging.

Acceptance Criteria—The acceptance criteria in 10 CFR 50.49 are that an inservice EQ component is maintained within its qualification including (1) its established aging limits and (2) continued qualification for the projected accident conditions. Compliance with 10 CFR 50.49 requires refurbishment, replacement, or requalification prior to exceeding the aging limits of each installed device. When monitoring is used to modify a component aging limit, plant-specific acceptance criteria are established based on applicable 10 CFR 50.49(f) qualification methods. The staff considers this acceptable since it is consistent with 10 CFR 50.49 requirements of refurbishment, replacement, or requalification prior to exceeding the qualified life of each installed device.

Operating Experience—The RNP Environmental Qualification Program includes consideration of operating experience to modify qualification bases and conclusions, including aging limits. Compliance with 10 CFR 50.49 provides evidence that the component will perform its intended functions during accident conditions after experiencing the detrimental effects of inservice aging. The staff finds that the applicant has adequately addressed operating experience.

The staff also reviewed the UFSAR Supplement to determine whether it provides an adequate description of the program.

4.4.1.3 Conclusions

On the basis of its review, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1), that, for the environmental qualification of electrical equipment TLAA, the analyses have been projected to the end of the period of extended operation, or the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The staff also concludes that the UFSAR Supplement contains an appropriate summary description of the environmental qualification of electrical equipment TLAA evaluation for the period of extended operation, required by 10 CFR 54.21(d). Therefore, the staff has concluded that the safety margins established and maintained during the current operating term will be maintained during the period of extended operation as required by 10 CFR 54.21(c)(1).

4.4.2 GSI-168, Environmental Qualification of Electrical Components

4.4.2.1 Summary of Technical Information in the Application

Environmental qualification evaluations of electrical equipment are identified as TLAAs for RNP. The evaluations of these TLAAs are considered the technical rationale that the CLB will be maintained during the period of extended operation. The evaluations are provided in Section 4.4.1 of the LRA. Consistent with the above NRC guidance, no additional information is required to address GSI-168 in a renewal application at this time.

4.4.2.2 Staff Evaluation

GSI-168 is now closed. The staff issued RIS 2003-09, "Environmental Qualification of Low-Voltage Instrumentation and Control Cables," on May 2, 2003, and indicated that no further action is required by the applicant.

4.4.2.3 Conclusions

The staff determined that no further action is required by the applicant because GSI-168 is closed.

4.5 Concrete Containment Tendon Loss of Prestress

4.5.1 Summary of Technical Information in the Application

The applicant describes the RNP containment building as a steel-lined concrete shell in the form of a vertical right cylinder with a hemispherical dome and a flat base. The dome and base are constructed of reinforced concrete. The cylinder walls are concrete, reinforced circumferentially and prestressed vertically.

The applicant points out that prestressing force (in vertical direction) is not constant; it decreases over time due to a variety of design conditions. The applicant identifies the factors affecting the prestressing force that were considered in the original evaluation of the containment prestressing tendons as steel relaxation, concrete shrinkage, concrete creep, elastic shortening of concrete, and 2 percent reduction for broken tendons.

For license renewal, the applicant states that the calculation of prestress was updated to address potential losses through the period of extended operation. The new calculation considers the factors listed above that influence loss of prestress. However, the value for concrete shrinkage was marginally reduced based on a comparison to estimated shrinkage values used in the original calculation, as well as reference to the time of application of loading compared to completion of the containment walls. Specifically, the original analysis used a shrinkage coefficient of 0.0003, and the original containment design information estimates the actual shrinkage to be 0.00005. The value used in the revised calculation is 0.0002. This is supported by the fact that shrinkage is a volume change in concrete that occurs with time rather than with load; as such, higher values are more realistic for pretensioned members where the prestress is transferred to the concrete at an early age, whereas the lower value is more appropriate for post-tensioned members. Moreover, the applicant makes a point that RNP tendons are considered to be post-tensioned because the tendons were not loaded until after the concrete was placed. This allowed a portion of the shrinkage to occur prior to tendon tensioning.

Furthermore, the applicant explains that no prestress losses were considered for elastic shortening, due to the retensioning of the tendons approximately a month after the initial tensioning. No reduction in prestress was taken for general corrosion based on review of the 5-year and 25-year surveillance tendon inspections. For example, based on visual examination of the 25-year tendon and upon removal of the grout surrounding the tendon, the applicant noted, "The surface of the bars was covered with a reddish-brown oxide that could be removed simply by wiping the surface clean by hand. No measurable metal loss or etching could be detected once the dust was removed." Therefore, grouting the tendons has proven to be effective for the prevention of corrosion.

The applicant indicates that the calculation projects the prestress losses for 60 years. The applicant also indicates that the tendons were originally tensioned a few months prior to the original licensing date of the plant. As such, the actual prestress period for the tendons is more than 60 years. Based on comparison of the evaluated margin to the required minimum prestress, the slight increase in duration will not allow the actual prestress to go below the required minimum. Based on the above analysis of tendon prestress, the applicant has determined that the final effective prestress at the end of 60 years exceeds the minimum required value. Consequently, the post-tensioning system will continue to perform its intended function throughout the period of extended operation. Therefore, the analysis associated with containment tendon loss of prestress has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

4.5.2 Staff Evaluation

The RNP is one of the few operating plants in which the containment prestressing tendons are protected from corrosion by means of cement grout. Though the cement grout provides a reliable alkaline medium for protecting the tendons, the tendon system cannot be monitored for either the remaining prestress level, or for the effectiveness of the cement grout in protecting the tendons. Also, some extraneous causes of early deterioration of prestressing tendon systems with greased tendons in the United States are to an extent applicable to the high hardness prestressing system components (e.g., American Iron and Steel Institute (AISI) 5160 bars, AISI 4130 couplers, and AISI 8620 grip nuts) of RNP containment.

In RAI 4.5-1, the staff requested information to understand the basis of the applicant's TLAA. From the TLAA provided, the relative magnitudes of the changes in the various factors affecting the prestressing loss and remaining prestressing force levels are not clear. The applicant was asked to provide a table showing the initial average prestressing force, losses due to the five factors (indicated by bullets in the TLAA), and the final average prestressing force originally considered at 40 years, and the values proposed at the end of the extended period of operation.

In response to RAI 4.5-1, the applicant provided the following table showing the calculated prestressing forces at the initial prestressing, at 40 years, and at 60 years after the installation of the forces.

Description	Initial Value	Value After 1 Year	Value After 50 Years	Value At 60 Years
Prestress losses due to concrete shrinkage	N/A	4002 psi	1998 psi	0
Prestress losses due to concrete creep	N/A	6317 psi	3152 psi	0

Prestress losses due to tendon relaxation	N/A	6000 psi	2400 psi	1800 psi
Prestress losses due to elastic shortening	2104 psi	N/A	N/A	N/A
Tendon prestress	120,000 psi	103,680 psi	96,128 psi	94,328 psi
Minimum required prestress	91,726 psi	91,726 psi	91,726 psi	91,726 psi

The staff reviewed the table in conjunction with the values estimated in the UFSAR. The staff also reviewed the modifications made by the applicant to the UFSAR values and discussed in 4.5.1 of this SER. The staff considers the modifications made to the concrete shrinkage value reasonable and acceptable. Based on the review of the applicant's estimated values at 40 and 60 years, the staff finds that the prestressing force imparted to the containment will be adequate during the period of extended operation.

Knowing the types of materials used for fabricating the tendons and their anchorage components, and their potential for corrosion, the staff in RAI 4.5-2 requested the following information from the applicant:

> Information Notice (IN) 99-10, Revision 1, "Degradation of Prestressing Tendon Systems in Prestressed Concrete Containments," describes the experience related to hydrogen stress cracking of ASTM A 421 wires, and breakage of AISI 4140 anchor-heads due to hydrogen stress cracking. However, these incidences were detected, and corrective actions were taken as the tendon components were amenable for in-service inspection, component replacement, and re-tensioning, as required.
>
> The RNP tendon components (i.e., AISI 5160 bars, AISI 4130 couplers, and AISI 8620 grip nuts) are high hardness components, subjected to sustained high stresses, and hydrogen stress cracking of the high hardness components is a plausible aging effect in the presence of galvanized tendon ducts around the grouted tendon components. As recognized by the applicant in Revision No. 15 of the UFSAR (page 3.8.1-56), the results of the two surveillance blocks cannot be relied upon to provide confidence regarding the plausibility of such aging effects, or the time dependent trending of prestressing forces. Moreover, no such surveillance blocks are available for the future prediction of the containment tendon behavior.
>
> In light of the above discussion, the applicant is requested to explore the methods that can be used to assess the containment prestressing levels during the extended period of operation.

The RAI essentially requested the applicant to explore the methods that could be used to assess and track the containment prestressing force and potential degradation of prestressing tendon components.

In response, the applicant provided the following information:

- Degradation (breakage) of prestressing wires (as discussed in Information Notice 99-10) was primarily attributed to the ability of moisture to reach unprotected areas; RNP tendons are completely encased in grout and are therefore not susceptible to moisture intrusion.

- Stress-corrosion cracking occurs when high stress, corrosive environment, and susceptible material are present. Only one element is present in RNP containment prestress components (i.e., high stress).

- Surveillance blocks examined at 5 and 25 years showed no corrosion of the embedded tendon material.

- Containment structural integrity tests were performed in 1970, 1974, and 1992, and comparisons are provided to the NRC in a letter dated October 7, 1992 (Serial No. NLS-92-262).

- The prestressing levels have been analytically determined to be sufficient through the period of extended operation. IWL examination will be continued during the EPO.

- To provide additional assurance of the tendon design capacity, tests (at integrated leak rate test pressure) similar to the structural integrity test performed in 1992, will be scheduled to coincide with the first and second Appendix J containment integrated leak rate test during the period of extended operation. The monitoring criteria of these tests will be limited to deformations and cracking associated with the vertical prestressed tendons and will not include radial or axial monitoring. The proposed tests will be performed in conjunction with the analytical determination of tendon prestress, the established corrosion resistance of the embedded tendons, the previously completed structural integrity tests, and the ongoing inspections of concrete.

The staff believes that stress corrosion of the tendon hardware components is a plausible aging effect, and means have to be found to assess the containment integrity during the period of extended operation. In the last bulleted item, the applicant commits to perform structural integrity pressure tests of the RNP containment two times during the extended period of operation. However, the applicant is not clear as to what measurements will be taken during the tests. The staff believes that observing the crack pattern of the containment and measuring the containment deformations during the recommended pressure tests provide a gross means of confirming that a widespread degradation of the prestressing tendon components has not occurred. The staff believes that all means available during the pressure tests should be employed to assess the integrity of the prestressing tendons and the containment.

In Item 45 of the RNP license renewal commitments, the applicant incorporates the staff's recommendations for performing structural integrity testing and making the necessary observations during the tests. The staff finds the applicant's commitment acceptable as it would assess the integrity of the prestressing tendons and the RNP containment during the period of extended operation.

In RAI 4.5-3, the applicant is requested to justify why the information sought in RAI 4.5-1 should not be inserted in the UFSAR Supplement. Having such a table would clearly show the expected average prestressing force level in the tendons and in the concrete of the containment during the extended period of operation.

In Appendix A2 of the LRA, the applicant indicates changes to Section 3.8.1.4.7 of the UFSAR related to the changes in the value of shrinkage and tendon relaxation loss for estimating the final prestress force in the containment at the end of the period of extended operation. The staff recommends that the table provided in response to RAI 4.5-1 be inserted in the UFSAR Supplement or in Section 3.8 of the UFSAR.

In Item 46 of the RNP license renewal commitments, the applicant agrees to incorporate the table in Section 3.8.1.4.7 of the RNP UFSAR.

4.5.3 Conclusions

On the basis of the information provided in Section 4.5 and Appendix A2 of the LRA and in the responses to the staff's RAIs, the staff has concluded that the TLAA for tendon prestressing force performed in accordance with the requirement of 10 CFR 54.21(c)(1)(ii) will be valid for the period of extended operation. This conclusion is based on the assumption that the applicant will be indirectly monitoring the condition of the tendon hardware components by pressure testing of the containment.

4.6 Other TLAAs

4.6.1 Thermal Aging Embrittlement

In Section 4.6.1 of the LRA, the applicant provides its TLAA for assessing the effect of 60-year operation on the thermal aging embrittlement and leak-before-break (LBB) analyses for cast austenitic stainless steel (CASS) materials in the RNP reactor coolant main loop piping and for demonstrating that the LBB analysis for the RNP reactor coolant main loop piping would remain acceptable for service through the expiration of the extended period of operation for RNP, as evaluated in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

4.6.1.1 Summary of Technical Information in the Application

In Section 4.6.1 of the LRA, the applicant states that the fracture mechanics analyses for the CASS components in the RCS are considered to be TLAAs because of the effects of thermal aging, and that for RNP, these analyses are the LBB analysis of RCS piping and welds and the analysis of RCPs in support of ASME Code, Section XI, Code Case N-481. In this section of the LRA, the applicant summarizes the effects that thermal aging of the CASS reactor coolant piping and pump casing components will have on the LBB analysis for the RNP main RCS piping and Code Case N-481 inspection analyses for RNP RCPs.

In Section 4.6.1 of the LRA, the applicant stated that an LBB analysis was performed to demonstrate that any potential leaks that develop in the RCS loop piping would be detected by plant leak monitoring systems before a postulated throughwall crack (resulting in a leak of the reactor coolant) would grow to unstable proportions during the 40-year plant life. In this section of the LRA, the applicant explained that the RNP LBB assumes the existence of a throughwall crack of sufficient size, such that the resultant leakage can be easily detected by the existing leakage monitoring system, and demonstrates that, even under maximum faulted loads, the assumed crack size is much smaller (with margin) than a critical flaw size that could grow to pipe failure. The applicant stated that the aging effects that need to be addressed during the period of extended operation include thermal aging of CASS materials in the primary loop piping components and fatigue crack growth.

In regard to the applicant's evaluation of the effect of thermal aging on the integrity of the RNP RCPs, the applicant stated that, following ASME approval of Code Case N-481, "Alternate Examination Requirements for Cast Austenitic Pump Casings, Section XI, Division 1," in March 1990, the Westinghouse Owner's Group sponsored WCAP-13045, which provided a generic fracture mechanics analysis and demonstrated generic compliance with the code case for the fleet of Westinghouse-designed light-water reactors. The applicant stated that Code Case N-481 permits surface examination methods to be used in lieu of volumetric examination methods for inspections of RCP casings[2], provided a fracture mechanics analysis is prepared which meets specified requirements. The applicant also stated that the code case requires a plant-specific evaluation to demonstrate safety and serviceability of the pumps and that, therefore, WCAP-15363, Revision 0, was issued in April 2000 as the plant-specific analysis to support use of the alternate inspection techniques for the Westinghouse Model 93 pumps at RNP. The applicant also stated that the plant-specific loadings were compared to the generic loadings in WCAP-13045, and plant-specific materials were compared to generic materials data used in WCAP-13045, demonstrating the requirements of the code case were met for the 40-year operation of the plant.

The applicant stated that, to support the license renewal process, a new report, WCAP-15363, Revision 1, was prepared which supersedes WCAP-15363, Revision 0, and includes an evaluation of the plant-specific pump casing material properties to account for reduced fracture toughness due to thermal embrittlement during the 60-year extended operational period. The applicant stated that WCAP-15363, Revision 1, uses the limiting transients from the 40-year design transient set provided in WCAP-15363, Revision 0, and that the 40-year design transients have been shown to be conservative for 60 years of plant operation. The applicant stated that WCAP-15363, Revision 1, demonstrates that the safety margin requirements for leakage and crack stability of the RNP RCP casings have been met and justify the use of the surface examination of pump casings in lieu of volumetric examination in accordance with the code case throughout the period of extended operation. The applicant stated that, therefore, the ASME Code Case N-481 analysis has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

4.6.1.2 Staff Evaluation

Thermal aging refers to the gradual change in the microstructure and properties of a material due to its exposure to elevated temperatures for an extended period of time. Thermal aging may reduce the fracture toughness for a given material.[3] When this occurs, the material's critical crack size, which is a bounding material property for any given material, is smaller. Should cracks exist in a component and grow to sizes larger than the critical crack size for the component's material of fabrication, the cracks are considered to be unstable and will propagate rapidly through the component. This phenomenon is referred to by materials and mechanical engineers as crack growth by fast fracture. Cracks that propagate unstably by this phenomenon may lead to catastrophic failure of the component. CASS components are known to be particularly susceptible to reduction in fracture toughness as a result of thermal aging; neutron embrittlement of CASS internals may enhance this effect. When this occurs, a CASS component' s tolerance to withstand the presence of existing flaws (cracks) is significantly reduced.

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[2]The applicant's statement is slightly in error. ASME Code Case N-481 actually provides alternative visual examination requirements for Class 1 pump casings fabricated from CASS. Licensees seeking to apply the alternative requirements in the Code Case to their RCP casings are required by the alternative provision requirements of 50.55a(a)(3)(i) to Title 10, *Code of Federal Regulations*, to submit the methods for NRC review and approval. The alternative inspection visual methods include alternative VT-1, VT-2, and VT-3 requirements. The alternative requirements in Code Case N-481 also require the licensee applying to use the code case methods to submit an alternative fracture mechanics analysis for the pump casings that supports use of the alternative inspection requirements.

[3]Fracture toughness refers to a material property that is an indication of a material's resistance to rapid unstable crack propagation. For metallic alloys, fracture toughness properties are, in part, dependent upon an alloy's microstructural configuration and alloying content.

The RNP Class 1 RCS main loop piping includes some piping, valve, and pump casings fabricated from CASS. The only significant effects of the additional period of operation on the structural integrity of the Class 1 RCS at RNP are on the LBB analysis for the RCS main loop piping components fabricated from CASS, and on the fracture mechanics analysis that is required to support use of alternative inspection methods proposed for the RNP RCP casings fabricated from CASS. The staff evaluates the effect of the additional period of operation on the structural integrity assessment for these items in the paragraphs that follow.

<u>The RNP LBB Analysis for the Main Loop RCS Piping and Components</u>

In Section 4.6.1 of the LRA, the applicant indicated that it performed a new LBB analysis to assess the effect of 60 years of operation on the acceptability of the previous LBB analysis for RNP. The applicant stated that the new LBB analysis and calculation is contained in proprietary Class 2 report WCAP-15628, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the H.B. Robinson Unit 2 Nuclear Power Plant for the License Renewal Program [July 2001]," and that this report includes allowances for reduction of fracture toughness of CASS due to thermal embrittlement during a 60-year operating period. The applicant stated that the new LBB analysis meets the requirements for LBB required by 10 CFR 50, Appendix A, General Design Criterion 4, and uses the recommendations and criteria from the NRC Standard Review Plan for LBB evaluations. The applicant stated that the new LBB analysis uses the prior 40-year design basis thermal transients as input for the fatigue crack growth analysis and that these transients have been shown to be conservative for the 60-year operating period. The applicant therefore concluded that the RCS primary loop piping LBB analysis has been projected to the end of the period of extended operation, and has been demonstrated to be acceptable through the expiration of the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

The applicant's TLAA for the LBB for the RCS main loop piping did not indicate whether WCAP-15628 was reviewed and approved by the NRC. The applicant's TLAA for LBB also did not discuss why the applicant considered the 40-year design basis thermal transients to be conservative and bounding for the LBB analysis through the expiration of the extended operating period for RNP or discuss how the LBB analysis accounted for potential loss of fracture toughness properties that could result from thermal aging of RCS main loop piping, pump, or valve components made from CASS. Therefore in RAI 4.6.1-1, the staff requested that the applicant submit WCAP-15628 for review and approval.

In response to RAI 4.6.1-1 and by letter dated May 7, 2003, the applicant submitted Westinghouse proprietary Class 2 report WCAP-15628 for review and approval. The staff has completed its review of WCAP-15628. Regarding the adequacy of the fatigue crack growth analysis through the expiration of the extended operating period for RNP using the original 40-year design basis thermal transients, the applicant summarized RNP's 40-year thermal fatigue design transients, the number of actual plant transients that have occurred through 2000, and the 60-year projection methods and basis for the LBB analysis. This summary indicates that the projected number of occurrences through 60 years of licensed life are bounded by the number of transients originally assumed in the 40-year fatigue analysis. In regard to the concern about the thermal aging of RCS main loop piping and components made from CASS, the staff has verified that the applicant considered appropriate, fully-aged toughness for CASS in the original 40-year LBB analysis. Based on the above evaluation, the staff agrees with the applicant's conclusion that this TLAA is in accordance with

10 CFR 54.21(c)(1)(ii), and the LBB application for the primary loop piping and components is acceptable for the period of extended operation.

Effect of Thermal Aging on the Inspection Methods Proposed for the RNP Reactor Coolant Pumps

The 1995 edition of the ASME Boiler and Pressure Vessel Code, Section XI, Table IWB-2500-1, Examination Category B-L-1, Item B12.10, requires that volumetric examinations be performed on ASME Class 1 pump casing welds once every 10-year inservice inspection (ISI) interval. ASME Code Case N-461 provides alternative ISI techniques for examinations of RCP casings in PWR-designed light-water reactors. The methods of the code case allow a licensee to use the following alternative requirements for assuring the integrity of RCP casings made from CASS in lieu of performing the volumetric examination methods required by ASME Section XI, Table IWB-2500-1, Examination Category B-L-1, Item B12.10:

- perform a VT-2 visual examination of the exterior of all pumps during the hydrostatic
- pressure test required by Table IWB-2500-2, Examination Category B-P
- perform a VT-1 visual examination of the external surfaces of the weld on one casing
- perform a VT-3 visual examination of the internal surfaces whenever a pump is disassembled for maintenance
- perform an evaluation that includes the following elements and that is required to be submitted to the NRC for review:

  —an analysis of the material properties of the pump casing, including the fracture toughness value

  —a stress analysis for the pump casing

  —a review of the operating history for the pump

  —postulation of an existing reference flaw that has a flaw depth equal to one-quarter the pump casing thickness and a flaw length equal to six times the postulated flaw depth (i.e., a quarter-thickness flaw that has an aspect ratio of 6:1)

  —establishment of stability criteria for the postulated flaw under the governing stress conditions

  —consideration of the effects of thermal aging embrittlement and any other processes or mechanisms that may degrade the properties of the pump casing during service

Pursuant to 10 CFR 54.21(c)(1)(i), in order to demonstrate that the TLAA for the RNP RCP casing will remain valid for the period of extended operation, the applicant stated that WCAP-15363, Revision 0, was issued by Westinghouse to justify use of the Code Case N-481 for the inspections of the RNP RCP casings during the current operating term and that WCAP-15636, Revision 1, was issued to justify use of the Code Case N-481 for the inspections of the RNP RCP casings through the expiration of the extended operating term for RNP. In response to RAI 4.6.1-2, by letter dated May 7, 2003, the applicant submitted Westinghouse proprietary Class 2 report WCAP-15363, Revision 1, "A Demonstration of Applicability of ASME Code Case N-481 to the Primary Loop Pump Casings of H.B. Robinson Unit 2 for the License Renewal Program," for review and approval.

In Section B.4.2 of Appendix B to the LRA, the applicant has stated that the program attributes for the CASS Program are consistent with those specified in AMP XI.M12 of the Generic Aging Lessons Learned (GALL) Report. In AMP XI.M12, it is stated that the existing ASME Section XI requirements, including the alternative requirements of ASME Code Case N-481 for RCP casings, are adequate for all RCP casings and valve bodies. It is also stated in the program element for *Detection of Aging Effects* that, for RCP casings and valve bodies but not susceptible piping, no additional inspection or evaluations are required to demonstrate that the material has adequate fracture toughness.

The staff notes that the ASME Code Section XI Inservice Inspection Program is required to be updated by the applicant and reviewed by the staff every 10-year ISI interval. The acceptability of using Code Case N-481 as an alternative requirement for the ISI of RCP casings will be evaluated by the staff during the review of the applicant's Inservice Inspection Program, which is required to be submitted for NRC approval every 10years. Therefore, it is more appropriate for the staff to review the applicant's fracture mechanics analysis during the staff's review of the applicant's ISI program for the 10-year interval. Based on the consideration discussed above, the staff has determined that there is no need to review the applicant's fracture mechanics analysis as documented in WCAP-15636, Revision 1, to support the use of Code Case N-481 for inservice inspection of RCP casings during the extended period of operation for RNP. Therefore, the staff concludes that a TLAA on the fracture toughness analysis used for supporting the application of Code Case N-481 to the in-service inspections of the RCP casings is not necessary for the RNP LRA, as would otherwise be mandated by 10 CFR 54.21(c)(1).

4.6.1.3 Updated Final Safety Analysis Report Supplement

The applicant provides the following UFSAR Supplement summary description for the LBB analysis of RCS piping In Section A.3.2.5.1 of Appendix A of the LRA:

> WCAP-15628 . . . is a new leak-before-break (LBB) calculation applicable to RNP large bore reactor coolant system (RCS) piping and components that includes allowances for reduction of fracture toughness of cast austenitic stainless steel due to thermal embrittlement during a 60-year operating period. The new analysis meets the requirements for LBB required by 10 CFR 50, Appendix A, General Design Criterion 4, and uses the recommendations and criteria from the NRC Standard Review Plan for LBB evaluations. The new analysis uses the 40-year design basis thermal transients as input for the fracture mechanics analyses. These transients have been shown to be conservative for the 60-year operating period. Therefore, the RCS primary loop piping Leak-Before-Break analysis has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

The applicant provides an UFSAR Supplement summary description for the fracture mechanics analysis of the RNP RCP casing in Section A.3.2.5.2 of Appendix A of the LRA. However, as discussed in Section 4.6.1.2, the UFSAR Supplement for the fracture mechanics analysis of the RNP RCP casing, as documented in WCAP-15363, Revision 1, is not needed for the applicant's LRA, because this analysis will be reviewed during the staff's review of the applicant's Inservice Inspection Program, which will be submitted by the applicant for NRC approval every 10 years.

The applicant's UFSAR Supplement summary description of the TLAA on thermal aging of CASS indicates that the TLAA is in compliance with the requirements of 10 CFR 54.21(c)(1)(ii). This TLAA is based on WCAP-15628, which was issued to demonstrate the validity of the existing 40-year LBB analysis for the period of extended operation for RNP. Therefore, in RAI 4.6.1-3, the staff requested clarification as to whether the UFSAR Supplement summary description for the TLAA of thermal aging of CASS, as given in Section A.3.2.5.1 of Appendix A of the LRA, should indicate compliance with the requirements of 10 CFR 54.21(c)(1)(i) instead of with the requirements of 10 CFR 54.21(c)(1)(ii). In the RAI, the staff also requested that the UFSAR Supplement summary descriptions for the TLAA of LBB analysis for the main RCS loop piping at RNP, as given in Sections A.3.2.5.1 of Appendix A of the LRA, be amended to reflect the information provided in Carolina Power and Light Company's (CP&L's) response to RAI 4.6.1-1, when the response is submitted under oath and affirmation to the NRC document control desk.

In its response to RAI 4.6.1-3, dated April 28, 2003, the applicant clarified that the LBB analysis performed for license renewal incorporates plant-specific material property data and adjustments to material property data to account for changes projected to occur during the license renewal period. Therefore, the LBB analysis has been performed to demonstrate that the margins of safety on acceptable flaw size and stability are acceptable, as projected through the expiration of the extended period of operation for RNP and evaluated against the criterion stated in 10 CFR 54.21(c)(1)(ii).

The UFSAR Supplement summary description on the TLAA for LBB (as given in Section A.3.2.5.1 of Appendix A of the LRA) provides a summary description of the 60-year LBB analysis for the RNP primary loop piping. Since the UFSAR Supplement summary description refers to the applicable safety assessments for this analysis, and since the applicant's response to RAI 4.6.1-3 provides the applicant's basis for assessing this analysis against the criterion stated in 10 CFR 54.21(c)(1)(ii), the staff concludes that this UFSAR Supplement summary description for the TLAA on LBB provides sufficient details as to how the analysis will remain valid, as projected through the expiration of the extended period of operation for RNP.

The staff therefore concludes that the UFSAR Supplement summary description provided in Section A3.2.5.1 of Appendix A of the LRA is acceptable, and RAI 4.6.1-3 is resolved.

4.6.1.4 Conclusions

On the basis of its review, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that for the TLAAs on thermal aging of CASS RCS components, the analyses remain valid through the end of the period of extended operation. The staff also concludes that the UFSAR Supplement contains an appropriate summary description of the TLAA on thermal aging of CASS for the period of extended operation, as required by 10 CFR 54.21(d). Therefore, the staff has concluded that the safety margins established and maintained during the current operating term for the primary reactor coolant loop piping will be maintained until the expiration of the period of extended operation as required by 10 CFR 54.21(c)(1).

4.6.2 Foundation Pile Corrosion

4.6.2.1 Summary of Technical Information in the Application

The applicant identified in the LRA that corrosion of Class 1 structure foundation piles is a TLAA based on the evaluation of the piles for a 40-year corrosion loss. The applicant indicated the original analysis determined that the possibility of active corrosion is minimal and corrosion losses would be negligible because the measured soil resistivity values are so high. This analysis relies on plant-specific data regarding soil resistivity and industry data from NUREG-1557 and EPRI TR-103842.

The RNP UFSAR states that, "Any steel structure in soil (even without the protection afforded by concrete) is progressively less susceptible to corrosion as the electrical resistivity of the soil increases. Soil resistivity measurements taken in August 1958, prior to construction of Unit 1 and as reconfirmed by measurements taken at the construction site in December, 1966, have established that the soil resistivity is so high that the possibility of active corrosion is minimal."

The applicant stated in the LRA that it performed a reanalysis of foundation pile corrosion for license renewal and determined that corrosion losses would continue to remain nonsignificant for the period of extended operation. It concluded that corrosion will not prevent the foundation piles from performing their license renewal intended functions. Furthermore, the applicant stated that its conclusion is consistent with the recommendations and findings of NUREG-1557 and EPRI TRA 103842 and is in accordance with the estimated corrosion losses developed in the original analysis.

4.6.2.2 Staff Evaluation

The staff notes that NUREG-1557, "Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal," identifies corrosion of steel piles as a "Nonsignificant ARDM." It further states, "Steel piles driven in undisturbed soils have been unaffected by corrosion & those driven in disturbed soil experience minor to moderate corrosion to a small area of metal." The staff also reviewed EPRI TRA 103842, "Class I Structures License Renewal Industry Report," and found the following statement:

> Romanoff examined corrosion data from 43 piling installations and on that basis drew some general conclusions regarding the corrosion of driven steel piles. These test installations had pile depths of up to 136 feet and time of exposure varying from 7 to 50 years in a wide variety of soil conditions. Romanoff's review of this data indicates that the type and amount of corrosion observed on steel pilings driven into undisturbed natural soil, regardless of the soil characteristics and properties, is not sufficient to significantly affect the strength of pilings as load bearing structures. The data also indicate that undisturbed soils are so deficient in oxygen at levels a few feet below the ground surface or below the water table, that steel piles are not appreciably affected by corrosion, regardless of the soil type or the soil properties.

Based on the recommendations and findings of NUREG-1557 and EPRI TRA 103842, and results of the applicant's reanalysis of foundation pile corrosion for license renewal, the staff concurs that corrosion losses would continue to remain insignificant for the period of extended operation.

4.6.2.3 Conclusions

The staff reviewed the TLAA regarding the foundation pile corrosion in accordance with the estimated corrosion losses developed in the original analysis and projected in the reanalysis. The conclusion of the reanalysis is consistent with the recommendations and findings of NUREG-1557 and EPRI TRA 103842. The staff finds that the foundation pile corrosion reanalysis results have been projected to the end of the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1).

The staff also reviewed the UFSAR Supplement for this TLAAs and finds that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

4.6.3 Elimination of Containment Penetration Coolers

4.6.3.1 Summary of Technical Information in the Application

As stated in the LRA, in 1995, an evaluation was performed to justify eliminating the need for cooling water flow to the hot pipe containment penetration coolers to the maximum extent possible. As part of this effort, insulation was credited to reduce the temperature of the concrete surrounding the hot pipe penetrations. The performance requirement for the hot pipe penetrations was to maintain the surrounding concrete temperature below 200 °F under normal operating conditions and other long term conditions.

As part of this effort, insulation was credited to reduce the temperature of the concrete surrounding the hot pipe penetrations. The performance requirement for the hot pipe penetrations was to maintain the surrounding concrete temperature below 200 °F under normal operating conditions and other long term conditions.

Residual heat removal (RHR) system penetration S-15 did not require cooling water to be maintained because the concrete temperature around S-15 only exceeded 200 °F during short duration transients and the temperature then was less than 350 °F. In addition, the steady-state temperature without cooling water and continuous RHR flow at 380 °F results in the temperature of the surrounding concrete of approximately 210 °F.

The analysis of concrete temperature determined that the allowable number of cycles of heatup and cooldown, at 40 hours or less per cycle, was 252 cycles. This is the total number of heatup/cooldown cycles the concrete surrounding the S-15 RHR penetration could experience temperatures greater than 200 °F over the balance of plant life figured from 1995. The balance of plant life was projected as 16 years (out of 40 years total plant life) when this calculation was issued in 1995. The allowable number of cycles was compared to the maximum number of heatup/cooldown cycles projected to the end of the period of extended operation.

Because the projected number of cycles for 60-years of operation (120 cycles) is less than the allowed number of cycles for penetration S-15 (252 cycles), the evaluation concluded that the analysis remains conservative and bounding for the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

4.6.3.2 Staff Evaluation

The LRA states that "the analysis of concrete temperature determined that the allowable number of cycles of heatup and cooldown, at 40 hours or less per cycle, was 252 cycles." The LRA further states, "Because the projected number of cycles for 60-years of operation (120 cycles) is less than the allowed number of cycles for penetration S-12 (252 cycles), the evaluation concluded that the analysis remains conservative and bounding for the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(i)." The staff requested the applicant to describe how the analysis was performed and submit the analysis results of concrete properties at the end of 252 cycles. The applicant provided the following response to RAI 4.6.3-2:

- The concrete heatup and cooldown temperatures range from 200 °F to 210 °F during reactor coolant system heatup and 210 °F to 200 °F during reactor coolant system cooldown.

- A thermal fatigue analysis was not performed.

- An evaluation was developed that justified operation with cooling water isolated to the RHR penetrations for a continuous period of approximately 18 months. Cooling water was actually isolated to the RHR penetration for less than 4 months between RFO-15 and -16, leaving the equivalent of 14 months (or 10,080 hours) of "unused" operation with cooling water isolated. The available time of 10,800 hours is equivalent to 252 cycles of heatup/cooldown based on 40 hours per cycle. The 252 cycles of heatup/cooldown bound the projected number of heatup/cooldown cycles (120) and the design heatup/cooldown cycles (200) shown in LRA Section A.2.1.1. The RHR penetrations are subject to high temperatures only during RHR operation, because the RHR system operates only during the heatup and cooldown cycles, not during normal plant operation. No disintegration or physical degradation of the concrete was predicted under the above-described operating conditions. The subject evaluation determined a 25 percent reduction in compressive strength due to temperature effects; however, the reduced compressive strength was still greater than the concrete design strength (3000 psi) that was used in original concrete calculations. The reduced concrete strength (3010 psi) at the penetration was determined to be acceptable. This determination was conservative because the actual concrete compressive strengths from field testing were higher than that used in the evaluation, and the actual temperatures are less than the 277 °F used in the evaluation.

The staff also requested the applicant to clarify whether the conclusion of 252 cycles was obtained from its operating experience. During a teleconference call on June 10, 2003, the applicant stated it had found an analysis result indicating that the temperature in concrete around the containment penetration would always remain below 200 °F. Therefore, the applicant proposed to withdraw this TLAA item in LRA Section 4.6.3. The staff agreed with the applicant's approach of withdrawing this TLAA issue because its analysis results indicate that there is no need for the TLAA. The applicant submitted a letter dated August 14, 2003, to withdraw this TLAA item from the LRA.

4.6.3.3 Conclusions

Since the applicant's analysis results indicate that the concrete temperature around the containment penetration will always remain below 200 °F with the elimination of containment penetration coolers, the applicant has withdrawn this TLAA issue from LRA Section 4.6.3. The staff finds the applicant's response to be acceptable, and Confirmatory Item 4.6.3-1 is closed.

4.6.4 Aging of Boraflex

4.6.4.1 Summary of Technical Information in the Application

In LRA Section 4.6.4, the applicant describes the TLAA for the degradation of Boraflex, which is a boron carbide dispersion, in an elastomeric silicone that is currently used in the spent fuel storage racks as a neutron absorber. The base polymer of Boraflex has been shown to degrade in the borated water environment of the spent fuel pool and under the influence of gamma radiation. Degradation effects include leaching of boron from the polysiloxane matrix, which results in diminished neutron absorption capability of the Boraflex panels.

The applicant references the following NRC INs and Generic Letter (GL) that have identified the concern of the aging of Boraflex neutron-absorbing material:

- IN 87-43, "Gaps in Neutron-Absorbing Material in High-Density Spent Fuel Storage Racks"
- IN 93-70, "Degradation of Boraflex Neutron Absorber Coupons"
- IN 95-38, "Degradation of Boraflex Neutron Absorber in the Spent Fuel Storage Racks"
- GL 96-04, "Boraflex Degradation in Spent Fuel Pool Storage Racks"

In its response to GL 96-04, the applicant commits to continue monitoring and performing analyses of the Boraflex degradation at RNP. In the LRA, Section 4.6.4, the applicant states that it will continue the existing coupon monitoring program as required during the period of extended operation. The applicant also commits to continue monitoring spent fuel pool silica levels and performing silica evaluations.

In the LRA, the applicant has identified aging of Boraflex in the spent fuel pool racks plate as a TLAA. The staff evaluates the TLAA for aging of Boraflex based on the information presented in Section 4.6.4 of the LRA and the applicant's response to the staff's RAI.

4.6.4.2 Staff Evaluation

In LRA Section 4.6.4, the applicant describes the TLAA for the degradation of Boraflex, which is a boron carbide dispersion, in an elastomeric silicone that is currently used in the spent fuel storage racks as a neutron absorber. The base polymer of Boraflex has been shown to degrade in the borated water environment of the spent fuel pool and under the influence of gamma radiation. Degradation effects include leaching of boron from the polysiloxane matrix, which results in diminished neutron absorption capability of the Boraflex panels.

In LRA Section 4.6.4, the applicant stated that prior to the extended period of operation, either an analysis will be performed to permit the elimination of the credit for the Boraflex panels in the spent fuel racks in determining K_{eff} for the spent fuel array, or credit will be taken for the current Boraflex monitoring program which will be evaluated against the GALL Report.

In its April 28, 2003, letter, in Commitment No. 47, the applicant stated that the current Boraflex monitoring program will be evaluated against the requirements for a license renewal AMP, and the results of the evaluation will be documented in the UFSAR. The applicant may withdraw this commitment if its planned analysis to credit soluble boron successfully eliminates credit for the Boraflex sheets in the spent fuel racks.

In its response to RAI 4.6.4-1 dated April 28, 2003, the applicant stated that it currently intends to request a technical specifications (TS) change to eliminate the credit for the Boraflex monitoring program. The proposed TS change is expected to be consistent with similar changes that have been approved for other licensees and represents a reasonable approach for resolution of Boraflex degradation. The applicant also stated that the revised analysis is expected to credit soluble boron and fuel assembly burnup in the reactivity analysis and is based on an approved methodology. Upon NRC approval of the proposed TS change, the license renewal intended function provided by Boraflex panels will no longer be applicable, and the current Boraflex monitoring procedure will be terminated.

By letter dated May 28, 2003, the applicant submitted for staff review a license amendment to change the TS to credit a combination of soluble boron and controlled fuel loading patterns and therefore remove Boraflex monitoring procedures. The staff asked for confirmation that the license amendment to remove the requirements to credit the Boraflex panels from the RNP TS has been approved and that the Boraflex panels will no longer be needed to maintain the K_{eff} for the geometry of the spent fuel rods stored in the spent fuel pool within acceptable levels. As part of this confirmatory item, the staff asked the applicant to provide a reference regarding the staff's safety evaluation to CP&L approving the license amendment for the Boraflex panels. The staff required a commitment statement from the applicant, saying that, "if the NRC staff denies the applicant's request to eliminate and modify, if necessary, the current boraflex monitoring procedure to satisfy the NRC's requirement for the license renewal Boraflex TLAA, and the results of the evaluation will be documented in the UFSAR and the Boraflex monitoring TLAA will be implemented as a part of license renewal." This is Confirmatory Item 4.6.4-1. By letter dated December 22, 2003, License Amendment 198, the staff approved the applicant's request to eliminate the need to credit the Boraflex neutron absorbing material for reactivity control in the spent fuel storage pool. In place of Boraflex material (i.e., panels), the staff approved the applicant's request to take credit for a combination of soluble boron and controlled fuel loading patterns in the spent fuel pool to maintain the required subcriticality margins in the spent fuel storage pool. On the basis of License Amendment 198, the staff finds that Confirmatory Item 4.6.4-1 is closed. In addition, the applicant may eliminate its Commitment No. 47 and eliminate any discussion in the RNP UFSAR regarding the Boraflex TLAA or the Boraflex monitoring program.

4.6.4.3 Updated Final Safety Analysis Report Supplement

As indicated in the applicant's response to RAI 4.6.4-1, the applicant has indicated that it plans to stop taking credit for the Boraflex program and that, therefore, it will not be necessary for the applicant to include a summary description of the Boraflex TLAA in the UFSAR Supplement.

On the basis of License Amendment 198, issued on December 22, 2003, the applicant may at its own volition, eliminate the UFSAR Supplement summary description for the TLAA for the boraflex panels .

4.6.4.4 Conclusions

As discussed in License Amendment 198, issued on December 22, 2003, the staff approved the applicant's request to credit soluble boron and controlled fuel loading patterns to maintain the required subcriticality margins in the spent fuel storage pool. The staff also approved the applicant's request to eliminate the need to credit the Boraflex neutron absorbing material for reactivity control in the spent fuel storage pool. The Boraflex panels will no longer be used. Therefore, it is not necessary for the applicant to include a TLAA on degradation of Boraflex as part of the LRA.

5 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The NRC staff issued its safety evaluation report (SER) with open items related to the renewal of operating licences for H.B. Robinson Steam Electric Plant, Unit 2, on August 25, 2003. On September 30, 2003, the applicant presented its license renewal application, and the staff presented its review findings, to the ACRS Plant License Renewal Subcommittee. The staff reviewed the applicant's responses to SER open and confirmatory items and completed its review of the license renewal application. The staff's evaluation is documented in an SER that was issued by letter dated January 20, 2004.

During the 510th meeting of the ACRS, March 3-6, 2004, the ACRS completed its review of the Robinson license renewal application and the NRC staff's SER. The ACRS documented its findings in a letter to the Commission dated March 18, 2004. A copy of this letter is provided on the following pages of this SER Chapter.

March 18, 2004

The Honorable Nils J. Diaz
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT 2

Dear Chairman Diaz:

During the 510th meeting of the Advisory Committee on Reactor Safeguards, March 3-6, 2004, we completed our review of the License Renewal Application (LRA) for the H. B. Robinson Steam Electric Plant, Unit 2, known as Robinson Nuclear Plant, and the related final Safety Evaluation Report (SER) prepared by the NRC staff. Our Plant License Renewal Subcommittee reviewed this application and the staff's initial SER during a meeting on September 30, 2003. During these reviews, we had the benefit of discussions with representatives of the NRC staff and the Carolina Power and Light Company (CP&L). We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The programs instituted and committed to by CP&L to manage age-related degradation are appropriate and provide reasonable assurance that the Robinson Nuclear Plant can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public.

2. The CP&L application for renewal of the operating license for Robinson Nuclear Plant should be approved.

BACKGROUND AND DISCUSSION

This report fulfills the requirements of 10 CFR 54.25, which states that the ACRS should review and report on all license renewal applications. In its application, CP&L requested renewal of the operating license for the Robinson Nuclear Plant for 20 years beyond the current license term, which expires on July 31, 2010. Robinson Nuclear Plant is a Westinghouse-designed, three-loop, pressurized-water reactor rated at 2,339 megawatts-thermal (MWt) with replacement steam generators installed in 1984. It is located adjacent to Unit 1 of the H.B. Robinson Steam Electric Plant, a coal fired steam power plant. The LRA was prepared in accordance with NUREG 1801, The Generic Aging Lessons Learned Report.

The Robinson Nuclear Plant final SER documents the results of the staff's review of the information submitted by the applicant, including commitments that were necessary to resolve open and confirmatory items identified by the staff in the initial SER and those identified during onsite NRC inspections and audits. In particular, the staff reviewed the completeness of the applicant's identification of structures, systems, and components that are within the scope of license renewal, the integrated plant assessment process, the identification of the plausible aging mechanisms associated with passive long-lived components, the adequacy of the aging management programs, and the identification and assessment of time limited aging analyses (TLAAs) requiring review.

Several design features that are unique to Robinson Nuclear Plant, such as grouted tendons, containment liner insulation, and some shared systems with a fossil unit, were identified. All shared systems are included in the scope of the LRA.

Robinson Nuclear Plant site has aggressive ground water due to a low pH. The applicant has committed to inspect the dam spillway and the intake structures every 10-years and will also perform opportunistic inspections of inaccessible concrete structures.

The pressurizer spray head is not in scope and, given its importance for cooldown, we questioned its omission. The applicant responded that the accident-basis analysis for plant operation does not include pressurizer spray so its exclusion is permissible. The applicant further stated that degradation of the nozzle would be noticed during normal operation.

The applicant stated that the plant has 37 existing aging management programs, of which 27 have been enhanced, and 10 new programs have been added. Several of these programs have yet to be developed and they will require NRC approval. As with other applicants, we encouraged CP&L to establish a schedule for implementing these commitments well ahead of the beginning of the license renewal period so as not to place an unreasonable demand on both the applicant and NRC resources. CP&L has committed to have 18 of these programs in place by mid 2004.

Time limited aging analyses were performed by the applicant to evaluate reactor vessel neutron embrittlement, metal fatigue for certain components, environmental qualification, grouted concrete containment tendon prestress, boraflex aging, and foundation pile corrosion. All these issues have been resolved satisfactorily. In the case of reactor vessel neutron embrittlement, the staff performed independent calculations and found the applicant's analysis acceptable.

We agree with the staff's conclusion that all open and confirmatory items have been closed appropriately. We conclude that on the basis of our review of the final SER, the LRA, and the NRC inspection and audit reports, there are no issues, specifically related to the matters described in 10 CFR 54.29(a)(1) and (a)(2), that preclude renewal of the operating license for the plant. The programs instituted and committed to by CP&L to manage age-related degradation are appropriate and provide reasonable assurance that the plant can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public. The CP&L application for renewal of the operating license for the Robinson Nuclear Plant should be approved.

Sincerely,

/RA/

Mario V. Bonaca
Chairman

References:
1. U.S. Nuclear Regulatory Commission, "Final Safety Evaluation Report Related to H. B. Robinson Steam Electric Plant, Unit 2," January 2004.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report with open items Related to the License Renewal of the H. B. Robinson Steam Electric Plant, Unit 2," August 2003.
3. Letter from J. W. Moyer, Carolina Power and Light Company, to the U.S. Nuclear Regulatory Commission, Subject: Application for Renewal of Operating License, H. B. Robinson Steam Electric Plant, Unit 2, June 14, 2002.
4. NRC Inspection Report 50-261/03-08, H.B. Robinson Steam Electric Plant, May 8, 2003.
5. NRC Inspection Report 50-261/03-09, H.B. Robinson Steam Electric Plant, July 31, 2003.

6 CONCLUSIONS

The staff reviewed Robinson Nuclear Plant (RNP) license renewal application in accordance with Commission regulations and NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," dated July, 2001. In 10 CFR 54.29, the staff identifies the standards for issuance of a renewed license.

On the basis of its evaluation of the application as discussed above, the staff has determined that the requirements of 10 CFR 54.29 have been met.

The staff notes the requirements of Subpart A of 10 CFR Part 51 are documented in NUREG-1437, Supplement 13, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," dated December 12, 2003.

Appendix A: Commitment Listing

During the review of the RNP LRA by the NRC staff, the applicant made commitments to provide aging management programs (AMPs) to manage aging effects of structures and components (SCs) prior to the period of extended operation. The following table lists these commitments, along with the implementation schedule and the source of the commitment.

ITEM NUMBER	COMMITMENT	UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) SUPPLEMENT LOCATION	IMPLEMENTATION SCHEDULE	SOURCE
1. Quality Assurance Program	Quality Assurance Program. Existing program is credited. See note below.	A.3.1		
2. 10 CFR 54.37(b) Requirements	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.	A.3.1	Following issuance of renewed license	Request for Additional Information (RAI) 2.1.1-2

3. NUREG-1801 GALL Report	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 AMP with programs defined in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report." 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."	A.3.1	Prior to the period of extended operation	RAI B.1-1
4. ASME Section XI, Subsection IWB, IWC, and IWD Program	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	Water Chemistry Program. Existing program is credited. No changes required. See note below.	A.3.1.2		
6. Reactor Head Closure Studs Program	Reactor Head Closure Studs Program. Existing program is credited. No changes required. See note below.	A.3.1.3		
7. Steam Generator Tube Integrity Program	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	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	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	10 CFR 50, Appendix J Program. Existing program is credited. See note below.	A.3.1.7		
11. Flux Thimble Eddy Current Inspection Program	Flux Thimble Eddy Current Inspection Program. Existing program is credited. See note below.	A.3.1.8		
12. Fire Protection Program	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.	A.3.1.9	Prior to the period of extended operation	LRA, Appendix B, Section B.3.1

13. Boric Acid Corrosion Program	The scope of the Boric Acid Corrosion Program will be expanded to (1) ensure that 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.	A.3.1.10	Prior to the period of extended operation	LRA, Appendix B, Section B.3.2
14. Flow-Accelerated Corrosion Program	The Flow-Accelerated Corrosion Program will be modified to (1) include additional components potentially susceptible to flow-accelerated corrosion and/or erosion, and (2) clarify when condition reports shall be initiated.	A.3.1.11	Prior to the period of extended operation	LRA, Appendix B, Section B.3.3
15. Bolting Integrity Program	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.	A.3.1.12	Prior to the period of extended operation	LRA, Appendix B, Section B.3.4
16. Open-Cycle Cooling Water System Program	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.	A.3.1.13	Prior to the period of extended operation	LRA, Appendix B, Section B.3.5

17. Inspection of Overhead Heavy Load and Light Load Handling	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.	A.3.1.14	Prior to the period of extended operation	LRA, Appendix B, Section B.3.6 RAI B.3.6-2

18. Fire Water System Program	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 inservice 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	A.3.1.15	As noted in the commitment	LRA, 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

	of Concerned Scientists) Proposed Staff Guidance on Aging Management of Fire Protection Systems for License Renewal, January 28, 2002.			
19. Buried Piping and Tanks Surveil-lance Program	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.	A.3.1.16	Prior to the period of extended operation	LRA, Appendix B, Section B.3.8
20. Above Ground Carbon Steel Tanks Program	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.	A.3.1.17	Prior to the period of extended operation	LRA, Appendix B, Section B.3.9
21. Fuel Oil Chemistry Program	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.	A.3.1.18	Prior to the period of extended operation	LRA, Appendix B, Section B.3.10

22. Reactor Vessel Surveil-lance Program	Reactor Vessel Surveillance Program administrative controls will be revised to require surveillance test samples to be stored in lieu of optional disposal.	A.3.1.19	Prior to the period of extended operation	LRA, Appendix B, Section B.3.11
23. Buried Piping and Tanks Inspection Program	The Buried Piping and Tanks Inspection Program will be enhanced to (1) require that an appropriate as-found pipe coating and material condition inspection is performed whenever buried piping within the scope of this program is exposed, (2) add precautions to ensure backfill with material that is free of gravel or other sharp or hard material that can damage the coating, (3) require that the 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.	A.3.1.20	Prior to the period of extended operation	LRA, Appendix B, Section B.3.12
24. ASME Section XI, Subsection IWE Program	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.	A.3.1.21	Prior to the period of extended operation	LRA, Appendix B, Section B.3.13

25. ASME Section XI, Subsection IWL Program	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.	A.3.1.22	Prior to the period of extended operation	LRA, 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	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) to 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, and (8) include trending requirements for structures based on aggressive ground water and lake water.	A.3.1.23	Prior to the period of extended operation	LRA, 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. Dam Inspection Program	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..	A.3.1.24	Prior to the period of extended operation	LRA, 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 Confirmatory Item 3.5-1
28. Systems Monitoring Program	Systems Monitoring Program administrative controls will be enhanced to (1) include aging effects identified in the aging management reviews (AMRs), (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.	A.3.1.25	Prior to the period of extended operation	LRA, Appendix B, Section B.3.17 RAI B.3.17-1

29. Preventive Main-tenance Program	Preventive Maintenance Program administrative controls will be enhanced to (1) include aging effects/mechanisms identified in the AMRs and (2) incorporate specific aging management activities identified in the AMRs into the program.	A.3.1.26	Prior to the period of extended operation	LRA, Appendix B, Section B.3.18
30. Metal Fatigue of Reactor Coolant Pressure Boundary (Fatigue Monitoring Program)	The Fatigue Monitoring Program load/unload transient limit will be reduced to provide the margin needed for consideration of reactor water environmental effects.	A.3.1.27	Prior to the period of extended operation	LRA, Appendix B, Section B.3.19

31. Nickel-Alloy Nozzles and Penetrations Program	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 systematically assess for implementation applicable programmatic enhancements, 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	LRA, Appendix B, Section B.4.1 RAI B.4.1-1 RNP-RA/03-0154

32. Thermal Aging Embrittle-ment and Cast Austenitic Stainless Steel (CASS) Program	The Thermal Aging Embrittlement and 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.	A.3.1.29	Prior to the period of extended operation	LRA, Appendix B, Section B.4.2

33. Pressurized Water Reactor 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.	A.3.1.30	As noted in the commitment	LRA, Appendix B, Section B.4.3 RAI B.4.3-2

34. One-Time Inspection Program	One-Time Inspection Program activities consist of inspections of the following. (1) The 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) The 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, nondestructive 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 material condition. (6) The diesel fire pump fuel oil tank. (7) Steam Generator feed ring/J-nozzles.	A.3.1.31	Prior to the period of extended operation	LRA, Appendix B, Section B.4.4 RAI 3.5.1-1 RAI B.3.10-6 Open Item 2.3.1.6-1

35. Selective Leaching of Materials Program	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.	A.3.1.32	Prior to the period of extended operation	LRA, Appendix B, Section B.4.5
36. Non-Environmentally Qualified Insulated Cables and Connections Program	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.	A.3.1.33	As noted in the commitment Prior to the period of extended operation	LRA, Appendix B, Section B.4.6 RAI 3.6.1-2 B4.6-3 Confirmatory Item 3.6.2.3.1.2-1

37. Aging Management Program for Non-EQ Electrical Cables Used in Instrumentation Circuits	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.	A.3.1.34	As noted in the commitment Prior to the period of extended operation	RAI 3.6.1-2 RAI B.4.6-3
38. Aging Management Program for Neutron Flux Instrumentation Circuits	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.	A.3.1.35	As noted in the commitment Prior to the period of extended operation	RAI 3.6.1-2 RAI B.4.6.-3

39. Aging Management Program for Fuse Holders	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.	A.3.1.36	As noted in the commitment Prior to the period of extended operation	RAI 2.5.2-1
40. Aging Management Program for Bus Duct	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 to all nonsegregated 4.16 kV and 480 V bus duct within the scope of license renewal. The program has a 10-year frequency.	A.3.1.37	As noted in the commitment Prior to the period of extended operation	RAI 2.5.2-2
41. Environmental Qualification of Electric Equipment Program	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.	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 Embrittle-ment	Time-Limited Aging Analysis (TLAA) - Reactor Vessel Neutron Embrittlement. Existing program is credited. 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 time 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.	A.3.2.2	As noted in the commitment	LRA, Section 4.3 RAI 4.3-2 RAI 4.3-7 RAI 4.3-10

	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.			
44. TLAA - Environ-mental Quali-fication	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 qualifications or they will be replaced.	A.3.2.3	As noted in the commitment	LRA, 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.	A.3.2.4	As noted in 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 supersedes the proposed changes shown on LRA Page A-6 for UFSAR Section 3.8.1.4.7.	A.3.2.4	Prior to the period of extended operation	RAI 4.5-3

47. TLAA - Aging of Boraflex in Spent Fuel Pool	Prior to the period of extended operation, the Boraflex Monitoring Program will be modified to (1) include neutron attenuation testing, called blackness testing, to determine gap formation in Boraflex panels; (2) include trending the results for silica levels by using the EPRI RACKLIFE predictive code or equivalent, and (3) include measurements of boron areal density by techniques such as the BADGER device, RNP has requested, by letter dated May 28, 2003, Serial: RNP-RA/03-0038, an amendment to the Technical Specifications to eliminate the need to credit Boraflex neutron absorbing material for reactivity control. The Boraflex Monitoring Program will be eliminated upon NRC approval of this amendment or upon implementation of another option(such as re-racking the spent fuel pool) which eliminates the need to credit Boraflex for reactivity control. Revised commitment	A.3.2.8	Prior to the period of extended operation	LRA, Section 4.6.4 RNP-RA/03-0154

NOTE: Not listed in this table. 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.

Appendix B: Chronology

This appendix contains a chronological listing of routine licensing correspondence between the U.S. Nuclear Regulatory Commission (NRC) staff and Carolina Power & Light Company (CP&L). This appendix also contains other correspondence regarding the NRC staff's review of the H.B. Robinson Power Station, Unit 2 (under docket No. 50-261).

June 14, 2002	In a letter (signed by J.W. Moyer), CP&L submitted its application to renew the operating license of RNP, Unit 2. In its submittal, CP&L provided an original signed hard copy of the application and 81 additional electronic copies of applications on CDs. (ADAMS Accession Number: ML021690663)
June 14, 2002	In a letter (signed by B.L. Fletcher III), CP&L submitted eight sets of boundary drawings to the NRC.
July 15, 2002	In a letter (signed by S.K. Mitra), the NRC informed CP&L that the NRC had received its application to renew the operating license of H.B. Robinson Power Station Unit 2, June 17, 2002, and that Mr. Mitra was appointed as the project manager for the RNP LRA. (ADAMS Accession Number: ML021970121)
Aug 1, 2002	In a letter (signed by R. Prato), the applicant responded to question originated by Mr. S.K. Mitra regarding HVAC damper housings and structural sealant identification in the RNP LRA. (ADAMS Accession Number: ML022140212)
Aug 6, 2002	In a letter (signed by S.K. Mitra), a summary of meeting between the NRC staff and CP&L representatives to discuss the RNP LRA. (ADAMS Accession Number: ML 022180732)
Aug 8, 2002	In a letter (signed by S.K. Mitra), a summary of conference call between the NRC staff and CP&L representatives to discuss the RNP LRA. (ADAMS Accession Number: ML 022200373)
Aug 12, 2002	In a letter (signed by P.T. Kuo), the NRC published that CP&L provided enough information for the acceptance and docketing to the RNP LRA. (ADAMS Accession Number: ML 022240731)
Aug 14, 2002	In a letter (signed by B.L. Fletcher), CP&L provided additional information to support the NRC's review of the RNP LRA. (ADAMS Accession Number: ML 022310271)
Aug 16, 2002	In the *Federal Register*, a "Notice of Acceptance for Docketing of the Application and Notice of Opportunity for a Hearing Regarding H.B. Robinson Nuclear Plant LRA." is published.

Sept 13, 2002	In a letter (signed by S.K. Mitra), to CP&L representatives asking them to provide a revised schedule for the RNP LRA. (ADAMS Accession Number: ML 022590085)
Oct 23, 2002	In a letter (signed by J.W. Moyer), CP&L provided additional information concerning the Interim Staff Guidance (ISG) regarding fire protection system aging management, station blackout, aging management of concrete components, and 10 CFR 54.4(a)(2) in support to the NRC's review of the RNP LRA. (ADAMS Accession Number: ML 023020463)
Nov 06, 2002	In a letter (signed by B.L. Fletcher), CP&L provided CD-ROM as a review tool which contains information concerning the mechanical and civil systems to facilitate the NRC's review of the RNP LRA. (ADAMS Accession Number: ML 023170509)
Nov 20, 2002	In a letter (signed by S.K. Mitra) to CP&L representatives asking them to provide a revised schedule for the review of the RNP LRA. (ADAMS Accession Number: ML 023240495)
Nov 20, 2002	In a letter (signed by S.K. Mitra), a summary of meetings between the NRC staff and CP&L representatives to discuss the RNP LRA. (ADAMS Accession Number: ML 023240516)
Jan 02, 2003	In a letter (signed by B.L. Fletcher), CP&L provided response to request for additional information regarding "severe accident mitigation alternatives analysis" in support of the NRC's review of the RNP LRA. (ADAMS Accession Number: ML 030060112)
Jan 15, 2003	In a letter (signed by C.T. Baucom), CP&L provided a schedule to respond to NRC's Request No. 9 regarding "severe accident mitigation alternatives analysis" in support of the NRC's review of the RNP LRA. (ADAMS Accession Number: ML 030220231)
Jan 20, 2003	In a letter (signed by B.L. Fletcher), CP&L provided response to NRC's Request No. 9 regarding "severe accident mitigation alternatives analysis" in support of the NRC's review of the RNP LRA. (ADAMS Accession Number: ML 030220231)
Feb 11, 2003	In a letter (signed by S.K. Mitra), the NRC staff issued requests for additional information (RAIs) regarding the RNP LRA. (ADAMS Accession Number: ML030420424)
Feb 21, 2003	In a letter (signed by S.K. Mitra), the NRC staff issued a modification to the February 11, 2003, RAIs to include additional requests related to the RNP LRA. (ADAMS Accession Number: ML030550625)

Mar 04, 2003	In a letter (signed by C.T. Baucom), CP&L submitted a request for exemption from 10 CFR 54.21(b) which allows the submittal of a single LRA amendment for RNP. (ADAMS Accession Number: ML 030650477)
Mar 05, 2003	In a letter (signed by S.K. Mitra), a summary of meetings between the NRC staff and CP&L representatives to discuss the draft requests for additional information (RAIs) for the RNP LRA. (ADAMS Accession Number: ML 030640168)
Apr 28, 2003	In a letter (signed by C.T. Baucom), CP&L submitted a response to the RAI regarding application for renewal of operating license. (ADAMS Accession Number: ML 031210062)
May 7, 2003	In a letter (signed by C.T. Baucom), CP&L submitted proprietary documents as part of the response for additional information in support of license renewal application. (ADAMS Accession Number: ML 031320378)
May 8, 2003	In a letter (signed by C.A. Casto), the NRC issued an inspection report (NRC Inspection Report 50-261/03-08) that discusses the examination of the process of scoping and screening of plant equipment to select equipment subject to an aging management review in support of the LRA. (ADAMS Accession Number: ML031320011)
May 15, 2003	In a letter (signed by C.T. Baucom), CP&L submitted a withdrawal of request for exemption from 10 CFR 54.21(b). (ADAMS Accession Number: ML 031390022)
May 20, 2003	In a letter (signed by S.K. Mitra), the NRC held a meeting with representatives from CP&L to discuss and clarify the final RAI in support of LRA. (ADAMS Accession Number: ML 0313280379)
June 13, 2003	In a letter (signed by C. T. Baucom), CP&L submitted supplemental information regarding the LRA and in support of the answers to the RAI. (ADAMS Accession Number: ML 0313280379)
June 25, 2003	In a letter (signed by C. T. Baucom), CP&L submitted the annual review of the RNP current licensing basis (CLB). (ADAMS Accession Number: ML 031820165)
July 24, 2003	In a letter (signed by C. T. Baucom), CP&L submitted the comments on the draft supplemental environmental impact statement.
July 30, 2003	In a letter (signed by S.K. Mitra) the staff informed CP&L that the NRC had received and plans to withhold from the public the proprietary version of Westinghouse Electric Company's Topical Reports WCAP-15628 and WCAP-15363, Revision 1. (ADAMS Accession Number: ML 032120706)

July 31, 2003	In a letter (signed by S.K. Mitra), a summary of conference call between the NRC staff and CP&L representatives to discuss the responses to a request for additional information for the RNP LRA. (ADAMS Accession Number: ML 032120368)
July 31, 2003	In a letter (signed by S.K. Mitra), a summary of conference call between the NRC staff and CP&L representatives to clarify final response to a request for additional information for the RNP LRA. (ADAMS Accession Number: ML 032130258)
July 31, 2003	In a letter (signed by C.A. Casto), the NRC issued an inspection report (NRC Inspection Report 50-261/03-09) that discusses the evaluation of aging management programs in support of the RNP LRA. (ADAMS Accession Number: ML032130040)
August 12, 2003	In a letter (signed by S.K. Mitra), the NRC issued an audit report that discusses the verification of the consistencies between the applicant's aging management programs (AMPs) described in the RNP LRA and the AMPS in NUREG-1801, "Generic Lessons Learned (GALL) Report." (ADAMS Accession Number: ML032250040)
August 14, 2003	In a letter (signed by J.F. Lucas), CP&L submitted a letter that lists revisions made to the RNP license renewal commitments included in the original LRA. (ADAMS Accession Number: ML 032300478)
August 15, 2003	In a letter (signed by R.L. Emch), a summary of meeting between the NRC staff, CP&L representatives, and the general public to discuss the environmental review and gather comments on the draft supplemental environmental impact statement (DSEIS) in support of the RNP, Unit 2 license renewal process. (ADAMS Accession Number: ML 032270603)
August 25, 2003	In a letter (signed by P.T. Kuo), the NRC staff issued a safety evaluation report with open items that discusses the staff safety evaluations in support of the RNP, Unit 2 license renewal process. (ADAMS Accession Number: ML 032370382)
September 2, 2003	In a letter (signed by S.K. Mitra) the NRC issued a revised schedule for the review of the RNP LRA. (ADAMS Accession Number: ML 032460755)
September 3, 2003	In a letter (signed by S.K. Mitra), a summary of meetings between the NRC staff and CP&L representatives to clarify final response to the request for additional information for the RNP LRA. (ADAMS Accession Number: ML 032461542)
September 16, 2003	In a letter (signed by J.F. Lucas), CP&L submitted a letter that provides responses to the RNP open and confirmatory items listed in the SER with open items issued on August 25, 2003. (ADAMS Accession Number: ML 032650884)

October 9, 2003	In a letter (signed by J.F. Lucas), CP&L submitted a letter that provides annual update of changes in the current licensing basis that affect the license renewal application submitted June 14, 2002. (ADAMS Accession Number: ML 032880498)
November 7, 2003	In a letter (signed by C.T. Baucom), CP&L submitted a letter that provides technical comments on the safety evaluation report with open items published August 25, 2003. (ADAMS Accession Number: ML 0331400150)
November 12, 2003	In a letter (signed by J.F. Lucas), CP&L submitted a letter that provides confirmation that PEC is developing guidance regarding Archaeological, Cultural, and Historic (AC&H) Resources to be incorporated into the Environmental Compliance Manual prior to the end of 2004. (ADAMS Accession Number: ML033180546)
December 22, 2003	In a letter (signed by C.P. Patel), the NRC issued Amendment No. 198 regarding the changes in Technical Specifications on Boraflex neutron-absorbing material. (ML033560622)
March 18, 2003	In a letter (signed by M. Bonaca), the Advisory Committee on Reactor Safeguards provided its conclusions and recommendations on the renewal of the operating license for H.B. Robinson, Unit 2.

APPENDIX C: PRINCIPAL CONTRIBUTORS

LICENSE RENEWAL AND ENVIRONMENTAL IMPACTS PROGRAM

NAME	RESPONSIBILITY
Pao-Tsin Kuo	Program Director
Sam Lee	Section Chief
S.K. Mitra	Project Manager
Sonary Chey	Clerical Support
Nina M. Barnett	Clerical Support
Thelma Davis	Clerical Support
Hai-Boh Wang	Technical Support
Mario G. Cora	Backup Project Manager
Kimberly A. Corp	Technical Support
Tomeka Terry	Technical Support
Kamishan O. Martin	Technical Support
Brian Lee	Technical Support
Quynh Nguyen	Technical Support
Zahira Cruz-Perez	Technical Support
Melissa Jenkins	Administrative Support
Yvonne Edmonds	Administrative Support
Antoinette Walker	Clerical Support
Jessie Delgado	Clerical Support
Gwen Davis	Clerical Support

PRINCIPAL TECHNICAL CONTRIBUTORS

NAME	RESPONSIBILITY
Amar Pal	Electrical Engineering
Arnold Lee	Mechanical Engineering
Billy Rogers	DIPM\IQPB
Caudle Julian	Inspection Support
Clifford G. Munson	Structural Engineering
Daniel Frumkin	Fire Protection Engineering
Daniele Oudinot	DSSA/SPLB
David Jeng	Civil Engineering
David H. Shum	Plant Systems Engineering
Desai Benoi	Resident Inspector
Greg Galletti	Quality Assurance
Hansrai Ashar	Civil Engineering
Harold Walker	Plant Systems Engineering
Jai Rajan	Mechanical Engineering
James Medoff	Materials Engineering
James Strnisha	Mechanical Engineering
John Lehning	DSSA/APLB
John Ma	Civil Engineering
John Tsao	Materials Engineering
Kenneth Chang	Mechanical Engineering

Mark Hartzman	DE/EMEB
Pei-Ying Chen	Mechanical Engineering
Prakash Patnaik	Materials Engineering
Raj Goel	DSSA/SPLB
Ralph Architzel	Plant Systems Engineering
Samuel Miranda	DSSA/SRXB
Steve Jones	Plant Systems Engineering
Stewart Bailey	Mechanical Engineering
Vincent S. Klco	DNS/SPES
Yamir Diaz	DE/EMCB
Yong Kim	Structural Engineering
Yueh-Li Li	Mechanical Engineering

APPENDIX D: REFERENCES

American Society of Mechanical Engineers (ASME)

ASME Boiler and Pressure Vessel Code Requirements and Code Cases

ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" (acceptable editions endorsed by 10 CFR 50.55a are those through the 1995 Edition, inclusive of the 1996 Addenda).

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ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWD, Requirements for Class 3 Components of Light-Water Cooled Power Plants.

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Carolina Power and Light Company (CP&L)

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This safety evaluation report (SER) documents the technical review of the H.B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, known as Robinson Nuclear Plant (RNP), license renewal application (LRA) by the U.S.Nuclear Regulatory Commission (NRC) staff. Letter dated June 14, 2002, Carolina Power & Light Company (CP&L or the applicant) submitted the LRA for RNP in accordance with Title 10 ofthe Code of Federal Regulations Part 54 (10 CFR Part 54 or the Rule). RNP is requesting renewal of the operating license for Unit 2 (license number DPR-23) for a period of 20 years beyond the current expiration date of midnight, July 31, 2010. The construction permit for RNP was issued by the NRC on April 13, 1967, and the operating license was issued September 23, 1970, pursuant to Section 104b of the Atomic Energy Act of 1954, as amended.

RNP is adjacent to Unit 1 of the H.B. Robinson Steam Electric Plant (SEP), a coal-fired steam power plant. The plant is located on the edge of Lake Robinson, a man-made lake in Darlington and Chesterfield Counties, South Carolina. RNP is a pressurized light-water moderated and cooled system. The nuclear plant incorporates a three-loop closed-cycle, pressurized water, nuclear steam supply system (NSSS) designed by Westinghouse Electrical Corporation and licensed to generate 2339 MW-thermal, or approximately 769 MW-electric.

This SER presents the status of the staff's review of information submitted to the NRC through January 21, 2004. In it's SER issued August 25, 2003, the staff has identified open and confirmatory items that had to be resolved before the staff could make a final determination on the application. These items and their resolutions are summarized in Sections 1.5 and 1.6 of this report. The staff's final conclusion of it's review of the RNP LRA can be found in Section 6 of this SER.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

SER, safety evaluation report, H. B. Robinson Steam Electric Plant (HBRSEP), license renewal application (LRA), open items, confirmatory items, RAI, request for additional information, 10 CFR Part 54, Carolina Power and Light Company, license number DPR-23

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