SECTION 4

TIME-LIMITED AGING ANALYSES

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 beltine 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," 6^{th} Edition. According to the applicant, EOCI-61 did not require a reduction in allowable stresses for fatigue. However, the AISC 6^{th} 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.

<u>Spent Fuel Cask Crane</u>

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

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

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.

The RNP LBB Analysis for the Main Loop RCS Piping and Components

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.