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NUREG-1916, Vol. 2

Safety Evaluation Report

Related to the License Renewal of Shearon Harris Nuclear Power Plant, Unit 1

Docket No. 50-400

Carolina Power & Light Company

Office of Nuclear Reactor Regulation

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ABSTRACT

This safety evaluation report (SER) documents the technical review of the Shearon Harris Nuclear Power Plant (HNP), Unit 1, license renewal application (LRA) by the United States (US) Nuclear Regulatory Commission (NRC) staff (the staff). By letter dated November 14, 2006, Carolina Power & Light (CP&L) Company, doing business as Progress Energy Carolinas, Inc., submitted the LRA in accordance with Title 10, Part 54, of the *Code of Federal Regulations*, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." CP&L requests renewal of the Unit 1 operating license (Facility Operating License Number NPF-63) for a period of 20 years beyond the current expiration at midnight October 24, 2026, for Unit 1.

HNP is located approximately 16 miles southwest of Raleigh, NC., and 15 miles northeast of Sanford, NC. The NRC issued the construction permit for Unit 1 on January 27, 1978, and operating license on January 12, 1987. Unit 1 is of a dry ambient pressurized water reactor design. Westinghouse supplied the nuclear steam supply system and Daniel International originally designed and constructed the balance of the plant with the assistance of its agent, Ebasco. The Unit 1 licensed power output is 2900 megawatt thermal with a gross electrical output of approximately 900 megawatt electric.

This SER presents the status of the staff's review of information submitted through July 21, 2008, the cutoff date for consideration in the SER. The staff identified an open item and two confirmatory items that were resolved before the staff made a final determination on the application. SER Sections 1.5 and 1.6 summarizes these items and their resolution. Section 6.0 provides the staff's final conclusion on the review of the HNP LRA.

TABLE OF CONTENTS

Appendices

Tables

ABBREVIATIONS

ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Document Access and Management System
AERM	aging effect requiring management
AFW	auxiliary feedwater
AMP	aging management program
AMR	aging management review
AMSAC	ATWS mitigating system actuation circuitry
ANSI	American National Standards Institute
ART	adjusted reference temperature
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
BMV	bare metal visual
BOP	balance of plant
BTP	Branch Technical Position
BTRS	boron thermal regeneration system
BWR	boiling water reactor
CASS	cast austenitic stainless steel
CCW	component cooling water
CFR	*Code of Federal Regulations*
CI	confirmatory item
CIV	containment isolation valve
CLB	current licensing basis
CMAA	Crane Manufacturers Association of America
CP&L	Carolina Power & Light Company, a Progress Energy Company
CRDM	control rod drive mechanism
CSI	charging and safety injection
CSIP	charging and safety injection pump
CSS	containment spray system
CST	condensate storage tank
CTMU	cooling tower makeup
CUF	cumulative usage factor
CVCS	chemical and volume control system
C_VUSE	upper shelf energy determined by charpy v-notch test results
CWS	circulating water system
DBA	design basis accident
DBD	design basis document
DBE	design basis event
DEH	digital-electric hydraulic

ECCS	emergency core cooling system
EDB	(PassPort) equipment database
EDG	emergency diesel generator
EFPY	effective full-power year
EOL	end of life
EPRI	Electric Power Research Institute
EQ	environmental qualification
EQML	environmental qualification master list
ESF	engineered safety feature
ESW	emergency service water
FAC	flow-accelerated corrosion
FERC	Federal Energy Regulatory Commission
FHB	fuel handling building
FR	*Federal Register*
FSAR	final safety analysis report
ft.	foot, feet
GALL	Generic Aging Lessons Learned Report
GDC	general design criteria or general design criterion
GEIS	Generic Environmental Impact Statement
GL	generic letter
GSI	generic safety issue
HEPA	high efficiency particulate air
HHSI	high head safety injection
HNP	Shearon Harris Nuclear Power Plant
HVAC	heating, ventilation, and air conditioning
I&C	instrumentation and controls
IASCC	irradiation assisted stress corrosion cracking
IEEE	Institute of Electrical and Electronics Engineers
IGSCC	intergranular stress corrosion cracking
IN	Information Notice
INPO	Institute of Nuclear Power Operations
IPA	integrated plant assessment
ISG	interim staff guidance
ISI	inservice inspection
KV	kilovolt
LBB	leak-before-break
LHSI	low head safety injection
LOCA	loss of coolant accident
LRA	license renewal application
LRBD	license renewal boundary drawing
LTOP	low-temperature over-pressure protection

MEB	metal enclosed bus
MeV	million electron volts
MFIV	main feedwater isolation valve
MIC	microbiologically influenced corrosion
MSLB	main steam line break
MSR	moisture separator reheater
N/A	not applicable
NCR	nuclear condition reports
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NESC	National Electrical Safety Code
NFPA	National Fire Protection Association
NPS	nominal pipe size
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
NSW	normal service water
NUREG	designation of publications prepared by the NRC staff
OI	open item
OM	operation and maintenance
OPB	outside the power block
PASS	post-accident sampling system
pH	concentration of hydrogen ions
PMID	preventive maintenance identification number
PORV	power-operated relief valve
PRT	pressurizer relief tank
PSI	passive safety injection
PSS	primary sampling system
P-T	pressure-temperature
PTS	pressurized thermal shock
PVC	polyvinyl chloride
PWR	pressurized water reactor
PWSCC	primary water stress corrosion cracking
QA	quality assurance
RAB	reactor auxiliary building
RAI	request for additional information
RCCA	rod cluster control assembly
RCDT	reactor coolant drain tank
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RFO	refueling outage
RG	Regulatory Guide
RHR	residual heat removal

RPV	reactor pressure vessel
RPVH	reactor pressure vessel head
RT_{NDT}	reference temperature nil ductility transition
RT_{PTS}	reference temperature for pressurized thermal shock
RVI	reactor vessel internals
RVLIS	reactor vessel level indicating system
RWST	refueling water storage tank
SBO	station blackout
SC	structure and component
SCC	stress-corrosion cracking
SER	safety evaluation report
SRP-LR	Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants
SSC	system, structure, and component
SSE	safe-shutdown earthquake
SUT	startup transformer
TAC	technical assignment control (internal NRC work management tool)
TLAA	time-limited aging analysis
TS	technical specification
UHS	ultimate heat sink
US	United States
USE	upper-shelf energy
UT	ultrasonic testing
VCT	volume control tank
VHP	vessel head penetration
WCAP	Westinghouse Commercial Atomic Power
WOG	Westinghouse Owners Group
WPB	waste processing building
WPS	waste processing system

SECTION 4

TIME-LIMITED AGING ANALYSES

4.1 Identification of Time-Limited Aging Analyses

This section of the safety evaluation report (SER) addresses the identification of time-limited aging analyses (TLAAs). In license renewal application (LRA) Sections 4.2 through 4.7, the applicant addressed the TLAAs for Shearon Harris Nuclear Power Plant (HNP), Unit 1. SER Sections 4.2 through 4.8 document the review of the TLAAs conducted by the staff of the United States (US) Nuclear Regulatory Commission (NRC) (the staff).

TLAAs are certain plant-specific safety analyses that involve time-limited assumptions defined by the current operating term. Pursuant to Title 10, Section 54.21(c)(1), of the *Code of Federal Regulations* (10 CFR 54.21(c)(1)), applicants must list TLAAs as defined in 10 CFR 54.3.

In addition, pursuant to 10 CFR 54.21(c)(2), applicants must list plant-specific exemptions granted under 10 CFR 50.12 based on TLAAs. For any such exemptions, the applicant must evaluate and justify the continuation of the exemptions for the period of extended operation.

4.1.1 Summary of Technical Information in the Application

To identify the TLAAs, the applicant evaluated calculations for HNP against the six criteria specified in 10 CFR 54.3. The applicant indicated that it has identified the calculations that met the six criteria by searching the current licensing basis (CLB). The CLB includes the final safety analysis report (FSAR), Technical Specifications, technical reports, licensing correspondence, and applicable vendor reports. In LRA Table 4.1-1, "Time-Limited Aging Analyses," the applicant listed the applicable TLAAs using the categories from NUREG-1800:

- reactor vessel neutron embrittlement
- metal fatigue
- environmental qualification of electrical equipment
- concrete containment tendon prestress (Not applicable to HNP)
- containment liner plate, metal containments, and penetrations fatigue analysis
- other plant-specific time-limited aging analyses

Pursuant to 10 CFR 54.21(c)(2), the applicant identified exemptions granted under 10 CFR 50.12 that were based on a TLAA, as defined in 10 CFR 54.3. The applicant listed the following exemptions for TLAAs in LRA Section 4.1.3, "Identification of Exemptions:"

Two exemptions were listed as meeting the TLAA definition. The first involves an exemption from the provisions to 10 CFR Part 50, Appendix A, General Design Criterion 4, with respect to asymmetric blowdown loads from discrete breaks in the reactor coolant system (RCS) primary loop by use of leak-before-break analysis. The second involves an exemption to the requirements of 10 CFR 50.60(a) and 10 CFR Part 50, Appendix G, to permit the use of American Society of Mechanical Engineers (ASME) Code Case N-640 alternative fracture toughness analysis methods in the development of revised reactor vessel pressure-temperature (P-T) curves. The analyses supporting these exemptions meet all the criteria for TLAAs and have been included on Table 4.1-1. See SER Section 4.3.4 for the leak-before-break analysis and Section 4.2.4 for the operating P-T limits analyses which utilize the provisions of Code Case N-640. SER Section 4.2.5 addresses low-temperature overpressure limits for license renewal.

4.1.2 Staff Evaluation

LRA Table 4.1-1 lists the HNP TLAAs; the applicant also addressed exemptions based on these TLAAs. The staff reviewed the information to determine whether the applicant has provided sufficient information pursuant to 10 CFR 54.21(c)(1) and 10 CFR 54.21(c)(2).

As defined in 10 CFR 54.3, TLAAs meet the following six criteria:

(1) involve systems, structures, and components within the scope of license renewal, as described in 10 CFR 54.4(a)
(2) consider the effects of aging
(3) involve time-limited assumptions defined by the current operating term (40 years)
(4) are 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 described in 10 CFR 54.4(b)
(6) are contained or incorporated by reference in the CLB

The staff reviewed LRA Tables 4.1-1& 4.1-2 against SRP-LR Tables 4.1-2 & 4.1-3, which show potential TLAAs, to confirm that the applicant omitted no TLAAs as defined in 10 CFR 54.3.

During the audit and review, the staff asked the applicant why the fatigue analysis of the reactor coolant pump (RCP) flywheel did not meet TLAA criteria. The applicant responded that the evaluation supporting the interval for inservice inspections of the RCP flywheels based on a plant life of 60 years does not meet the 10 CFR 54.3(a)(3) criterion ("Involve time-limited assumptions defined by the current operating term, for example, 40 years"). The staff reviewed Plant Technical Specification Amendment No. 119, Section 4.4.10, which states, "Each Reactor Coolant Pump Motor Flywheel be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August, 1975," to confirm the inspection interval of 20 years for this component.

On the basis that the fatigue crack growth evaluation supports the inspection interval for inservice inspection instead of the current operating term (40 years), the staff agreed that the RCP flywheel fatigue crack growth analysis does not meet the 10 CFR 54.3(a)(3) criterion for TLAAs. Additionally, the plant technical specification supports the inspection requirement for the component.

As required by 10 CFR 54.21(c)(2), an applicant must list all exemptions granted under 10 CFR 50.12, based on a TLAA, and evaluated and justified for continuation through the period of extended operation. The LRA states that each active exemption was reviewed to determine whether the exemption was based on a TLAA. The applicant identified TLAA-based exemptions. Based on the information provided by the applicant regarding the process used to identify these exemptions and its results, the staff concludes that the two exemptions meet all TLAA criteria.

4.1.3 Conclusion

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 and that two exemptions have been granted on the TLAA basis TLAA as so defined.

4.2 Reactor Vessel Neutron Embrittlement

"Neutron embrittlement" is the term that describes changes in mechanical properties of reactor vessel materials that result from exposure to neutrons. The most pronounced material change is a reduction in fracture toughness. As fracture toughness decreases with cumulative fast neutron exposure the material's resistance to crack propagation decreases. The rate of neutron exposure is defined as neutron flux, and the cumulative degree of exposure over time is defined as neutron fluence.

Fracture toughness of ferritic materials depends upon fluence as well as temperature. The Reference Temperature for nil-ductility transition (RT_{NDT}) is a metric for embrittlement. For temperatures above the transition temperature, the material is ductile, and below is brittle. As fluence increases, the nil-ductility reference temperature increases and higher temperatures are required for the material to continue behaving in a ductile manner. This shift in reference temperature is the ΔRT_{NDT} plus a margin term added to account for uncertainties in the limited data available for the projections. Determination of the reactor pressure vessel (RPV) fluence and the projected reduction in fracture toughness as a function of fluence affects several analyses that support HNP operation:

- RPV Material Upper-Shelf Energy (USE)
- RPV Pressurized Thermal Shock (PTS)
- RPV Operating P-T Limits
- RPV Low-Temperature Overpressurization Setpoints

In evaluating an extension of the operating period from 40 years to 60 years, the 60-year peak fluence value and its impact upon the analyses that support operation must be determined. The aging effect within the TLAA will be managed during the period of extended operation.

4.2.1 Neutron Fluence

4.2.1.1 Summary of Technical Information in the Application

NRC regulations require projections showing the ΔRT_{NDT} expected at the end-of-life (EOL). A minimum USE value limits the amount of downward shift, and a PTS screening criterion RT_{NDT} limits. If a projection indicates that these limits may be exceeded, changes must be implemented to prevent this occurrence.

Framatome (now AREVA) has developed a fluence analysis methodology that can predict the fast neutron fluence in the reactor vessel. The methodology demonstrated that the calculated fluence value would be unbiased and have uncertainty within the NRC suggested limit of 20 percent. The AREVA fluence analysis methodology adheres to the guidance in Regulatory Guide (RG) 1.190 and has been benchmarked accordingly. The AREVA methodology has been reviewed by the staff and has been approved for referencing in licensing actions in Westinghouse built reactors. Capsule X was removed from the reactor vessel at the end of Cycle 8 for testing and evaluation. The capsule received an average fast fluence of 3.25×10^{19} n/cm^2 (E > 1.0 MeV). Based on the calculated eight-cycle average full-power flux and a 90-percent capacity factor, the projected 40-calendar year (EOL) of 36 effective full power year (EFPY) peak vessel fluence at the base metal-clad interface is 4.55×10^{19} n/cm^2, E > 1.0 MeV. An additional analysis considered the implementation of a 4.5 percent (to 2900 MWt) power uprate commencing with Cycle 11. Based on the calculated eight-cycle-average full power flux and a 90-percent capacity factor, the projected 40-calendar year peak vessel fluence is 4.59×10^{19} n/cm^2 (E > 1.0 MeV).

Using the AREVA methodology, the data from Capsule X, and a value of 55 EFPY to account for 60 years of operation, the applicant obtained projected values of neutron flux for use in the fluence-related analyses addressed later in this section. In addition, the RPV boundary components outside the beltline region have been evaluated to determine whether additional materials should be considered for analysis for the period of extended operation. The beltline, as defined by 10 CFR 50.61(a)(3), is the RPV region that directly surrounds the height of the active core and adjacent RPV regions predicted to experience sufficient neutron radiation damage for consideration in the selection for the most limiting material for radiation damage. The threshold fluence for material is 1×10^{17} n/cm^2 (E > 1.0 MeV). The existing AREVA neutron fluence models have been extended to facilitate this evaluation. The materials outside of the traditional beltline region expected to receive fluence values greater than 10^{17} n/cm^2 were evaluated but none determined to be limiting.

Therefore, the neutron fluence has been projected to the end of the period of extended operation by use of a methodology previously approved by the staff. The 55 EFPY fluence projections will be used for evaluation of fluence-based TLAAs for license renewal.

4.2.1.2 Staff Evaluation

The staff reviewed LRA Section 4.2.1, pursuant to 10 CFR 54.21(c)(1)(ii), to verify that the analyses have been projected to the end of the period of extended operation.

The staff reviewed the fluence calculations for the power uprate documented in BAW-2355, Supplement 1. Well over half of the final fluence value will accrue after the power uprate implemented at the beginning of Cycle 11. The applicant chose Cycle 18 as the representative equilibrium cycle for post-uprate loadings; thus, the Cycle 18 calculation parameters represent the equilibrium cycle for the post-uprate operation. As post-uprate cycles result in higher neutron leakage per EFPY (new or once-burned assemblies loaded on the periphery), it is conservative to assume the equilibrium cycle for all post-uprate fluence calculations.

The peak fluence locations (for this plant 0°and 45°azimuthal angles) affect the intermediate shell and the circumferential weld AB. The applicant stated (in the power uprate review) that fresh or once-burned assemblies would not be placed in locations different from those analyzed in Equilibrium Cycle 18, indicating that the 0° and 45° locations will not be affected by the use of fresh or once-burned assemblies and that the final fluence value (*i.e.*, the maximum value) will not exceed that at 0° azimuth.

With these assumptions, the applicant determined the fluence value and the adjusted reference temperature (ART) for 60 calendar years of operation as listed in LRA Table 4.2-3.

The pressure vessel critical element is the intermediate shell plate B4197-2 for which the end of period of extended operation peak fluence value is 6.905×10^{19} n/cm^2 and the ART = 195.3 °F; therefore, the pressure vessel has a large margin for PTS (10 CFR 50.61) because the screening criterion for plates is 270 °F.

In summary, the staff confirmed that calculation of the proposed fluence values to the end of the period of extended operation (55 EFPY) used an approved methodology. The applicant's assumptions for expected operation of the plant are conservative; therefore, the staff finds the proposed values acceptable but, if the loading patterns differ from the Equilibrium Cycle 18 pattern assumed in the analysis, the applicant must submit for staff review a revised loading pattern analysis of the effect on the vessel fluence values. This is the third licensed Condition as stated in Section 1.7 of this SER.

4.2.1.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of neutron fluence in LRA Section A.1.2.1. On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address neutron fluence is adequate.

4.2.1.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that, for neutron fluence, the analyses have been projected to the end of the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.2.2 Upper Shelf Energy Analysis

4.2.2.1 Summary of Technical Information in the Application

LRA Section 4.2.2 summarizes the evaluation of USE analysis for the period of extended operation. Fracture toughness is a measure of the amount of energy a material can absorb before fracturing. Charpy V-notch tests estimate fracture toughness, and one of the units of measure is ft.-lbs. of absorbed energy. The more ductile a material, the higher the fracture toughness and more ft.-lbs. of energy absorbed before fracture. The fracture toughness of reactor vessel steels is temperature-dependent. At low temperatures, the vessel material toughness is relatively low and constant and the material behaves in a brittle fashion. Rising temperatures reach a point where the toughness increases rapidly until another plateau where the toughness is relatively high and constant. In this high toughness region, the material is ductile. These regions of the curve are the lower shelf, transition zone, and upper shelf, respectively. The USE is the toughness value (absorbed energy) from the upper shelf portion of the curve (ductile region) for a material at a time in its service life; 10 CFR Part 50, Appendix G, screening criteria limit the degree that an RPV material USE value may drop due to neutron irradiation. The regulation requires the initial RPV material USE to be greater than 75 ft.-lb. when the material is in the unirradiated condition and for the USE to remain above 50 ft.-lb. in the fully irradiated condition throughout the licensed life of the vessel, unless lower values of energy can be demonstrated to provide margins of safety against fracture equivalent to those required by ASME Code Section XI, Appendix G.

An evaluation of the RPV for the period of extended operation (55 EFPY) USE for the reactor vessel beltline materials used RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2 guidelines. The reactor vessel USE evaluations were at the 1/4T wall location of each beltline material using the respective copper contents and Figure 2 of RG 1.99, Revision 2. The reactor vessel beltline material with the lowest predicted USE is the intermediate shell plate, heat number B4197-2; however, the predicted value for this material is not projected to fall below the required 50 ft-lb limit; therefore, the analyses for reactor vessel USE decreases projected to the end of the 60-year period of extended operation demonstrate that, for the most limiting material, the lowest predicted USE is greater than the 10 CFR Part 50, Appendix G, limit of 50 ft-lbs.

4.2.2.2 Staff Evaluation

The staff reviewed LRA Section 4.2.2, to verify pursuant to 10 CFR 54.21(c)(1)(ii) that the analysis has been projected to the end of the period of extended operation.

Part 50 of 10 CFR, Appendix G, Section IV.A., provides NRC requirements for demonstrating that reactor vessels in US light-water reactor facilities will have fracture toughness requirements throughout their service lives. The section requires for reactor vessel beltline materials USE values equal to or above 75 ft-lb when in unirradiated condition and equal to or above 50 ft-lb throughout the licensed life of the reactor vessel. RG 1.99, Revision 2, expansively addresses calculations of USE values and describes two methods for determining them for reactor vessel beltline materials depending on whether they are under the Reactor Vessel Material Surveillance Program.

LRA Table 4.2-1 shows for reactor vessel beltline materials USE assessments based on the listed 1/4T neutron fluence values based on projected values at the end of the period of extended operation (*i.e.*, at 55 EFPY).

According to NUREG-1801, Revision 1, "Generic Aging Lessons Learned Report," (GALL Report) Table IV A-2, ferritic materials are subject to neutron embrittlement when exposed to a neutron fluence greater than 1 x 10^{17} n/cm^2 (E >1 MeV) at the end of the period of extended operation.

The staff's review of LRA Section 4.2.2 found an area in which additional information was necessary to complete the review of the applicant's TLAA evaluation. The applicant responded to the staff's request for additional information (RAI) as follows.

In RAI 4.2.6 dated July 20, 2007, the staff requested from the applicant USE values for all ferritic materials and their welds exposed to a neutron fluence value greater than 1 x 10^{17} n/cm^2 (E > 1 MeV).

In its response dated August 16, 2007, the applicant stated that further study determined that five additional reactor vessel materials will be exposed to a neutron fluence value greater than 1 x 10^{17} n/cm^2 (E > 1 MeV) at the end of the period of extended operation; therefore, the USE evaluation in Table 4.2-4 of the applicant's RAI response dated August 16, 2007, was a part of the neutron embrittlement analyses for these five reactor vessel materials. The applicant further stated that, as their projected USE values are greater than those of the limiting beltline material (Intermediate Shell Plate Heat No. B4197-2), these five materials (upper to intermediate circumferential weld AC, upper shell, inlet nozzle, inlet nozzle weld and upper shell longitudinal welds BE/BF) are not limiting for the USE analysis.

The staff's independent calculations of the USE values for the reactor vessel beltline materials through the period of extended operation applied as their basis the 1/4T neutron fluence values listed in LRA Table 4.2-1 for the reactor vessel. Applying the methods of RG 1.99, Revision 2, for its independent USE calculations, the staff determined that Intermediate Shell Plate Heat No. B4197-2 is the limiting beltline material. The staff's calculated 55 EFPY USE value of 52.0 ft-lb was in close agreement with the applicant's calculation (*i.e.*, 52.8 ft-lb) for this plate material. Both values meet the 10 CFR Part 50, Appendix G acceptance criterion of USE values of reactor vessel beltline materials above 50 ft-lb throughout the licensed life of the plant.

The staff also evaluated the USE values for the five additional materials to be exposed to a neutron fluence value greater than 1 x 10^{17} n/cm^2 (E > 1 MeV) at the end of the period of extended operation. The staff finds the USE values for these materials acceptable because they comply with 10 CFR Part 50, Appendix G and because they are bounded by the USE value of the limiting beltline material, Intermediate Shell Plate Heat No. B4197-2.

Based on this review, the staff's finds the applicant's response to RAI 4.2.6 acceptable. The staff's concern described in RAI 4.2.6 is resolved.

Based on its technical assessments, the staff determines that the reactor vessel will maintain an acceptable level of USE values throughout the period of extended operation; therefore, the staff concludes that the applicant's TLAA for USE, as in LRA Section 4.2.2 and in the applicant's response to RAI 4.2.6, is in compliance with 10 CFR Part 50, Appendix G, and, therefore, acceptable.

4.2.2.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of the USE analysis in LRA Section A.1.2.1.1. On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address the USE analysis is adequate.

4.2.2.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that, for USE analysis, the analyses have been projected to the end of the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.2.3 Pressurized Thermal Shock Analysis

4.2.3.1 Summary of Technical Information in the Application

LRA Section 4.2.3 summarizes the evaluation of PTS analysis for the period of extended operation. Section 50.61 of 10 CFR defines screening criteria for embrittlement of RPV materials in pressurized-water reactors as well as actions required if these screening criteria are exceeded. The screening criteria limit the degree that vessel material reference temperature may increase for PTS - RT_{PTS} following RPV neutron irradiation. For circumferential welds, the PTS screening criterion is 300 °F maximum, for plates, forgings, and axial weld materials 270 °F maximum. Projected EOL reference temperature for pressurized-thermal shock (RT_{PTS}) values must be shown to remain below the applicable screening temperature.

A 10 CFR 50.61 PTS evaluation for the reactor vessel beltline materials accounted for 40 years of operation (36 EFPY). Before power uprate, the controlling reactor vessel beltline material for PTS was the intermediate shell plate, heat number B4197-2, with an RT_{PTS} value of 196.1 °F, well below the PTS screening criterion of 270 °F. The results of the PTS evaluation demonstrate that the reactor vessel beltline material RT_{PTS} values will not exceed the PTS screening criteria before EOL (36 EFPY). The results of the PTS evaluation to account for the 4.5 percent (to 2900 MWt) power uprate commencing with Cycle 11 demonstrate that the reactor vessel beltline material RT_{PTS} values will not exceed the PTS screening criteria before EOL (36 EFPY). The reactor vessel controlling beltline material is the intermediate shell plate, heat number B4197-2, with a RT_{PTS} value of 196.2 °F.

A PTS evaluation for the reactor vessel beltline materials was in accordance with 10 CFR 50.61. Calculation of PTS reference temperature RT_{PTS} values is by addition of the initial RT_{NDT} to the predicted radiation-induced ΔRT_{NDT} and the margin term to account for uncertainties in the values of initial RT_{NDT} copper and nickel contents, fluence, and calculation procedures. Calculation of the predicted radiation-induced ΔRT_{NDT} is by use of the respective reactor vessel beltline material copper and nickel contents and the neutron fluence applicable to the reactor vessel for license renewal at 55 EFPY.

Evaluations of the RT_{PTS} values for each reactor vessel beltline material were with chemistry factors determined from Tables 1 and 2 in 10 CFR 50.61. In addition, the chemistry factors for the intermediate shell plate, heat number B4197-2, and the intermediate shell to lower shell circumferential weld were recalculated with available surveillance data.

The RT_{PTS} values for the reactor vessel beltline materials at 55 EFPY were determined. The results of the PTS evaluation demonstrate that the reactor vessel beltline materials will not exceed the PTS screening criteria before the end of the period of extended operation. The reactor vessel controlling beltline material for PTS is the intermediate shell plate, heat number B4197-2, with an RT_{PTS} value of 199.9 °F, well below the PTS screening criterion of 270 °F.

4.2.3.2 Staff Evaluation

The staff reviewed LRA Section 4.2.3, to verify pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses have been projected to the end of the period of extended operation.

Section 50.61 of 10 CFR provides NRC requirements for reactor vessels in US pressurized water reactor (PWR) facilities with adequate protection against PTS consequences throughout their service lives. The section requires applicants to calculate RT_{PTS} values for each base metal and weld material in reactor vessel beltline regions and sets maximum limits of 270 °F for RT_{PTS} values calculated for base metals (*i.e.*, forging and plate materials) and axial weld materials and 300 °F for RT_{PTS} values calculated for circumferential weld materials. Section 50.61 also expansively addresses how RT_{PTS} values should be calculated and describes two methods for determining them for reactor vessel beltline materials depending on whether they are under the Reactor Vessel Material Surveillance Program.

LRA Table 4.2-2 lists for the reactor vessel beltline materials RT_{PTS} values based on the neutron fluence values at the clad-base metal surface of the reactor vessel. To determine the RT_{PTS} values the applicant used neutron fluence values based on the values projected to the end of the period of extended operation (*i.e.*, at 55 EFPY). As the limiting material for PTS the applicant reported Intermediate Shell Plate Heat No. B4197-2 with a RT_{PTS} value of 199.9 °F at 55 EFPY based on credible surveillance capsule data. Calculation of this value used the chemistry factor from the chemical composition of the limiting beltline material.

Reviewing the applicant's use of surveillance capsule test data, the staff found the nickel and copper values shown in LRA Table 4.2-2 for the limiting beltline material and the surveillance test coupons identical but their chemistry factors different.

The staff's review of LRA Section 4.2.3 found areas in which additional information was necessary to complete the review of the applicant's TLAA evaluation. The applicant responded to the staff's RAIs as follows.

In RAI 4.2.3 dated July 20, 2007, the staff asked the applicant to clarify where the chemistry factor for the surveillance capsule test sample was derived from.

In its response dated August 16, 2007, the applicant added a footnote in LRA Table 4.2-2 to show that the chemistry factor for the surveillance capsule test sample was derived from the surveillance data.

Based on its review, the staff finds the applicant's response to RAI 4.2.3 acceptable. The staff's concern described in RAI 4.2.3 is resolved.

According to GALL Report Table IV A-2, ferritic materials are subject to neutron embrittlement when exposed to a neutron fluence greater than 1×10^{17} n/cm^2 (E > 1 MeV) at the end of the period of extended operation. In RAI 4.2.6, dated July 20, 2007, the staff requested from the applicant RT_{PTS} values for all the ferritic materials and their welds exposed to neutron fluence values greater than 1×10^{17} n/cm^2 (E > 1 MeV).

In its response dated August 16, 2007, the applicant stated that five additional reactor vessel materials will be exposed to a neutron fluence value greater than 1×10^{17} n/cm^2 (E > 1 MeV) at the end of the period of extended operation; therefore, the applicant calculated RT_{PTS} values, shown in Table 4.2-5 of the applicant's RAI response dated August 16, 2007, as a part of PTS analysis for these five materials. These calculated RT_{PTS} values are less than the RT_{PTS} value of the limiting beltline material (Intermediate Shell Plate Heat No. B4197-2).

To verify the validity of the applicant's calculation of the RT_{PTS} value at 55 EFPY for the limiting beltline material, the staff's independent calculations per 10 CFR 50.61 found the RT_{PTS} value acceptable. The staff also evaluated the RT_{PTS} values for the five additional reactor vessel materials to be exposed to neutron fluence greater than 1×10^{17} n/cm^2 (E greater than 1 MeV) at the end of the period of extended operation and found their RT_{PTS} values in compliance with specific 10 CFR 50.61 requirements and acceptable. In addition, the predicted RT_{PTS} value for the limiting beltline material Intermediate Shell Plate Heat No. B4197-2 bounds the RT_{PTS} values of these five reactor vessel materials.

Based on its review, the staff finds the applicant's response to RAI 4.2.6 acceptable. The staff's concern described in RAI 4.2.6 is resolved.

Based on its technical assessments, the staff concludes that the reactor vessel will maintain acceptable RT_{PTS} values throughout the period of extended operation. The staff, therefore, concludes that the applicant's TLAA for PTS in LRA Section 4.2.3 and in the applicant's RAI response dated August 16, 2007, complies with specific 10 CFR 50.61 screening criteria. The staff concludes that the reactor vessel will be acceptable for PTS through the period of extended operation.

4.2.3.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of the PTS analysis in LRA Section A.1.2.1.2. On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address the PTS analysis is adequate.

4.2.3.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that, for PTS analysis, the analyses have been projected to the end of the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.2.4 Operating Pressure-Temperature Limits Analysis

4.2.4.1 Summary of Technical Information in the Application

LRA Section 4.2.4 summarizes the evaluation of operating P-T limits analysis for the period of extended operation. The Adjusted Reference Temperature (ART) is the value of Initial RT_{NDT} + ΔRT_{NDT} + margins for uncertainties at a specific location. Neutron embrittlement increases the ART; thus, the minimum temperature at which an reactor vessel is allowed to be pressurized increases over the licensed period. The ART of the limiting beltline material is for correction of beltline P-T limits to account for radiation effects. In accordance with 10 CFR Part 50, Appendix G, reactor vessel thermal limit analyses must determine operating P-T limits for boltup, hydrotest, pressure tests, normal operation, and anticipated operational occurrences. P-T operating limits are required for three categories of operation: (1) hydrostatic pressure tests and leak tests, (2) nonnuclear heat-up/cool-down and low-level physics tests, and (3) core critical operation.

Reactor vessel P-T limits and minimum temperature requirements in accordance with 10 CFR Part 50, Appendix G, are defined by operating condition, vessel pressure, the presence of fuel in the vessel, and core criticality. The P-T limits must be at least as conservative as limits obtained by the methods of analysis and margins of safety of Appendix G of Section XI of the ASME Code. The minimum temperature requirements pertain to the controlling material, which is the material in either the closure flange or the beltline region with the highest reference temperature.

Calculation of ART values for the reactor vessel beltline region materials in accordance with RG 1.99, Revision 2 is by addition of the initial RT_{NDT} to the predicted radiation-induced ΔRT_{NDT} and a margin term to account for uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence, and the calculation procedures. Calculation of the predicted radiation-induced ΔRT_{NDT} is by the respective reactor vessel beltline material copper and nickel contents and the neutron fluence applicable to 55 EFPY. The evaluations for the ART were at the 1/4T and 3/4T wall locations of each beltline material with chemistry factors determined from Tables 1 and 2 in

RG 1.99, Revision 2. In addition, chemistry factors for the intermediate shell plate, heat number B4197-2, and the intermediate shell to lower shell circumferential weld were recalculated with available surveillance data.

In this manner, ART results for the reactor vessel beltline region materials applicable to 55 EFPY are determined. Calculation of P-T operating limits was by approved procedures and established methods and techniques in accordance with the requirements of 10 CFR 50 Appendix G, ASME Code Section XI Appendix G, and ASME Code Cases N-588 and N-640. These results show the reactor vessel controlling beltline material as the intermediate shell plate, heat number B4197-2.

4.2.4.2 Staff Evaluation

The staff reviewed LRA Section 4.2.4, to verify pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses have been projected to the end of the period of extended operation.

Paragraph IV.A.2 of 10 CFR Part 50, Appendix G, provides the staff's requirements and criteria for P-T limits for commercial US light-water reactors. Section 50.36 of 10 CFR requires nuclear power production facility licensees to include the P-T limits and low-temperature over-pressure protection (LTOP) system setpoints among the limiting conditions for operation in plant technical specifications.

The staff, in its safety evaluation dated July 28, 2000, approved the current HNP P-T limits as valid for 32 EFPY. Revision of the P-T limits is based on the extent to which the beltline materials are exposed to the neutron fluence during the period of extended operation.

The staff's review of LRA Section 4.2.4 found areas in which additional information was necessary to complete the review of the applicant's TLAA evaluation. The applicant responded to the staff's RAIs as follows.

In RAI 4.2.4(A) dated July 20, 2007, the staff requested from the applicant a statement in LRA Section 4.2.4 indicating how it will manage future P-T limits during the period of extended operation.

In its response dated August 16, 2007, the applicant stated that it will add the following statement to LRA Section 4.2.4:

> The current P-T limits are valid through 36 EFPY. The P-T limits for the extended period of operation will be managed by using approved fluence calculations when there are changes in power or core design in conjunction with surveillance capsule results.

Based on its review, the staff finds the applicant's response to RAI 4.2.4(A) acceptable because it complies with the staff's request; therefore, the staff's concern described in RAI 4.2.4 is resolved.

In RAI 4.2.4(B) dated July 20, 2007, the staff asked the applicant to clarify how it will comply with regulatory criteria while changing P-T limits.

In its response dated August 16, 2007, the applicant indicated that it will add the following statement to LRA Section 4.2.4:

> P-T limits have been imposed on operational parameters at HNP, thereby assuring that the reactor vessel is operated within required safety margins in accordance with the requirements of 10 CFR 50.60 and 10 CFR 50 Appendix G. HNP has implemented changes in the P-T curves throughout the current period of operation using the license amendment process, and expects to continue to use the license amendment process to implement future changes in P-T curves for the remainder of the current period of operation and for the extended period of operation.

Based on its review, the staff finds the applicant's response to RAI 4.2.4(B) acceptable because the change in P-T limits will be implemented by the license amendment process, which meets the regulatory requirements of 10 CFR 50.60 and 10 CFR Part 50, Appendix G. The staff's concern described in RAI 4.2.4(B) is resolved.

According to GALL Report Table IV A-2, ferritic materials are subject to neutron embrittlement when exposed to neutron fluences greater than 1×10^{17} n/cm^2 (E > 1 MeV) at the end of the period of extended operation.

In RAI 4.2.6, dated July 20, 2007, the staff requested from the applicant ART values for ferritic materials and their welds exposed to a neutron fluence value greater than 1×10^{17} n/cm^2 (E > 1 MeV). The ART value for the limiting beltline material determines beltline P-T limits that account for neutron embrittlement in the development of P-T limits pursuant to 10 CFR Part 50, Appendix G requirements.

Table 4.2-6 of the applicant's response dated August 16, 2007, shows ART values as parts of neutron embrittlement analyses for these five reactor vessel materials.

The staff reviewed the ART values listed in LRA Table 4.2-3, independently calculated the ART values for the reactor vessel beltline materials by the method specified in RG 1.99, Revision 2, and verified the ART value of the limiting beltline material, Intermediate Shell Plate Heat No. B4197-2, per RG 1.99, Revision 2, Regulatory Position C.1 (without surveillance data) and per RG 1.99, Revision 2, Regulatory Position C.2 (with surveillance data).

The calculated ART value according to Regulatory Position C.1 is higher than that according to Regulatory Position C.2. Consistent with RG 1.99, Revision 2, Section 2.1, the applicant calculated the ART value using the surveillance data and the staff finds this calculation acceptable. Because the method for calculating the beltline materials ART values meets the requirements of the RG 1.99, Revision 2, the staff accepts the ART values listed in LRA Table 4.2-3.

The staff verified ART values for the five additional reactor vessel materials to be exposed to a neutron fluence greater than 1×10^{17} n/cm^2 (E > 1 MeV) at the end of the period of extended operation. The staff finds the ART values for these materials acceptable because they comply with specific 10 CFR Part 50 Appendix H requirements. In addition, the ART values of these five reactor vessel materials are less than the value of the limiting beltline material (Intermediate Shell Plate Heat No. B4197-2). Because the ART evaluation of the limiting beltline material bounds the evaluation of these five reactor vessel materials, the staff concludes that the neutron embrittlement ART analysis for the reactor vessel materials is still valid.

Based on its review, the staff finds the applicant's response acceptable; therefore, the staff's concern described in RAI 4.2.6 is resolved.

Based on its technical assessments, the staff concludes that the ART values for the reactor vessel beltline materials, as projected through the period of extended operation, are consistent with the guidelines of RG 1.99, Revision 2; therefore, the staff concludes that the applicant's TLAA for P-T limits is acceptable.

4.2.4.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of the operating P-T limits analysis in LRA Section A.1.2.1.3. On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address the operating P-T limits analysis is adequate.

4.2.4.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that, for operating P-T limits analysis, the analyses have been projected to the end of the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.2.5 Low-Temperature Overpressure Limits Analysis

4.2.5.1 Summary of Technical Information in the Application

LRA Section 4.2.5 summarizes the evaluation of low-temperature overpressure limits analysis for the period of extended operation. ASME Section XI, Appendix G, establishes RCS P-T procedures and limits primarily for low-temperature conditions to protect against reactor vessel nonductile failure. When enabled at low temperatures, the low-temperature overpressure protection system assures that these limits are not exceeded. This temperature is conservatively selected at < 325°F.

There has been no analysis of low-temperature overpressure setpoints to support operation to the end of the period of extended operation for license renewal. The low-temperature overpressure setpoint analysis will be recalculated following the removal of one of the remaining surveillance capsules from the vessel when the calculated fast neutron fluence on the capsule meets or exceeds the calculated fast neutron fluence on the vessel wall at the end of the period of extended operation.

4.2.5.2 Staff Evaluation

The staff reviewed LRA Section 4.2.5, pursuant to 10 CFR 54.21(c)(1). License amendment request No. 100 dated April 12, 2000, was for staff approval of HNP's LTOP setpoint settings for 32 EFPY. The staff's safety evaluation dated July 28, 2000, approving this request required a minimum enabling temperature of 325 °F to be maintained for reactor vessel pressures above 450 psig (pounds per square inch gauge).

The staff's review of LRA Section 4.2.5 found an area in which additional information was necessary to complete the review of the applicant's TLAA evaluation. The applicant responded to the staff's RAI as follows.

In RAI 4.2.5 dated July 20, 2007, the staff requested that the applicant address any new LTOP setpoints analysis and its implementation due to any change in P-T limits during the period of extended operation.

In its response dated August 16, 2007, the applicant stated that the following text would be added to LRA Section 4.2.5:

> HNP will submit the appropriate analysis for LTOP set points that will be valid for the period of extended operation. LTOP set points have been imposed on operational parameters at HNP, thereby assuring that the reactor vessel is operated within required safety margins in accordance with the requirements of 10 CFR 50.60 and 10 CFR 50, Appendix G. HNP has implemented changes in the LTOP set points throughout the current period of operation using the license amendment process, and expects to continue to use the license amendment process to implement future changes in LTOP set points for the remainder of the current period of operation and for the extended period of operation.

Based on its review, the staff finds the applicant's response to RAI 4.2.5 acceptable because the applicant's plan to manage LTOP setpoints complies with the staff's request and because any change in LTOP set points will be implemented by the license amendment process, which is consistent with 10 CFR 50.60 and 10 CFR Part 50, Appendix G requirements; therefore, the staff's concern described in RAI 4.2.5 is resolved.

On the basis of its review, the staff concludes that the applicant's TLAA for LTOP setpoints is acceptable.

4.2.5.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of low-temperature overpressure limits analysis in LRA Section A.1.2.1.4. On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address the low-temperature overpressure limits analysis is adequate.

4.2.5.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses have been projected to the end of the period of extended operation that 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 FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3 Metal Fatigue

By letter dated August 31, 2007, the applicant revised LRA Section 4.3 to summarize several thermal and mechanical fatigue analyses of plant mechanical components presented as TLAAs addressed in the following subsections.

4.3.1	Explicit Fatigue Analyses (Nuclear Steam Supply System (NSSS) Components)
4.3.1.1	Reactor Vessel
4.3.1.2	Reactor Vessel Internals
4.3.1.3	Control Rod Drive Mechanism
4.3.1.4	Reactor Coolant Pumps
4.3.1.5	Steam Generators
4.3.1.6	Pressurizer
4.3.1.7	Reactor Coolant Pressure Boundary Piping (ASME Class 1)
4.3.2	Implicit Fatigue Analysis (ASME Class 2, Class 3, and American National Standards Institute (ANSI) B31.1 Piping)
4.3.2.1	ASME Class 2 and Class 3 Piping
4.3.2.2	ANSI B31.1 Piping
4.3.3	Environmentally-Assisted Fatigue Analysis
4.3.4	RCS Loop Piping Leak-Before-Break Analysis
4.3.5	Cyclic Loads that Do Not Relate to RCS Transients
4.3.5.1	Primary Sample Lines
4.3.5.2	Steam Generator Blowdown Lines

4.3.1 Explicit Fatigue Analyses (NSSS Components)

The applicant submits the latest design fatigue analyses for each NSSS component within the reactor coolant pressure boundary (RCPB) to demonstrate that the design analyses will remain bounding through the period of extended operation. Components within the scope of this review include nonpressure-boundary reactor internals components.

Original fatigue design calculations assumed a large number of design transients from relatively severe system dynamics over the original 40-year design life. In general, actual plant operations have resulted in only a fraction of the originally expected fatigue duty.

A review to establish the current design basis for the major NSSS components showed that the use of transients from the steam generator replacement/uprating analysis is reasonable and limiting for the primary equipment except the pressurizer surge line and portions of the pressurizer lower head analyzed separately (LRA Subsections 4.3.1.6 and 4.3.1.7); therefore, the governing transients, "NSSS Design Transients," are those from the steam generator replacement/uprating analysis. Table 4.3-2 presents 40-year design cumulative usage factor (CUF) values compiled from design documents including the recent steam generator replacement/uprating analysis.

The next evaluation factored the effects of the reactor water environment on fatigue. The evaluation of NSSS components demonstrated compliance with 10 CFR 54.21(c)(1) by a combination of methods under 10 CFR 54(c)(1)(ii) and (iii).

The following sections summarize the results for each of the major NSSS components evaluated.

4.3.1.1 Reactor Vessel

4.3.1.1.1 Summary of Technical Information in the Application

LRA Section 4.3.1.1 summarizes the reactor vessel evaluation for the period of extended operation. There are TLAAs for several reactor vessel subcomponents. The use of transients from the steam generator replacement/uprating analysis is reasonable and limiting for the primary equipment with the exceptions of the pressurizer surge line and portions of the pressurizer lower head analyzed separately. Forty-year design CUF values were also parts of the steam generator replacement/uprating analysis. The reactor vessel fatigue analysis demonstrated that, if reactor vessel components were exposed to a bounding set of postulated transient cycles, their CUF values would not exceed 1.0.

The applicant stated that for the component parts of the reactor vessel, the highest 40-year design fatigue usage value is 0.37 for the closure studs. Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF of 0.56. This value does not exceed the design limit of 1.0 and is, therefore, acceptable. This 60-year fatigue usage bounds the maximum environmentally-adjusted usage factor of 0.1740 for the reactor vessel outlet nozzles in LRA Table 4.3-3; therefore, the analysis has been projected to the period of extended operation per 10 CFR 54.21(c)(1) (ii).

4.3.1.1.2 Staff Evaluation

The staff reviewed LRA Section 4.3.1.1 to verify pursuant to 10 CFR 54.21(c)(1)(ii) that the analysis has been projected to the end of the period of extended operation.

The staff reviewed LRA Table 4.3-1 for an adequate list of the assumed transients.

During the audit, the staff asked the applicant to address following questions:

(1) Describe the method for estimating the number of cycles for 60 years of operation for the transients listed in LRA Table 4.3-1 and explain why the cycles to date and the cycles projected for 60 years can be zero.

(2) The staff reviewed FSAR Table 3.9-1 ("Summary of Limiting Reactor Coolant Design Transients") and determined that LRA transients loop out of service shutdown, loop out of service startup, and inadvertent startup of an inactive loop may not be present at HNP. Why are those transients cycles in LRA Table 4.3-1?

(3) Does HNP address the inadvertent auxiliary spray cooling transient in FSAR Table 3.9-1?

On the first question, it was unclear why the applicant addressed the 60-year projected cycle of zero based on 18 years (cycles to date) operation. The applicant responded, "The cycle projections will be removed from the License Renewal Application. Cycle projections will not be used to justify acceptability of fatigue-related TLAAs by 10 CFR 54.21(c)(1)(i) - the analyses remain valid for the period of extended operation."

On the bases that the staff reviewed all metal fatigue TLAAs to confirm that the applicant will not use cycle projections to justify fatigue-related TLAAs under 10 CFR 54.21(c)(1)(i), that the analyses remain valid for the period of extended operation, and that the applicant's LRA Amendment 2 by letter dated, August 31, 2007 deleted cycle projections from the LRA, the staff finds this response acceptable.

On the second question, the applicant responded,

> Normal Transients 13, 14, and Upset Transient 8 were included in the qualifications performed by WCAP-14778, Revision 1, "Carolina Power and Light Harris Nuclear Plant Steam Generator Replacement/Uprating Analysis and Licensing Project NSSS Engineering Report," September 2000. As noted in the license renewal basis document, Normal Condition transients 13 and 14 (Loop Out of Service) are not applicable to the current HNP license. HNP is not currently licensed to operate with N-1 loops. The Loop Out of Service transients were included in the Westinghouse System Standard Design Criteria 1.3, Revision 2 so that the components are designed in case the plant is licensed to operate with N-1 loops. It was recommended by Westinghouse that the "Loop Out of Service" transients continue to be considered for the SGR/Uprating Project; therefore, the transients were carried forward to the License Renewal fatigue

evaluation. This also applies to Upset Transient 8 (Inadvertent Startup of an Inactive Loop).

The staff reviewed Westinghouse Commercial Atomic Power (WCAP)-14778 to confirm consideration of those loop out of service transients in the design analysis. On the basis that consideration of additional transients in the fatigue analysis generates conservative design results, the staff finds the use of transients from the steam generator replacement/uprating analysis for reactor vessel components acceptable.

On the third question, the applicant responded;

> The inadvertent auxiliary spray transient is a subcategory of the umbrella transient Inadvertent RCS Depressurization. The Inadvertent RCS Depressurization has 20 cycles with 10 of those cycles being the postulated as inadvertent auxiliary spray events. The inadvertent auxiliary spray events were not specifically listed, since the inadvertent auxiliary spray events were already included in the Inadvertent RCS Depressurization transients.

The staff reviewed the transient definition from the basis document, "Westinghouse System Standard Design Criteria 1.3," to confirm that the inadvertent auxiliary spray transient could be enveloped by the umbrella transient inadvertent RCS depressurization. On this basis, the staff finds this response acceptable.

The staff reviewed LRA Table 4.3-2 to confirm the 40-year design maximum reactor vessel CUF of 0.3744 for closure studs. The CUF value 0.562 accounts for the additional 20 years of extended operation by multiplying the 40-year design CUF of 0.3744 by 1.5. On this basis, the staff concluded that the analyses have been projected to the end of the period of extended operation per 10 CFR 54.21(c)(1) (ii).

4.3.1.1.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of the reactor vessel in LRA Section A1.2.2.1. On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address the reactor vessel is adequate.

4.3.1.1.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that, for, the analyses have been projected to the end of the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.1.2 Reactor Vessel Internals

4.3.1.2.1 Summary of Technical Information in the Application

LRA Section 4.3.1.2 summarizes the evaluation of reactor vessel internals for the period of extended operation. There is a TLAA for the reactor vessel internals. The NSSS design transients are those shown in the steam generator replacement/uprating analysis, in which 40-year design CUF values also were determined. The reactor vessel internals fatigue analysis demonstrated that, if exposed to a bounding set of postulated transient cycles, reactor vessel internals component CUF values would not exceed 1.0.

For the reactor vessel internals, the 40-year design fatigue usage value is 0.52 for the core internals. Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF of 0.78. This value does not exceed the design limit of 1.0; therefore, the analysis has been projected to the period of extended operation per 10 CFR 54.21(c)(1) (ii).

4.3.1.2.2 Staff Evaluation

The staff reviewed the applicant's basis document WCAP-16353-P, "Harris Nuclear Plant Fatigue Evaluation for License Renewal," and confirmed the core internal CUF of 0.52 for the 40-year design life. The staff accepted the projection of the 60-year CUF of 0.78 by multiplying the 40-year CUF of 0.52 by 1.5.

On this basis, the staff concluded the analysis has been projected to the end of the period of extended operation per 10 CFR 54.21(c)(1) (ii).

4.3.1.2.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of the reactor vessel internals in LRA Section A.1.2.2.2. On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address the reactor vessel internals is adequate.

4.3.1.2.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that, for reactor vessel internals, the analyses have been projected to the end of the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.1.3 Control Rod Drive Mechanism

4.3.1.3.1 Summary of Technical Information in the Application

LRA Section 4.3.1.3 summarizes the evaluation of the control rod drive mechanism for the period of extended operation. There are TLAAs for several Control Rod Drive Mechanism (CRDM) subcomponents. The NSSS design transients are those shown in the steam generator replacement/uprating analysis, in which 40-year design CUF values also were determined. The CRDM fatigue analysis demonstrated that, if exposed to a bounding set of postulated transient cycles, CRDM component CUF values would not exceed 1.0.

For the CRDM, the highest 40-year design fatigue usage value is 0.99 for the "Lower Joint Canopy Area" (LRA Table 4.3-2). Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF of 1.49. This value exceeds the design limit of 1.0 and, therefore, requires an AMP. The Reactor Coolant Pressure Boundary Fatigue Monitoring Program will keep fatigue usage within the design limit or take appropriate re-evaluation or corrective action to manage the effects of fatigue on the CRDM for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

4.3.1.3.2 Staff Evaluation

The GALL Report recommends a fatigue monitoring program to manage metal fatigue according to 10 CFR 54.21(c)(1)(iii). The staff has evaluated the applicant's AMP B3.1,"Reactor Coolant Pressure Boundary Fatigue Monitoring Program," for monitoring and tracking the number of critical thermal and pressure transients for RCS components, determined that this program is acceptable to address metal fatigue of RCS components according to 10 CFR 54.21(c)(1)(iii), and documented its evaluation and acceptance in SER Section 3.0. On the basis that the applicant's action is consistent with the GALL Report recommendation, the staff finds that management of the effects of aging on intended functions will be adequate for the period of extended operation.

4.3.1.3.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of the CRDM in LRA Section A.1.2.2.3. On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address the CRDM is adequate.

4.3.1.3.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that 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 FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.1.4 Reactor Coolant Pumps

4.3.1.4.1 Summary of Technical Information in the Application

LRA Section 4.3.1.4 summarizes the evaluation of RCPs for the period of extended operation. The RCPs have been designed and analyzed to meet the ASME Code of record. The original design fatigue analysis used fatigue waiver requirements and showed the pumps as having a TLAA. The RCP fatigue analysis demonstrated that, if the RCPs were exposed to a bounding set of postulated transient cycles, the fatigue waiver would remain valid.

The current design fatigue analysis for the RCPs used the ASME Code NB-3222.4(d) waiver of fatigue requirements; therefore, determination of a 40-year or 60-year fatigue usage factor for the RCPs was unnecessary. Using the general approach described in LRA Section 4.3.1, the applicant made 60-year fatigue cycle projections for license renewal. Based on the projections, the fatigue waiver remains valid for 60 years of operation.

4.3.1.4.2 Staff Evaluation

The staff reviewed LRA Section 4.3.1.4,ASME Code Section III and NB-3222.4(d), which defines components not requiring analysis for cyclic service, and concluded that there is no significant cyclic change in temperature, pressure, or mechanical loading. The conditions addressed in NB-3222.4(d), remain valid for the period of extended operation; therefore, the fatigue waiver remains valid for the period of extended operation.

4.3.1.4.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of the RCPs in LRA Section A.1.2.2.4. On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address the RCPs is adequate.

4.3.1.4.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that, for the RCPs, the analyses remain valid for the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.1.5 Steam Generators

4.3.1.5.1 Summary of Technical Information in the Application

LRA Section 4.3.1.5 summarizes the evaluation of steam generators for the period of extended operation. There are TLAAs for several steam generator subcomponents. The use of transients from the steam generator replacement/uprating analysis is reasonable and limiting for the primary equipment with the exceptions of the pressurizer surge line and portions of the pressurizer lower head analyzed separately; therefore, the NSSS design transients are those shown in the steam generator replacement/uprating analysis, in which 40-year design CUF values also were determined. The steam generator fatigue analysis demonstrated that, if steam generator subcomponents were exposed to a bounding set of postulated transient cycles, component CUF values would not exceed 1.0 with the exceptions of the secondary manway bolts and the 4-inch inspection port bolts addressed in more detail below.

Other than those for the secondary manway bolts and the 4-inch inspection port bolts, the highest 40-year design fatigue usage value is 0.98 for minor shell taps. Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF of 1.47. This value exceeds the design limit of 1.0, and, therefore, requires an AMP.

The Reactor Coolant Pressure Boundary Fatigue Monitoring Program will keep fatigue usage within the design limit or take appropriate re-evaluation or corrective action to manage the effects of fatigue on the steam generator (other than the secondary manway bolts and the 4-inch inspection port bolts) for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

The steam generator secondary manway bolts and 4-inch inspection port bolts have 40-year design fatigue usage factors over 1.0. These components were "to be replaced based on a replacement schedule;" however, the applicant reanalyzed the steam generator secondary manway cover bolts and 4-inch inspection port bolts to remove unnecessary conservatism. The update changed only the number of unit loading and unit unloading transient cycles in the previous design analysis. Each transient was to occur 2000 times over the life of the plant, a number still greater than the best estimate number in the previous design analysis. Reanalysis of the usage factor for the secondary manway bolts and the 4-inch inspection port bolts used 40-year design cycles for all transients except the unit-loading and unit-unloading transients. These transients were limited to 2,000 cycles each compared to the 18,300 cycles for normal condition transients 3 and 4. The calculated usage for the bolts based on this transient set is as follows:

- Secondary Manway Cover Bolts: Fatigue Usage = 0.83
- 4-inch inspection port bolts: Fatigue Usage = 0.81

Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields:

- Secondary Manway Cover Bolts: Fatigue Usage = 1.245
- 4-inch inspection port bolts: Fatigue Usage = 1.215

These values exceed the design limit of 1.0 and, therefore, require an AMP. The Reactor Coolant Pressure Boundary Fatigue Management Program will maintain the design allowable cycles for all transients (except unit-loading and unit-unloading) and the reduced number of unit loading and unit unloading transients or take appropriate re-evaluation or corrective action to manage the effects of fatigue on the secondary manway bolts and the 4-inch inspection port bolts for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

4.3.1.5.2 Staff Evaluation

The staff reviewed LRA Section 4.3.1.5 to verify, pursuant to 10 CFR 54.21(c)(1)(iii), that management of the effects of aging on intended functions will be adequate for the period of extended operation.

During audit and review, the staff confirmed that steam generator components will be managed under a cycle-based fatigue monitoring program. The staff also confirmed that analysis of the steam generator secondary manway cover bolts and 4-inch inspection port bolts fatigue evaluations was based on design transient cycles except the number of unit-loading and unit-unloading transient cycles assumed to occur 2000 times over the life of the plant; therefore, the enhanced Fatigue Management Program will track these cycles with a limit of 2000 cycles and an alarm limit of 1500 cycles. In the applicant's letter dated August 31, 2007, Commitment 32 stated that the enhanced fatigue monitoring program will address corrective actions through the Corrective Action Program for components exceeding alarm limits, including a revised fatigue analysis or repair or replacement of the component. In this letter, the applicant also set the cycle/transient alarm limit at around 75 percent of the design basis cycle/transient and provided an adequate time frame for corrective actions. On these bases, the staff concluded that the applicant's alarm limit for the cycle-based fatigue management program is adequate.

The GALL Report recommends a fatigue monitoring program to manage metal fatigue according to 10 CFR 54.21(c)(1)(iii). The staff has evaluated the applicant's AMP B3.1, "Reactor Coolant Pressure Boundary Fatigue Monitoring Program," for monitoring and tracking the number of critical thermal and pressure transients for RCS components, determined that this program is acceptable to address metal fatigue of RCS components according to 10 CFR 54.21(c)(1)(iii), and documented its evaluation and acceptance in SER Section 3.0. On the basis that the applicant's action is consistent with the GALL Report recommendation, the staff finds that management of the effects of aging on intended function will be adequate for the period of extended operation.

4.3.1.5.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of steam generators in LRA Section A.1.2.2.5. On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address steam generators is adequate.

4.3.1.5.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that 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 FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.1.6 Pressurizer

4.3.1.6.1 Summary of Technical Information in the Application

LRA Section 4.3.1.6 summarizes the evaluation of the pressurizer for the period of extended operation. There are TLAAs for several pressurizer subcomponents. The use of transients from the steam generator replacement/uprating analysis is reasonable and limiting for the primary equipment with the exceptions of the pressurizer surge line and portions of the pressurizer lower head analyzed separately; therefore, the NSSS design transients are those shown in the steam generator replacement/uprating analysis, in which 40-year design CUF values also were determined.

The pressurizer fatigue analysis demonstrated that, if pressurizer subcomponents were exposed to a bounding set of postulated transient cycles, CUF values would not exceed 1.0 for all components; however, certain pressurizer lower head locations are not bounded by the original design fatigue analysis because it did not consider insurge/outsurge transients discovered subsequently.

For the pressurizer (other than the lower head and surge line nozzle), the highest 40-year design fatigue usage value is 1.00 for the "Trunnion Bolt Hole" (LRA Table 4.3-2). Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF of 1.50.

The applicant used Westinghouse Owners Group (WOG) recommendations to address operational pressurizer insurge/outsurge transients by reviewing plant operating records in sufficient detail to determine pressurizer insurge/outsurge transients for past operation, updating pressurizer lower head and surge nozzle transients to reflect past and projected future operations, and evaluating the impact of the updated transients on the structural integrity of the pressurizer. The WOG also recommended operating strategies that may be useful in addressing the insurge/outsurge issue. On January 20, 1994, the applicant adopted the modified operating procedures recommended by the WOG to mitigate pressurizer insurge/outsurge transients.

The applicant used plant data from hot functional testing to January 20, 1994, to establish pre-modified operating procedure transients that represent past plant heat-up and cool-down operations and collected and processed plant data from July 19, 1999, to October 18, 2004, for post-modified operating procedures operations. The 5.26 years of data history with the pre-modified operating procedure transients was projected to predict 60-year fatigue usage based on current operating practices.

Fatigue evaluations of the pressurizer lower head and surge line nozzle used the online monitoring and Westinghouse proprietary design analysis features of the WESTEMS™ Integrated Diagnostics and Monitoring System. The fatigue evaluations follow the procedures of

ASME Code, Section III, NB-3200. Calculations of stress ranges, cycle pairing, and fatigue usage factors were by use of WESTEMS™ consistent with the ASME Code and WOG recommendations.

The fatigue evaluations at critical locations of the pressurizer lower head (including the pressurizer surge line nozzle) and of the surge line RCS hot leg nozzle were based upon pre-modified operating procedure transients with the post-modified operating procedure transients that include the effects of insurge/outsurge and surge line stratification. These transients were developed based upon plant-specific data and WOG information and guidelines. The predicted fatigue usage was determined assuming future operations following current operating procedures.

For 40 years of plant life, the pressurizer lower head has the highest fatigue usage of 0.36 at the inside surface of the lower head at the heater penetration region. Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a fatigue usage of 0.54. Evaluation of this location also accounted for the effects of reactor water environment on fatigue. The 60-year fatigue usage for this location is 1.35 as shown in LRA Table 4.3-3.

For the pressurizer, the maximum fatigue usage for 60 years of operation is 1.35. This value exceeds the design limit of 1.0 and, therefore, requires an AMP. The Reactor Coolant Pressure Boundary Fatigue Monitoring Program will maintain the design limit fatigue usage or take appropriate re-evaluation or corrective action to manage the effects of fatigue on the pressurizer for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

4.3.1.6.2 Staff Evaluation

During audit and review, the staff asked the applicant what components are in the stress-based fatigue monitoring portion of the HNP program. The applicant responded as follows:

> The HNP Fatigue Evaluation for License Renewal (WCAP-16353-P) resulted in the following locations recommended for inclusion into the program.
>
> - Pressurizer Lower Head
> - Pressurizer Surge Line
> - CVCS Piping and Heat Exchanger
>
> Based on the Westinghouse recommendations, the HNP fatigue monitoring program will be enhanced to include the above components by monitoring fatigue usage for these locations using online fatigue monitoring software.

In this letter, the applicant also indicated its stress-based fatigue monitoring locations and stress-based alarm limit of 0.9. On the basis that the 0.9 alarm limit will provide adequate time for actions, the staff concluded that the applicant's stress-based alarm limit is adequate. For all other locations managed through a cycle-based monitoring program, the applicant also provided its alarm limit. Commitment 32 states that the enhanced program will address corrective actions through the Corrective Action Program for components exceeding alarm limits, including a revised fatigue analysis or repair or replacement of the component.

LRA Amendment 2 states that the applicant used plant data from July 19, 1999, to October 18, 2004, to predict 60-year fatigue usage based on current operating practices. The staff does not agree with this prediction, which used 5.26 years of data to determine the next 40 years of operation transients; however, the applicant, by letter date January 17, 2008, committed to a stress-based fatigue monitoring program to manage those components. On this basis, the staff finds this LRA amendment acceptable. Therefore the applicant projections will not be used. The applicant will manage the effects of aging for the period of extended operation.

LRA Amendment 2 also states that the pressurizer lower head heater penetration region has the highest fatigue usage (0.36) for the 40 years of plant life. LRA Table 4.3-2 lists a design fatigue usage factor of 0.909 for this location. The staff asked the applicant to address the difference. This item was confirmatory item (CI) 4.3 and needed the applicant's docketed response to complete the staff's review.

In letter dated April 23, 2008, the applicant stated that HNP will update the piping design specification to reflect the current design basis operational transients used in the Time-Limited Aging Analyses for the reactor coolant pressure boundary (See Commitment No. 37). The applicant also amended LRA FSAR Supplement Section A.1.2.2.2.10 to indicate that the TLAA on metal fatigue of the charging nozzle, surge line, and pressurizer lower head and surge nozzle will be managed in accordance with the 10 CFR 54.21(c)(1)(iii). This is consistent with the applicant's TLAA on metal fatigue of the Class 1 piping components (as provided in LRA Section 4.3.5), which indicates that the Fatigue Monitoring Program will be used to manage the effects of aging for these components in accordance with the TLAA acceptance criterion requirement in 10 CFR 54.21(c)(1)(iii).

Based on this review, the staff finds that the applicant has appropriately addressed the staff's confirmatory item on the TLAA on metal fatigue of the reactor coolant pressure boundary. Confirmatory Item 4.3 is closed.

During the audit and review, the staff asked the applicant to explain the input of stresses to apply the stress transfer function of fatigue analysis software, WESTEMS™, to the stressed components or the stress intensity and asked for input and results of any benchmarking problems for pressure, temperature, or moment loadings.

The applicant's response is in pages 67 to 93 of Enclosure 3 of LRA Amendment 2 by letter dated August 31, 2007.

The staff reviewed the applicant's response explaining the method for the stress transfer function of fatigue analysis software WESTEMS. On the basis of its review, the staff confirmed that the applicant superimposed stress at the component stress level for each time step and for each applied loading type. The staff concluded that the method is in accordance with ASME Section III, Division 1, NB-3200 criteria.

The applicant also stated,

> The verification of fatigue analysis software thermal and mechanical stress calculations have been performed in the programs verification and validation documentation. However, each application verification of the finite element model and of the final thermal transfer function databases should be performed in order to show applicability to the problem being modeled. To do this for mechanical loads, Westinghouse verifies the finite element model results by comparing them to the expected theoretical values. For the time varying thermal results, the applicant performs thermal stress analyses using both the finite element program and WESTEMS™."

On the basis that verified fatigue analysis software stress results had the theoretical values and traditional finite element analysis, the staff finds the applicant's transfer function method for evaluating stress results acceptable.

The staff also reviewed the applicant's benchmark verification results plotted in Figures B-1 through B-11 and additional results of samples 1 and 2 all indicating that the stress results generated from fatigue analysis software and those generated from traditional finite element ANSYS analysis have negligible differences. On this basis, the staff concludes that stress evaluation by fatigue analysis software is acceptable.

The GALL Report recommends a fatigue monitoring program to manage metal fatigue according to 10 CFR 54.21(c)(1)(iii). The staff has evaluated the applicant's AMP B3.1, "Reactor Coolant Pressure Boundary Fatigue Monitoring Program," for monitoring and tracking the number of critical thermal and pressure transients for RCS components, determined that this program is acceptable to address metal fatigue of RCS components according to 10 CFR 54.21(c)(1)(iii), and documented its evaluation and acceptance in SER Section 3.0. On the basis that the applicant's action is consistent with the GALL Report recommendation, the staff finds that management of the effects of aging on intended functions will be adequate for the period of extended operation.

4.3.1.6.3 FSAR Supplement

The applicant provided an FSAR supplement summary description, as amended by letter dated April 23, 2208, of its TLAA evaluation of the pressurizer in LRA Section A.1.2.2.6. On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address pressurizer is adequate.

4.3.1.6.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that management of the effects of aging on intended functions will be adequate for the period of extended operation. The staff also concludes that the FSAR supplement is an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.1.7 Reactor Coolant Pressure Boundary Piping (ASME Class 1)

4.3.1.7.1 Summary of Technical Information in the Application

LRA Section 4.3.1.7 summarizes the evaluation of RCPB piping (ASME Class 1) for the period of extended operation. There are TLAAs for RCPB piping components. The use of transients from the steam generator replacement/uprating analysis is reasonable and limiting for the primary equipment with the exceptions of the pressurizer surge line and portions of the pressurizer lower head analyzed separately; therefore the NSSS design transients are those shown in the steam generator replacement/uprating analysis, in which 40-year design CUF values also were determined. The RCPB piping fatigue analysis demonstrated that, if the RCPB piping components were exposed to a bounding set of postulated transient cycles, their CUF values would not exceed 1.0; however, the pressurizer surge line is not bounded by the original design fatigue analysis.

In response to NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," the applicant evaluated the pressurizer surge line stratification transients separately for 40 years of operation.

For component parts of the RCPB piping, the highest 40-year design fatigue usage value is 0.98 for the pressurizer spray piping (LRA Table 4.3-2) before evaluation of the effects of reactor water environments on fatigue (LRA Subsection 4.3.3). Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF of 1.47.

Accounting for the effects of reactor water environments on fatigue, the highest 60-year fatigue usage is 2.120 for the pressurizer surge line as shown in LRA Table 4.3-3.

As these values exceed the design limit of 1.0, they require an AMP. The Reactor Coolant Pressure Boundary Fatigue Monitoring Program will maintain the design limit fatigue usage or take appropriate re-evaluation or corrective action to manage the effects of fatigue on the pressurizer for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

4.3.1.7.2 Staff Evaluation

The staff reviewed LRA Section 4.3.1.7 and LRA Table 4.3-2, which lists design fatigue usage factors. Section 4.3.1.7 addresses the pressurizer spray piping and surge line piping fatigue management only and not other Class 1 piping fatigue management. The staff requested from the applicant clarification addressing all the Class 1 piping.

In a letter dated January 17, 2008, the applicant clarified that the basis for aging management in LRA Section 4.3.17 should have applied to the entire scope of the Class 1 piping for HNP, and should not have been limited to only pressurizer spray piping and surge line piping. In this response, the applicant amended its LRA to state that:

Therefore, the effects of fatigue on the reactor coolant pressure boundary piping will be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

This LRA amendment expands the scope of the applicant's metal fatigue assessment in LRA Section 4.3.1.7 to the entire Class 1 piping in the reactor coolant pressure boundary and addresses the staff's issue.

The staff noted that Footnote C of LRA Table 4.3-3 had indicated that the design basis transients for the surge line, charging nozzle, and pressurizer lower head and surge nozzle had been redefined. The staff's position is that an ASME design report should follow design specification and that if the design conditions change, an updated design specification should reflect the change(s). In a supplemental question (followup question), the staff asked the applicant to: (1) clarify what the redefined transients are that had been mentioned in Footnote C of LRA Table 4.3-3 and (2) clarify whether the piping design specification had been updated to address the redefined transients mentioned in this footnote.

The applicant responded to the staff's followup question by letter dated January 17, 2007. In this letter (Audit Question LRA 4.3.3-5 [Followup] Response in Enclosure 1), the applicant provided a summary of the transients that were redefined for the surge line, charging nozzle, and pressurizer. The applicant stated that the design specification had not been updated to reflect the redefined transients for the surge line, charging nozzle, and pressurizer lower head and surge nozzle.

The staff position is that an ASME design report should follow design specification. If design conditions change, an updated design specification should reflect the change(s). The applicant has not updated the piping design specification. The LRA does not currently include a commitment to update the design specification for the surge line, charging nozzle, and pressurizer lower head and surge nozzle based on the reanalyses that were performed by the applicant (as discussed in the followup response to Question 4.3.3-6). Thus, the issue on whether the applicant currently reflects the redefined transients in the design basis CUF calculations for the surge line, charging nozzle, and pressurizer lower head and surge nozzle remains a confirmatory item. This was CI 4.3.

In letter dated April 23, 2008, the applicant stated that HNP will update the piping design specification to reflect the current design basis operational transients used in the Time-Limited Aging Analyses for the reactor coolant pressure boundary (See Commitment No. 37). The applicant also amended LRA FSAR Supplement Section A.1.2.2.2.10 to indicate that the TLAA on metal fatigue of the charging nozzle, surge line, and pressurizer lower head and surge nozzle will be managed in accordance with the 10 CFR 54.21(c)(1)(iii). This is consistent with the applicant's TLAA on metal fatigue of the Class 1 piping components (as provided in LRA Section 4.3.5), which indicates that the Fatigue Monitoring Program will be used to manage the effects of aging for these components in accordance with the TLAA acceptance criterion requirement in 10 CFR 54.21(c)(1)(iii).

Based on this review, the staff finds that the applicant has appropriately addressed the staff's confirmatory item on the TLAA on metal fatigue of the reactor coolant pressure boundary. Confirmatory Item 4.3 is closed.

4.3.1.7.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of RCPB Piping (ASME Class 1) in LRA Section A.1.2.2.7 stating that the effects of fatigue on the pressurizer will be managed for the period of extended operation. The staff asked the applicant to clarify whether all Class 1 piping will be managed instead of the pressurizer only.

By letter dated January 17, 2008, the applicant clarified that the basis for aging management in LRA Section 4.3.1.7 should have applied to the entire scope of the Class 1 piping for HNP, and should not have been limited to only pressurizer spray piping and surge line piping. In this response, the applicant amended LRA Section A.1.2.2.7 to state that:

> Therefore, the effects of fatigue on the reactor coolant pressure boundary piping will be managed for the period of extended operation.

This amendment of LRA Section A.1.2.2.7 expands the scope of the applicant's FSAR supplement on the metal fatigue assessment in LRA Section 4.3.1.7 to the entire Class 1 piping in the reactor coolant pressure boundary.

In SER Section 4.3.1.7, the staff determined that the applicant had redefined the design basis transients for the surge line, charging nozzle, and pressurizer lower head and surge nozzle but had not updated the design specification for these components to reflect the redefined transients used in the fatigue assessment for these components. The applicant, in a teleconference, agreed to add Commitment No. 37 to update, prior to the period of extended operation, the design specifications to reflect current design basis transients. This is to be formalized in a docketed correspondence. This was CI 4.3.

In letter dated April 23, 2008, the applicant stated that HNP will update the piping design specification to reflect the current design basis operational transients used in the Time-Limited Aging Analyses for the reactor coolant pressure boundary (See Commitment No. 37). The applicant also amended LRA FSAR Supplement Section A.1.2.2.2.10 to indicate that the TLAA on metal fatigue of the charging nozzle, surge line, and pressurizer lower head and surge nozzle will be managed in accordance with the 10 CFR 54.21(c)(1)(iii). This is consistent with the applicant's TLAA on metal fatigue of the Class 1 piping components (as provided in LRA Section 4.3.5), which indicates that the Fatigue Monitoring Program will be used to manage the effects of aging for these components in accordance with the TLAA acceptance criterion requirement in 10 CFR 54.21(c)(1)(iii).

Based on this review, the staff finds that the applicant has appropriately addressed the staff's confirmatory item on the TLAA on metal fatigue of the reactor coolant pressure boundary. Confirmatory Item 4.3 is closed.

On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address RCPB piping (ASME Class 1) is inadequate.

4.3.1.7.4 Conclusion

On the basis of its review, as discussed above, with the resolution of the confirmatory item, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that 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 FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.2 Implicit Fatigue Analysis (ASME Class 2, Class 3, and ANSI B31.1 Piping)

4.3.2.1 ASME Class 2 and 3 Piping

4.3.2.1.1 Summary of Technical Information in the Application

LRA Section 4.3.2.1 summarizes the evaluation of ASME Classes 2 and 3 piping for the period of extended operation. Auxiliary piping designed to ASME Section III, Code Classes 2 and 3 requirements required no explicit fatigue evaluation. Instead, for such piping the code implicitly treats fatigue using a stress range reduction factor (f), which is a function of the total number of thermal expansion stress range cycles, equal to 1.0 for up to 7,000 cycles. For greater numbers of cycles, f may be reduced further, reducing the thermal expansion range stress allowable. The applicant's fatigue evaluation for Classes 2 and 3 piping shows the original design evaluations for Classes 2 and 3 components remain valid for 60 years.

The affected Classes 2 and 3 piping are effectively extensions of the adjacent Class 1 piping; therefore, the cycle count depends closely on reactor operating cycles and can be estimated by a review of the limiting reactor coolant system design transients in FSAR Table 3.9.1-1. Of those listed normal conditions likely to produce full-range thermal cycles in a 40-year plant lifetime are the 200 heatup and cooldown cycles. The assumption that all upset conditions lead to full-range thermal cycles adds 980 cycles for a total of 1180 occurrences. The 980 cycles are equal to the summation of upset condition transients 1 through 12 plus five operating-basis earthquakes at 10 cycles each. For the 60-year period of extended operation, the number of full-range thermal cycles for these piping analyses would be increased proportionally to 1770, only a fraction of the 7000 full-range thermal cycles for a stress range reduction factor of 1.0; therefore, the analysis for Classes 2 and 3 piping has been projected to the period of extended operation per 10 CFR 54.21(c)(1) (ii).

4.3.2.1.2 Staff Evaluation

During the audit and review, the staff asked the applicant why the Class 1 piping thermal transients are relevant to Classes 2 and 3 piping. LRA Amendment 2 dated August 31, 2007, states, "The CL 2 & 3 piping are the extension of Class piping and subject to same cycle counting; therefore, the cycle count depends closely on reactor operating cycles."

The staff sought supplement information on this response and, in a supplemental (followup) question, asked the applicant to clarify whether the LRA amendment in LRA Amendment 2 postulates that the Class 2 and 3 piping is subject to the same design transients as that for Class 1 piping.

In its response dated January 17, 2008, the applicant clarified that the assessment of the Class 2 and 3 piping is based on an assessment of the number of full thermal transient cycles (full temperature cycles) that the piping is projected to be subjected to. This is consistent with the staff's basis for evaluating ASME Code Class 2 and 3 piping in SRP-LR Sections 4.3.2.1.2 and 4.3.2.1.4, and is acceptable. The staff's supplemental question on the Class 2 and 3 piping is resolved.

In LRA Amendment 2 dated August 31, 2007, the applicant clarified how its projections of the full thermal transient cycles for the Class 2 and 3 piping was performed. In this response, the applicant clarified that the full thermal transient cycles for the Class 2 and 3 piping are considerably less frequent and of a smaller temperature range than those analyzed for the plant's heatups and cooldowns of the reactor coolant pressure boundary (i.e., for the Class 1 pressure boundary components) and that as a result, the applicant uses the heatups and cooldowns as a conservative basis for estimating the full thermal transients that are applicable to the Class 2 and 3 piping components. The applicant also clarified that it conservatively included all assumed upset transients for the plant in 60-year projections of the full thermal transients for the Class 2 and 3 piping components and that it applied a factor 1.5 (i.e. a factor of 60/40) to these 40-year totals, arriving at a 60-year full thermal transient projection of 1770 cycles for the Class 2 and 3 piping components. The applicant stated that, based on this projection, the number of full thermal transient cycles for the Class 2 and 3 piping over a 60-year life is still less 7000 cycles and that, based on this number, the maximum allowable stress range for the Class 2 and 3 piping would not need to be reduced and that the original design basis fatigue calculation for these components remains valid for the period of extended operation. The staff finds this to be acceptable because it is in conformance with the staff's metal fatigue criteria for evaluating these components in SRP-LR Sections 4.3.2.1.2 and 4.3.2.1.4.

On this basis, the staff finds the Class 2 and 3 piping fatigue analyses to be acceptable because: (1) the applicant has used a conservative basis for estimating the 60-year projections for full thermal transients that apply to the Class 2 and 3 piping components, (2) based on these projections, the applicant has demonstrated that design basis fatigue analysis for the Class 2 and 3 piping components will remain valid for the period of extended operation, and (3) applicant's basis for evaluating the fatigue analysis for the Class 2 and 3 is in conformance with the staff's criteria in SRP-LR Sections 4.3.2.1.2 and 4.3.2.1.4.

On the basis of this review, the staff concludes that the applicant has demonstrated that the fatigue analysis for the Class 2 and 3 piping remains valid for the period of extended operation in accordance with the criterion in 10 CFR 54.21(c)(1)(i).

In the applicant's response dated January 17, 2008, the applicant also amended LRA Section 4.3.2.1 to verify that the metal fatigue Class 2 and 3 piping was determined to be acceptable in accordance with the criterion in 10 CFR 54.21(c)(1)(i), in that the current TLAA analysis has been determined to be valid for the period of extended operation.

4.3.2.1.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of ASME Classes 2 and 3 piping in LRA Section A.1.2.2.8. By letter dated January 17, 2008, the applicant amended the LRA to indicate that the fatigue analysis for the Class 2 and 3 piping would be dispositioned and found acceptable in accordance with the criterion in 10 CFR 54.21(c)(1)(i) in that the applicant has provided a valid basis for demonstrating that the number of full thermal transient cycles for the Class 2 and 3 piping will be less than 7000 cycles over a 60-year licensed plant life. The staff also verified that the amendment of the LRA in the applicant's response dated January 17, 2008, included an amendment of FSAR supplement Section A.1.2.2.8 to reflect the change in the LRA.

In SER Section 4.3.2.1.3, the staff provided its basis for concluding that the applicant had provided an acceptable basis for accepting the TLAA on metal fatigue of the Class 2 and 3 piping in accordance with the TLAA acceptance criterion in 10 CFR 54.21(c)(1)(i). On the basis of this review, the staff concludes that FSAR supplement Section A.1.2.2.9 with respect to the applicant's TLAA on metal fatigue of the Class 2 and 3 piping, as amended in the applicant's response dated January 17, 2008, is adequate.

4.3.2.1.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that, for ASME Code Class 2 and 3 piping , the analyses remain valid for the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.2.2 ANSI B31.1 Piping

4.3.2.2.1 Summary of Technical Information in the Application

LRA Section 4.3.2.2 summarizes the evaluation of ANSI B31.1 piping for the period of extended operation. In addition to ASME Classes 2 and Class 3 piping, the scope of license renewal includes nonsafety-related piping designed to ANSI B31.1. Auxiliary piping designed to ANSI B31.1 requirements required no explicit fatigue evaluation. Instead, for ANSI B31.1 piping, the "power piping" code implicitly treats fatigue using a stress allowable reduction factor (f), which is a function of the total number of thermal expansion stress range cycles, equal to 1.0 for up to 7,000 cycles. For greater number of cycles, f may be reduced further, reducing the thermal expansion range stress allowable.

For the main feedwater system and associated systems (*e.g.*, condensate system) and main steam system and associated systems (*e.g.*, steam generator system), anticipated thermal cycles correspond to heatup and cooldown cycles. For the 60-year period of extended operation, the number of full-range thermal cycles for these piping analyses would be increased proportionally to 300; therefore, main feedwater and main steam system components will not experience 7000 cycles during the period of extended operation.

The auxiliary feedwater system supplies feedwater to the secondary side of the steam generators when the normal feedwater system is not available to maintain the heat sink capabilities of the steam generator. The system is an alternative to the feedwater system during startup, hot standby, and cooldown and also functions as an engineered safeguards system. HNP relies directly on the auxiliary feedwater system to prevent core damage during plant transients caused by loss of normal feedwater flow, steam line rupture, main feedwater line rupture, loss of coolant accidents (LOCAs), loss of offsite power, or any combination of these causes by supplying feedwater to the unaffected steam generators to maintain their inherent heat sink capability. The total numbers of cycles projected for 40 years of operation are as follows: 200 heatup and cooldown cycles, 2000 cycles of feedwater cycling at hot standby, 980 cycles for all upset conditions, 240 cycles of quarterly auxiliary feedwater pump tests in accordance with ASME Code Section XI, and 40 cycles of tests per plant technical specifications for a total of 3460 cycles. For the 60-year period of extended operation, the number of full-range thermal cycles for these piping analyses would increase proportionally to 5,190; therefore, auxiliary feedwater components will not experience 7000 cycles during the period of extended operation.

The diesel generators in the emergency diesel generator system undergo monthly surveillance tests in accordance with plant technical specifications. For the 60-year period of extended operation, the number of full-range thermal cycles for these piping analyses would increase proportionally to 720; therefore, the emergency diesel generator diesel exhaust piping will experience significantly fewer than 7000 equivalent full-temperature cycles during the period of extended operation.

The diesel generator in the security power system undergoes a monthly surveillance test to satisfy fire protection program surveillance requirements. For the 60-year period of extended operation, the number of full-range thermal cycles for these piping analyses would increase proportionally to 720; therefore, the security diesel generator diesel exhaust piping will experience significantly fewer than 7000 equivalent full-temperature cycles during the period of extended operation.

The diesel-driven fire pump in the fire protection system undergoes a monthly test to satisfy fire protection program surveillance requirements. For the 60-year period of extended operation, the number of full-range thermal cycles for these piping analyses would increase proportionally to 720; therefore, the diesel-driven fire pump piping will experience significantly fewer than 7000 equivalent full-temperature cycles during the period of extended operation, and the analysis for ANSI B31.1 piping has been projected to the period of extended operation using per 10 CFR 54.21(c)(1) (ii).

4.3.2.2.2 Staff Evaluation

The staff reviewed the technical information in LRA Section 4.3.2, pertaining to the non-Class 1 fatigue analysis of piping, against the criteria contained in SRP-LR Section 4.3.2.1.2 and documented the results in the Audit Report.

SRP-LR Section 4.3.2.1.2.1 states that for piping designed or analyzed to ANSI B31.1 standards, the acceptance criteria is the existing fatigue strength reduction factors remain valid because the number of cycles would not be exceeded during the period of extended operation. Although ANSI B31.1 Code does not require explicit fatigue analysis, it considers fatigue implicitly in the design calculation by applying an allowable stress range reduction factor. Fatigue also can depend on the number of design thermal expansion cycles.

The staff reviewed the applicant's basis document which provided the basis and calculations for the metal fatigue. In the basis document, the applicant discussed the operating cycles for the piping, piping components, or piping elements in B31.1 piping systems, including but not limited to those in the main steam system, main feedwater system, condensate system, auxiliary feedwater system, and steam generator system. This also includes B31.1 piping components associated with the diesel generators in the emergency diesel generator system and the security power system and associated with the diesel-driven fire pump in the fire protection system. For these B31.1 piping systems, the applicant concluded that B31.1 piping, piping components, and piping elements will experience less than 7000 full thermal transient cycles for 60-years of licensed operation and that, based on this determination, the maximum allowable stress range for these components would not need to be reduced.

By letter dated August 31, 2007), the applicant supplemented the LRA and clarified that the number of startups and shutdowns for the Class 1 piping in the reactor coolant pressure boundary (i.e., 300 cycles) could be used as a conservative basis for estimating the number of full thermal transients that are projected for the B31.1 piping, piping components, and piping elements in the main steam, main feedwater, condensate, and steam generator systems through 60-years of licensed operations.

The staff finds this to be a valid basis for projecting the number of full thermal transient cycles for these B31.1 piping, piping components, and piping elements through 60-years of licensed operations because: (1) the full temperature range for startup/shutdown cycling of the reactor coolant pressure boundary is bounding for the full temperature ranges associated with operational/isolational cycling of these B31.1 systems, and (2) over the life of the plant, the number of times the reactor coolant pressure boundary is thermally cycled during plant startup/shutdowns will exceed the number of operational/isolational cycles that occur in these B31.1 systems. Thus, the staff concludes that the applicant has provided an acceptable basis for concluding that the number of full thermal transients for the B31.1 piping in these systems will be less 7000 cycles through 60 years of licensed operations and that the metal fatigue analysis for these systems will remain valid for the period of extended operation. This is acceptable because it is in conformance with the recommendations in SRP-LR Section 4.3.2.1.2.1.

By letter dated August 31, 2007, the applicant supplemented the LRA and provided its basis for concluding that 5190 cycles represents a conservative estimate of the number of full thermal transients that are projected for the B31.1 piping, piping components, and piping elements in the auxiliary feedwater system through 60 years of licensed operations. The applicant has based its 60-year full thermal transient projection for the auxiliary feedwater system piping on the number of plant startups and shutdowns that are projected to occur through 60 years of operation, as well as on the number of upset transients, the number of feedwater cycles during hot standby, the number of auxiliary feedwater pump tests that are required by the plant's inservice testing program (IST) program, and the number of auxiliary feedwater system functional tests that are required by technical specifications that are projected to occur through 60 years of operation.

The staff finds this to be an acceptable basis because: (1) the applicant's 60-year projection for the auxiliary system B31.1 piping is based not only the projected number of plant startups and shutdowns, but also on the number of auxiliary system actuations that are projected to occur during anticipated operational transients, required system testing, and system operation during hot standby, and (2) the applicant's projection includes a margin of 1.5 on the cycle projection to account for the period of extended operation. Thus, the staff concludes that the applicant has provided an acceptable basis for concluding that the number of full thermal transients for the B31.1 piping in the auxiliary feedwater system will be less than 7000 cycles through 60 years of licensed operations and that the metal fatigue analysis for this system will remain valid for the period of extended operation. This is acceptable because it is in conformance with the recommendations in SRP-LR Section 4.3.2.1.2.1.

The B31.1 piping associated with the emergency diesel generator system, security power system, and diesel-driven fire protection pump are not normally in service, but undergo a monthly system test in accordance plant technical specifications. The applicant estimated that the number of full thermal transients associated with these systems corresponds to the number of monthly actuations that are projected to occur in the system tests through 60 years of licensed operation (i.e., 720 full thermal cycle actuations).

The staff was of the opinion that the applicant should have included the number of time these systems were projected to actuate during system operational transients or other testing. However, the staff determined that, even if the number of plant trips represented in LRA Table 4.3-1 for upset conditions were accounted for in the projection with a safety factor of two (i.e., bringing the total to 1140), the number of full thermal transients for these systems would still be less than 7000 full thermal transient cycles. Thus, the staff concludes that the applicant has provided an acceptable basis for concluding that the metal fatigue assessment for the B31.1 piping, piping components, and piping elements associated with the emergency diesel generator system, security power system, and diesel-driven fire protection pump will remain valid for the period of extended operation. This is acceptable because it is in conformance with the recommendations in SRP-LR Section 4.3.2.1.2.1.

Based on this assessment, the staff concludes that: (1) the applicant has provided an acceptable basis to demonstrate that the number of full thermal transients for the B31.1 piping, piping components, and piping elements associated with the main steam, main feedwater, condensate, steam generator, auxiliary feedwater, emergency diesel generator, and security power systems, and with the diesel-driven fire protection pumps will be less than 7000 full thermal transient cycles through 60 years of licensed operation, and (2) this is acceptable because it is in conformance with the staff's criterion for acceptance in SRP-LR Section 4.3.2.1.2.1. On the basis of its audit and review, the staff concludes that the applicant has demonstrated that the metal fatigue analyses for these ANSI B31.1 piping systems will remain valid for the period of extended operation, as required by 10 CFR 54.21(c)(1)(i).

By letter dated January 17, 2008, the applicant amended the LRA to indicate that the fatigue analysis for the ANSI B31.1 piping would be dispositioned and found acceptable in accordance with the criterion in 10 CFR 54.21(c)(1)(i) in that the number full thermal transient cycles for the ANSI B31.1 piping are projected to be less than 7000 over a 60-year licensed plant life. The staff has verified that the applicant has used a conservative estimate of the number of full thermal transient cycles that are projected to occur in the ANSI B31.1 piping components through 60 years of licensed operations. Based on this assessment, the staff concludes that the applicant has provided an acceptable basis for accepting the TLAA on metal fatigue fo the ANSI B31.1 piping in accordance with 10 CFR 54.21(c)(1)(i).

4.3.2.2.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of ANSI B31.1 piping in LRA Section A.1.2.2.9. By letter dated January 17, 2008, the applicant amended the LRA to indicate that the fatigue analysis for the ANSI B31.1 piping would be dispositioned and found acceptable in accordance with the criterion in 10 CFR 54.21(c)(1)(i) in that the applicant has provided a valid basis for demonstrating that the number of full thermal transient cycles for the ANSI B31.1 piping will be less than 7000 cycles over a 60-year licensed operating period. The staff also verified that the amendment of the LRA in the applicant's letter dated January 17, 2008, included an amendment of FSAR supplement Section A.1.2.2.9 to reflect the change that the applicant is accepting this TLAA in accordance with the TLAA acceptance criterion in 10 CFR 54.21(c)(1)(i).

In SER Section 4.3.2.2.3, the staff provided its basis for concluding that the applicant had provided an acceptable basis for accepting the TLAA on metal fatigue of the ANSI B31.1 piping in accordance with the TLAA acceptance criterion in 10 CFR 54.21(c)(1)(i). Based on this assessment, the staff concludes that the applicant has provided an acceptable basis for accepting the TLAA on metal fatigue fo the ANSI B31.1 piping in accordance with 10 CFR 54.21(c)(1)(i). On the basis of this review, the staff concludes that FSAR supplement Section A.1.2.2.9 on the applicant's TLAA on metal fatigue of the ANSI B31.1 piping, as amended in the applicant's letter dated January 17, 2008, is adequate.

4.3.2.2.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that, for ANSI B31.1 piping, the analyses remain valid for the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.3 Environmentally-Assisted Fatigue Analysis

4.3.3.1 Summary of Technical Information in the Application

LRA Section 4.3.3 summarizes the evaluation of environmentally-assisted fatigue analysis for the period of extended operation. Reactor water environment effects on fatigue were evaluated for a subset of representative components selected based upon the evaluations in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Design Curves to Selected Nuclear Power Plant Components." Because the Class 1 piping was designed in the more recent history of Westinghouse plant design, locations selected corresponded to the Westinghouse newer vintage plant. Representative components evaluated are as follows:

- Reactor Vessel Shell and Lower Head
- Reactor Vessel Inlet and Outlet Nozzles
- Pressurizer Surge Line
- Charging Nozzle
- Safety Injection Nozzle
- Residual Heat Removal (RHR) System Class 1 Piping

In addition to these representative NUREG/CR-6260 locations, locations in the pressurizer lower head potentially subject to insurge/outsurge transients also were evaluated for reactor water environmental effects.

The methods for evaluating environmental effects on fatigue were based on NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels," NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," and NUREG/CR-6717, "Environmental Effects of Fatigue Crack Initiation in Piping and Pressure Vessel Steels." The applicant used environmental fatigue life correction factors to obtain adjusted cumulative fatigue usage, which includes the effects of reactor water environments.

For the charging nozzle, additional analyses for several "partial cycle" transients accounted for transients much less severe than design so they would not be counted as full design cycles. The ANSI B31.1 Power Piping Code, 1967 Edition, Section 102.3.2, provides the following equation and methodology for the mathematical determination of the number of equivalent full temperature range changes from the number of lesser temperature range changes:

$$N = N_E + r_1^5 N_1 + r_2^5 N_2 + \ldots r_n^5 N_n$$

Where: N = the number of equivalent full temperature cycles,
N_E = number of cycles at full temperature change for which expansion stress has been calculated,
$N_1, N_2 \ldots N_n$ = number of cycles at lesser temperature changes,
$r_1, r_2 \ldots r_n$ = ratio of lesser temperature cycles to the cycle for which the expansion stress has been calculated.

4.3.3.2 Staff Evaluation

The staff reviewed LRA Section 4.3.3 to verify (1) pursuant to10 CFR 54.21(c)(1)(ii) that the analyses have been projected to the end of the period of extended operation or (2) pursuant to

10 CFR 54.21(c)(1)(iii) that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

The staff reviewed LRA Section 4.3.3 against SRP-LR Section 4.3.3.2, "Generic Safety Issue." The SRP-LR recommends that license renewal applicants address Generic Safety Issue 190. To assess the impact of the reactor coolant environment on a sample of critical components, the SRP-LR states that applicants should address the recommendations as follows:

(1) The critical components include, as a minimum, those selected in NUREG/CR-6260.

(2) Evaluation of the sample of critical components has applied environmental correction factors to the ASME Code fatigue analyses.

(3) Formulas for calculating the environmental life correction factors are those in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels or approved technical equivalents.

In LRA Table 4.3-3, the applicant has evaluated the sample of critical components by applying environmental correction factors to the ASME Code fatigue analysis.

The staff confirmed that the critical components include those selected in NUREG/CR-6260 and that calculations of environmental life correction factors use NUREG/CR-6583 formulas for carbon and low-alloy steels and NUREG/CR-5704 formulas for austenitic stainless steels; therefore, the staff confirmed that the applicant has followed staff recommendations to assess the impact of the reactor coolant environment consistently with the SRP-LR.

The methodology described in ANSI B31.1 Power Piping Code, 1967 Edition, Section 102.3.2 for partial cycle counting does not apply to ASME Code Class 1 components and that ANSI B31.1 power piping thermal qualification does not consider the ranges of pressure, temperature, and moment as for Class 1 piping. The staff asked the applicant to justify use of the ANSI B31.1 code method for cycle reduction. In LRA Amendment 2, the applicant responded that an independent ASME Code Section III, Division I, Subsection NB fatigue evaluation has established a quantitative basis for application of the ANSI B31.1 cycle reduction methodology to cycle counting of HNP charging nozzle transients. The staff reviewed the result of the CUF evaluation. On the basis that the applicant's calculation results demonstrate a conservative fatigue usage factor, the staff finds this approach acceptable for this location and specific transient reduction only.

During the review of LRA Amendment 2, dated August 31, 2007, the staff noted that Column C of LRA Table 4.3-3 states that for the surge line, charging nozzle, and pressurizer lower head at heater penetration the CUF evaluation used redefined transients. The staff asked the applicant which transients had been redefined for the environmental fatigue analyses for these component locations and whether the design specification for these component locations had been updated based on the redefined transients for these components.

The applicant responded to the staff's follow-up question by letter dated January 17, 2007. In this letter (refer to the Audit Question LRA 4.3.3-5 [Followup] Response in Enclosure 1), the applicant provided a summary of the transients that were redefined for the surge line, charging nozzle, and pressurizer. In its response, the applicant also indicated that the design specification had not been updated to reflect the redefined transients for the surge line, charging nozzle, and pressurizer lower head and surge nozzle.

The staff position is that an ASME design report should follow design specification and that if design conditions change, an updated design specification should reflect the change(s). The applicant has not updated the piping design specification to reflect the redefinition of the design transients that are applicable to the surge line, the charging nozzle, and the pressurizer lower head and surge nozzle. The LRA does not currently include a commitment to update the design specification for these components based on the reanalyses that were performed by the applicant (as discussed in the followup response to Question 4.3.3-5). Thus, the issue on whether the applicant currently reflects the redefined transients in the design basis and environmental CUF calculations for the surge line, charging nozzle, and pressurizer lower head and surge nozzle was not properly addressed in the applicants response.

The applicant, in a teleconference, agreed to add Commitment No. 37 to update, prior to the period of extended operation, the design specifications to reflect current design basis transients. This is to be formalized in a docketed correspondence. This was CI 4.3.

In letter dated April 23, 2008, the applicant stated that HNP will update the piping design specification to reflect the current design basis operational transients used in the Time-Limited Aging Analyses for the reactor coolant pressure boundary (See Commitment No. 37). The applicant also amended LRA FSAR Supplement Section A.1.2.2.2.10 to indicate that the TLAA on metal fatigue of the charging nozzle, surge line, and pressurizer lower head and surge nozzle will be managed in accordance with the 10 CFR 54.21(c)(1)(iii). This is consistent with the applicant's TLAA on metal fatigue of the Class 1 piping components (as provided in LRA Section 4.3.5), which indicates that the Fatigue Monitoring Program will be used to manage the effects of aging for these components in accordance with the TLAA acceptance criterion requirement in 10 CFR 54.21(c)(1)(iii).

Based on this review, the staff finds that the applicant has appropriately addressed the staff's confirmatory item on the TLAA on metal fatigue of the reactor coolant pressure boundary. Confirmatory Item 4.3 is closed.

The staff reviewed LRA Table 4.3-3 to confirm that the applicant has evaluated bottom head junction, reactor vessel nozzles, and RHR piping CUFs by multiplying environmental correction factors by design fatigue usage factors and further multiplying by 1.5 to account for 60 years. Based on this review, the staff concluded that reactor vessel lower head and nozzles fatigue TLAAs have been projected through the period of extended operation in accordance with 10 CFR 54.21(C)(1)(ii). The other four components, surge line, charging nozzle and pressurizer lower head at heater penetration, will be within the scope of the applicant's Reactor Coolant Pressure Boundary Fatigue Monitoring Program to manage environmentally-assisted metal fatigue of the surge line, charging nozzle, safety-injection nozzle, and pressurizer lower head and surge nozzle in accordance with 10 CFR 54.21(c)(1)(iii). LRA Amendment 2, as provided in the applicant's letter dated August 31, 2007, does not indicate the method for management of the fatigue effects. The applicant, in a teleconference, agreed to provide the method of management for these components.

By letter dated January 17, 2008, the applicant clarified that the TLAA on environmentally-assisted metal fatigue of the surge line, charging line, safety injection nozzle, and pressurizer lower head in accordance with the TLAA acceptance criterion in 10 CFR 54.21(c)(1)(iii) and that the Reactor Coolant Pressure Boundary Fatigue Monitoring Program is credited to manage environmentally-assisted metal fatigue in these components for the period of extended operation.

The GALL Report recommends a fatigue monitoring program to manage metal fatigue in accordance with 10 CFR 54.21(c)(1)(iii). The staff has evaluated the applicant's AMP B3.1, "Reactor Coolant Pressure Boundary Fatigue Monitoring Program," for monitoring and tracking the number of critical thermal and pressure transients (cycle-based monitoring) for RCS components and for evaluating stress-based fatigue, determined that this program is acceptable to address metal fatigue of RCS components according to 10 CFR 54.21(c)(1)(iii), and documented its evaluation and acceptance in SER Section 3.0. On the basis that the applicant's action is consistent with the GALL Report recommendation, the staff finds that management of the effects of aging on intended functions will be adequate for the period of extended operation.

4.3.3.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of environmentally-assisted fatigue analysis in LRA Section A.1.2.2.10. The staff has determined that the current version of FSAR supplement Section A.1.2.2.10 indicates that the TLAA on environmentally-assisted metal fatigue of reactor coolant pressure boundary components was found acceptable for the period of extended operation. However, the staff has verified that the applicant has credited its Reactor Coolant Pressure Boundary Fatigue Monitoring Program to manage environmentally-assisted metal fatigue in the HNP surge line, charging nozzle, and pressurizer lower head and surge nozzle in accordance with 10 CFR 54.21(c)(1)(iii). Thus, FSAR Supplement Section A.1.1.38 and Commitment No. 32 are also applicable to the evaluation of this TLAA and the summary description in FSAR Supplement Section A.1.2.2.10 does not reflect this information.

The staff has verified that the applicant's Reactor Coolant Pressure Boundary Fatigue Monitoring Program, as enhanced in Commitment No. 32 is an AMP that is consistent with the staff's recommended program element criteria in GALL AMP X.M1, "Metal Fatigue of the Reactor Coolant Pressure Boundary." The staff has verified that the applicant included this acceptance criterion in FSAR Supplement Section A.1.1.38 and has included its commitment to manage the effects of aging in the surge line, charging nozzle, and pressurizer lower head and surge nozzle within the scope of Commitment No. 32, as provided in the applicant's letter dated January 17, 2008.

Based on this assessment, the staff concludes that the summary description in FSAR Supplement Section A.1.1.38 and the applicant's enhancement of the Reactor Coolant

Pressure Boundary Fatigue Monitoring Program, as given in LRA Commitment No. 32, tie in appropriately to the applicant's basis for accepting the TLAA on environmentally-assisted metal fatigue of the surge line, charging nozzle, and pressurizer lower head and surge nozzle. This is an acceptable basis for accepting the TLAA on environmentally-assisted metal fatigue, as assessed relative to the surge line, charging nozzle, and pressurizer lower head and surge nozzle, because it is in compliance with the staff acceptance basis in 10 CFR 54.21(c)(1)(iii). The staff's evaluation of the Reactor Coolant Pressure Boundary Fatigue Monitoring Program is given in SER Section 3.0.3.2.26.

In CI 4.3, the staff requested additional information to ensure that the applicant would provide a design specification for the surge line, the charging nozzle, and the pressurzier lower head and surge nozzle that was based on the redefined transients for these components, as discussed in the applicants follow-up response to Audit Question 4.3.3-5, dated January 17, 2008. The CI included a request to update FSAR supplement Section A.1.2.2.10 to reflect that the applicant is crediting its Reactor Coolant Pressure Boundary Fatigue Monitoring Program to manage environmentally-assisted metal fatigue in the HNP surge line, charging nozzle, and pressurizer lower head and surge nozzle in accordance with 10 CFR 54.21(c)(1)(iii). The staff's resolution of CI 4.3 on the acceptability of FSAR supplement Section A.1.2.2.10 was pending formalized docketed correspondence.

In letter dated April 23, 2008, the applicant stated that HNP will update the piping design specification to reflect the current design basis operational transients used in the Time-Limited Aging Analyses for the reactor coolant pressure boundary (See Commitment No. 37). The applicant also amended LRA FSAR Supplement Section A.1.2.2.2.10 to indicate that the TLAA on metal fatigue of the charging nozzle, surge line, and pressurizer lower head and surge nozzle will be managed in accordance with the 10 CFR 54.21(c)(1)(iii). This is consistent with the applicant's TLAA on metal fatigue of the Class 1 piping components (as provided in LRA Section 4.3.5), which indicates that the Fatigue Monitoring Program will be used to manage the effects of aging for these components in accordance with the TLAA acceptance criterion requirement in 10 CFR 54.21(c)(1)(iii).

Based on this review, the staff finds that the applicant has appropriately addressed the staff's confirmatory item on the TLAA on metal fatigue of the reactor coolant pressure boundary. Confirmatory Item 4.3 is closed.

On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address environmentally-assisted fatigue analysis is adequate.

4.3.3.4 Conclusion

On the basis of its review, the staff concludes that, pursuant to 10 CFR 54.21(c)(1)(ii), the applicant has demonstrated that the analyses have been projected to the end of the period of extended operation and, that pursuant to 10 CFR 54.21(c)(1)(iii), with resolution of CI 4.3, the applicant has demonstrated that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The staff also concludes that with resolution of CI 4.3, the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.4 RCS Loop Piping Leak-Before-Break Analysis

4.3.4.1 Summary of Technical Information in the Application

LRA Section 4.3.4 summarizes the evaluation of the RCS loop piping leak-before-break analysis for the period of extended operation. In accordance with the CLB, a leak-before-break (LBB) analysis showed that any potential leak that develops in the RCS loop piping can be detected by plant leak monitoring systems before a postulated crack causing the leak would grow to unstable proportions during the 40-year plant life. LBB evaluations postulate a surface flaw at a limiting stress location and demonstrate that a through-wall crack will not be the result of exposure to a lifetime of design transients. A separate evaluation assumes a through-wall crack of sufficient size for the resultant leakage to be detected easily by the existing leakage monitoring system and then demonstrates that, even under maximum faulted loads, the crack is much smaller (with margin) than a critical flaw size that could grow to pipe failure. The aging effects to be addressed during the period of extended operation include thermal aging of the primary loop piping components and fatigue crack growth.

WCAP-14549-P, Addendum 1, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Harris Nuclear Plant for the License Renewal Program," is a new LBB calculation applicable to large-bore RCS piping and components with allowances for reduction of fracture toughness of cast austenitic stainless steel due to thermal embrittlement during a 60-year operating period, concluded that:

- Stress corrosion cracking is precluded by use of fracture-resistant materials in the piping system and controls on reactor coolant chemistry, temperature, pressure, and flow during normal operation. An Electric Power Research Institute material reliability program is underway to address the Alloy 82/182 primary water stress corrosion cracking issue for the industry due to the V. C. Summer cracking incident; however, per calculations for Alloy 82/182 locations this material is not bounding.
- Water hammer should not occur in the RCS piping because of system design, testing, and operational considerations.
- The effects of low- and high-cycle fatigue on primary piping integrity are negligible. The fatigue crack growth evaluated is insignificant.
- There is a margin of 10 between the leak rate of small stable leakage flaws and the capability (1 gpm) of the RCS pressure boundary leakage detection System.
- There is a margin of two or more between the small stable leakage flaw sizes and the larger critical stable flaws.

The new analysis meets LBB requirements required by 10 CFR Part 50, Appendix A, General Design Criterion 4 and uses the recommendations and criteria from the NRC Standard Review Plan for LBB evaluations; therefore, the RCS primary loop piping LBB analysis has been projected to the end of the period of extended operation. When the EPRI Materials Reliability Program methodology described in MRP-140, "Materials Reliability Program:

Leak-Before-Break Evaluation for PWR Alloy 82/182 Welds," is reviewed and approved by the staff, the applicant will review its plant-specific calculation for consistency with the approved approach.

4.3.4.2 Staff Evaluation

The staff reviewed LRA Section 4.3.4, to verify pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses have been projected to the end of the period of extended operation.

The staff reviewed the final licensing basis LBB document, WCAP-14549-P, Addendum 1, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Harris Nuclear Plant for the License Renewal Program," and confirmed the use of saturated material fracture toughness in the LBB analysis. The staff also confirmed the fatigue crack growth evaluation for 60 years that no through-wall crack will occur. No flaw growth evaluation due to primary water stress corrosion cracking was considered but the applicant monitors for such cracking and will address the issue under current licensing requirements. In LRA Amendment 2, Commitment 35 states that when the EPRI Materials Reliability Program methodology described in MRP-140, "Material Reliability Program: Leak-Before-Break Evaluation for PWR Alloy 82/182 Welds," is reviewed and approved by the staff, the applicant will review its plant-specific calculation for consistency with the approved approach. On this basis, the staff finds the applicant's analysis acceptable.

4.3.4.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of RCS loop piping in LRA Section A.1.2.2.11. On the basis of its review of the FSAR supplement and Commitment 35, the staff concludes that the summary description of the applicant's actions to address RCS loop piping is adequate.

4.3.4.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that, for the RCS loop piping LBB analysis, the analyses have been projected to the end of the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.5 Cyclic Loads That Do Not Relate to RCS Transients

This section addresses components listed with thermal fatigue TLAAs where the number of thermal cycles may not correspond to Class 1 component transient cycles. These components were designed originally in accordance with ASME Section III, Class 2 or Class 3 or the ANSI B31.1 Power Piping Code, which requires instead of explicit CUF values, implicit fatigue analyses using stress range reduction factors. These design codes account for cyclic loading by reducing the allowable stress for the component if the number of anticipated cycles exceeds certain limits. It requires the designer to determine the overall number of anticipated thermal cycles for the component and apply stress range reduction factors if this number exceeds 7,000. This implicit fatigue analysis method effectively reduces the allowable stress for the component to keep the applied loads below the endurance limit for the material.

The basic strategy in the following subsections considers the number of transient cycles postulated for 40 years and for license renewal determines whether the number of cycles for 60 years would require a reduction in stress beyond that applied during the original design process. These determinations can be made by a comparison of the design cycles projected for 60 years against the 7,000-cycle criterion for a stress range reduction factor. If the total number of cycles projected for 60 years does not exceed 7,000, then the original design considerations remain valid.

4.3.5.1 Primary Sample Lines

4.3.5.1.1 Summary of Technical Information in the Application

LRA Section 4.3.5.1 summarizes the evaluation of primary sample lines for the period of extended operation. System equipment in the scope of this TLAA are system piping and valves (a) parts of the RCPB and (b) normally or automatically isolated from the RCPB. Part (a) is the portions of piping upstream of the piping anchor for the outboard isolation valves for penetrations M-78A, B, and C. These portions are essentially the safety-related system piping component and a small portion of the nonsafety-related tubing up to the first anchor. Part (b) is the portion of piping downstream from the anchor on the nonsafety-related tubing is not relevant to the applicant for safety determinations. There are three sample line penetrations involved: RCS hot legs (M-78A), pressurizer liquid space (M-78B), and pressurizer steam space (M-78C). The following analyses determined the number of cycles to which the equipment would be subject and compared it to the implicit fatigue analysis acceptance criterion of 7,000 cycles. The applied cycles are determined on the manner of equipment use.

Penetration M-78A - RCS hot legs: The piping downstream of M-78A has three parallel branch lines that supply the post-accident sample panel in the post-accident sampling system, the primary sample panel in the reactor coolant sample system, and the gross failed fuel detector in the gross failed fuel detection system. The gross failed fuel detector operates continuously during reactor startup, operation, and shutdown and the base load follows the reactor thermal cycles; however, as a result of this configuration, the safety-related portion of the reactor coolant sample lines may experience additional thermal cycles whenever flow through the detector is interrupted.

This experience would occur when the containment isolation valves are closed, when flow is swapped between RCS Hot Leg 2 and Hot Leg 3, or when flow to the letdown line, volume control tank, and boron thermal regeneration system is isolated. The cyclic operation of the primary sample panel has no effect on the thermal cycles experienced by the flow through Penetration M-78A due to the continuous flow through the gross failed fuel detector. Interruption of flow through the detector from downstream equipment would require isolation of the letdown line, volume control tank, and boron thermal regeneration system. This latter possibility happening is very rare and a negligible contributor to the consideration of the number of cycles.

Based on this consideration, the total number of cycles experienced by the RCS hot leg sample lines can be estimated by adding to the number of RCS thermal cycles the number of times the hot leg is swapped and the number of cycles caused by Penetration M-78A isolations of sufficient duration to permit cool-down of the sample lines. This evaluation conservatively considers a penetration isolation lasting more than 10 minutes while the RCS hot leg temperature exceeds 500°F one thermal cycle.

Currently RCS flow is swapped between Hot Legs 2 and 3 on an approximate monthly schedule. Even though this swap results in six cycles on each supply from the hot legs, this evaluation conservatively considers twelve cycles each year and simplifies the evaluation. Over 60 years of operation with shutdowns ignored the result is 720 cycles. Rounding up this number to 1,000 cycles accounts for uncertainty in early plant operating practice.

The estimated number of cycles due to reactor shutdowns and the number of Penetration M-78A isolations that would result in a thermal cycle were based on plant data over a period of approximately 6.75 years when there were 9 cycles due to reactor shutdowns and 30 thermal cycles due to penetration isolation valve closure. A ratio of 60 to 6.75 years yields 8.88 rounded up to 9 multiplied by 9 shutdown cycles and 30 penetration isolation cycles yields the following 60-year projections:

- 81 reactor thermal cycles
- 270 thermal cycles due to penetration isolations

Therefore, the total number of hot leg thermal cycles for penetration M-78A is 1,351 cycles, fewer than the requisite 7,000 cycles. As the total number of thermal cycles for the sample lines is fewer than 7,000 cycles, no reanalysis of the piping design calculations is necessary; therefore, an evaluation as required by 10 CFR 54.21(c)(1) successfully demonstrated under 10 CFR 54.21(c)(1) (i) that the reactor coolant sample line design analyses of record remain valid for the period of extended operation (60 years).

4.3.5.1.2 Staff Evaluation

The staff reviewed LRA Section 4.3.5.1 to verify pursuant to 10 CFR 54.21(c)(1)(i), that the analyses remain valid for the period of extended operation.

The staff confirmed design of these primary sample lines in accordance with ASME Code Classes 2 and 3. On the basis that the total number of thermal cycles for these lines is less than 7000 for 60 years, the staff concluded that the primary sample lines analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.3.5.1.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of primary sample lines in LRA Section A.1.2.2.12. On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address primary sample lines is adequate.

4.3.5.1.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that, for the primary sample lines fatigue analysis, the analyses remain valid for the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.5.2 Steam Generator Blowdown Lines

4.3.5.2.1 Summary of Technical Information in the Application

LRA Section 4.3.5.2 summarizes the evaluation of steam generator blowdown lines for the period of extended operation. The steam generator blowdown lines included in this TLAA are listed in FSAR Table 3.2.1-1 as the system portion designed to ASME Section III, Class 2 and ANSI B31.1 codes. This FSAR table lists these components as (a) "the system piping and valves from the steam generator to and including outboard containment isolation valves," (b) "from containment isolation valves to RAB Wall," and (c) "Other." Components in the turbine building also may be designed to ANSI B31.1 as noted in the "Other" listing, but these have no bearing on equipment within the scope of license renewal..

Blowdown flow normally is maintained during operation to maintain steam generator water chemistry. A thermal cycle in the blowdown lines may result whenever blowdown flow to the flash tank is interrupted. There are many potential reasons for interruption of blowdown flow during periods of operation. For example, blowdown flow would be interrupted by an auxiliary feedwater pump actuation signal, a safety injection signal, high-condenser hotwell level signal, steam generator flash tank hi-hi level, containment isolation, or other testing purposes. These interruptions could result in thermal cycles in addition to reactor heat-up and cool-down cycles.

The method of estimating the number of cycles is to review data over a recent time period and count the number of cycles in which blowdown flow was interrupted. This number of cycles multiplied by a ratio based on years estimates the total number of cycles expected over 60 years of operation. The potential to undercount comes from the assumption that the number of cycles counted for the period reviewed represents past and future operations. Additionally, no partial cool-down cycles are counted. To offset the potential undercount, a conservative count extrapolates the total number of cycles to 60 years.

The conservative method counts one cycle when blowdown flow is interrupted for more than 30 minutes. For the purposes of thermal fatigue, a complete thermal cycle is defined as a heat-up from ambient to operating temperature followed by a cool-down to ambient temperature. The thermal cycle counting is conservative because it includes interruptions of blowdown flow in which a significant decrease in temperature is not expected based on the operating practice for re-establishing blowdown flow following a blowdown isolation valve closure. This operating practice states that if the isolation valves are closed for more than 30 minutes the downstream piping must be warmed up before the isolation valves are opened; therefore, an isolation valve closed for less than 30 minutes does not constitute a significant cool-down period.

The number of cycles due to reactor shutdowns is included in the blowdown cycles counted. Based on plant data over a period of approximately 5.5 years, the estimated number blowdown flow interruptions that would result in thermal cycles is 37 cycles. Application of a ratio for 60 and 100 years yields 404 and 673 cycles, respectively. As the total number of thermal cycles for the steam generator blowdown lines is fewer than 7,000 cycles, no reanalysis of the piping design calculations is necessary; therefore, an evaluation as required by 10 CFR 54.21(c)(1) successfully demonstrated under 10 CFR 54.21(c)(1) (i) that the steam generator blowdown line design analyses of record remain valid for the period of extended operation (60 years).

4.3.5.2.2 Staff Evaluation

The staff reviewed LRA Section 4.3.5.2 to verify pursuant to 10 CFR 54.21(c)(1)(i), that the analyses remain valid for the period of extended operation.

The staff confirmed design of the steam generator blowdown lines in accordance with ASME Code Class 2 and ANSI B31.1. On the basis that the total number of thermal cycles for these lines is less than 7000 for 60 years, the staff concluded that the steam generator blowdown lines analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.3.5.2.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of steam generator blowdown lines in LRA Section A.1.2.2.13. On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address steam generator blowdown lines is adequate.

4.3.5.2.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that, for steam generator blowdown lines, the analyses remain valid for the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.4 Environmental Qualification of Electrical Equipment

The 10 CFR 50.49 EQ program is a TLAA for purposes of license renewal. The TLAA of the environmental qualification (EQ) electrical components includes all long-lived, passive, and active electrical and instrumentation and control components that are important to safety and located in a harsh environment. The harsh environments of the plant are those areas subject to environmental effects by LOCAs or high-energy line breaks. EQ equipment comprises safety-related and Q-list equipment, nonsafety-related equipment the failure of which could prevent satisfactory accomplishment of any safety-related function, and necessary post-accident monitoring equipment.

As required by 10 CFR 54.21(c)(1), the applicant must provide a list of EQ TLAAs in the LRA. The applicant shall demonstrate that for each type of EQ equipment, one of the following is true: (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.

4.4.1 Summary of Technical Information in the Application

LRA Section 4.4 summarizes the evaluation of EQ of electrical equipment for the period of extended operation. Thermal, radiation, and cyclical aging analyses of plant electrical and instrumentation and control components required to meet 10 CFR 50.49 qualification are TLAAs.

The NRC has established nuclear station EQ requirements in 10 CFR Part 50, Appendix A, General Design Criterion 4 and in 10 CFR 50.49, which specifically requires establishment of an EQ program to demonstrate that electrical components in harsh plant environments (plant areas that could be subject to environmental effects of LOCAs, high-energy line breaks, or post-LOCA radiation) are qualified to perform safety functions in such environments despite the effects of inservice aging. Section 50.49 requires EQ to address the effects of significant aging mechanisms.

4.4.2 Staff Evaluation

The staff reviewed LRA Section 4.4, to verify pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

The staff reviewed the program basis calculation for adequate information for 10 CFR 54.21(c)(1). For the electrical equipment shown in LRA Table 4.1-1, the applicant demonstrated per 10 CFR 54.21(c)(1)(iii) that the aging effects of EQ equipment will be adequately managed during the period of extended operation. The staff reviewed the Environmental Qualification (EQ) Program for whether it maintain electrical and instrumentation and control component performance of intended functions consistent with the CLB for the period of extended operation. The staff's evaluation of the qualification of these components focused on how the Environmental Qualification (EQ) Program manages the aging effects for 10 CFR 50.49 requirements.

The staff's audit of the information in LRA Section B3.2 and the program bases documents is documented in SER Section 3.0.3.1.13. On the basis of its audit, the staff finds that the Environmental Qualification (EQ) Program, for which the applicant claimed consistency with GALL AMP X.E1, "Environment Qualification of Electrical Components," is consistent with the GALL Report; therefore, the staff finds the program capable of programmatically managing the qualified life of components within the scope of license renewal. The continued implementation of the Environmental Qualification (EQ) Program reasonably assures management of the aging effects for continued performance by components within the scope of the program of intended functions for the period of extended operation.

4.4.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of EQ of electrical equipment in LRA Section A.1.2.3. On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address EQ of electrical equipment is adequate.

4.4.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that, for EQ of electrical equipment, 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 FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.5 Concrete Containment Tendon Prestress

4.5.1 Summary of Technical Information in the Application

LRA Section 4.5 summarizes the evaluation of concrete containment tendon prestress for the period of extended operation. NUREG-1800 assigns TLAA Section 4.5 to the issue of Concrete Containment Tendon Prestress. The Unit 1 containment structures have no prestressed tendons; therefore, this section is not applicable.

4.5.2 Staff Evaluation

The containment has no prestressed tendons; therefore, the staff finds this TLAA not required.

4.5.3 FSAR Supplement

The staff concludes that no FSAR supplement is required because the containment building has no pre-stressed tendons.

4.5.4 Conclusion

On the basis of its review, as discussed above, the staff concludes this TLAA is not required.

4.6 Containment Liner Plate, Metal, Metal Containments, and Penetrations Fatigue Analysis

4.6.1 Containment Mechanical Penetration Bellows Fatigue

4.6.1.1 Mechanical Penetration Bellows - Valve Chambers

4.6.1.1.1 Summary of Technical Information in the Application

LRA Section 4.6.1.1 summarizes the evaluation of mechanical penetration bellows - valve chambers for the period of extended operation. The four mechanical penetration bellows addressed by this section are the containment spray and safety injection system recirculation valve chamber bellows (two each) for containment penetrations M-47 through M-50. These penetrations are illustrated in FSAR Table 6.2.4-1. Each line has motor-operated gate valves enclosed in valve chambers leak-tight at containment design pressure. Each line from the containment sump to the valve is enclosed in a separate concentric guard pipe also leak-tight. A seal keeps both the chamber and the guard pipe from connecting directly to the containment sump or to the containment atmosphere.

Per plant specifications, the valve chamber bellows expansion joint design is in accordance with ASME Section III, Paragraph NC-3649.1 so no single corrugation is permitted to deflect more than its maximum allowable amount. Each bellows is designed to withstand over a lifetime of 40 years a total of 7,000 expansion and compression cycles due to maximum normal operating conditions and 10 cycles of movement due to safe shutdown earthquake conditions.

This TLAA addresses the requirement that the 40-year lifetime may be extended to 60 years without exceeding the design criterion of 7,000 expansion and compression cycles. The 10 cycles of movement due to safe shutdown earthquake conditions are still available because no earthquake of such magnitude has been experienced.

Operating cycles of expansion and compression due to maximum normal operating conditions are calculated conservatively by addition of RCS (Class 1) design cycles corresponding to containment heat-up and cool-down to the number of times the containment is pressurized during Type A integrated leak rate testing plus the number of Type B local leak rate tests.

The expansion bellows is the barrier between the valve chamber and the reactor auxiliary building. The containment isolation valves for these chambers isolate the containment sumps from the containment spray and RHR systems and therefore normally experience no fluid flow. RHR operation during RCS cool-down would have a negligible impact on the bellows due to the piping configuration but is included because RHR operation typically corresponds to RCS (Class 1) cycles.

The number of reactor thermal cycles projected over 60 years is 81. Containment integrated leak rate testing is infrequent (*i.e.*, every 10 years). A conservative assumption of integrated leak rate testing every 5 rather than 10 years yields 12 cycles. In the Type B local leak rate test program the maximum test interval for this equipment is 24 months. A conservative assumption is a minimum of yearly with an additional 60 cycles and a total number of 153 cycles anticipated for 60 years.

The total number of thermal cycles for the containment spray and safety injection system recirculation valve chamber bellows is fewer than 7,000 so no reanalysis of the design calculations is necessary. An evaluation as required by 10 CFR 54.21(c)(1) successfully demonstrated under 10 CFR 54.21(c)(1)(i) that the containment spray and safety injection system recirculation valve chamber bellows design analyses of record remain valid for the period of extended operation.

4.6.1.1.2 Staff Evaluation

The staff reviewed LRA Section 4.6.1.1 to verify pursuant to 10 CFR 54.21(c)(1)(i), that the analyses remain valid for the period of extended operation.

The staff confirmed design of the bellows in accordance with ASME Class 2 to withstand 7000 cycles of thermal expansion and compression and 10 cycles of safe shutdown earthquake movement. The staff reviewed the applicant's conservative estimation of the thermal cycle for the bellows. On the basis that the total number of thermal cycles for these bellows is less than 7000 for 60 years with 10 cycles of safe shutdown earthquake movement still available, the staff concluded that the bellows design analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.6.1.1.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of mechanical penetration bellows - valve chambers in LRA Section A.1.2.4.1. On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address mechanical penetration bellows - valve chambers is adequate.

4.6.1.1.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that, for mechanical penetration bellows - valve chambers the analyses remain valid for the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.6.1.2 Mechanical Penetration Bellows - Fuel Transfer Tube Bellows Expansion Joint

4.6.1.2.1 Summary of Technical Information in the Application

LRA Section 4.6.1.2 summarizes the evaluation of mechanical penetration bellows - fuel transfer tube bellows expansion joint for the period of extended operation. The fuel transfer tube is essentially a tubular passageway connecting the transfer canal in the containment building with that in the spent fuel pit building. Per plant specifications, the fuel transfer tube bellows-expansion-joint design is in accordance with ASME Section III, Paragraph NC-3649.1, with no single corrugation permitted to deflect more than its maximum allowable amount. Each bellows is designed to withstand a total of 7,000 cycles of expansion and compression over a lifetime of 40 years of maximum normal operating conditions and 10 cycles of movement due to safe shutdown earthquake conditions.

This TLAA addresses the requirement that the 40-year lifetime extend to 60 years without exceeding the design criterion of 7,000 cycles of expansion and compression. The 10 cycles of movement due to safe shutdown earthquake are still available as no earthquake of such magnitude has been experienced.

The expansion cycles would occur when the tube is flooded between the transfer canal in the containment building and the fuel handling building. This operation typically occurs twice every refueling outage; therefore, the maximum number of operating cycles projected over a 60-year period is 80 cycles.

The total number of thermal cycles for the fuel transfer tube bellows expansion joint is fewer than 7,000 so no reanalysis of the design calculations is necessary. An evaluation as required by 10 CFR 54.21(c)(1) successfully demonstrated that the fuel transfer tube bellows expansion joint design analyses of record remain valid for the period of extended operation (60 years).

4.6.1.2.2 Staff Evaluation

The staff reviewed LRA Section 4.6.1.2 to verify pursuant to 10 CFR 54.21(c)(1)(i), that the analyses remain valid for the period of extended operation.

The staff confirmed design of the bellows in accordance with ASME Class 2 to withstand 7000 cycles of thermal expansion and compression and 10 cycles of safe shutdown earthquake movement. The staff reviewed the applicant's conservative estimation of the thermal cycle for the bellows. On the basis that the total number of thermal cycles for these bellows is less than 7000 for 60 with 10 cycles of safe shutdown earthquake movement still available, the staff concluded that the bellows design analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.6.1.2.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of mechanical penetration bellows - fuel transfer tube bellows expansion joint in LRA Section A.1.2.4.2. On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address mechanical penetration bellows - fuel transfer tube bellows expansion joint is adequate.

4.6.1.2.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that, for mechanical penetration bellows - fuel transfer tube bellows expansion joint, the analyses remain valid for the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7 <u>Other Plant-Specific Time-Limited Aging Analyses</u>

4.7.1 Turbine Rotor Missile Generation Analysis

4.7.1.1 Summary of Technical Information in the Application

LRA Section 4.7.1 summarizes the evaluation of turbine rotor missile generation analysis for the period of extended operation.

According to 10 CFR Part 50, Appendix A, General Design Criterion 4, nuclear power plant safety-related structures, systems, and components must be protected appropriately against dynamic effects, including those of missiles. Failures of large steam turbines of the main turbine generator could eject large high-energy missiles that can damage plant structures, systems, and components. The overall safety objective is to protect safety-related structures, systems, and components adequately from potential turbine missiles.

RG 1.115 describes methods acceptable to the staff for protecting safety-related structures, systems, and components against low-trajectory missiles from turbine failure by appropriate orientation and placement of the turbine generator set. The applicant complies with RG 1.115, Revision 1 with the exception of Position C.2.

FSAR Section 3.5.1.3.2, "Probability of Turbine Missile Generation," describes a Westinghouse study based upon mechanics to obtain a rough estimate of turbine-generator reliability based on expected operating conditions. The study determined the number of cycles required to cause a crack (flaw) to grow larger and calculated as 140,000 the number of cold start-up cycles (worst-case stress environment) required for the undetectable flaw of maximum size to grow to 1/3 of the critical crack size. A estimated reasonable upper limit for the number of this type of stress cycle is five per year or 200 per 40 years plant life; thus, the maximum undetectable crack poses no threat to the integrity of a turbine-generator with the designed mechanical properties.

The original analysis estimated five cycles per year for 40 years of plant operation. For the period of extended operation, the estimate of 5 cycles per year yields 300 cycles for 60 years of plant life, well below the 140,000 cycles required by the maximum size undetectable flaw to grow to 1/3 of the critical crack size; therefore, this analysis projects to the end of the period of extended operation.

4.7.1.2 Staff Evaluation

The staff reviewed LRA Section 4.7.1, to verify pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses have been projected to the end of the period of extended operation.

The staff reviewed FSAR Section 3.5.1.3.2. and the applicant's analyses in LRA Section 4.7.1 to confirm that the number of projected cycles of 300 is well below the 140,000 required by the maximum undetectable flaw to grow to 1/3 of the critical crack size. On this basis, the staff concluded that this analysis remains valid for period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

The applicant stated that the fracture mechanics crack growth analysis of the number of turbine start-up cycles that could result in critical flaw size is projected to the end of the period of extended operation. The staff noted that the fracture mechanics analysis remains valid but did not project to critical flaw size; therefore, the method should be that of 10 CFR 54(c)(1)(i) instead of (ii).

By letter dated January 17, 2008. In its response, the applicant agreed that the basis for accepting the TLAA on the turbine rotor missile generation analysis should have been dispositioned in accordance with the staff's acceptance criterion in 10 CFR 54.21(c)(1)(i), in that the existing analysis has been demonstrated to be valid for the period of extended operation. The applicant stated that an amendment of LRA Section 4.7.1 would be made to reflect this. The staff verified that the applicant included the appropriate amendment of LRA Section 4.7.1 in Enclosure 2 of the letter dated January 17, 2008. Thus, dispositioning this TLAA in accordance with the criterion in 10 CFR 54.21(c)(1)(i) and appropriately reflecting this in an amendment of LRA Section 4.7.1 is resolved.

4.7.1.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of turbine rotor missile generation analysis in LRA Section A.1.2.5. On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address turbine rotor missile generation analysis is adequate.

By letter dated January 17, 2008, the applicant agreed that the basis for accepting the TLAA on the turbine rotor missile generation analysis should have been dispositioned in accordance with the staff's acceptance criterion in 10 CFR 54.21(c)(1)(i), in that the existing analysis has been demonstrated to be valid for the period of extended operation. The applicant stated that an amendment of FSAR supplement Section A.1.2.5 would be made to reflect this. The staff verified that the applicant included the appropriate amendment to FSAR supplement Section A.1.2.5 in Enclosure 2 of the letter dated January 17, 2008. Thus, dispositioning this TLAA in accordance with the criterion in 10 CFR 54.21(c)(1)(i) and appropriately reflecting this in an amendment of FSAR supplement Section A.1.2.5 is resolved.

On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the TLAA on the turbine rotor missile generation analysis, as given in LRA Section A.1.2.5 and amended in the applicant's letter dated January 17, 2008, is adequate.

4.7.1.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that, for turbine rotor missile generation analysis, the analyses remain valid to the end of the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.2 Crane Cyclic Analyses

The applicant indicated load cycle limits for cranes as potential TLAAs. The following cranes within the scope of license renewal have TLAAs, which require evaluation for 60 years.

- Polar Crane
- Jib Cranes
- Reactor Cavity Manipulator Crane
- Fuel Cask Handling Crane
- Fuel Handling Bridge Crane
- Fuel Handling Building Auxiliary Crane

The method of review for the crane cyclic load limit TLAA involves:

- review of the existing 40-year design basis to determine the number of load cycles in the design of each of the cranes within the scope of license renewal

- development of 60-year load cycle projections for each of the cranes within the scope of license renewal compared to the number of design cycles for 40 years

4.7.2.1 Polar Crane

4.7.2.1.1 Summary of Technical Information in the Application

LRA Section 4.7.2.1 summarizes the polar crane evaluation for the period of extended operation. The overhead crane in the containment (250-ton / 50-ton) for reactor servicing operations is of the polar configuration and seated on a girder bracketed off the containment wall.

The polar crane purchasing specification required conformance to Crane Manufacturers Association of America (CMAA) Specification 70, 1971 edition, for electric overhead traveling cranes. The purchasing specification did not state a service classification but the crane meets the Service Class A requirement. The crane, therefore, was designed for 20,000 to 100,000 maximum-rated load cycles for a 40-year life.

The number of maximum rated load cycles for the 250-ton (main hook) originally projected for 40 years was 2,720. The number of maximum rated cycles for a 60-year life based on 40 refueling outages is 4,020, fewer than the 20,000 to 100,000 permissible cycles and, therefore, acceptable.

The number of maximum rated load cycles for the 50-ton (auxiliary hook) originally projected for 40 years was 1,080. The number of maximum rated cycles for a 60-year life based on 40 refueling outages is 1,600, fewer than the 20,000 to 100,000 permissible cycles and, therefore, acceptable.

The polar crane main hook and auxiliary hook ultimately share the same structure and therefore their cycles should be combined as follows: 4020 + 1,600 = 5,620 cycles, fewer than the 20,000 to 100,000 permissible cycle range and, therefore, acceptable.

Therefore, the Polar Crane fatigue analysis has been projected successfully for 60 years of plant operation.

4.7.2.1.2 Staff Evaluation

The staff reviewed LRA Section 4.7.2.1, to verify pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses have been projected to the end of the period of extended operation.

The staff reviewed the applicant's estimate of the number of the maximum rated load cycles for the 60 years operation compared to the number of permissible design cycles. On the basis that the 60-year number of operation cycles, 5620, is much less than the permissible number, 20,000 to 100,000, the staff concluded that this analysis remains valid for period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

The staff noted that the design analysis remains valid but does not project the analysis result to 60 years; therefore, the method should be that of 10 CFR 54(c)(1)(i) instead of (ii).

By letter dated January 17, 2008, the applicant agreed that the basis for accepting the TLAA on the polar crane should have been dispositioned in accordance with the staff's acceptance criterion in 10 CFR 54.21(c)(1)(i), in that the existing analysis has been demonstrated to be valid for the period of extended operation. The applicant stated that an amendment of LRA Section 4.7.2.1 would be made to reflect this. The staff verified that the applicant included the appropriate amendment of LRA Section 4.7.2.1 in Enclosure 2 of the letter dated January 17, 2008. Thus, dispositioning this TLAA in accordance with the criterion in 10 CFR 54.21(c)(1)(i) and appropriately reflecting this in an amendment of LRA Section 4.7.2.1 is resolved.

4.7.2.1.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of the polar crane in LRA Section A.1.2.6.1. On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address the polar crane is adequate.

By letter dated January 17, 2008 the applicant agreed that the basis for accepting the TLAA on the polar crane should have been dispositioned in accordance with the staff's acceptance criterion in 10 CFR 54.21(c)(1)(i), in that the existing analysis has been demonstrated to be valid for the period of extended operation. The applicant stated that an amendment of FSAR supplement Section A.1.2.6.1 would be made to reflect this. The staff verified that the applicant included the appropriate amendment of FSAR supplement Section A.1.2.6.1 in Enclosure 2 of the letter dated January 17, 2008. Thus, the reflected item in the amendment of FSAR supplement Section A.1.2.6.1 is resolved.

On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the TLAA on the polar crane, as given in LRA Section A.1.2.6.1 and amended in the applicant's letter of January 17, 2008, is adequate.

4.7.2.1.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that, for the polar crane, the analyses remain valid to the end of the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.2.2 Jib Cranes

4.7.2.2.1 Summary of Technical Information in the Application

LRA Section 4.7.2.2 summarizes the evaluation of jib cranes for the period of extended operation. The two containment jib cranes (5-ton) support low-load capacity refueling and maintenance and have the flexibility to be mounted on any of six base plates to relieve and increase availability for the ever-critical path polar crane.

The jib crane purchasing specification required conformance to CMAA Specification 74 for under-running single-girder electric overhead traveling cranes, Service Class A1 (standby). The crane, therefore, was designed for 20,000 to 100,000 maximum rated load cycles for a 40-year life.

The number of maximum rated load cycles originally projected for 40 years was 12,690. The number of maximum rated load cycles for a 60-year life based on 40 refueling outages is 18,800, fewer than the 20,000 to 100,000 permissible cycles and, therefore, acceptable.

Therefore, the jib crane fatigue analysis has been projected successfully for 60 years of plant operation.

4.7.2.2.2 Staff Evaluation

The staff reviewed LRA Section 4.7.2.2, to verify pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses have been projected to the end of the period of extended operation. The staff reviewed the applicant's estimate of the number of maximum-rated load cycles for the 60 years of operation compared to the number of permissible design cycles. On the basis that the

60-year number of operation cycles, 18,800, is less than the permissible number, 20,000 to 100,000, the staff concluded that this analysis remains valid for period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

The staff noted that the design analysis remains valid but does not project the analysis result to 60 years; therefore, the method should be that of 10 CFR 54.21(c)(1)(i) instead of (ii).

By letter dated January 17, 2008, the applicant agreed that the basis for accepting the TLAA on the jib cranes should have been dispositioned in accordance with the staff's acceptance criterion in 10 CFR 54.21(c)(1)(i), in that the existing analysis has been demonstrated to be valid for the period of extended operation. The applicant stated that an amendment of LRA Section 4.7.2.2 would be made to reflect this. The staff verified that the applicant included an appropriate amendment of LRA Section 4.7.2.2 in Enclosure 2 of the letter dated January 17, 2008. Thus, the TLAA is in accordance with 10 CFR 54.21(c)(1)(i) and appropriately reflecting this in an amendment of LRA Section 4.7.2.2 is resolved.

4.7.2.2.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of jib cranes in LRA Section A.1.2.6.2. On the basis of its review of the FSAR supplement, the staff did not initially agree with the fatigue analysis projected. The staff concludes that the summary description of the applicant's actions to address jib cranes is not adequate.

By letter dated January 17, 2008 the applicant agreed that the basis for accepting the TLAA on the jib cranes should have been dispositioned in accordance with the staff's acceptance criterion in 10 CFR 54.21(c)(1)(i), in that the existing analysis has been demonstrated to be valid for the period of extended operation. The applicant stated that an amendment of FSAR supplement Section A.1.2.6.2 was made to reflect this. The staff verified that the applicant included the applicable amendment of FSAR supplement Section A.1.2.6.2 in Enclosure 2 of the letter dated January 17, 2008. Thus, the applicant appropriately reflected this in an amendment of FSAR supplement Section A.1.2.6.2.

On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the TLAA on the jib cranes, as given in LRA Section A.1.2.6.2 and amended in the applicant's letter dated January 17, 2008, is adequate.

4.7.2.2.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that, for jib cranes, the analyses remain valid to the end of the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.2.3 Reactor Cavity Manipulator Crane

4.7.2.3.1 Summary of Technical Information in the Application

LRA Section 4.7.2.3 summarizes the evaluation of the reactor cavity manipulator crane for the period of extended operation. The rectilinear bridge and trolley crane with a vertical mast extending down into the refueling water flexibly grips, removes, and replaces fuel assemblies to support refueling operations. Only the passive bridge structure manufactured from carbon steel is within the scope of license renewal.

The reactor cavity manipulator crane purchasing specification required the maximum design stress for the crane structure to be 1/5 of ultimate tensile strength. The low maximum design stress for the crane structure indicates stress marginally below the fatigue limit for the carbon steel material, which is estimated to be acceptable for 10^7 cycles; therefore, the estimated number of lifts for 40 years is 10^7 cycles.

The number of load cycles originally projected for 40 years was 11,390. The number of maximum rated load cycles for a 60-year life based on 40 refueling outages is 16,824, fewer than the 10^7 permissible cycles and, therefore, acceptable.

Therefore, the reactor cavity manipulator crane fatigue analysis has been projected successfully for 60 years of plant operation.

4.7.2.3.2 Staff Evaluation

The staff reviewed LRA Section 4.7.2.3, to verify pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses have been projected to the end of the period of extended operation.

The staff reviewed the applicant's estimate of the number of the maximum-rated load cycles for 60 years of operation compared to the permissible number of design cycles. On the basis that the 60-year number of operation cycles, 16,824, is much less than the permissible number, 10^7, the staff concluded that this analysis remains valid for period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

The staff noted that design analysis remains valid but does not project the analysis result to 60 years; therefore, the method should be that of 10 CFR 54(c)(1)(i) instead of (ii).

By letter dated January 17, 2008, the applicant agreed that the basis for accepting the TLAA on the reactor cavity manipulator crane should have been dispositioned in accordance with the staff's acceptance criterion in 10 CFR 54.21(c)(1)(i), in that the existing analysis has been demonstrated to be valid for the period of extended operation. The applicant stated that an amendment of LRA Section 4.7.2.3 would be made to reflect this. The staff verified that the applicant included the appropriate amendment of LRA Section 4.7.2.3 in Enclosure 2 of the letter dated January 17, 2008.

4.7.2.3.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of reactor cavity manipulator crane in LRA Section A.1.2.6.3. On the basis of its review of the FSAR supplement, the staff does not agree with the fatigue analysis projected. The applicant agreed that the basis for accepting the TLAA on the reactor cavity manipulator crane should have been dispositioned in accordance with the staff's acceptance criterion in 10 CFR 54.21(c)(1)(i), in that the existing analysis has been demonstrated to be valid for the period of extended operation. The applicant stated that an amendment of FSAR supplement Section A.1.2.6.3 would be made to reflect this. The staff verified that the applicant included the applicable amendment of FSAR supplement Section A.1.2.6.3 in Enclosure 2 of the letter dated January 17, 2008. Thus, the applicant appropriately reflected this in an LRA amendment of FSAR supplement Section A.1.2.6.3.

On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the TLAA on the reactor cavity manipulator crane, as given in LRA Section A.1.2.6.3 and amended in the applicant's letter dated January 17, 2008, is adequate.

4.7.2.3.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that, for the reactor cavity manipulator crane, the analyses remain valid to the end of the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.2.4 Fuel Cask Handling Crane

4.7.2.4.1 Summary of Technical Information in the Application

LRA Section 4.7.2.4 summarizes the evaluation of the fuel cask handling crane for the period of extended operation. The fuel cask handling crane (150-ton) transfers the spent fuel cask between the railroad car and the spent fuel cask loading pool. The fuel cask handling crane and the fuel handling auxiliary crane share the same rails supported from the fuel handling building in the overhead.

The fuel cask handling crane purchasing specification required conformance to CMAA Specification 70 for electric overhead traveling cranes. The purchasing specification did not state a service classification but the crane meets the Service Class A requirement and, therefore, was designed for 20,000 to 100,000 maximum-rated load cycles for a 40-year life.

The number of load cycles originally projected for 40 years was 7,350. The number of load cycles based on 40 refueling outages for a 60-year life is 8,750, fewer than the 20,000 to 100,000 permissible cycles and, therefore, acceptable.

Therefore, the fuel cask handling crane fatigue analysis has been projected successfully for 60 years of plant operation.

4.7.2.4.2 Staff Evaluation

The staff reviewed LRA Section 4.7.2.4, to verify pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses have been projected to the end of the period of extended operation.

The staff reviewed the applicant's estimate of the number of maximum-rated load cycles for 60 years of operation compared to the permissible number of design cycles. On the basis that the 60-year number of operation cycles, 8,750, is much less than the permissible number, 20,000 to 100,000, the staff concluded that this analysis remains valid for period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

The staff noted that the design analysis remains valid but does not project the analysis result to 60 years; therefore, the method should be that of 10 CFR 54(c)(1)(i) instead of (ii).

By letter dated January 17, 2008, the applicant agreed that the basis for accepting the TLAA on the fuel cask handling crane should have been dispositioned in accordance with the staff's acceptance criterion in 10 CFR 54.21(c)(1)(i), in that the existing analysis has been demonstrated to be valid for the period of extended operation. The applicant stated that an amendment of LRA Section 4.7.2.4 would be made to reflect this. The staff verified that the applicant included appropriate amendment of LRA Section 4.7.2.4 in Enclosure 2 of the letter dated January 17, 2008. The applicant appropriately reflected this in an amendment of LRA Section 4.7.2.4.

4.7.2.4.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of the fuel cask handling crane in LRA Section A.1.2.6.6. On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address the fuel cask handling crane is adequate.

By letter dated January 17, 2008, the applicant agreed that the basis for accepting the TLAA on the fuel cask handling crane should have been dispositioned in accordance with the staff's acceptance criterion in 10 CFR 54.21(c)(1)(i), in that the existing analysis has been demonstrated to be valid for the period of extended operation. The applicant stated that an amendment of FSAR supplement Section A.1.2.6.4 would be made to reflect this. The staff verified that the applicant included the appropriate amendment of FSAR supplement Section A.1.2.6.4 in Enclosure 2 of the letter dated January 17, 2008. The applicant appropriately reflected this in an LRA amendment of FSAR supplement Section A.1.2.6.4.

On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the TLAA on the fuel cask handling crane, as given in LRA Section A.1.2.6.4 and amended in the applicant's letter of January 17, 2008, is adequate.

4.7.2.4.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that, for the fuel cask handling crane, the analyses remain valid to the end of the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.2.5 Fuel Handling Bridge Crane

4.7.2.5.1 Summary of Technical Information in the Application

LRA Section 4.7.2.5 summarizes the evaluation of the fuel handling bridge crane for the period of extended operation. The fuel handling bridge crane (1.25-ton) is a wheel-mounted walkway spanning the width of the fuel handling building. The crane carries an electric monorail hoist on an overhead structure.

The fuel handling bridge crane purchasing specification required the maximum design stress for all load-bearing parts, design load plus structural weight, to be 1/5 of the ultimate strength of the material. Westinghouse specified neither a permissible number of cycles for the lifetime of the crane nor a service class. Material of construction for this crane conforms to American Society for Testing and Materials Specification A-36. The low maximum design stress for the carbon steel crane structure above the refueling water elevation indicates the stress is marginally below the fatigue limit for the carbon steel material, which, therefore, is acceptable for an estimated 10^7 cycles; therefore, the estimated acceptable number of maximum-rated load cycles for 40 or 60 years was 10^7 cycles.

The number of load cycles originally projected for 40 years was 18,602 based on crane usage for the original fuel load, fuel movements during 27 refueling outages, usage for fuel and fuel insert shuffles, and movement of spent fuel from other applicant facilities. The number of load cycles projected for 60 years is 27,558, assuming 40 refueling outages and projected crane use for fuel handling activities, fewer than the 10^7 permissible cycles and, therefore, acceptable.

Therefore, the fuel handling bridge crane fatigue analysis has been projected successfully for 60 years of plant operation.

4.7.2.5.2 Staff Evaluation

The staff reviewed LRA Section 4.7.2.5, to verify pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses have been projected to the end of the period of extended operation.

The staff reviewed the applicant's estimate of the number of the maximum-rated load cycles for 60 years of operation compared to the permissible number of design cycles. On the basis that the 60-year number of operation cycles, 27,558, is much less than the permissible number, 10^7, the staff concluded that this analysis remains valid for period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

The staff noted that design analysis remains valid but does not project the analysis result to 60 years; therefore, the method should be that of 10 CFR 54(c)(1)(i) instead of (ii).

By letter dated January 17, 2008, the applicant agreed that the basis for accepting the TLAA on the fuel handling bridge crane should have been dispositioned in accordance with the staff's acceptance criterion in 10 CFR 54.21(c)(1)(i), in that the existing analysis has been demonstrated to be valid for the period of extended operation. The applicant stated that an amendment of LRA Section 4.7.2.5 would be made to reflect this. The staff verified that the applicant included appropriate amendment of LRA Section 4.7.2.5 in Enclosure 2 of the letter dated January 17, 2008. The applicant appropriately reflected this in an amendment of LRA Section 4.7.2.5.

4.7.2.5.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of fuel handling bridge crane in LRA Section A.1.2.6.5.

By letter dated January 17, 2008, the applicant agreed that the basis for accepting the TLAA on the fuel handling bridge crane should have been dispositioned in accordance with the staff's acceptance criterion in 10 CFR 54.21(c)(1)(i), in that the existing analysis has been demonstrated to be valid for the period of extended operation. The applicant stated that an amendment of FSAR supplement Section A.1.2.6.5 would be made to reflect this. The staff verified that the applicant included the appropriate amendment of FSAR supplement Section A.1.2.6.5 in Enclosure 2 of the letter dated January 17, 2008. The applicant appropriately reflected this in an LRA amendment of FSAR supplement Section A.1.2.6.5.

On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the TLAA on the fuel handling bridge crane, as given in LRA Section A.1.2.6.5 and amended in the applicant's letter dated January 17, 2008, is adequate.

4.7.2.5.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that, for the fuel handling bridge crane, the analyses remain valid to the end of the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.2.6 Fuel Handling Building Auxiliary Crane

4.7.2.6.1 Summary of Technical Information in the Application

LRA Section 4.7.2.6 summarizes the evaluation of fuel handling building auxiliary crane for the period of extended operation. The fuel handling building auxiliary crane (12-ton) supports the refueling process and shares with the fuel cask handling crane the same rails supported from the fuel handling building in the overhead.

The fuel handling building auxiliary crane purchasing specification required conformance to CMAA Specification 70 for electric overhead traveling cranes. The purchasing specification did not state a service classification but the crane meets the Service Class A requirement and, therefore, was designed for 20,000 to 100,000 maximum-rated load cycles for a 40-year life.

The number of load cycles originally projected for 40 years is 12,280. Based on 40 refueling outages, the number of load cycles projected for 60-year life is 15,380, fewer than the 20,000 to 100,000 permissible cycles and, therefore, acceptable.

Therefore, the fuel handling building auxiliary crane fatigue analysis has been projected successfully for 60 years of plant operation.

4.7.2.6.2 Staff Evaluation

The staff reviewed LRA Section 4.7.2.6, to verify pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses have been projected to the end of the period of extended operation.

The staff reviewed the applicant's estimate of the number of maximum-rated load cycles for 60 years of operation compared to the permissible number of design cycles. On the basis that the 60-year number of operation cycles, 15,380, is less than the number of permissible cycles, 20,000 to 100,000, the staff concluded that this analysis remains valid for period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

The staff noted that design analysis remains valid but does not project the analysis result to 60 years; therefore, the method should be that of 10 CFR 54(c)(1)(i) instead of (ii).

By letter dated January 17, 2008, the applicant agreed that the basis for accepting the TLAA on the fuel handling building auxiliary crane should have been dispositioned in accordance with the staff's acceptance criterion in 10 CFR 54.21(c)(1)(i), in that the existing analysis has been demonstrated to be valid for the period of extended operation. The applicant stated that an amendment of LRA Section 4.7.2.6 would be made to reflect this. The staff verified that the applicant included appropriate amendment of LRA Section 4.7.2.6 in Enclosure 2 of the letter dated January 17, 2008.

4.7.2.6.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of the fuel handling building auxiliary crane in LRA Section A.1.2.6.6.

By letter dated January 17, 2008, the applicant agreed that the basis for accepting the TLAA on the fuel handling building auxiliary crane should have been dispositioned in accordance with the staff's acceptance criterion in 10 CFR 54.21(c)(1)(i), in that the existing analysis has been demonstrated to be valid for the period of extended operation. The applicant stated that an amendment of FSAR supplement Section A.1.2.6.6 would be made to reflect this. The staff verified that the applicant included the appropriate amendment of FSAR supplement

Section A.1.2.6.6 in Enclosure 2 of the letter dated January 17, 2008. The applicant appropriately reflected this in an LRA amendment of FSAR supplement Section A.1.2.6.6.

On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the TLAA on the fuel handling building auxiliary crane, as given in LRA Section A.1.2.6.6 and amended in the applicant's letter dated January 17, 2008, is adequate.

4.7.2.6.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that, for the fuel handling building auxiliary crane, the analyses remain valid to the end of the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.3 Main and Auxiliary Reservoir Sedimentation Analyses

4.7.3.1 Summary of Technical Information in the Application

LRA Section 4.7.3 summarizes the evaluation of main and auxiliary reservoir sedimentation analyses for the period of extended operation. The auxiliary reservoir functions as the ultimate heat sink and the main reservoir as a backup when the auxiliary reservoir is not available. The FSAR states that for 40 years of plant life the volume of potential sediment amounted to 0.4 percent in the auxiliary reservoir and to 0.7 percent in the main reservoir of the reservoir capacity at the normal water level. The FSAR concludes that the effects of sediment deposit on reservoir operations and cooling capacities will be negligible for the current 40-year period of operation. The FSAR considers sedimentation in the main and auxiliary reservoirs as a TLAA with sedimentation effects based on a 40-year plant life.

During the original licensing review the applicant made a commitment to use RG 1.127, "Inspection of Water-Control Structures Associated with Nuclear Power Plants," to monitor sedimentation effects on water control structures. The RG 1.127 inspections monitor in the main and auxiliary reservoirs sedimentation which could reduce reservoir capacity at normal water levels. In addition, HNP technical specifications require a daily check for minimum water level in the main and auxiliary reservoirs for the ultimate heat sink to operate.

For the extended life of 60 years, the applicant expects sediment effects of increased vegetation, paving, and control of storm runoff by catch basins and storm drains at the plant island also to be negligible. A simple calculation of sedimentation based on the ratio of 60 years to 40 years projects values with negligible effects on the capabilities of the reservoirs; however, the applicant intends to use a monitoring program to address this TLAA. The plant-specific Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants Aging Management Program monitors the main and auxiliary reservoirs, shorelines, and drainage areas for landslides, excessive sedimentation, or drainage basin developments that could cause a sudden increase in sediment load that would reduce reservoir capacity. The frequency of the inspection of the auxiliary and main reservoirs is every five years. Inspection results to date have found no excessive sedimentation or changes leading to excessive sedimentation that could cause a sudden increase in sediment load; therefore, continued implementation of the Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants Aging Management Program will manage sedimentation effects in the main and auxiliary reservoirs during the period of extended operation.

4.7.3.2 Staff Evaluation

The staff reviewed LRA Section 4.7.3, to verify pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

The staff has evaluated the applicant's AMP B2.32, "RG 1.127, Inspection of Control Structures Associated with Nuclear Power Plants Program," for managing aging effects for dams and spillways, dikes, canals, reservoirs, and the intake, screening, and discharge structures of plant cooling water systems, determined that this program is acceptable to address aging effects for the main and auxiliary reservoirs in accordance with 10 CFR 54.21(c)(1)(iii), and documented its evaluation and acceptance in SER Section 3.0. On the basis that the applicant's action is consistent with the GALL Report recommendation, the staff finds that management of the effects of aging on intended functions will be adequate for the period of extended operation.

4.7.3.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of main and auxiliary reservoir sedimentation analyses in LRA Section A.1.2.7. On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address main and auxiliary reservoir sedimentation analyses is adequate.

4.7.3.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that, for main and auxiliary reservoir sedimentation analyses, 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 FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.4 High-Energy Line Break Location Postulation Based on Fatigue Cumulative Usage Factor

4.7.4.1 Summary of Technical Information in the Application

LRA Section 4.7.4 summarizes the evaluation of high-energy line break location postulation based on fatigue CUF for the period of extended operation. FSAR Section 3.6 describes the design bases and measures demonstrating that systems, components, and structures required to shut down and maintain the reactor in a cold shutdown condition safely are protected adequately against the effects of blow-down jets, reactive forces, and pipe whip from postulated rupture of piping both inside and outside containment.

CUFs have been useful in determining break locations of high-energy Class 1 piping systems except the RCS main loop piping. The applicant used guidance from RG 1.46, "Protection Against Pipe Whip Inside Containment," and from NRC Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment." FSAR Section 3.6.2.1.1.2 states that RG 1.46 has been followed in all matters except for the postulation of break points. MEB 3-1 criteria for Class 1 piping have been adapted to postulate pipe breaks occurring at:

- terminals
- intermediate locations where the maximum stress range as calculated by Eqs. (10) and either (12) or (13) exceeds 2.4 Sm.
- intermediate locations where the CUF exceeds 0.1.

Because the calculation of CUFs used design cycles of a 40-year design life the high-energy line-break postulation based on CUF is a TLAA.

As discussed in SER Subsection 4.3.1, original fatigue design calculations assumed a large number of design transients corresponding to relatively severe system dynamics over the original 40-year design life. Using the general approach described in LRA Subsection 4.3.1, the applicant made for license renewal 60-year fatigue cycle projections based on which the current design fatigue usage factors remain valid for 60 years of operations; therefore, the current CUFs for the postulation of break locations in Class 1 lines may be used for the 60-year operating term.

4.7.4.2 Staff Evaluation

The staff reviewed LRA Section 4.7.4, to verify pursuant to 10 CFR 54.21(c)(1)(i), that the analyses remain valid for the period of extended operation and, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

By letter dated January 17, 2008, the applicant amended LRA Section 4.7.4 to clarify its basis for managing the TLAA on high-energy line breaks in accordance with the aging management criterion in 10 CFR 54.21(c)(1)(iii) and to withdraw 10 CFR 54.21(c)(1)(i) as a basis for TLAA acceptance. In its response, the applicant made the following specific amendments of the LRA:

> Revise the Analysis and Disposition discussions of LRA Subsection 4.7.4 to read as follows:

Analysis

Original fatigue design calculations assumed a large number of design transients, corresponding to relatively severe system dynamics over the original 40-year design life. The current design fatigue usage factors will remain valid during the period of extended operation as long as the number of design transients is not exceeded.

The HNP Fatigue Monitoring Program will identify when piping systems are approaching the original 40-year number of design transients. Prior to any piping system exceeding its original number of design transients, the pertinent design calculations for that system will be reviewed to determine if any additional locations should be designated as postulated high energy line breaks, under the original criteria of Section 3.6 of the FSAR. If other locations are determined to require consideration as postulated break locations, appropriate actions will be taken to address the new break locations.

Disposition: 10 CFR 54.21(c)(1)(iii) – The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Make a conforming change to LRA Table 4.1-1 to revise the method used to comply with10 CFR 54.21(c)(1) for Subsection 4.7.4 to be 54.21(c)(1)(iii). Also, revise the final two paragraphs of LRA Subsection A.1.2.8 to read verbatim with the two paragraphs in the Analysis subsection of LRA 4.7.4 above.

In addition, revise enhancement (5) of LRA Subsection A.1.1.38 to read:

(5) address corrective actions, to be implemented through the Corrective Action Program, for components that have exceeded alarm limits, with options to include a revised fatigue analysis or repair or replacement of the component and for piping systems that have exceeded their cyclic alarm limit to require a review of the pertinent design calculations to determine if any additional locations should be designated as postulated high energy line breaks.

Revise LRA Subsection B.3.1 to address potential high energy line break locations by revising the following enhancement in LRA Subsection B.3.1:

Program Elements Affected

- Corrective Actions

Enhance the program to address corrective actions if an analyzed component is determined to have exceeded the alarm limit, with options to revise the fatigue analysis, repair, or replace the component. Corrective actions, if required, will be implemented through the HNP Corrective Action Program. Enhance the program to address if a piping system is determined to have exceeded its cyclic alarm limit to require a review of the pertinent design calculations to determine if any additional locations should be designated as postulated high energy line breaks.

This changed enhancement impacts License Renewal Commitment #32.

The staff verified that the applicant has amended Commitment No. 32 on the LRA to include this corrective action for analyzed components that exceed the metal fatigue alarm limits and that it has the potential to be high energy line breaks, and that this included this as provision (5) in LRA Commitment No. 32, as provided in the applicant's letter dated January 17, 2008. This amendment of LRA Section 4.7.4 and of enhancement of the Fatigue Monitoring Program, as given in provision (5) of Commitment No. 32, will ensure that those Class 1 piping locations that exceed metal fatigue alarms limits will be analyzed further to see if they need to be identified as high energy line break locations that will require additional analysis by the license. The staff finds this to be a conservative approach. This amendment of Commitment No. 32 is consistent with the recommended "corrective actions" program element criterion in GALL AMP X.M1, "Metal Fatigue of the Reactor Coolant Pressure Boundary," and is acceptable.

Based on the applicant's amendment of LRA Section 4.7.4 and of Commitment No. 32 to implement appropriate corrective actions for Class 1 pipe locations that are determined to exceed the metal fatigue alarm limit, the staff concludes that the applicant's Fatigue Monitoring Program is capable of the high energy line break locations already identified in the LRA and that it may be identified if the alarm limit for a particular Class 1 piping location is exceeded. This program will maintain the validity of the design fatigue value. On this basis, the staff determined that, as long as the design CUF values remain valid, so will the high-energy line break locations, and that, if the design CUF values exceed the CUF limit of 1.0, the applicant's implementation of the Fatigue Monitoring Program will initiate appropriate corrective actions in accordance with provision (5) in LRA Commitment No. 32. On this basis, the staff finds that the applicant has provided an acceptable basis for using the Fatigue Monitoring Program to manage its TLAA on high energy line breaks in accordance with 10 CFR 54.21(c)(1)(iii).

4.7.4.3 FSAR Supplement

The applicant provided an FSAR supplement summary description of its TLAA evaluation of high-energy line break location postulation based on fatigue cumulative usage factor in LRA Section A.1.2.8.

The staff has verified that FSAR supplement Section A.1.2.8 ties the basis for accepting applicant's TLAA on high energy line breaks to FSAR Supplement Section A.1.1.38 on the applicant's Reactor Coolant Pressure Boundary Fatigue Monitoring Program. Thus, FSAR supplement Section A.1.1.38 on the applicant's Reactor Coolant Pressure Boundary Fatigue Monitoring Program is also applicable to this TLAA. The staff has also verified that the applicant has amended FSAR Section A.1.1.38 to incorporate the enhancement of the applicant's Reactor Coolant Pressure Boundary Fatigue Monitoring Program to incorporate the new corrective action for Class 1 piping location that exceed the applicant's metal fatigue alarm limit and that this enhancement has been incorporated into the revision of LRA Commitment No. 32 that was provided in the applicant's letter dated January 17, 2008.

On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address high-energy line break location postulation based on its crediting of the Reactor Coolant Pressure Boundary Fatigue Monitoring Program is acceptable and that FSAR supplement Section A.1.28 is acceptable in accordance with 10 CFR 54.21(d).

4.7.4.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that, for high-energy line break location postulation based on fatigue CUF, the analyses remain valid for the period of extended operation. The applicant also has demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that 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 FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.8 Conclusion for TLAAs

The staff reviewed the information in LRA Section 4, "Time-Limited Aging Analyses." On the basis of its review, the staff concludes, that the applicant has provided a sufficient list of TLAAs, as defined in 10 CFR 54.3 and that the applicant has demonstrated that: (1) the TLAAs will remain valid for the period of extended operation, as required by 10 CFR 54.21(c)(1)(i); (2) the TLAAs have been projected to the end of the period of extended operation, as required by 10 CFR 54.21(c)(1)(ii); or (3) that the effects of aging on intended function(s) will be adequately managed for the period of extended operation, as required by 10 CFR 54.21(c)(1)(iii). The staff also reviewed the FSAR supplement for the TLAAs and finds that the supplement contains descriptions of the TLAAs sufficient to satisfy the requirements of 10 CFR 54.21(d).

SECTION 5

REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The NRC staff issued its safety evaluation report (SER) with open item related to the renewal of operating license for Shearon Harris Nuclear Power Plant, Unit 1 on March 18, 2008. On May 7, 2008, the applicant presented its license renewal application, and the staff presented its review findings to the ACRS Plant License Renewal Subcommittee. The staff reviewed the applicant's comments on the SER and completed its review of the license renewal application. The staff's evaluation is documented in an SER that was issued by letter dated August 21, 2008.

During the 556th meeting of the ACRS, October 2, 2008, the ACRS completed its review of the HNP license renewal application and the NRC staff's SER. The ACRS documented its findings in a letter to the Commission dated October 16, 2008. A copy of this letter is provided on the following pages of this SER section.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

October 16, 2008

The Honorable Dale E. Klein
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

Dear Chairman Klein:

During the 556th meeting of the Advisory Committee on Reactor Safeguards, October 2-3, 2008, we completed our review of the license renewal application for the Shearon Harris Nuclear Power Plant (HNP), Unit 1 and the final Safety Evaluation Report (SER) prepared by the NRC staff. Our Plant License Renewal Subcommittee also reviewed this matter during a meeting on May 7, 2008. During these reviews, we had the benefit of discussions with representatives of the NRC staff and the applicant, Carolina Power & Light Company (CP&L), doing business as Progress Energy Carolinas, Inc. We also had the benefit of the documents referenced. This report fulfills the requirement of 10 CFR 54.25 that the ACRS review and report on all license renewal applications.

CONCLUSION AND RECOMMENDATIONS

1. The programs established and committed to by the applicant to manage age-related degradation provide reasonable assurance that HNP, Unit 1 can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public.

2. The application for renewal of the operating license of HNP, Unit 1 should be approved.

3. Prior to the period of extended operation, the staff should inspect the applicant's programs for managing water intrusion into underground cable vaults and cable insulation testing.

BACKGROUND AND DISCUSSION

HNP, Unit 1 is a three-loop Westinghouse pressurized water reactor with a large, dry, steel-lined, reinforced concrete containment. The current power rating of 2900 MWt includes a 4.5 percent power uprate that was implemented in 2001. The original HNP steam generators were replaced in 2001. Pressurizer nozzle weld overlays and enlargement of the containment sump screen were completed in 2007. CP&L requested renewal of the HNP, Unit 1 operating license for 20 years beyond the current license term, which expires on October 24, 2026.

In the final SER, the staff documented its review of the license renewal application and other information submitted by CP&L and obtained during the audits and an inspection conducted at the plant site. The staff reviewed the completeness of the applicant's identification of structures, systems, and components (SSCs) that are within the scope of license renewal; the integrated plant assessment process; the applicant's identification of the plausible aging mechanisms associated with passive, long-lived components; the adequacy of the applicant's Aging Management Programs (AMPs); and the identification and assessment of time-limited aging analyses (TLAAs) requiring review.

The applicant identified the SSCs that fall within the scope of license renewal and performed an aging management review for these SSCs. The applicant will implement 40 AMPs for license renewal. These include 28 existing programs, 19 of which have been enhanced, and 11 new programs that are consistent with guidance in the Generic Aging Lessons Learned (GALL) Report. In addition, one new site-specific AMP for testing of underground high-voltage oil-filled cable connections to the plant switchyard was developed as a result of the staff's audits and review.

The applicant identified the systems and components requiring TLAAs and reevaluated them for the period of extended operation. The staff concluded that the applicant has provided an acceptable list of TLAAs, as defined in 10 CFR 54.3. Further, the staff concluded that in all cases the applicant has met the requirements of the license renewal rule by demonstrating that: (a) the TLAAs will remain valid for the period of extended operation; (b) the TLAAs have been projected to the end of the period of extended operation; or (c) the aging effects will be adequately managed for the period of extended operation. We concur with the staff's conclusion that HNP TLAAs have been properly identified and that the required criteria will be met for the period of extended operation.

The staff conducted four license renewal audits and one inspection at the HNP site. The audits verified the appropriateness of the scoping and screening methodology, AMPs, aging management reviews, and TLAAs. The inspection verified that the license renewal requirements are appropriately implemented. Based on the audits and inspection, the staff concludes in the SER that the proposed activities will effectively manage the aging of SSCs identified in the application and that the intended functions of these SSCs will be maintained during the period of extended operation. We agree with this conclusion.

Corrosion of the containment liner at the base slab was detected in 1997. The moisture barrier was replaced in 1998, and only minor corrosion has been observed during subsequent inspections. Minor corrosion and pitting was recorded in 1993, 2000, and 2004 on exterior and interior surfaces of the containment spray and residual heat removal valve enclosures in the Auxiliary Building, which form part of the containment

pressure boundary. No significant material loss has been reported. We agree with the staff's conclusion that the HNP ASME Section XI, Subsection IWE Program will adequately detect and manage the effects of containment liner corrosion.

Prior to the period of extended operation, the staff should confirm that the applicant has properly implemented the testing program intended to ensure that long-term cable insulation properties are maintained. Due to the plant-specific history of water intrusion into underground cable vaults, the staff should also confirm the adequacy of the applicant's programs to monitor and control water levels to minimize wetting of the cables.

The HNP current licensing basis (CLB) analyses include credit for closure of the feedwater regulating valves and bypass valves as a redundant method for main feedwater isolation during a main steamline break inside containment. According to the CLB, the feedwater regulating and bypass valves are not classified as safety-related components. The valves are located in the Turbine Building and close automatically from a main feedwater isolation signal, a loss of power signal from the reactor protection system, loss of control air, or loss of DC power. The staff raised a concern that the license renewal requirements for safety-related components, specified under 10 CFR 54.4(a)(1) should apply to these valves, due to their main feedwater isolation function.

The applicant responded that Section 15.1.5 of the Standard Review Plan specifically allows credit for backup nonsafety-related components to mitigate the consequences of a main steamline break inside containment, following a single failure of an active safety-related isolation valve. As a result, the staff concludes in the final SER that the feedwater regulating and bypass valves are properly categorized as nonsafety-related components, that the requirements of 10 CFR 54.4(a)(2) apply to these valves, and that no additional SSCs need to be included within the HNP license renewal scope to ensure the isolation function of these valves. We agree with these conclusions.

We agree with the staff that there are no issues related to the matters described in 10 CFR 54.29(a)(1) and (a)(2) that preclude renewal of the operating license for HNP, Unit 1. The programs established and committed to by CP&L provide reasonable assurance that HNP, Unit 1 can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public. The CP&L application for renewal of the operating license for HNP, Unit 1 should be approved.

Sincerely,

/RA/

William J. Shack,
Chairman

REFERENCES

1. Letter dated November 14, 2006, from C. J. Gannon, Jr., Carolina Power & Light Company, doing business as Progress Energy Carolinas, Inc. to U.S. Nuclear Regulatory Commission, transmitting the Shearon Harris Nuclear Power Plant, Unit 1, Application for Renewal of Operating License (ML063350270).

2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of Shearon Harris Nuclear Power Plant, Unit 1," August 2008 (ML082340985).

3. Letter dated March 26, 2008, from M. Heath, U.S. Nuclear Regulatory Commission to R. J. Duncan II, Carolina Power & Light Company, transmitting the "Audit Summary Regarding the License Renewal Application for the Shearon Harris Nuclear Power Plant, Unit 1" (ML080800243).

4. Letter dated September 10, 2007, from J. W. Shea, U.S. Nuclear Regulatory Commission to R. J. Duncan II, Carolina Power & Light Company, transmitting Shearon Harris Nuclear Power Plant NRC Inspection Report 05000400/2007007 (ML072530894).

5. U.S. Nuclear Regulatory Commission, NUREG-1801, Volumes 1 & 2, Revision 1, "Generic Aging Lessons Learned Report," September 2005.

6. Letter dated May 30, 2008, from C. L. Burton, Progress Energy Carolinas, Inc. to U.S. Nuclear Regulatory Commission, transmitting "Resolution of Open Item and License Renewal Application Amendment 8" (ML081570346).

7. U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 15, Accident Analysis, Section 15.1.5, Steam System Piping Failures Inside and Outside of Containment PWR, Revision 3, March 2007.

SECTION 6

CONCLUSION

The staff of the United States (US) Nuclear Regulatory Commission (NRC) (the staff) reviewed the license renewal application (LRA) for Shearon Harris Nuclear Power Plant, Unit 1, in accordance with NRC regulations and NUREG-1800, Revision 1, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," dated September 2005. Title 10, Section 54.29, of the *Code of Federal Regulations* (10 CFR 54.29) sets the standards for issuance of a renewed license.

On the basis of its review of the LRA, the staff determines that the requirements of 10 CFR 54.29(a) have been met.

The staff noted that any requirements of 10 CFR Part 51, Subpart A, are documented in NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants (GEIS)," Supplement 33, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants, Supplement 3 Regarding Shearon Harris Nuclear Power Plant, Unit 1," dated August 13, 2008.

APPENDIX A

HNP UNIT 1 LICENSE RENEWAL COMMITMENTS

During the review of the Shearon Harris Nuclear Power Plant (HNP), Unit 1, license renewal application (LRA) by the staff of the United States (US) Nuclear Regulatory Commission (NRC) (the staff), Carolina Power & Light Company (the applicant) made commitments related to aging management programs (AMPs) to manage aging effects for structures and components. The following table lists these commitments along with the implementation schedules and sources for each commitment.

APPENDIX A: HNP UNIT 1 LICENSE RENEWAL COMMITMENTS				
Item Number	Commitment	FSAR Supplement Section/ LRA Section	Enhancement or Implementation Schedule	Source
(1)	In accordance with the guidance of NUREG-1801, Revision 1, regarding aging management of reactor vessel internals components, HNP will: (1) participate in the industry programs for investigating and managing aging effects on reactor internals (such as Westinghouse Owner's Group and Electric Power Research Institute materials programs), (2) evaluate and implement the results of the industry programs as applicable to the reactor internals, and (3) upon completion of these programs, but not less than 24 months before entering tile period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	A.1.1	As stated in the commitment	Reactor Vessel Internals Aging Management Activities LRA Section A.1.1
(2)	In accordance with the guidance of NUREG-1801, Revision 1, regarding aging management of nickel alloy and nickel-clad components susceptible to primary water stress corrosion cracking, HNP will comply with applicable NRC Orders and will implement: (1) applicable Bulletins and Generic Letters, and (2) staff-accepted industry guidelines.	A.1.1	As stated in the commitment	Primary Water Stress Corrosion Cracking of Nickle Alloys LRA Section A.1.1

APPENDIX A: HNP UNIT 1 LICENSE RENEWAL COMMITMENTS				
Item Number	Commitment	FSAR Supplement Section/ LRA Section	Enhancement or Implementation Schedule	Source
(3)	Program inspections are performed as augmented inspections in the HNP Inservice Inspection (ISI) Program. The ISI Program administrative controls will be enhanced to specifically identify the requirements of NRC Order EA-03-009.	A.1.1.5	Prior to the period of extended operation	Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors Program LRA Section B.2.5
(4)	The Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program is a new program to be implemented.	A.1.1.6	Prior to the period of extended operation	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program LRA Section B.2.6

APPENDIX A: HNP UNIT 1 LICENSE RENEWAL COMMITMENTS				
Item Number	**Commitment**	**FSAR Supplement Section/ LRA Section**	**Enhancement or Implementation Schedule**	**Source**
(5)	The Program will be enhanced to provide a consolidated exclusion bases document (i.e., a FAC susceptibility analysis). The exclusion bases document will include an evaluation of the Steam Generator Feedwater Nozzles to determine their susceptibility to FAC.	A.1.1.7	Prior to the period of extended operation	The Flow-Accelerated Corrosion (FAC) Program LRA Section B.2.7
(6)	A precautionary note will be added to plant bolting guidelines to prohibit the use of molybdenum disulfide lubricants.	A.1.1.8	Prior to the period of extended operation	Bolting Integrity Program LRA Section B.2.8
(7)	The Program implementing procedure will be enhanced to include a description of the instructions for implementing corrective actions if tube plugs or secondary-side components (e.g., tube supports) are found to be degraded.	A.1.1.9	Prior to the period of extended operation	Steam Generator Tube Integrity Program LRA Section B.2.9

APPENDIX A: HNP UNIT 1 LICENSE RENEWAL COMMITMENTS				
Item Number	Commitment	FSAR Supplement Section/ LRA Section	Enhancement or Implementation Schedule	Source
(8)	The Program will be enhanced to: 1) include measurements of actual boron areal density using in-situ techniques, 2) include neutron attenuation testing ("blackness testing"), to determine gap formation in Boraflex panels, and 3) include the use of the EPRI RACKLIFE predictive code or its equivalent.	A.1.1.12	Prior to the period of extended operation, unless an approved analysis exists that eliminates credit for the Boraflex in the BWR fuel racks	Boraflex Monitoring Program LRA Section B.2.12
(9)	The Program will be enhanced to: (1) include in the Program all cranes within the scope of license renewal; (2) require the responsible engineer to be notified of unsatisfactory crane inspection results; (3) specify an annual inspection frequency for the Fuel Cask Handling Crane, Fuel Handling Bridge Crane, and Fuel handling Building Auxiliary Crane, and every refuel cycle for the Polar Crane, Jib Cranes, and Reactor Cavity Manipulator Crane, and (4) include a requirement to inspect for bent or damaged members, loose bolts/components, broken welds, abnormal wear of rails, and corrosion (other than minor surface corrosion) of steel members and connections.	A.1.1.13	Prior to the period of extended operation	Inspection of Overhead Heavy Load and Light Load Handling Systems Program LRA Section B.2.13 Response to Audit Question B.2.13-JW-01

APPENDIX A: HNP UNIT 1 LICENSE RENEWAL COMMITMENTS				
Item Number	Commitment	FSAR Supplement Section/ LRA Section	Enhancement or Implementation Schedule	Source
(10)	The program will be enhanced to: (1) include inspection criteria as described in NUREG-1801 for penetration seals, (2) provide specific procedural guidance for inspecting fire barrier walls, ceilings and floors, (3) include a visual inspection of the diesel-driven fire pump fuel oil supply piping for signs of leakage, and (4) include minimum qualification requirements for inspectors performing inspections required by this Program.	A.1.1.14	Prior to the period of extended operation	Fire Protection Program LRA Section B.2.14
(11)	The Program will be revised to: (1) incorporate a requirement to perform one or a combination of the following two activities: (a) Perform non-intrusive baseline pipe thickness measurements at various locations, prior to the expiration of current license and trended through the period of extended operation. The plant-specific inspection intervals will be determined by engineering evaluation performed after each inspection of the fire protection piping to detect degradation prior to the loss of intended function, or (b) Perform flow testing meeting the general flow requirements (intent) of NFPA 25, (2) either replace the sprinkler heads prior to reaching their 50-year service life or revise site procedures to perform field service testing, by a recognized testing laboratory, of representative samples from one or more sample areas, and (3) for in-scope spray nozzles, either (a) add a requirement to perform flow testing to ensure proper spray pattern or (b) add a modification to prevent blockage from external sources.	A.1.1.15	Prior to the period of extended operation	Fire Water System Program LRA Section B.2.15 Commitment (1)(b) and the option of using a combination of (l)(a) and (1)(b) were added in the response to Audit Question B.2.15-PB-01. Commitment (3) was added per Audit Question 3.3.1-70-MK-0 1.
(12)	Program administrative controls will be enhanced to: (1) add requirements to enter an item into the corrective action program whenever an administrative value or control limit for parameters	A.1.1.16	Prior to the period of extended	Fuel Oil Chemistry Program

APPENDIX A: HNP UNIT 1 LICENSE RENEWAL COMMITMENTS				
Item Number	Commitment	FSAR Supplement Section/ LRA Section	Enhancement or Implementation Schedule	Source
	relevant to this program are exceeded or water is drained from a fuel oil tank in the scope of this program; (2) establish administrative values for fuel oil chemistry parameters relating to corrosion; (3) require Diesel Fuel Oil System chemistry controls to include semiannual monitoring and trending of water and sediment and particulates from an appropriate sample point for the day tanks and semiannual monitoring and trending of biological growth in the main storage tanks; (4) require Security Power System fuel oil chemistry controls to include semiannual monitoring and trending of biological growth in the fuel oil in the buried storage tank and periodic inspecting of the internal surfaces of the buried storage tank and the aboveground day tank or require UT or other NDE of the tanks if inspection proves inadequate or indeterminate; (5) require Site Fire Protection System fuel oil chemistry controls for the Diesel Driven Fire Pump fuel oil storage tank to include quarterly monitoring and trending of particulates and semiannual monitoring and trending of biological growth, to check and remove water quarterly, to periodically inspect the tank or require UT or other NDE of the tank if inspection proves inadequate or indeterminate; and to revise chemistry sampling procedures to address positive results for biological growth including as one option the use of biocides; and (6) verify the condition of the Diesel Fuel Oil Storage Tank Building Tank Liners by means of bottom thickness measurements under the One Time Inspection Program. Day tank sampling for water, sediment, and particulate contamination is considered to be confirmatory of components outside the main storage tanks, and its frequency may be adjusted based on site operating experience.		operation	LRA Section B.2.16 Response to Audit Question B.2.16-MK-12

APPENDIX A: HNP UNIT 1 LICENSE RENEWAL COMMITMENTS				
Item Number	Commitment	FSAR Supplement Section/ LRA Section	Enhancement or Implementation Schedule	Source
(13)	The Program will be enhanced to: (1) include a provision that tested and untested specimens from all capsules pulled from the reactor vessel must be kept in storage to permit future reconstitution use, and that the identity, traceability, and recovery of the capsule specimens shall be maintained throughout testing and storage, (2) include a provision that withdrawal of the next capsule (i.e., Capsule W) will occur during Refueling Outage 16, at which time the capsule fluence is projected to be equivalent to the 60-year maximum vessel fluence of 6.8×10^{19} n/cm2 in accordance with ASTM E 185-82, (3) include a provision that analysis of Capsule W be used to evaluate neutron exposure for remaining Capsules Y and Z, as required by 10 CFR Part 50 Appendix H. The withdrawal schedule for one of the remaining capsules will be adjusted, based on the analysis of Capsule W, so that the capsule fluence will not exceed twice the 60-year maximum vessel fluence in accordance with ASTM E 185-82. The neutron exposure and withdrawal schedule for the last capsule will be optimized to provide meaningful metallurgical data. If the last capsule is projected to significantly exceed a meaningful fluence value, it will either be relocated to a lower flux position or withdrawn for possible testing or re-insertion. Capsules Y and Z and archived test specimens available for reconstitution will be available for the monitoring of neutron exposure if additional license renewals are sought, and (4) include a provision that, if future plant operations exceed the limitations in Section1.3 of Regulatory Guide 1.99, Revision 2, or the applicable bounds, e.g., cold leg operating temperature and neutron fluence, as applied to the surveillance capsules, the impact of these plant operation changes on the extent of reactor vessel embrittlement	A.1.1.17	Prior to the period of extended operation	Reactor Vessel Surveillance Program LRA Section B.2.17, Response to RAI B.2.17

APPENDIX A: HNP UNIT 1 LICENSE RENEWAL COMMITMENTS				
Item Number	Commitment	FSAR Supplement Section/ LRA Section	Enhancement or Implementation Schedule	Source
	will be evaluated, and the NRC will be notified.			
(14)	The One-Time Inspection Program is a new program to be implemented.	A.1.1.18	Prior to the period of extended operation	One Time Inspection Program LRA Section B.2.18
(15)	The Selective Leaching of Materials Program is a new program to be implemented.	A.1.1.19	Prior to the period of extended operation	Selective Leaching of Materials Program LRA Section B.2.19
(16)	The Buried Piping and Tanks Inspection Program is a new program to be implemented.	A.1.1.20	Prior to the period of extended operation	Buried Piping and Tanks Inspection Program LRA Section B.2.20

APPENDIX A: HNP UNIT 1 LICENSE RENEWAL COMMITMENTS				
Item Number	Commitment	FSAR Supplement Section/ LRA Section	Enhancement or Implementation Schedule	Source
(17)	The One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program is a new program to be implemented.	A.1.1.21	Prior to the period of extended operation	One-Time Inspection of ASME Code Class 1 Small Bore Piping Program LRA Section B.2.21
(18)	The program will be enhanced to: (1) include a specific list of systems managed by the program for License Renewal, (2) provide specific guidance for insulated/jacketed pipe and piping components to identify signs of leakage and provide criteria for determining whether the insulation/jacket should be removed to inspect for corrosion, (3) provide inspection criteria for components not readily accessible during plant operations or refueling outages, (4) provide specific guidance for visual inspections of elastomers for cracking, chafing, or changes in material properties due to wear, and (5) incorporate a checklist for evaluating inspection findings, with qualified dispositions.	A.1.1.22	Prior to the period of extended operation	External Surfaces Monitoring Program LRA Section B.2.22
(19)	The Program will be enhanced: (1) to require an evaluation of historic plant-specific test data in order to ensure that conservative wear rates are used so that a loss of intended function will not occur, (2) to provide guidance for treatment of flux thimbles that could not be inspected due to restriction, defect or other reason, and (3) to require test results and evaluations be formally documented as QA records.	A.1.1.23	Prior to the period of extended operation	Flux Thimble Tube Inspection Program LRA Section B.2.23

APPENDIX A: HNP UNIT 1 LICENSE RENEWAL COMMITMENTS				
Item Number	Commitment	FSAR Supplement Section/ LRA Section	Enhancement or Implementation Schedule	Source
(20)	The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program is a new program to be implemented.	A.1.1.24	Prior to the period of extended operation	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program LRA Section B.2.24
(21)	The Program will be enhanced as follows: (1) a review and revision of work documents and analysis requirements will be performed to ensure that the used oil from appropriate component types in the scope of license renewal is analyzed to determine particle count and moisture, and if oil is not changed in accordance with the manufacturer's recommendation, then additional analyses for viscosity, neutralization number, and flash point will be performed. This activity will ensure that used oil is visually checked for water; and (2) the program administrative controls will be enhanced to include a requirement to perform ferrography or elemental analysis to identify wear particles or products of corrosion when particle count exceeds an established level or when considered appropriate.	A.1.1.25	Prior to the period of extended operation	Lubrication Oil Analysis Program LRA Section B.2.25

APPENDIX A: HNP UNIT 1 LICENSE RENEWAL COMMITMENTS				
Item Number	**Commitment**	**FSAR Supplement Section/ LRA Section**	**Enhancement or Implementation Schedule**	**Source**
(22)	The Program implementing procedure will be enhanced to: (1) include additional recordable conditions, (2) include moisture barrier and applicable aging effects, (3) include pressure retaining bolting and aging effects, and (4) include a discussion of augmented examinations.	A.1.1.26	Prior to the period of extended operation	ASME Section XI, Subsection IWE Program LRA Section B.2.26
(23)	The Program will be enhanced to describe in the implementing procedures the evaluation and corrective actions to be taken when leakage rates do not meet their specified acceptance criteria.	A.1.1.29	Prior to the period of extended operation	10 CFR Part 50, Appendix J Program LRA Section B.2.29
(24)	Program administrative controls will be enhanced to identify the structures that have masonry walls in the scope of License Renewal.	A.1.1.30	Prior to the period of extended operation	Masonry Wall Program LRA Section B.2.30
(25)	The Program implementing procedures will be enhanced to: (1) identify the License Renewal structures and systems that credit the program for aging management, (2) require notification of the responsible engineer when below-grade concrete is exposed so an inspection may be performed prior to backfilling, (3) require periodic groundwater chemistry monitoring including consideration for potential seasonal variations., (4) define the term "structures of a system" in the	A.1.1.31	Prior to the period of extended operation	Structures Monitoring Program LRA Section B.2.31

APPENDIX A: HNP UNIT 1 LICENSE RENEWAL COMMITMENTS				
Item Number	Commitment	FSAR Supplement Section/ LRA Section	Enhancement or Implementation Schedule	Source
	system walkdown procedure and specify the condition monitoring parameters that apply to "structures of a system," (5) include the corporate structures monitoring procedure as a reference in the plant implementing procedures and specify that forms from the corporate procedure be used for inspections, (6) identify additional civil/structural commodities and associated inspection attributes required for License Renewal, and (7) require inspection of inaccessible surfaces of reinforced concrete pipe when exposed by removal of backfill.			
(26)	The Program will be enhanced to: (1) require an evaluation of any concrete deficiencies in accordance with the acceptance criteria provided in the corporate inspection procedure, (2) require initiation of a Nuclear Condition Report (NCR) for degraded plant conditions and require, as a minimum, the initiation of an NCR for any condition that constitutes an "unacceptable" condition based on the acceptance criteria specified, and (3) require documentation of a visual inspection of the miscellaneous steel at the Main Dam and Spillway.	A.1.1.32	Prior to the period of extended operation	RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants Program LRA Section B.2.32

APPENDIX A: HNP UNIT 1 LICENSE RENEWAL COMMITMENTS				
Item Number	Commitment	FSAR Supplement Section/ LRA Section	Enhancement or Implementation Schedule	Source
(27)	The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program to be implemented.	A.1.1.33	Prior to the period of extended operation	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program LRA Section B.2.33
(28)	The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program is a new program to be implemented.	A.1.1.34	Prior to the period of extended operation	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used In Instrumentation Circuits Program LRA Section B.2.34

APPENDIX A: HNP UNIT 1 LICENSE RENEWAL COMMITMENTS				
Item Number	**Commitment**	**FSAR Supplement Section/ LRA Section**	**Enhancement or Implementation Schedule**	**Source**
(29)	The Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program to be implemented.	A.1.1.35	Prior to the period of extended operation	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program LRA Section B.2.35
(30)	The Metal Enclosed Bus Program is a new program to be implemented.	A.1.1.36	Prior to the period of extended operation	Metal Enclosed Bus Program LRA Section B.2.36
(31)	The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program to be implemented.	A.1.1.37	Prior to the period of extended operation	Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Program LRA Section B.2.37

APPENDIX A: HNP UNIT 1 LICENSE RENEWAL COMMITMENTS				
Item Number	Commitment	FSAR Supplement Section/ LRA Section	Enhancement or Implementation Schedule	Source
(32)	The Program will be enhanced to: (1) expand the program scope to include an evaluation of selected RCPB components beyond the reactor pressure vessel (including auxiliary system components such as the pressurizer lower head, pressurizer surge line, and CVCS piping and heat exchanger), and to include the NUREG/CR-6260 locations analyzed for environmental effects, (2) provide preventive actions to include, prior to a monitored location exceeding a cumulative usage factor limit of 1.0, evaluation of operational changes to reduce the number or severity of future transients, (3) include a provision to utilize online fatigue analysis software for the periodic updating (not to exceed once every 18 months) of cumulative usage, (4) describe the acceptance criteria for maintaining fatigue usage below the design limit, and (5) address corrective actions, to be implemented through the Corrective Action Program, for components that have exceeded alarm limits, with options to include a revised fatigue analysis or repair or replacement of the component and for piping systems that have exceeded their cyclic alarm limit to require a review of the pertinent design calculations to determine if any additional locations should be designated as postulate high energy line breaks.	A.1.1.38	Prior to the period of extended operation	Reactor Coolant Pressure Boundary (RCPB) Fatigue Monitoring Program LRA Section B.3.1
(33)	The Low Temperature Overpressure (LTOP) setpoint analysis will be recalculated following removal of one of the remaining surveillance capsules from the reactor vessel.	A.1.2.1.4	Prior to the period of extended operation	TLAA - Low temperature Over-Pressure Limits LRA Section 4.2.5

APPENDIX A: HNP UNIT 1 LICENSE RENEWAL COMMITMENTS				
Item Number	**Commitment**	**FSAR Supplement Section/ LRA Section**	**Enhancement or Implementation Schedule**	**Source**
(34)	The Oil-Filled Cable Testing Program is a new program to be implemented.	A.1.1.40	Prior to the period of extended operation	Oil-Filled Cable Testing Program LRA Section B.2.38, Response to Audit Question LRA-3.6.2-1-RM-02
(35)	When the EPRI MRP methodology described in MRP-140, "Materials Reliability Program: Leak-Before-Break Evaluation for PWR Alloy 82/182 Welds," has been reviewed and approved by the NRC, HNP will review its plant-specific calculation for conformance to the endorsed approach.	A.1.2.2.11	As stated in the commitment	TLAA - Leak-Before-Break evaluation for Alloy 82/182 Welds LRA Section 4.3.4, Response to Audit Question LRA 4.3.4-1

APPENDIX A: HNP UNIT 1 LICENSE RENEWAL COMMITMENTS				
Item Number	Commitment	FSAR Supplement Section/ LRA Section	Enhancement or Implementation Schedule	Source
(36)	HNP will replace the subject elastomeric and thermoplastic components referenced in RAIs 3.4-2, 3.4-3, 3.4-4, 3.4-5, and 3.4-7 and add them to the Preventive Maintenance Program. HNP will perform an evaluation to determine the frequency of periodic replacement of the components during the period of extended operation based on the guidance in the HNP Preventive Maintenance Program.	A.1.1	Prior to the period of extended operation	Section 3.4 AMR Tables for Main Steam Supply, Feedwater, and Secondary Sampling Systems Response to Confirmatory Item 3.4-1
(37)	HNP will update the piping design specification to reflect the current design basis operational transients used in the Time-Limited Aging Analyses for the reactor coolant pressure boundary.	A.1.1	Prior to the period of extended operation	Table 4.3-3 60-year Environmentally Adjusted CUF Valves Response to Confirmatory Item 4.3

APPENDIX B

CHRONOLOGY

This appendix lists chronologically the routine licensing correspondence between the staff of the United States (US) Nuclear Regulatory Commission (NRC) (the staff) and Carolina Power & Light Company (CP&L). This appendix also lists other correspondence on the staff's review of the Shearon Harris Nuclear Power Plant (HNP), Unit 1 license renewal application (LRA) (under Docket No. 50-400).

APPENDIX B: CHRONOLOGY	
Date	**Subject**
November 14, 2006	In a letter (signed by C. J. Gannon), CP&L submitted an application to renew the operating license of Shearon Harris Nuclear Power Plant, Unit 1. In its submittal, CP&L provided an original signed hard copy of the application and additional electronic copies of the application on CDs. (ADAMS Accession No. ML063350267)
November 14, 2006	In a letter (signed by C. J. Gannon), CP&L submitted three sets of reference drawing to the NRC. (ADAMS Accession No. ML063240168)
December 5, 2006	In a letter (signed by P. T. Kuo), the NRC acknowledged receipt and availability of the license renewal application for Shearon Harris Nuclear Power Plant, Unit 1. (ADAMS Accession No. ML063210237)
January 8, 2007	In a letter (signed by P. T. Kuo), the NRC determined the acceptability and sufficiency for docketing the application from CP&L, for renewal of the operating license for Shearon Harris Nuclear Power Plant, Unit 1. (ADAMS Accession No. ML063520336)
February 22, 2007	In a letter (signed by M. Heath), the NRC staff issued RAIs associated with the review of the LRA for HNP Unit 1. (ADAMS Accession No. ML070510124)
March 14, 2007	In a letter (signed by P. T. Kuo), the NRC proposed a review schedule, intent to prepare an environmental impact statement and opportunity for a hearing regarding the application from CP&L, for renewal of Shearon Harris Nuclear Power Plant, Unit 1. (ADAMS Accession No. ML070230076)
March 23, 2007	In a letter (signed by D. Corlett), CP&L provided responses to RAIs associated with the review of the HNP LRA. (ADAMS Accession No. ML070880738)

APPENDIX B: CHRONOLOGY	
Date	**Subject**
June 11, 2007	In a letter (signed by M. Heath), the NRC staff issued RAIs associated with the review of the LRA for HNP Unit 1. (ADAMS Accession No. ML071590147)
July 10, 2007	In a letter (signed by C. Burton), CP&L provided responses to RAIs associated with the review of the HNP LRA (ADAMS Accession No. ML071980380)
July 20, 2007	In a letter (signed by M. Heath), the NRC staff issued RAIs associated with the review of the LRA for HNP Unit 1. (ADAMS Accession No. ML071860407)
August 7, 2007	In a letter (signed by M. Heath), the NRC staff issued RAIs associated with the review of the LRA for HNP Unit 1. (ADAMS Accession No. ML072140246)
August 7, 2007	In a letter (signed by M. Heath), the NRC staff issued RAIs associated with the review of the LRA for HNP Unit 1. (ADAMS Accession No. ML072180069)
August 7, 2007	In a letter (signed by M. Heath), the NRC staff issued RAIs associated with the review of the LRA for HNP Unit 1. (ADAMS Accession No. ML072140043)
August 13, 2007	Summary of a telephone conference held on August 13, 2007 with NRC staff and CP&L (ADAMS Accession No. ML072430282)
August 16, 2007	In a letter (signed by T. J. Natale), CP&L provided responses to RAIs associated with the review of the HNP LRA (ADAMS Accession No. ML072350080)
August 20, 2007	In a letter (signed by C. Burton), CP&L provided amendment 1 identifying changes from RAIs associated with the review of the LRA of HNP Unit 1. (ADAMS Accession No. ML072350552)
August 20, 2007	In a letter (signed by M. Heath), the NRC staff issued RAIs associated with the review of the LRA for HNP Unit 1. (ADAMS Accession No. ML072130460)
August 21, 2007	Summary of a telephone conference held on July 19, 2007 with NRC staff and CP&L (ADAMS Accession No. ML072260087)
August 27, 2007	In a letter (signed by M. Heath), the NRC staff issued RAIs associated with the review of the LRA for HNP Unit 1. (ADAMS Accession No. ML072260118)

APPENDIX B: CHRONOLOGY	
Date	**Subject**
August 31, 2007	In a letter (signed by T. J. Natale), CP&L provided amendment 2, identifying changes regarding Time-Limited Aging Analyses associated with the review of the LRA for HNP Unit 1. (ADAMS Accession No. ML072540804)
September 5, 2007	In a letter (signed by T. J. Natale), CP&L provided responses to RAIs associated with the review of the HNP LRA (ADAMS Accession No. ML072560017)
September 18, 2007	In a letter (signed T. J. Natale), CP&L provided responses to RAIs associated with the review of the HNP LRA (ADAMS Accession No. ML072680944)
September 24, 2007	In a letter (signed by T. J. Natale), CP&L provided amendment 3, identifying changes regarding Aging Management Review associated with the review of the LRA for HNP Unit 1. (ADAMS Accession No. ML072750528)
November 5, 2007	In a letter (signed by T. J. Natale), CP&L provided amendment 4, license renewal 10 CFR 54.21 (b) annual update associated with the LRA for HNP Unit 1. (ADAMS Accession No. ML073180491)
December 11, 2007	In a letter (signed by T. J. Natale), CP&L provided amendment 5, additional questions regarding fire protection1 and aging management of pressurizer and steam generator with HNP Unit 1. (ADAMS Accession No. ML073531235)
January 7, 2008	In a letter (signed by M. Heath), the NRC staff issued RAIs associated with the review of the LRA for HNP Unit 1. (ADAMS Accession No. ML073511866)
January 14, 2008	In a letter (signed by M. Heath), the NRC staff issued RAIs associated with the review of the LRA for HNP Unit 1. (ADAMS Accession No. ML080070509)
January 17, 2008	In a letter (signed by T. J. Natale), CP&L provided amendment 6, additional questions regarding aging management review and time-limited again analysis with HNP Unit 1. (ADAMS Accession No. ML080230467)
January 22, 2008	In a letter (signed by T. J. Natale), CP&L provided amendment 6, additional questions regarding aging management review and time-limited again analysis with HNP Unit 1. (ADAMS Accession No. ML080290646)

APPENDIX B: CHRONOLOGY	
Date	**Subject**
February 19, 2008	In a letter (signed by T. J. Natale), CP&L provided License Renewal Application - Revision 4 to the License Renewal Commitments for HNP Unit 1. (ADAMS Accession No. ML080580195)
April 23, 2008	In a letter (signed by C. Burton), CP&L provided License Renewal - Resolution of Open Item and License Renewal Application Amendment 7. (ADAMS Accession No. ML081200755)
May 30, 2008	In a letter (signed by C. Burton), CP&L provided License Renewal - Resolution of Open Item and License Renewal Application Amendment 8. (ADAMS Accession No. ML081570346)
July 21, 2008	Email: Clarification on SBO Recovery Path. (ADAMS No. ML082310661)
October 16, 2008	Letter from William J. Shack, ACRS Chairman, to Honorable Dale E. Klein, NRC Chairman. Report on the Safety Aspects of the License Renewal Application for the Shearon Harris Nuclear Power Plant, Unit 1. (ADAMS No. ML082810713)

APPENDIX C

PRINCIPAL CONTRIBUTORS

This appendix lists the principal contributors for the development of this safety evaluation report (SER) and their areas of responsibility.

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Name	**Responsibility**
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S. Weerakkody	Management Oversight
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D. Wrona	SER Support
Z. Xi	Structural Engineering

APPENDIX D

REFERENCES

This appendix lists the references used throughout this safety evaluation report (SER) for review of the license renewal application (LRA) for Shearon Harris Nuclear Power Plant, Unit 1.

APPENDIX D: REFERENCES
NUREG-1800, Revision 1, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, " September 2005.
NUREG-1801, Revision 1, "Generic Aging Lessons Learned (GALL) Report, " September 2005.
NEI 95-10, Revision 6, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," June 2005.
NRC Bulletin 87-01, "Thinning of Pipe Walls in Nuclear Power Plants, " November 1987.
NRC Generic Letter 89-08 "Erosion/Corrosion-Induced Pip Wall Thinning," May 1999.
NRC Bulletin 04-01, "Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors," May 2004.
NRC Bulletin 03-02, "Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity," August 2003.
NRC Order EA-03-009, "Issuance of Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors," February 2003.
Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," March 1988.

NRC FORM 335 (9-2004) NRCMD 3.7 — U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG-1916, Vol. 2
2. TITLE AND SUBTITLE Safety Evaluation Report Related to the License Renewal of Shearon Harris Nuclear Power Plant, Unit 1	3. DATE REPORT PUBLISHED MONTH: November — YEAR: 2008 4. FIN OR GRANT NUMBER
5. AUTHOR(S) Maurice Heath	6. TYPE OF REPORT Technical 7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of License Renewal
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This safety evaluation report (SER) documents the technical review of the Shearon Harris Nuclear Power Plant (HNP), Unit 1, license renewal application (LRA) by the United States (US) Nuclear Regulatory Commission (NRC) staff (the staff). By letter dated November 14, 2006, Carolina Power & Light (CP&L) Company, doing business as Progress Energy Carolinas, Inc., submitted the LRA in accordance with Title10, Part 54, of the Code of Federal Regulations, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." CP&L requests renewal of the Unit 1 operating license (Facility Operating License Number NPF-63) for a period of 20 years beyond the current expiration at midnight October 24, 2026, for Unit 1. HNP is located approximately 16 miles southwest of Raleigh, NC. The NRC issued the construction permit for Unit 1 on January 27, 1978, and operating license on January 12, 1987. Unit 1 is of a dry ambient pressurized water reactor design. Westinghouse supplied the nuclear steam supply system and Daniel International originally designed and constructed the balance of the plant with the assistance of its agent, Ebasco. The Unit 1 licensed power output is 2900 megawatt thermal with a gross electrical output of approximately 900 megawatt electric.
This SER presents the status of the staff's review of information submitted through July 21, 2008, the cutoff date for consideration in the SER. The staff identified an open item and two confirmatory items that were resolved before the staff made a final determination on the application. SER Sections1.5 and1.6 summarizes these items and their resolution. Section 6.0 provides the staff's final conclusion on the review of the HNP LRA.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	
10 CFR 54, license renewal, Shearon Harris, scoping and screening, aging management, time-limited aging analysis, TLAA, safety evaluation report	13. AVAILABILITY STATEMENT: unlimited 14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 16. PRICE

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