SECTION 4

TIME-LIMITED AGING ANALYSES

4.1 Identification of Time-Limited Aging Analyses

This section discusses the identification of time-limited aging analyses (TLAAs). Nuclear Management Company, LLC (NMC or the applicant), discusses the TLAAs in Sections 4.2 through 4.10 of its license renewal application (LRA). Sections 4.2 through 4.11 of this safety evaluation report (SER) document the review of the TLAAs conducted by the staff of the U.S. Nuclear Regulatory Commission (NRC or 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)), the applicant for license renewal must provide a list of TLAAs, as defined in 10 CFR 54.3(a), and demonstrate that (i) the analyses will remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended functions will be adequately managed for the period of extended operation. In accordance with 10 CFR 54.3(a), TLAAs are those licensee calculations and analyses that meet the following six criteria:

(1) involve systems, structures, and components (SSCs) within the scope of license renewal, as delineated in 10 CFR 54.4(a)

(2) consider the effects of aging

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

(4) were considered to be relevant by the licensee in making a safety determination

(5) involve conclusions or provide the basis for conclusions related to the capability of the SSC to perform its intended functions, as delineated in 10 CFR 54.4(b)

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

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

4.1.1 Summary of Technical Information in the Application

To identify the TLAAs, the applicant evaluated calculations for the Monticello Nuclear Generating Plant (MNGP) against the six criteria specified in 10 CFR 54.3, "Definitions." The applicant indicated that it had identified the calculations that met the six criteria by searching the CLB, which includes the Updated Safety Analysis Report (USAR), engineering calculations, technical reports, engineering work requests, licensing correspondence, and applicable vendor reports. In LRA Table 4.1-1, "List of MNGP Time-Limited Aging Analyses (TLAAs)," the applicant listed the applicable TLAAs in the following categories:

- neutron embrittlement of the reactor vessel and internals
- metal fatigue—RPV, internals and pressure boundary
- neutron embrittlement
- environmental fatigue
- fatigue of primary containment, piping, and components
- environmental qualification
- loss of preload
- plant-specific TLAAs

Pursuant to 10 CFR 54.21(c)(2), the applicant stated that it did not identify any exemptions granted under 10 CFR 50.12 that were based on a TLAA, as defined in 10 CFR 54.3.

4.1.2 Staff Evaluation

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

The applicant provided a list of common TLAAs from NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," dated July 2001. The applicant listed those TLAAs that are applicable to MNGP in LRA Table 4.1-1.

As required by 10 CFR 54.21(c)(2), an applicant must provide a list of all the exemptions granted under 10 CFR 50.12 that are based on a TLAA and evaluated and justified for continuation through the period of extended operation. In its LRA, the applicant stated that it reviewed each active exemption to determine whether the exemption was based on a TLAA. The applicant did not identify any TLAA-based exemptions. On the basis of the information provided by the applicant with regard to the process used to identify TLAA-based exemptions, as well as the results of the applicant's search, the staff concluded that the applicant did not identify any TLAA-based exemptions that are justified for continuation through the period of extended operation, in accordance with 10 CFR 54.21(c)(2).

4.1.3 Conclusion

On the basis of its review, the staff concluded that the applicant provided an acceptable list of TLAAs, as required by 10 CFR 54.21(c)(1). The staff also confirmed that no exemptions under 10 CFR 50.12 have been granted on the basis of a TLAA, as required by 10 CFR 54.21(c)(2).

4.2 Neutron Embrittlement of the Reactor Vessel and Internals

The materials of the reactor pressure vessel (RPV) and internals are subject to embrittlement resulting from high energy (E > 1 million electron volts (MeV)) neutron exposure. Embrittlement means the material has lower toughness (i.e., will absorb less strain energy during a crack or rupture), thus allowing a crack to propagate more easily under thermal and/or pressure loading.

Toughness (indirectly measured in foot-pounds (ft-lb) of absorbed energy in a Charpy impact test) is temperature dependent in ferritic materials. An initial nil-ductility reference temperature (RT_{NDT}), the temperature associated with the transition from ductile to brittle behavior, is determined for vessel materials through a combination of Charpy and drop weight testing. Toughness increases with temperature up to a maximum value called the "upper-shelf energy" (USE). Neutron embrittlement causes an increase in the RT_{NDT} and a decrease in the USE of RPV steels. The increase or shift in the initial nil ductility reference temperature (ΔRT_{NDT}) means higher temperatures are required for the material to continue to act in a ductile manner. To reduce the potential for brittle fracture during RPV operation by accounting for the changes in material toughness as a function of neutron radiation exposure (fluence), operating pressure-temperature (P-T) limit curves are included in plant technical specifications (TSs). The P-T curves account for the decrease in material toughness associated with a given fluence, which is used to predict the loss in toughness of the RPV materials. Based on the projected drop in toughness for a given fluence, the P-T curves are generated to provide a minimum temperature limit associated with the vessel pressure. The P-T curves are determined by the RT_{NDT} and ΔRT_{NDT} values for the licensed operating period, along with appropriate margins.

4.2.1 RPV Materials USE Reduction Due to Neutron Embrittlement

4.2.1.1 Summary of Technical Information in the Application

In LRA Section 4.2.1, the applicant summarized the evaluation of the RPV materials USE reduction from neutron embrittlement for the period of extended operation. USE is the standard industry parameter used to indicate the maximum toughness of a material at high temperature. Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires the predicted end-of-life Charpy impact test USE for RPV materials to be at least 50 ft-lb (absorbed energy), unless an approved analysis supports a lower value. Initial unirradiated test data are available for only one plate heat for the MNGP RPV to demonstrate a minimum 50 ft-lb USE by standard methods. End-of-life fracture energy was evaluated by using an equivalent margin analysis (EMA) methodology approved by the NRC in NEDO-32205-A, Revision 1, "10 CFR 50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2-6 Vessels," February 1994. This analysis confirmed that an adequate margin of safety against fracture, equivalent to the requirements of Appendix G to 10 CFR Part 50, does exist. The end-of-life USE calculations satisfy the criteria of 10 CFR 54.3(a), as described in SER Section 4.1. As such, these calculations are a TLAA.

Fluence was calculated for the MNGP RPV for the extended 60-year (54 effective full-power years (EFPY)) licensed operating period using the methodology of NEDC-32983P, "General Electric (GE) Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation," approved by the NRC in a letter dated September 14, 2001, from S.A. Richards, NRC, to J.F. Klapproth, GE. The NRC found that, in general, this methodology adheres to the guidance in Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001, for neutron flux evaluation. For MNGP, 54 EFPY is equivalent to 3.90×10^8 megawatt hours (MWh) through the end of Cycle 22 at 1775 megawatts thermal (MWt) plus 4.76×10^8 MWh at 1880 MWt. Peak fluence was calculated at the RPV inner surface (inner diameter) to evaluate USE. The value of neutron fluence was also calculated for the 1/4-thickness (1/4T) location into the RPV wall measured radially from the inside diameter (ID), using Equation 3 from Paragraph 1.1 of RG 1.99, Revision 2, "Radiation

Embrittlement of Reactor Vessel Materials." This 1/4T depth is recommended in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, Appendix G, Subarticle G-2120 as the maximum postulated defect depth for calculating P-T curves.

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.

<u>Neutron Fluence Evaluation</u>

LRA Section 4.2.1 indicates that the applicant calculated neutron fluence for the MNGP RPV for the extended 60-year (54 EFPY) licensed operating period based on 3.90×10^{8} MWh through Cycle 22 at 1775 MWt plus 4.76×10^{8} MWh at 1880 MWt. This calculation results in a peak neutron fluence of 5.17×10^{18} neutrons per square centimeter (n/cm^2) (E>1.0 MeV), a peak 1/4T fluence of 3.82×10^{18} n/cm^2 (E>1.0 MeV) for the RPV, and a neutron fluence at the inside of the shroud of 3.84×10^{21} n/cm^2 (E>1.0 MeV) at the end of the extended operating period. Originally, MNGP was licensed for 1670 MWt and uprated to 1775 MWt in October 1998 during fuel cycle 19.

The staff's review of LRA Section 4.2.1 identified areas for which it needed additional information to complete its evaluation of the applicant's neutron fluence evaluation. The applicant responded to the staff's requests for additional information (RAIs) as discussed below.

In RAI 4.2-1, dated September 28, 2005, the staff requested that the applicant provide the basis for the neutron flux estimates in the TLAA.

In its response, by letter dated October 28, 2005, the applicant explained the following:

> Flux estimates for the MNGP were performed in accordance with the General Electric methodology for neutron flux calculation documented in Licensing Topical Report (LTR) NEDC-32983P-A which has been approved by the NRC. In general, this methodology adheres to the guidance in Regulatory Guide 1.190 for neutron flux evaluation. A key input to this calculation was the total integrated power (MWDth) through the first 22 cycles of operation. In addition, Cycle 22 core data was used as a basis for the calculation. Flux profiles were generated from this data and, using the maximum flux, the integrated fluence at 54 EFPY was determined. Fluence estimates at 54 EFPY were conservatively determined using 1775 MWt for Cycles 1 through 22 (previous to rerate implementation in the fall of 1998 the rated power was 1670 MWt) and 1880 MWt for the remainder of the license renewal period of extended operation (54 EFPY). This resulted in EFPYs of 25.09 and 28.91 respectively.
>
> In addition to the conservative methodology described above, a bias adjustment derived from extensive benchmarking of the methodology against measured data as well as an uncertainty related to the flux calculation was incorporated. To account for variations in operation (e.g. capacity factor, core design, etc.), a

multiplier of 1.3 was applied to the reactor pressure vessel to obtain a bounding fluence...

In addition, in a letter dated June 10, 2005, the applicant confirmed that the 54 EFPY used in the TLAA bounds plant-specific operation:

> NMC has determined that the 54 Effective Full Power Years (EFPY) used for Time-Limited Aging Analyses bounds the plant-specific EFPY for MNGP based on a conservative evaluation of plant history and projected capacity factors. This evaluation results in an expectation of less than 49.5 EFPY at the end of the license renewal period of extended operation. Assuming a 100 percent capacity factor over the same operating period also results in a projected less than 54 EFPY for MNGP.

Based on its review, the staff found the applicant's response to RAI 4.2-1 acceptable. Because the applicant projected neutron fluence at the expiration of the extended period of operation by an NRC-approved methodology using conservative inputs, the staff considered the neutron fluence projection adequate for TLAA use for the RPV and shroud. Therefore, the staff's concern described in RAI 4.2-1 is resolved.

USE Evaluation

Appendix G to 10 CFR Part 50, provides the staff's criteria for maintaining acceptable levels of Charpy USE for the RPV beltline materials throughout the licensed lives of operating facilities. The rule requires a minimum 75 ft-lb Charpy USE value for RPV beltline materials in the unirradiated condition and a 50 ft-lb minimum Charpy USE value throughout the life of the facility, unless analysis demonstrates that lower USE values will provide acceptable margins of safety against fracture equivalent to those required by ASME Code Section XI, Appendix G. The rule also requires methods for calculating Charpy USE values to account for the effects of neutron irradiation on those values for the materials and to incorporate any relevant RPV surveillance capsule data reported through a plant's RPV material surveillance program, created pursuant to Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50.

RG 1.99, Revision 2, expands the discussion regarding the calculation of Charpy USE values and describes two methods for calculating Charpy USE values for RPV beltline materials depending on whether a given RPV beltline material is included in the plant's RPV Material Surveillance Program (i.e., 10 CFR Part 50, Appendix H Program). If surveillance data are not available, Charpy USE is determined in accordance with regulatory position 1.2 in RG 1.99, Revision 2. If surveillance data are available, Charpy USE should be determined in accordance with regulatory position 2.2 in RG 1.99, Revision 2. These methods refer to RG 1.99, Revision 2, Figure 2, which indicates that the percentage drop in Charpy USE depends on the amount of copper in the material and the neutron fluence. Since the analyses performed in accordance with Appendix G to 10 CFR Part 50 are based on a flaw with a depth equal to 1/4T, the neutron fluence used in the Charpy USE analysis is at the 1/4T depth location.

By letter dated April 30, 1993, the Boiling Water Reactor Owners Group (BWROG) submitted NEDO-32205-A to demonstrate that boiling-water reactor (BWR) RPVs could meet margins of safety against fracture equivalent to those required by ASME Code Section XI, Appendix G, for

Charpy USE values less than 50 ft-lb. In a letter dated December 8, 1993, the staff concluded that the topical report demonstrated that the evaluated materials have margins of safety against fracture equivalent to ASME Code Section XI, Appendix G, in accordance with Appendix G to 10 CFR Part 50. In that report, the BWROG derived through statistical analysis the unirradiated Charpy USE values for materials that originally had no documented unirradiated Charpy USE values. Using these statistically derived Charpy USE values, the BWROG predicted the Charpy USE values through 40 years of operation, in accordance with RG 1.99, Revision 2. The BWROG analysis determined that the minimum allowable Charpy USE value in the transverse direction for base metal and along the weld for weld material was 35 ft-lb.

Electric Power Research Institute (EPRI) Topical Report (TR)-113596, "BWR Vessel and Internals Project (BWRVIP) BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines," BWRVIP-74, issued September 1999, documents the GE updated Charpy USE evaluation. An October 18, 2001, letter from Mr. C.I. Grimes to Mr. C. Terry documented staff review and approval of EPRI TR-113596. The analysis in EPRI TR-113596 used the methodology in RG 1.99, Revision 2, to determine the reduction in the unirradiated Charpy USE from neutron irradiation. Using this methodology and a correction factor of 65 percent for conversion of the longitudinal properties to transverse properties, the lowest Charpy USE at 54 EFPY for all BWR/3-6 plates was projected to be 45 ft-lb. The correction factor for specimen orientation in plates is based on NRC Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements." EMA acceptance criteria specified in the staff-approved report BWRVIP-74 using the methodology in RG 1.99, Revision 2, are based on the percent reduction in the unirradiated Charpy USE values from neutron radiation. The acceptance criteria specified in the BWRVIP-74 report indicate that the maximum allowable percent reduction in USE value is 23.5 percent for the plates and 39 percent for the welds.

In RAI 4.2-2, dated September 28, 2005, the staff noted that because the analysis in BWRVIP-74 is generic, the applicant submitted plant-specific information in LRA Tables 4.2.1-1 and 4.2.1-2 for the limiting MNGP plates and welds to demonstrate that the RPV limiting beltline materials meet the criteria in the BWRVIP-74 report for the end of the license renewal period. These tables do not include an evaluation of surveillance plate and weld data. Surveillance data were submitted to the NRC in a letter dated December 21, 1998, containing Report SIR-97-003, Revision 2, "Review of the Results of Two Surveillance Capsules, and Recommendations for the Materials Properties and P-T Curves to be Used for the Monticello Reactor Pressure Vessel," which indicates that unirradiated Charpy USE data were available for surveillance plates, but not for surveillance welds. Therefore, Charpy USE evaluations using surveillance data could be performed for the plates but not the welds. The staff requested that the applicant determine the impact of the surveillance plate data on the limiting beltline plate USE and evaluate what impact, if any, these data have on the validity of the plate EMA.

In its response, by letter dated October 28, 2005, the applicant stated the following:

> Using the '1st Capsule' data for plate C2220-2 identified in Table 2-1 of SIR-97-003, Revision 2 results in a measured decrease of 18.3 percent as opposed to an 11.5 percent predicted decrease using RG 1.99 Figure 2 as noted in LRA Table 4.2.1-1 at a fluence of 2.93×10^{17} n/cm^2. Correspondingly, at the 54 EFPY 1/4T fluence of 3.82×10^{18} with an 18.3 percent measured decrease the RG1.99 Position 2.2 adjusted decrease is 33.5 percent which exceeds the margin to safety requirement of 23.5 percent defined in BWRVIP-74-A.

Based on its review, the staff found the applicant's response to RAI 4.2-2 acceptable; however, in this response, the applicant demonstrated that the adjusted percent reduction obtained using RG 1.99, position 2.2, also results in a 54 EFPY USE greater than 50 ft-lb for plate C2220-2. As described above, using data from SIR-97-003 results in a position 2.2 Charpy USE reduction of 33.5 percent at the expiration of the extended period of operation. With a transverse unirradiated USE of 86.5 ft-lb (0.65 x 133 ft-lbs), a 33.5 percent reduction results in a 54 EFPY Charpy USE of 57.5 ft-lb, which exceeds the 50 ft-lb minimum identified in Appendix G to 10 CFR Part 50. Because the projected USE exceeds the minimum recommended by Appendix G to 10 CFR Part 50, the staff found the applicant's response acceptable. Therefore, the staff's concern described in RAI 4.2-2 is resolved.

There are four plates in the MNGP RPV beltline, one with USE greater than 50 ft-lbs, as discussed in the previous paragraph. The other three plates have no surveillance data; however, the applicant has used the RG 1.99, Revision 2, methodology to demonstrate that these three plates will have less than a 23.5 percent reduction in Charpy USE value at the expiration of the extended period of operation. Therefore, these plates satisfy BWRVIP-74-A criteria and the margins of safety against fracture equivalent to ASME Code Section XI, Appendix G, in accordance with Appendix G to 10 CFR Part 50.

In RAI 4.2-3, dated September 28, 2005, the staff noted that the weld materials used in the MNGP RPV beltline were fabricated using the shielded metal arc weld (SMAW) process. Such welds are low in copper because the weld electrodes used in this process are not copper coated; therefore, the staff requested that the applicant calculate the projected Charpy USE for the limiting weld and plate in the reactor vessel beltline at the 1/4T depth using the neutron fluence at the end of the period of extended operation.

In its response, by letter dated October 28, 2005, the applicant analyzed the impact of neutron radiation on the RPV beltline welds. In this analysis, the weld material was projected to have a Charpy USE at the expiration of the extended period of operation of 68 ft-lb. The applicant utilized an unirradiated Charpy USE of 84.5 ft-lb, which is the lower 95/95 confidence value for the SMAW database reported in BWRVIP-74-A. The drop in Charpy USE was calculated by the RG 1.99, Revision 2, methodology and a 0.10 percent copper. As the RPV beltline welds are projected to have a Charpy USE at the expiration of the extended period of operation greater than 50 ft-lb, the RPV beltline weld material meets the criteria of Appendix G to 10 CFR Part 50 criteria at that point.

LRA Table 4.2.2-1 indicates that N2 nozzles are within the beltline of the RPV. The MNGP N2 nozzles were fabricated as forgings. In a letter dated February 27, 2006, the applicant provided additional data to demonstrate that, at the end of the period of extended operation, the N2 nozzles will have Charpy USE values greater than 50 ft-lb. The applicant indicated:

> Given the hot working normally associated with the fabrication of forgings (resulting in a more refined grain structure), it is expected that the fracture toughness properties of the A 508 Class 2 forging materials would be equivalent, if not better than, the corresponding A 533 Grade B plate materials typically used to fabricate beltline shell courses. 508 Class 2 forging materials (or equivalent) have been used throughout the industry for fabrication of reactor vessel components, including the MNGP recirculation inlet (N2) nozzles, and as such, a

significant amount of data has been reported on the fracture toughness of these materials.

The applicant performed a study using the NRC Reactor Vessel Integrity Database, Revision 2 (RVID2), to determine a generic Charpy USE for A 508 Class 2 forgings. The study indicates that the mean of the USE data for the forgings is 108 ft-lb, with a minimum observed USE of 70 ft-lb and a standard deviation of 24 ft-lb. As defined in NUREG-1475, "Applying Statistics," for 95/95 confidence with a data set consisting of 67 data points, the κ value is 1.9996. This results in a Mean–$\kappa\sigma$ of 60 ft-lb. Using the RG 1.99 methodology for determining the impact of neutron radiation on Charpy USE, the applicant determined that at the expiration of the extended license the Charpy USE will be 52 ft-lb. The staff has confirmed this value.

The applicant compared the generic Charpy USE data from forgings with the generic Charpy USE from plate material. The mean equivalent transverse Charpy USE was reported as 82.5 ft-lb for plate material in BWRVIP-74-A. The minimum observed Charpy USE was 59 ft-lb and the Mean–$\kappa\sigma$ was 64.5 ft-lb for the plate material.

The applicant also evaluated the RVID2 database surveillance capsule results for forging materials with respect to plate materials. These results indicate that application of the RG 1.99 prediction to forgings adequately predicts the irradiated behavior of these materials.

The applicant concluded the following:

> Therefore, it has been demonstrated that the forging materials meet or exceed the requirements for plate materials, and that the MNGP N2 nozzle case is bounded by the [equivalent margins analysis] EMA plate requirements described in BWRVIP-74-A. Further, it has been demonstrated that, in general, irradiated forging materials behave in a manner consistent with the predictions of RG 1.99. Based on the results of this evaluation, the USE of the N2 nozzle forgings will be adequate for the period of extended operation.

The staff concluded that the analysis provided for the MNGP N2 nozzles demonstrates that the nozzles will have Charpy USE greater than 50 ft-lb and will meet the requirements of Appendix G to 10 CFR Part 50 at the expiration of the extended license. Therefore, the staff's concern described in RAI 4.2-3 is resolved.

Table 4.2.1-1 of this SER summarizes the staff's review of the calculated USE values.

Table 4.2.1-1 Reactor Vessel Upper-Shelf Energy Analysis Summary

RV Beltline Component	Acceptance Criterion for USE	Component Value for 54 EFPY
C2220-2 Limiting Plate	> 50 ft-lb	57.5 ft-lb
Welds—shielded metal arc	> 50 ft-lb	68 ft-lb
N2 Nozzle—forging	> 50 ft-lb	52 ft-lb

4.2.1.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA evaluation of RPV materials USE reduction from neutron embrittlement in LRA Section A3.1. On the basis of its review of the USAR supplement, the staff concluded that the summary description of the applicant's actions to address the RPV materials USE reduction from neutron embrittlement is adequate.

4.2.1.4 Conclusion

The staff concluded that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses of the RPV materials Charpy USE reduction from neutron embrittlement have been projected to the end of the period of extended operation. The staff also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation, sufficient to satisfy the requirements of 10 CFR 54.21(d).

4.2.2 Adjusted Reference Temperature for RPV Materials Due to Neutron Embrittlement

4.2.2.1 Summary of Technical Information in the Application

In LRA Section 4.2.2, the applicant summarized the evaluation of the adjusted reference temperature (ART) for RPV materials from neutron embrittlement for the period of extended operation. The initial RT_{NDT} is the temperature at which a nonirradiated metal (ferritic steel) changes in fracture characteristics, going from ductile to brittle behavior. The applicant evaluated the RT_{NDT} according to the procedures in ASME Code, Paragraph NB-2331. Neutron embrittlement raises the initial RT_{NDT}. Appendix G to 10 CFR Part 50 defines the fracture toughness requirements for the life of the vessel. The ΔRT_{NDT} is evaluated as the difference in the 30 ft-lb index temperatures from the average Charpy curves measured before and after irradiation. This increase (ΔRT_{NDT}) means that higher temperatures are required for the material to continue to act in a ductile manner. The ART is defined as RT_{NDT} + ΔRT_{NDT} + margin. The margin is defined in RG 1.99. The P-T curves are developed from the ART for the RPV materials. These are determined by the unirradiated RT_{NDT} and by the ΔRT_{NDT} calculations for the licensed operating period. RG 1.99 defines the calculation methods for ΔRT_{NDT}, ART, and end-of-life USE. The ΔRT_{NDT} and ART calculations meet the criteria of 10 CFR 54.3(a). As such, they are TLAAs.

4.2.2.2 Staff Evaluation

The staff reviewed LRA Section 4.2.2, 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.

LRA Table 4.2.2-1 provides the ART values for all beltline materials at the expiration of the extended operating period. The materials with the highest ART values are the C2220-1 and C2220-2 plates, which have 0.17 percent copper and 0.65 percent nickel. Using the RG 1.99, Revision 2, methodology and a neutron fluence of 3.82×10^{18} n/cm^2 (E>1 MeV) at the 1/4T location, the ART for these plates is 157 °F at the expiration of the extended operating period.

The weld material, which has 0.10 percent copper and 0.99 percent nickel, has an ART of 97 °F at the expiration of the extended operating period.

The N2 nozzles have an ART of 117 °F. The certified material test report includes nickel content and initial RT_{NDT} data, but not copper content data. The copper value (0.18 percent) in the analysis is generic, derived from data from nine nozzles in other BWR beltline nozzles. The value in the analysis is the mean plus one standard deviation value and is acceptable to the staff because it is consistent with the RG 1.99, Revision 2, criteria when copper is not reported for the material.

The copper and nickel values for the plates and weld material are consistent with those reported in RVID2. The staff confirmed the applicant's projected values of ART. These ART values are used in the P-T limits evaluation. P-T limits in the MNGP TSs are periodically updated (discussed in SER Section 4.2.5).

4.2.2.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA ART evaluation for RPV materials from neutron embrittlement in LRA Section A3.1. On the basis of its review of the USAR supplement, the staff concluded that the summary description of the applicant's actions to address the ART for RPV materials from neutron embrittlement is adequate.

4.2.2.4 Conclusion

The staff concluded that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses of the ART for RPV materials from neutron embrittlement have been projected to the end of the period of extended operation. The staff also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation, sufficient to satisfy the requirements of 10 CFR 54.21(d).

4.2.3 Reflood Thermal Shock Analysis of the RPV

4.2.3.1 Summary of Technical Information in the Application

In LRA Section 4.2.3, the applicant summarized the evaluation of the reflood thermal shock analysis of the RPV for the period of extended operation. The MNGP USAR includes an end-of-life thermal shock analysis performed on the RPV for a design-basis loss-of-coolant accident (LOCA) followed by a low-pressure coolant injection (LPCI). The effects of neutron embrittlement assumed by this thermal shock analysis will change with an increase in the licensed operating period. This analysis satisfies the criteria of 10 CFR 54.3(a). As such, this analysis is a TLAA.

4.2.3.2 Staff Evaluation

The staff reviewed LRA Section 4.2.3, 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 peak fluence at the RPV wall is 5.17×10^{18} n/cm^2 (E>1.0 MeV) for 54 EFPY of operation. Based on this fluence value, the previous reflood thermal shock analysis of the RPV is not bounding for the period of extended operation. The original analysis has been superseded by an analysis for BWR-6 RPVs that is applicable to the MNGP BWR-3 RPV.

The BWR-6 RPV analysis applies to MNGP because it uses a bounding main steamline break event and an RPV thickness similar to that of the MNGP RPV. This analysis assumes end-of-license material toughness, which in turn depends on the end-of-license ART. The critical location for the fracture mechanics analysis is at 1/4T RPV thickness. For the main steamline break event, the peak stress intensity occurs approximately 300 seconds after initiation of the event. The analysis shows that at that point in the thermal shock event, the temperature of the vessel wall at 1.5 inches deep (1/4T depth for the BWR-6 RPV) is approximately 400 °F. For the MNGP vessel, the 1/4T depth is 1.26 inches.

The staff's review of LRA Section 4.2.3 identified an area for which it needed additional information to complete its evaluation of the applicant's neutron fluence evaluation. The applicant responded to the staff's RAI as discussed below.

In RAI 4.2-4, dated September 28, 2005, the staff requested that the applicant provide the *fracture toughness (peak stress intensity value) required to prevent fracture of the RPV* resulting from reflood thermal shock.

In its response, dated October 28, 2005, the applicant identified the maximum applied stress intensity for the thermal shock event as 103 kilopounds per square inch times the square root of inches (ksi-in$^{1/2}$). Fracture toughness at approximately 300 seconds after initiation of the event was estimated to be 200 ksi-in$^{1/2}$. In its response, the applicant stated the following:

> Paper G1/5, 'Fracture Mechanics Evaluation of a Boiling Water Reactor Vessel Following a Postulated Loss of Coolant Accident,' Ranganath, S., Fifth International Conference on Structural Mechanics in Reactor Technology, Berlin, Germany, August 1979, defines the basis for this evaluation. As noted in the MNGP LRA submittal, the BWR/6 example in the paper referenced above bounds the conditions at the MNGP. This was demonstrated in the submittal by comparison of the parameters for the BWR/6 case versus the plant-specific MNGP case. As shown in the submittal, the plant-specific temperature at 1/4T depth into the vessel wall was determined to be 370 °F at 300 seconds into the thermal shock event. It was also stated that using the highest 60 year Adjusted Reference Temperature (ART), the beltline material reaches upper shelf (200 ksi-in$^{1/2}$) at 261°F. Since this temperature is significantly lower than 370°F, it is assured that the beltline material remains at upper shelf at 300 seconds into the thermal shock event. Figure 5 of 'Fracture Mechanics Evaluation of a Boiling Water Reactor Vessel Following a Postulated Loss of Coolant Accident,' Ranganath, S., Fifth International Conference on Structural Mechanics in Reactor Technology, Berlin, Germany, August 1979, Paper GI/5, further demonstrates that at 300 seconds into the thermal shock event and at 1/4T depth into the vessel wall, the maximum applied stress intensity is 103 ksi-in$^{1/2}$. Therefore, there is sufficient margin to prevent fracture due to reflood thermal shock.

On the basis of its review, the staff found the applicant's response to RAI 4.2-4 acceptable because the applicant demonstrated that the beltline materials will have adequate fracture toughness (applied stress intensity is less than upper-shelf fracture toughness) at 300 seconds into the event through the period of extended operation. The revised analysis demonstrates that the reflood thermal shock analysis of the RPV applies for the extended period of operation. Therefore, the staff's concern described in RAI 4.2-4 is resolved.

4.2.3.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA evaluation of reflood thermal shock analysis of the RPV in LRA Section A3.1. On the basis of its review of the USAR supplement, the staff concluded that the summary description of the applicant's actions to address the reflood thermal shock analysis of the RPV is adequate.

4.2.3.4 Conclusion

The staff concluded that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses of the reflood thermal shock analysis of the RPV have been projected to the end of the period of extended operation. The staff also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation, sufficient to satisfy the requirements of 10 CFR 54.21(d).

4.2.4 Reflood Thermal Shock Analysis of the RPV Core Shroud

4.2.4.1 Summary of Technical Information in the Application

In LRA Section 4.2.4, the applicant summarized the evaluation of the reflood thermal shock analysis of the RPV core shroud for the period of extended operation. Radiation embrittlement may affect the ability of RPV internals, particularly the core shroud, to withstand an LPCI thermal shock transient. The analysis of core shroud strain from reflood thermal shock is a TLAA because it is part of the CLB, supports a safety determination, and is based on the calculated lifetime neutron fluence.

4.2.4.2 Staff Evaluation

The staff reviewed LRA Section 4.2.4, 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.

Before license renewal, the RPV core shroud was evaluated for an LPCI reflood thermal shock transient considering embrittlement effects of a 40-year radiation exposure (32 EFPY). The core shroud receives the maximum irradiation on the inside surface opposite the midpoint of the fuel centerline. The total integrated neutron flux at the end of 40 years of operation was $2.7\text{x}10^{20}$ n/cm^2 (greater than 1 MeV). The maximum thermal shock stress in this region is 155,700 pounds per square inch (psi), equivalent to 0.57-percent strain. This strain range of 0.57 percent was calculated at the midpoint of the shroud, the zone of highest neutron irradiation.

However, using the approved fluence methodology discussed in SER Section 4.2.1.2, the applicant revised the analysis for the period of extended operation by calculating the 54 EFPY fluence at the most irradiated point on the core shroud to be 3.84×10^{21} n/cm^2. The applicant indicated that the measured value of percent elongation for stainless steel weld metal is 4 percent for a temperature of 297 °C (567 °F) with a neutron fluence of 8×10^{21} n/cm^2 (greater than 1 MeV), while the average value for base metal at 290 °C (554 °F) is 20 percent. The calculated strain range of 0.57 percent represents a considerable margin of safety relative to measured values of percent elongation for annealed Type 304 stainless steel irradiated to 8×10^{21} n/cm^2 (greater than 1 MeV). Because the measured value of elongation bounds the calculated thermal shock strain amplitude of 0.57 percent, the calculated thermal shock strain at the most irradiated location is acceptable, considering the embrittlement effects for a 60-year operating period.

The revised analysis demonstrates that the reflood thermal shock analysis of the RPV core shroud applies for the extended period of operation and satisfies 10 CFR 54.21(c)(1)(ii) because the applicant provided additional data to justify operation to a higher neutron fluence to the end of the period of extended operation.

4.2.4.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA evaluation of reflood thermal shock analysis of the RPV core shroud in LRA Section A3.1. On the basis of its review of the USAR supplement, the staff concluded that the summary description of the applicant's actions to address the reflood thermal shock analysis of the RPV core shroud is adequate.

4.2.4.4 Conclusion

The staff concluded that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that the reflood thermal shock analyses of the RPV core shroud have been projected to the end of the period of extended operation. The staff also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation, sufficient to satisfy the requirements of 10 CFR 54.21(d).

4.2.5 RPV Thermal Limit Analysis: Operating Pressure—Temperature Limits

4.2.5.1 Summary of Technical Information in the Application

In LRA Section 4.2.5, the applicant summarized the evaluation of the RPV thermal limit analysis: operating P-T limits for the period of extended operation. The RPV thermal limit analysis provides operating P-T limits for the period of extended operation and is dependent on the ART. The 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 metal temperature at which an RPV is allowed to be pressurized increases. The ART of the limiting beltline material is used to correct the beltline P-T limits to account for irradiation effects. Appendix G to 10 CFR Part 50 requires RPV thermal limit analyses to determine operating P-T limits for boltup, hydrotest, pressure tests, and normal operating and anticipated operational occurrences. Operating limits for pressure and temperature are required for three categories of operation—(1) hydrostatic pressure tests and leak tests, referred to as Curve A, (2) nonnuclear heatup/cooldown and low-level physics tests, referred to as Curve B, and (3) core-critical operation, referred to as Curve C. P-T limits are developed for three vessel regions, the upper vessel region, the core beltline region, and the lower vessel bottom head region. The calculations associated with generation of the P-T curves satisfy the criteria of 10 CFR 54.3(a). As such, this topic is a TLAA.

4.2.5.2 Staff Evaluation

The staff reviewed LRA Section 4.2.5, 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 MNGP TSs include P-T limit curves for core-critical operation, nonnuclear heatup/cooldown, inservice leakage, and hydrostatic testing. They also limit the maximum rate of change of reactor coolant temperature. The criticality curves provide limits for both heatup and criticality calculated for a 32-EFPY operating period. The current TSs contain P-T curves developed using the 1989 edition of the ASME Boiler and Pressure Vessel Code, incorporating the effects of the 1998 power uprate, and ASME Code Case N–640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves Section XI, Division 1."

P-T limit curves in the MNGP TS are updated periodically, most recently in a February 24, 2003, NRC letter. The staff's February 24, 2003, safety evaluation (SE) indicates that the staff performed an independent assessment of the proposed curves. The assessment concluded that the irradiated P-T limit curves for 32 EFPY generated at the plant will be at least as conservative as those that will be generated with ASME Code Section XI, Appendix G, criteria and methods, as modified by ASME Code Case N–640, and the limit curves met the minimum temperature requirements in Table 1 of Appendix G to 10 CFR Part 50. The assessment was performed for P-T limit curves in which the 1/4T ART value was 157 °F. Because SER Section 4.2.2.2 indicates a 1/4T ART value of 157 °F at the expiration of the extended operating period, the TS P-T limit curves apply to the end of the period of extended operation. This conclusion will be reevaluated when surveillance data for the RPV are withdrawn and tested as part of the BWRVIP Integrated Surveillance Program.

4.2.5.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA evaluation of RPV thermal limit analysis—operating P-T limits in LRA Section A3.1. On the basis of its review of the USAR supplement, the staff concluded that the summary description of the applicant's actions to address the RPV thermal limit analysis—operating P-T limits is adequate.

4.2.5.4 Conclusion

The staff concluded that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that the RPV thermal limit analysis—operating P-T limits analyses have been projected to the end of the period of extended operation. The staff also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation, sufficient to satisfy the requirements of 10 CFR 54.21(d).

4.2.6 RPV Circumferential Weld Examination Relief

4.2.6.1 Summary of Technical Information in the Application

In LRA Section 4.2.6, the applicant summarized the evaluation of the RPV circumferential weld examination relief for the period of extended operation. Relief from RPV circumferential weld examination requirements under Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Inspection," is based on probabilistic assessments that predict an acceptable probability of failure per reactor operating year. The analysis is based on RPV metallurgical conditions, as well as flaw indication sizes and frequencies of occurrence that are expected at the end of a licensed operating period. MNGP has received this relief for the remaining 40-year licensed operating period. The circumferential weld examination relief analysis meets the requirements of 10 CFR 54.3(a). As such, it is a TLAA.

4.2.6.2 Staff Evaluation

The staff reviewed LRA Section 4.2.6, 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 technical basis for relief is discussed in the staff's final SER concerning the BWRVIP-05 report, "BWR Vessel and Internals Project (BWRVIP), BWR Reactor Pressure Vessel Weld Inspection Requirements," enclosed in the letter dated July 28, 1998, from Mr. G.C. Laines, NRC, to Mr. C. Terry, the BWRVIP Chairman. In this letter, the staff concluded that, because the failure frequency for circumferential welds in BWR plants is significantly below the criterion specified in RG 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," and below the core damage frequency of any BWR plant, continued inspection of the RPV circumferential welds will result in a negligible decrease in an already acceptably low rate of RPV failure; therefore, elimination of the inservice inspection (ISI) for RPV circumferential welds is justified. The staff's letter indicated that BWR applicants may request relief from 10 CFR 50.55a(g) ISI requirements for volumetric examination of circumferential RPV welds by demonstrating that (1) through the expiration of the license period, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds in the NRC staff's July 28, 1998, evaluation and (2) implementation of operator training and established procedures that limit the frequency of cold overpressure events to the frequency specified in the staff's SER. The letter indicated that the requirements for inspection of circumferential RPV welds during an additional 20-year license renewal period will be reassessed, on a plant-specific basis, as part of any BWR LRA; therefore, the applicant must request relief from inspection of circumferential welds during the license renewal period, pursuant to 10 CFR 50.55a.

Section A.4.5 of the BWRVIP-74 report indicates that the staff's SER of the BWRVIP-05 report conservatively evaluated the BWR RPVs to 64 EFPY, which is 10 EFPY greater than realistically expected for the end of the license renewal period. In the July 28, 1998, SER, the staff used the mean RT_{NDT} value for materials to evaluate failure probability of BWR circumferential welds at 32 and 64 EFPY. The neutron fluence at the clad-weld (inner) interface was used for this evaluation.

Since the staff analysis discussed in the BWRVIP-74 report is generic, the applicant submitted plant-specific information to demonstrate that the MNGP RPV beltline materials meet the criteria specified in the report. To demonstrate that the MNGP RPV has not become embrittled beyond the basis for the relief, the applicant, in LRA Table 4.2.6.1, compared 54 EFPY material data for the limiting MNGP circumferential weld with that of the 64 EFPY reference case in Appendix E to the staff's SER on the BWRVIP-05 report. The MNGP material data included amounts of copper and nickel, chemistry factor, the neutron fluence, ΔRT_{NDT}, initial RT_{NDT}, and mean RT_{NDT} of the limiting circumferential weld at the end of the renewal period. The staff has verified the data for the copper and nickel contents and the initial RT_{NDT} values for the MNGP circumferential beltline weld material by comparing them with the corresponding data in RVID. The 54 EFPY mean RT_{NDT} value for the MNGP circumferential beltline weld is 47.4 °F. The staff checked the applicant's calculations for the 54 EFPY mean RT_{NDT} values for the limiting MNGP circumferential welds using the data presented in LRA Table 4.2.6.1 and found them to be accurate. This 54 EFPY mean RT_{NDT} value for MNGP is bounded by the 64 EFPY mean RT_{NDT} value of 70.6 °F used by the NRC to determine conditional failure probability of a circumferential weld in a Chicago Bridge and Iron (CB&I) fabricated RPV. The 64 EFPY mean RT_{NDT} value from the staff SER dated July 28, 1998, is for a CB&I weld because CB&I welded the circumferential welds in the RPV. Because the 54 EFPY mean RT_{NDT} value is less than the 64 EFPY value from the staff SER dated July 28, 1998, the staff concluded that the NRC analysis bounds the MNGP RPV conditional failure probability.

The applicant stated that the procedures and training used to limit cold overpressure events will be the same as those approved by the NRC when MNGP requested relief for the current license period. A request for relief during the period of extended operation will be submitted to the NRC before the period of extended operation.

SER Table 4.2.6-1 summarizes the results of the staff's evaluation regarding the RPV circumferential weld examination relief.

Table 4.2.6-1 Effects of Irradiation on RPV Circumferential Weld Properties for MNGP

Value	CB&I 64 EFPY	MNGP 54 EFPY
Cu (%)	0.10	0.10
Ni (%)	0.99	0.99
CF	134.9	138.5
Fluence x 10^{19} (n/cm^2)	1.02	0.52
DRT_{NDT} (°F)	135.6	113
RT_{NDT} (°F)	-65	-65.6
Mean RT_{NDT} (°F)	70.6	47.4
Probability of a failure event (NRC)	1.78×10^{-5}	Note 1

Note 1. If the plant-specific mean ΔRT_{NDT} is less than the mean ΔRT_{NDT} associated with the limiting case study, the staff concludes that the probability of failure for the plant-specific circumferential weld under review will be less than the conditional probability of failure value for the limiting circumferential weld in the limiting case study.

4.2.6.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA evaluation of RPV circumferential weld examination relief in LRA Section A3.1. On the basis of its review of the USAR supplement, the staff concluded that the summary description of the applicant's actions to address the RPV circumferential weld examination relief is adequate.

4.2.6.4 Conclusion

The staff concluded that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses of the RPV circumferential weld examination relief have been projected to the end of the period of extended operation. The staff also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation, sufficient to satisfy the requirements of 10 CFR 54.21(d).

4.2.7 RPV Axial Weld Failure Probability

4.2.7.1 Summary of Technical Information in the Application

In LRA Section 4.2.7, the applicant summarized the evaluation of the RPV axial weld failure probability for the period of extended operation. The BWRVIP recommendations for inspection of RPV shell welds contain generic analyses supporting an NRC SER conclusion that the generic-plant axial weld failure rate is no more than $5x10^{-6}$ per reactor year. BWRVIP-05 showed that this axial weld failure rate of $5x10^{-6}$ per reactor year is orders of magnitude greater than the 40-year end-of-life circumferential weld failure probability and this analysis justified relief from inspection of the circumferential welds, as described in Section 4.2.6. MNGP received relief from the circumferential weld inspections for the remaining 40-year licensed operating period. The axial weld failure probability analysis meets the requirements of 10 CFR 54.3(a). As such, it is a TLAA.

4.2.7.2 Staff Evaluation

The staff reviewed LRA Section 4.2.7, 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.

In its July 28, 1998, letter to Mr. C. Terry, the BWRVIP Chairman, the staff identified a concern about the failure frequency of axially oriented welds in BWR RPVs. In response to this concern, the BWRVIP supplied evaluations of axial weld failure frequency in letters dated December 15, 1998, and November 12, 1999. The staff's SER on these analyses is enclosed in a March 7, 2000, letter from Mr. J. Strosnider (NRC) to Mr. C. Terry, BWRVIP Chairman. The staff performed a generic analysis using Pilgrim as a model for BWR RPVs. The staff analysis identified as Mod 2 that the vessel failure frequency will be 5.02×10^{-6} at a mean RT_{NDT} at the vessel inside surface of 114°F.

LRA Table 4.2.7-1 compared 54 EFPY material data for the limiting RPV axial weld with that of Mod 2 from the staff's SE in the March 7, 2000, letter. The MNGP material data included copper and nickel amounts, chemistry factor, neutron fluence, ΔRT_{NDT}, initial RT_{NDT}, and mean RT_{NDT} of the limiting axial weld at the end of the renewal period. The applicant calculated, and the staff confirmed, that the limiting axial weld mean RT_{NDT} at the inside surface at the expiration of the extended operating period is 47.4 °F. Because the mean RT_{NDT} at the vessel inside surface for the limiting axial weld is less than the value in the staff's Mod 2 analysis, the failure frequencies for the MNGP RPV will be less than 5 x 10^{-6} per reactor year of operation at the end of the period of extended operation; therefore, this analysis is acceptable.

SER Table 4.2.7-1 summarizes the results of the staff's evaluation regarding the RPV axial weld failure probability.

Table 4.2.7-1 Effects of Irradiation on RPV Axial Weld Properties for MNGP

Value	Mod 2	MNGP 54 EFPY
Cu (%)	0.219	0.10
Ni (%)	0.996	0.99
CF		138.5
Fluence x 10^{19} (n/cm^2)	0.148	0.52
DRT_{NDT} (°F)	116	113
RT_{NDT} (°F)	-2	-65.6
Mean RT_{NDT} (°F)	114	47.4
Probability of a failure event (NRC)	5.02x10^{-6}	Note 1

Note 1. If the plant-specific mean ΔRT_{NDT} is less than the mean ΔRT_{NDT} associated with the limiting case study, the staff concluded that probability of failure for the plant-specific axial weld under review will be less than the conditional probability of failure value for the limiting axial weld in the limiting case study.

4.2.7.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA evaluation of RPV axial weld failure probability in LRA Section A3.1. On the basis of its review of the USAR supplement, the staff concluded that the summary description of the applicant's actions to address the RPV axial weld failure probability is adequate.

4.2.7.4 Conclusion

The staff concluded that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that the RPV axial weld failure probability analyses have been projected to the end of the period of extended operation. The staff also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation, sufficient to satisfy the requirements of 10 CFR 54.21(d).

4.3 <u>Metal Fatigue of the RPV and Internals, and Reactor Coolant Pressure Boundary Piping and Components</u>

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

The specific criterion for fatigue analysis of ASME Code Section III components involves calculating the cumulative usage factor (CUF). The fatigue damage in the component caused by each thermal or pressure transient depends on the magnitude of the stresses caused by the transient. The CUF sums the fatigue damage from each transient. The ASME Code Section III criterion requires that the CUF not exceed 1.0.

4.3.1 RPV Fatigue Analyses

4.3.1.1 Summary of Technical Information in the Application

In LRA Section 4.3.1, the applicant discussed that the RPV was designed to ASME Code, Section III. RPV fatigue analyses were performed for the vessel support skirt, shell, upper and lower heads, closure flanges, nozzles and penetrations, nozzle safe ends, and closure studs. The end of the 40-year license fatigue usage was determined for the normal and upset pressure and thermal cycle events. After the original stress analyses, several hardware changes, operational changes (e.g., the 1998 power rerate), and/or stress analysis revisions have affected usage factors. Calculation of fatigue usage factors is part of the CLB and used to support safety determinations. The RPV fatigue analyses are TLAAs.

The applicant stated that the 1998 MNGP power rerate included a reanalysis of the RPV. LRA Table 4.3.1-1 lists the limiting design CUFs for the RPV components. The applicant stated that the fatigue usage factors in Table 4.3.1-1 were determined using the actual transient cycles from its Fatigue Monitoring Program (FMP). On the basis of the actual transient accumulation rate, the applicant concluded that fatigue usage of the RPV components is not expected to exceed the allowable limit of 1.0 during the period of extended operation. The applicant also stated that the Fatigue Monitoring Program will monitor transients contributing to fatigue usage, as described in Appendix B to the LRA.

4.3.1.2 Staff Evaluation

The staff reviewed LRA Section 4.3.1, pursuant to 10 CFR 54.21(c)(1)(iii), to verify that the effects of aging on the intended functions will be adequately managed for the period of extended operation, and, 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 RPV components were analyzed using the ASME Code fatigue requirements. LRA Table 4.3.1-1 lists the design transients for the fatigue analysis of the RPV components. USAR Table 4.2-1 lists the design transients for the RPV fatigue analysis. The staff confirmed that the transients in LRA Table 4.3.1-1 are the same as those in USAR Table 4.2-1.

The staff's review of LRA Section 4.3.1 identified an area for which it needed additional information to complete its evaluation of the RPV fatigue analysis. The applicant responded to the staff's RAI as discussed below.

Table 4.3.1-1 also provides the estimated 60-year fatigue usage factors for the RPV components, which are all less than the ASME Code Section III allowable limit of 1.0. The applicant indicated that these usage factors include the results of a reanalysis of RPV components performed as part of the 1998 power rerate. The applicant also indicated that the fatigue usage factors were determined from the FMP.

In RAI 4.3.1-1, dated June 21, 2005, the staff requested that the applicant describe how it had calculated the revised fatigue usage factors.

In its response, dated July 21, 2005, the applicant stated that GE Document SASR 89-77, "Accumulated Fatigue Usage for the Monticello Nuclear Generating Station Reactor Pressure," tabulates thermal transient (TT) cycles experienced by MNGP through July 1989. As discussed in SASR 89-77, the number of transient cycles through July 1989 was determined from a review of operator log books and plant records. The applicant stated that its FMP updates the number of TT cycles once per refueling cycle. The applicant used these updated cycles to compute the fatigue usage factors in LRA Table 4.3.1-1.

Based on its review, the staff found the applicant's response to RAI 4.3.1-1 acceptable because use of the actual TT cycles to estimate the fatigue usage factors for the period of extended operation is reasonable; therefore, the staff's concern described in RAI 4.3.1-1 is resolved.

The applicant will rely on its FMP to assure that the fatigue usage of the RPV components will remain within ASME Code Section III allowable limits during the period of extended operation.

4.3.1.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA evaluation of RPV fatigue analyses in LRA Section A3.2. On the basis of its review of the USAR supplement, the staff concluded that the summary description of the applicant's actions to address the RPV fatigue analyses is adequate.

4.3.1.4 Conclusion

The staff reviewed the applicant's TLAA regarding the RPV fatigue analyses, as summarized in LRA Section 4.3.1. The staff concluded that the applicant has demonstrated that, pursuant to 10 CFR 54.21(c)(1)(iii), the effects of aging on the intended functions will be adequately managed for the period of extended operation, and, pursuant to 10 CFR 54.21(c)(1)(ii), the analyses have been projected to the end of the period of extended operation. The staff also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation sufficient to satisfy 10 CFR 54.21(d) requirements.

4.3.2 Fatigue Analysis of RPV Internals

4.3.2.1 Summary of Technical Information in the Application

In LRA Section 4.3.2, the applicant discussed the fatigue analysis of the reactor vessel internals (RIT), indicating that the analysis was performed using ASME Code, Section III criteria. The applicant stated that the most significant fatigue loading occurs at the jet pump diffuser-to-baffle plate weld location. The original 40-year calculation showed a CUF of approximately 0.33. The applicant estimated the 60-year RIT fatigue usage by multiplying the 40-year fatigue usage by 1.5. The applicant concluded that the fatigue usage of the RIT will remain below the allowable limit of 1.0 through the period of extended operation.

4.3.2.2 Staff Evaluation

The staff reviewed LRA Section 4.3.2, 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 applicant stated that the RIT fatigue analysis was guided by ASME Code Section III criteria. The applicant indicated that the most significant fatigue loading occurs at the jet pump diffuser-to-baffle plate weld location. The applicant's evaluation included three transients, (1) normal startup and shutdown, (2) improper start of a recirculation loop, and (3) design-basis accident (DBA).

The applicant stated that the 60-year fatigue usage of the RIT was estimated by multiplying the original fatigue usage by a factor of 1.5.

The staff's review of LRA Section 4.3.2 identified an area for which it needed additional information to complete its evaluation of the RIT fatigue analysis. The applicant responded to the staff's RAI as discussed below.

In RAI 4.3.2-1, dated June 21, 2005, the staff requested that the applicant confirm that the extrapolation bounded the number of startup/shutdown design cycles listed in LRA Section 4.3.1.

In its response, dated July 21, 2005, the applicant reiterated that the most significant fatigue location for the RIT is at the jet pump diffuser-to-baffle plate weld. The applicant stated that the startup/shutdown cycles had a negligible impact on fatigue usage at this location. The applicant stated that the only significant contributor to fatigue for the jet pump-to-baffle plate weld is the transient that includes improper recirculation pump startup and post-DBA flooding. USAR Section 3.6.3.3 indicates that the RIT were originally evaluated for three improper recirculation pump starts. The applicant evaluated the impact of the updated number of design cycles listed in LRA Section 4.3.1 and found that the increase in the number of improper recirculation pump starts had no significant impact on fatigue usage. Therefore, the applicant concluded that the use of the 1.5 factor to estimate the 60-year fatigue usage was conservative.

Since the number of postulated DBA events does not increase for the period of extended operation and the increase in the number of improper recirculation pump starts has no significant impact on fatigue usage, the staff concluded that the applicant had adequately evaluated the RIT.

Based on its review, the staff found the applicant's response to RAI 4.3.2-1 acceptable because the applicant performed an acceptable evaluation of the RIT for the period of extended operation, in accordance with the 10 CFR 54.21(c)(1)(ii) requirements; therefore, the staff's concern described in RAI 4.3.2-1 is resolved.

4.3.2.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA evaluation of fatigue analysis of RPV internals in LRA Section A3.2. On the basis of its review of the USAR supplement, the staff concluded that the summary description of the applicant's actions to address the fatigue analyses of RPV internals is adequate.

4.3.2.4 Conclusion

The staff reviewed the applicant's TLAA regarding the fatigue analysis of RPV internals summarized in LRA Section 4.3.2 and concluded that the applicant has provided an acceptable demonstration, 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 also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation sufficient to satisfy 10 CFR 54.21(d) requirements.

4.3.3 ASME Section III Class 1 Reactor Coolant Pressure Boundary (RCPB) Piping and Fatigue Analysis

4.3.3.1 Summary of Technical Information in the Application

In LRA Section 4.3.3, the applicant summarized the evaluation of the ASME Code Section III Class 1 reactor coolant pressure boundary (RCPB) piping and fatigue analysis for the period of extended operation. Piping systems were originally designed in accordance with American Standards Association (ASA) B31.1 and United States of America Standard (USAS) B31.1.0 which did not require an explicit fatigue analysis. The applicant concluded that the analyses demonstrate that the 40-year CUFs for the limiting components in all affected systems are below the ASME Code Section III allowable value of 1.0.

4.3.3.2 Staff Evaluation

The staff reviewed LRA Section 4.3.3, pursuant to 10 CFR 54.21(c)(1)(iii), to verify that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

The applicant indicated that RCPB piping was originally designed in accordance with ASA B31.1 and USAS B31.1.0, which did not require explicit fatigue analyses of piping components. The applicant stated that portions of the RCPB required fatigue analysis, in accordance with ASME Code Section III for Nuclear Class 1 piping.

The staff's review of LRA Section 4.3.3 identified an area for which it needed additional information to complete its evaluation of the fatigue analysis. The applicant responded to the staff's RAI as discussed below.

In RAI 4.3.3-1, dated June 21, 2005, the staff requested that the applicant provide the basis for the requirement that RCPB portions be analyzed for fatigue, in accordance with the ASME Code Section III for Nuclear Class 1 piping. The staff also requested that the applicant indicate whether the number of TT cycles used to estimate the 60-year fatigue usage of the core spray (CSP) valve joint is consistent with the number of TT cycles obtained from the FMP and used to estimate the 60-year fatigue usage of the CSP nozzle.

In its response, dated July 21, 2005, the applicant indicated that replaced portions of the REC, CSP, and residual heat removal (RHR) systems were evaluated using ASME Code Section III fatigue analysis guidelines. Replacement of ASA B31.1 components with ASME Code Section III components is acceptable. Replaced components must then satisfy ASME Code requirements, including those for fatigue.

The applicant stated that the design fatigue usage at the limiting location for RCPB CSP piping is less than 0.65 (CSP valve joint). The applicant estimated the 60-year fatigue usage by multiplying the design value by 1.5 to obtain a fatigue usage slightly below the allowable limit of 1.0; however, in LRA Table 4.3.1-1, the applicant indicated that the projected 60-year fatigue usage of the CSP nozzle is 0.65 based on the number of TT cycles counted by the FMP. In responding to RAI 4.3.3-1, the applicant stated that portions of the CSP piping were replaced in 1986 and that, as discussed above, the replaced piping was evaluated using ASME Code Section III fatigue guidelines. The applicant indicated that the evaluation considered 100 startup/shutdown cycles; therefore, extrapolation of the CSP piping usage factor by a factor of 1.5 should represent 150 startup/shutdown cycles. LRA Section 4.3.1 projects 207 startup/shutdown cycles for 60 years of plant operation. The staff reviewed data provided in SASR 89-77 indicating that the most significant TT affecting the CSP nozzle fatigue usage is startup/shutdown cycles. SASR 89-77 also indicated that 86 startup/shutdown cycles had accumulated before the CSP piping replacement. Therefore, the number of expected startup/shutdown cycles for the replaced CSP piping is 121 through the period of extended operation.

Based on its review, the staff found the applicant's response to RAI 4.3.3-1 acceptable because the applicant's evaluation of the CSP valve joint represents a conservative estimate for the number of startup/shutdown cycles; therefore, the staff's concern described in RAI 4.3.3-1 is resolved.

The applicant's FMP tracks the number of design cycles for RPV components. As discussed in Appendix B to the LRA, the FMP scope also includes RCPB piping. The staff concluded that the FMP provides an acceptable program to manage the fatigue usage of the RCPB components during the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

4.3.3.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA evaluation of ASME Code Section III Class 1 RCPB piping and fatigue analysis in LRA Section A3.3. On the basis of its review of the USAR supplement, the staff concluded that the summary description of the applicant's actions to address the fatigue analysis of ASME Code Section III Class 1 RCPB piping is adequate.

4.3.3.4 Conclusion

The staff reviewed the applicant's TLAA regarding the fatigue analysis of ASME Code Section III Class 1 RCPB piping, as summarized in LRA Section 4.3.3, and concluded that the applicant has provided an acceptable demonstration that, pursuant to 10 CFR 54.21(c)(1)(iii), the effects of aging on the intended functions will be adequately managed for the period of extended operation. The staff also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation, sufficient to satisfy 10 CFR 54.21(d) requirements.

4.3.4 RCPB Section III Class 2 and 3 Piping and Components

4.3.4.1 Summary of Technical Information in the Application

In LRA Section 4.3.4, the applicant summarized the evaluation of the RCPB ASME Code Section III Class 2 and 3 piping and components for the period of extended operation. These components were designed to ASA B31.1 and USAS B31.1.0 criteria, which did not require explicit fatigue analyses of the piping components, but did require application of a stress reduction factor to the allowable thermal bending stress range, if the number of full-range cycles exceeds 7000. The applicant stated that the number of thermal cycles experienced by these systems is not expected to exceed 7000 during the period of extended operation; therefore, the applicant concluded that the analyses will remain valid for that period.

4.3.4.2 Staff Evaluation

The staff reviewed LRA Section 4.3.4, pursuant to 10 CFR 54.21(c)(1)(i), to verify that the analyses will remain valid for the period of extended operation.

The applicant indicated that the remaining piping and components were designed to codes that did not require explicit fatigue analyses. As discussed previously, the applicant performed fatigue analyses of the replaced portions of the RCPB piping. The design of the remaining piping systems is governed by the ASA B31.1 and USAS B31.1.0 criteria that limit the number of full-range stress cycles from thermal bending to 7000. The applicant stated that the projected number of thermal bending cycles will not exceed 7000 for any non-ASME Code Section III Class 1 piping during the period of extended operation based on an assessment of the number of thermal cycles for the FW nozzle. The applicant selected the FW nozzle because it was subject to the largest number of TT cycles in the RPV nozzle fatigue analyses. The applicant multiplied the number of FW nozzle TT design cycles by 1.5 to provide a bounding estimate for the non-ASME Class 1 piping.

On the basis of its review, the staff concluded that the applicant's evaluation provides a reasonable upper bound estimate of the number of full-range thermal bending cycles for non-ASME Class 1 piping systems because the evaluation bounds the expected number of TTs, including the number of expected startup/shutdown cycles for the facility. Therefore, the staff concluded that the applicant has adequately demonstrated that the analyses of the non-ASME Class 1 piping and components will remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

4.3.4.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA evaluation of ASME Code Section III Class 2 and 3 piping and components in LRA Section A3.4. On the basis of its review of the USAR supplement, the staff concluded that the summary description of the applicant's actions to address the RCPB Section III Class 2 and 3 piping and components is adequate.

4.3.4.4 Conclusion

The staff reviewed the applicant's TLAA regarding the RCPB Section III Class 2 and 3 piping and components, summarized in LRA Section 4.3.4, and concluded that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses will remain valid for the period of extended operation. The staff also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation, sufficient to satisfy 10 CFR 54.21(d) requirements.

4.4 Irradiation-Assisted Stress-Corrosion Cracking (IASCC)

4.4.1 Summary of Technical Information in the Application

In LRA Section 4.4, the applicant summarized the evaluation of irradiation-assisted stress corrosion cracking (IASCC) for the period of extended operation. Austenitic stainless steel RPV internal components exposed to a neutron fluence greater than 5×10^{20} n/cm^2 (E > 1 MeV) are susceptible to IASCC in the BWR environment. As described in the SER to BWRVIP-26, IASCC of RPV internals is a TLAA.

4.4.2 Staff Evaluation

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

The staff reviewed the information in the LRA and noted that the austenitic stainless steel components exposed to a neutron fluence greater than 5×10^{20} n/cm^2 (E >1 MeV) are considered susceptible to IASCC. These RPV internal components include the top guide, the shroud, and the incore instrumentation dry tubes and guide tubes. The staff reviewed the fluence calculations for the RPV and verified that other RPV internal components (e.g., the core plate) are not expected to exceed a neutron fluence of 5×10^{20} n/cm^2 and thus are considered not to be susceptible to IASCC. In the LRA, the applicant stated that the aging effects from IASCC of these RPV components are managed by three aging management programs (AMPs), B2.1.2, ASME Section XI In-Service Inspection, Subsections IWB, IWC, and IWD; B2.1.12, BWR Vessel Internals; and B2.1.25, Plant Chemistry. The applicant stated that implementation of these three AMPs will manage the aging effects from IASCC such that the RPV internal components will continue to perform their intended functions consistently with the licensing basis for the period of extended operation.

The staff reviewed other applicant documents pertaining to the RPV, BWRVIP documents, and EPRI topical reports applying to generic RPVs. The staff observed that, while fluence level was the primary contributor to IASCC, additional factors also contributed or increased component susceptibility to IASCC. The staff observed that BWRVIP-41, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines," states that materials like austenitic stainless steel used in jet pumps are not greatly susceptible to IASCC due to the low fluence levels in the annulus region." The staff also observed that the June 5, 2001, SER that accepted BWRVIP-41 stated that materials in a nonoxygenated environment are also not greatly susceptible to IASCC, which becomes a concern only when cracks are already present in a component. Thus, the SER stated that, when an applicant can show that cracks have not occurred in components, loss of fracture toughness from IASCC will not be a significant aging effect.

The staff asked the applicant to clarify its actions regarding the above additional factors. As to the aggressive oxygenated environment, the applicant responded that it had implemented hydrogen water chemistry (HWC) in 1989, which reduces the oxidizing environment of the reactor coolant system (RCS) by injecting excess hydrogen to combine with free oxygen produced by radiolysis. The dissolved oxygen content of FW is regulated to 20–50 parts per billion (ppb) during power operation, which minimizes corrosion potential. The staff reviewed historical data from the Water Chemistry Program and verified the low dissolved oxygen content.

In a letter, dated June 10, 2005, the applicant stated that, in addition to those examinations required by the ISI Program, which includes all pertinent examinations required by the BWRVIP program, it will examine the top guide grid high-fluence locations using the EVT-1 visual examination method. In the same letter, the applicant committed to inspections of 10 percent of these locations within 12 years. The staff reviewed the applicant's operational experience and observed that, to date, it has inspected 25 percent of the high-fluence locations of the top guide grid and detected no evidence of cracking.

The staff reviewed the fluence calculations for the RPV internals and observed that there was a factor of 30 percent that was added to the calculated fluence level results. The staff asked the applicant to clarify the purpose of this added factor. The applicant stated that this factor was added for conservatism.

The staff reviewed the RPV components for IASCC, considering that (1) these components were composed of a material that was identified in BWRVIP-41 as not highly susceptible to IASCC, (2) these components are in a nonaggressive, low-dissolved-oxygen environment, so, as stated in the SER, the susceptibility of these components to IASCC is reduced, (3) no evidence of cracks has been detected in the RPV inspections to date, so as stated in the SER, significant loss of fracture toughness will not result, and (4) the fluence calculations that determined the three RPV components susceptible to IASCC add a factor of 30 percent, for conservatism. The staff concluded that the applicant's AMP B2.1.2, ASME Section XI In-Service Inspection, Subsections IWB, IWC, and IWD; B2.1.12, BWR Vessel Internals; and B2.1.25, Plant Chemistry, will adequately manage the aging effects from IASCC for the period of extended operation.

During the audit and review, the staff identified an additional issue that required further clarification by the applicant. The applicant has committed to perform additional top guide examinations within the first 12 years of the period of extended operation; however, there is no commitment to perform examinations during the remaining period of extended operation, nor a commitment as to what the applicant will do if any RPV examination detects an indication. In RAI 4.1-1, the staff requested that the applicant describe its actions for the remainder of the period of extended operation.

In its response, by letter dated November 22, 2005, the applicant stated that it will perform an inspection of a sampling of top guide high-fluence locations (i.e., where fluence exceeds 5.0×10^{20} n/cm^2) consistent with the lower plenum inspection and flaw evaluation guidelines described in BWRVIP-47. Ten percent of the total high-fluence population will be inspected within 12 years, with a minimum of 5 percent inspected within the first 6 years. If flaws are detected, inspection of an additional 5 percent of the total high-fluence population will be completed. This process will be repeated until no new flaws are detected. Any flaw exceeding inspection limits will be evaluated and necessary corrective actions made that may include, but are not limited to, accept as-is, accept as-is with required periodic reinspection, or remove indication by metal removal. All corrective actions will be performed in accordance with approved procedures. Indication mapping and sizing will be documented for use in industry resolution of any related concerns. Reinspection scope and frequency during the entire period of extended operation will depend on initial inspection results, as well as on related industry experience. Therefore, the staff concluded this TLAA is acceptable and consistent with the GALL Report, and the staff's concern described in RAI 4.1-1 is resolved.

4.4.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA evaluation of IASCC in LRA Section A3.5. On the basis of its review of the USAR supplement, the staff concluded that the summary description of the applicant's actions to address the IASCC is adequate.

4.4.4 Conclusion

The staff concluded that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the aging effects from IASSC on the intended functions will be adequately managed for the period of extended operation. The staff also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation, sufficient to satisfy the requirements of 10 CFR 54.21(d).

4.5 Effects of Reactor Coolant Environment

4.5.1 Summary of Technical Information in the Application

In LRA Section 4.5, the applicant summarized its evaluation of the effects of the reactor coolant environment for the period of extended operation. The applicant evaluated the impact of the reactor coolant environment on the fatigue life of the locations addressed in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves for Selected Nuclear Power Plant Components." The applicant's evaluation indicated that the environmental fatigue usage for all locations is less than the allowable limit of 1.0 for the period of extended operation. The applicant concluded that the effects of environmentally assisted fatigue were shown to be acceptable through the period of extended operation. The applicant further indicated that the FMP periodically reviews and updates fatigue analyses to ensure continued compliance with the fatigue acceptance criteria.

4.5.2 Staff Evaluation

The staff reviewed LRA Section 4.5, 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 applicant stated that the FMP will continue during the period of extended operation to assure that design cycle limits are not exceeded. The applicant's FMP tracks transients and cycles of RCS components with explicit design transient cycles to assure that these components remain within their design bases. Generic Safety Issue (GSI)-166, "Adequacy of the Fatigue Life of Metal Components," raised concerns about the conservatism of the fatigue curves used in the design of the RCS components. Although GSI-166 was resolved for the current 40-year design life of operating components, the staff identified GSI-190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life," to address license renewal. The NRC closed GSI-190 in December 1999, after concluding the following:

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

The staff compared the usage factors provided by the applicant with those in NUREG/CR-6260 for the older vintage BWR. NUREG/CR-6260 identified several locations for which the environmental usage factor was projected to exceed 1.0, including the CSP nozzle safe end, the FW nozzle, the FW line reactor core isolation cooling (RCIC) tee connection, and the RHR return line tee.

The staff's review of LRA Section 4.5 identified an area for which it needed additional information to complete its evaluation of the effects of the reactor coolant environment. The applicant responded to the staff's RAI as discussed below.

The environmental fatigue usage for the CSP nozzle (safe end) in LRA Section 4.5 is much lower than the fatigue usage of the CSP nozzle (without environmental effects) in LRA Table 4.3.1-1. In RAI 4.5-1, dated June 21, 2005, the staff requested that the applicant provide the basis for the reported usage factors in LRA Section 4.5. In addition, the staff requested that the applicant discuss the calculation of the F_{en} multipliers used for each of the NUREG/CR-6260 locations.

In its response, dated August 16, 2005, the applicant stated that the fatigue usage reported in LRA Table 4.3.1-1 for the CSP nozzle was based on cycle counting, whereas, the environmental usage factor in LRA Section 4.5 was based on a detailed stress analysis. During a followup discussion on September 1, 2005, the applicant stated that the fatigue usage in Table 4.3.1-1 resulted from considering all load cycles at the maximum stress, and the usage factor in LRA Section 4.5 resulted from separating the individual load cycles by stress level. The staff found this explanation reasonable.

The applicant indicated that MNGP used HWC. The NUREG/CR-6260 components were evaluated for a high oxygen environment without HWC. Oxygen concentration has a significant impact on the fatigue life of carbon and low-alloy steel components. HWC lowers the oxygen concentration in BWRs to reduce the stress-corrosion cracking potential of stainless steel components. The reduced oxygen concentration significantly reduces the environmental impact on the fatigue life of carbon and low-alloy steel components compared to equivalent NUREG/CR-6260 carbon and low-alloy steel components.

NUREG/CR-6260 identified high environmental fatigue usage at the stainless steel CSP nozzle safe end. The applicant replaced the CSP safe ends in 1986. The applicant stated that the replaced CSP safe ends are carbon steel. Because the applicant has implemented an HWC program, the environmental impact on the fatigue usage of the carbon steel safe ends is not significant; therefore, the staff concluded that the applicant's calculated environmental fatigue usage for the CSP nozzle safe ends is reasonable and acceptable.

Because the applicant uses HWC, usage factors for the FW nozzle and the FW line RCIC tee connection are not directly comparable to the NUREG/CR-6260 values. The applicant stated that it had recently evaluated these locations in detail for environmental fatigue. The applicant also stated that the environmental factors for the evaluations considered both the times HWC had and had not been in operation. Since the FW nozzle safe ends were replaced in the 1980s, the lower environmental factor reflects the greater operating exposure to HWC. Considering the applicant's use of HWC and replacement of the FW nozzle safe ends in the 1980s, the reported environmental factors are reasonable; therefore, the staff found the applicant's evaluation of the environmental fatigue usage of the FW nozzle and the FW line RCIC tee connection acceptable.

The applicant also evaluated the RHR piping tapered transition and RHR return line tee in detail for environmental fatigue. The RHR return line tee was the bounding fatigue usage location. Since the RHR return line tee was replaced in the 1980s, it will be subject to fewer years of service than the component evaluated in NUREG/CR-6260. In addition, the environmental fatigue criteria for the stainless evaluations in NUREG/CR-6260 were independent of temperature. Later criteria in NUREG/CR-5704 found that the environmental effect on fatigue usage is insignificant at temperatures less than 200 °C. The staff noted that RHR shutdown cooling initiates at less than 200 °C. Considering the items discussed above, the applicant's environmental fatigue usage of the RHR return line tee is reasonable. Therefore, the staff found the applicant's evaluation of the environmental fatigue usage of the RHR return line tee acceptable.

Based on its review, the staff found the applicant's response to RAI 4.5-1 acceptable because the applicant reasonably evaluated the environmental impact on the fatigue life of RCPB components for the period of extended operation; therefore, the staff's concern described in RAI 4.5-1 is resolved.

4.5.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA evaluation of the effects of the reactor coolant environment in LRA Section A3.7. On the basis of its review of the USAR supplement, the staff concluded that the summary description of the applicant's actions to address the effects of the reactor coolant environment is adequate.

4.5.4 Conclusion

The staff reviewed the applicant's TLAA regarding the effects of reactor coolant environment, as summarized in LRA Section 4.5, and concluded that the applicant provided an acceptable demonstration, 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 also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation, sufficient to satisfy the requirements of 10 CFR 54.21(d).

4.6 Fatigue Analyses of the Primary Containment, Attached Piping, and Components

The Mark I containment consists of a freestanding steel containment drywell, vent system, and steel pressure suppression chamber (torus). Large-scale testing of the Mark III containment and in-plant testing of Mark I primary containment systems identified additional hydrodynamic loads not considered in the original containment design. The Mark I Owners Group initiated the Mark I Containment Program to develop a generic load definition and structural analysis techniques. The staff evaluation of the generic load definition and structural assessment techniques is in NUREG-0661, "Safety Evaluation Report, Mark I Containment Long Term Program, Resolution of Generic Technical Activity A-7," July 1980. The Mark I Containment Long-Term Program evaluation of hydrodynamic loads included fatigue analyses of the torus and vent system and of the torus attached piping (TAP).

The containment liner plates, penetration sleeves (including dissimilar metal welds), and penetration bellows may be designed in accordance with the ASME Code, Section III, requirements. If a plant's code of record requires a fatigue analysis, it may be a TLAA and must be evaluated in accordance with 10 CFR 54.21(c)(1) to ensure adequate management of the effects of aging on the intended functions for the period of extended operation.

The staff reviewed the adequacy for the period of extended operation of the fatigue analyses of the metal containment, containment liner plates (including welded joints), penetration sleeves, dissimilar metal welds, and penetration bellows. SER Section 4.3 reviews the fatigue analyses of the pressure boundary of process piping, following the guidance in SRP-LR Section 4.3.

4.6.1 Fatigue Analysis of the Suppression Chamber, Vents, and Downcomers

4.6.1.1 Summary of Technical Information in the Application

In LRA Section 4.6.1, the applicant discussed the suppression chamber and vent system fatigue analysis. New hydrodynamic loads were identified subsequent to the original design of the containment suppression chamber vents. These loads result from blowdown into the suppression chamber during a postulated LOCA and during safety relief valve (SRV) operation for plant transients. The applicant identified the vent header-downcomer intersection and the torus shell as the limiting locations in terms of fatigue usage. The applicant stated that the only contribution to fatigue usage during normal operation is from SRV operation and that the number of SRV actuations is not expected to exceed the design number through the period of extended operation.

4.6.1.2 Staff Evaluation

The staff reviewed LRA Section 4.6.1, pursuant to 10 CFR 54.21(c)(1)(i), to verify that the analyses will remain valid for the period of extended operation, and, pursuant to 10 CFR 54.21(c)(1)(iii), to verify that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

The applicant stated that the Mark I Containment Program evaluated the suppression chamber and vent header system, including fatigue analyses of the torus shell and vent header system. The applicant's Mark I Containment Program Plant Unique Analysis Report summarized these analyses. The applicant subsequently reevaluated these locations for the increased number of SRV actuations postulated as a result of the 1998 power rerate. The resulting fatigue usage, considering the increase in SRV cycles, was less than the 1.0 allowable limit. The applicant estimated that the number of SRV cycles will not exceed the number used for the evaluation of the suppression chamber and vent header system during the period of extended operation. In addition, the applicant indicated that the Fatigue Monitoring Program monitors the number of SRV lifts to assure that the usage factor remains below 1.0 for the limiting components.

The staff found that the applicant's FMP will ensure that fatigue usage of the suppression chamber vents and downcomers will remain below 1.0 for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

4.6.1.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA evaluation of fatigue analysis of the suppression chamber, vents, and downcomers in LRA Section A3.8. On the basis of its review of the USAR supplement, the staff concluded that the summary description of the applicant's actions to address the fatigue analysis of the suppression chamber, vents, and downcomers is adequate.

4.6.1.4 Conclusion

The staff reviewed the applicant's TLAA regarding the fatigue analysis of the suppression chamber, vents, and downcomers, as summarized in LRA Section 4.6.1, and concluded that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses will remain valid for the period of extended operation. In addition, the staff concluded that, pursuant to 10 CFR 54.21(c)(1)(iii), the effects of aging on the intended functions will be adequately managed for the period of extended operation. The staff also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation, sufficient to satisfy 10 CFR 54.21(d) requirements.

4.6.2 Fatigue Analysis of the SRV Piping Inside the Suppression Chamber and Internal Structures

4.6.2.1 Summary of Technical Information in the Application

In LRA Section 4.6.2, the applicant discussed the suppression chamber piping and internals structure fatigue evaluations. The reactor pressure relief system includes SRVs located on the main steamlines within the drywell between the reactor vessel and the first isolation valve. The applicant stated that it had not performed fatigue analyses for torus internal structures (i.e., catwalk and monorail). The applicant indicated that it had performed fatigue analyses for the SRV piping inside the torus. The applicant also stated that the SRV piping analyses will remain valid for the period of extended operation.

4.6.2.2 Staff Evaluation

The staff reviewed LRA Section 4.6.2, 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 applicant stated that the fatigue analyses of the torus internal SRV piping had been part of the Mark I Containment Program. The applicant also indicated that the piping had been evaluated for the 26-percent increase in the number of SRV cycles resulting from the 1998 power rerate. The resulting 40-year fatigue usage was well below the allowable limit of 1.0. The applicant multiplied the resulting 40-year fatigue usage by 1.5 to estimate the fatigue usage for 60 years of plant operation.

Because the applicant indicated that the number of SRV cycles used in the power rerate evaluation is conservative for 40 years of plant operation, the 1.5 factor provides a conservative estimate for the period of extended operation; therefore, the staff found that the applicant adequately demonstrated that the fatigue usage of the torus SRV piping will remain within acceptable limits for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

4.6.2.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA evaluation of fatigue analysis of the SRV piping inside the suppression chamber and internal structures in LRA Section A3.8. On the basis of its review of the USAR supplement, the staff concluded that the summary description of the applicant's actions to address the fatigue analysis of the SRV piping inside the suppression chamber and internal structures is adequate.

4.6.2.4 Conclusion

The staff reviewed the applicant's TLAA regarding the fatigue analysis of the SRV piping inside the suppression chamber and internal structures, as summarized in LRA Section 4.6.2, and concluded that the applicant provided an acceptable demonstration, 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 also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation, sufficient to satisfy 10 CFR 54.21(d) requirements.

4.6.3 Fatigue Analysis of Suppression Chamber External Piping and Penetrations

4.6.3.1 Summary of Technical Information in the Application

In LRA Section 4.6.3, the applicant discussed the fatigue analysis of suppression chamber external piping and penetrations. These analyses included the large- and small-bore TAP, suppression chamber penetrations, and the emergency core cooling system (ECCS) suction header and were based on cycles postulated to occur within the 40-year operating life of the plant. The applicant stated that these analyses will remain valid for the period of extended operation.

4.6.3.2 Staff Evaluation

The staff reviewed LRA Section 4.6.3, 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 and, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

The applicant stated that fatigue effects were specifically addressed for the suppression chamber TAP penetrations and the suction header in the Mark I Containment Program. The applicant indicated that it evaluated the TAP penetration fatigue analyses for the 26-percent increase in SRV cycles resulting from the 1998 power uprate. The resulting 40-year fatigue usage was less than the allowable limit of 1.0. The applicant indicated that the Fatigue Monitoring Program monitors the number of SRV lifts to assure that the usage factor remains below 1.0 for the limiting components.

Because the number of SRV lifts is monitored, the staff found that the applicant's FMP will assure that the TAP penetrations fatigue usage will remain below 1.0 for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

The applicant also stated that the Mark I Owner's Group had generically addressed TAP piping for all Mark I plants. The applicant identified the SRV piping as the limiting location and evaluated it for a 26-percent increase in the number of SRV cycles as a result of the 1998 power rerate. The resulting 40-year fatigue usage was well below the 1.0 allowable limit. The applicant multiplied the resulting 40-year fatigue usage by 1.5 to estimate fatigue usage for 60 years of plant operation.

Because the applicant indicated that the number of SRV cycles used in the power rerate evaluation is conservative for 40 years of plant operation, the 1.5 factor provides a conservative estimate for the period of extended operation; therefore, the staff found that the applicant adequately demonstrated that the fatigue usage of the torus SRV piping will remain within acceptable limits for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

4.6.3.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA evaluation of the fatigue analysis of suppression chamber external piping and penetrations in LRA Section A3.8. On the basis of its review of the USAR supplement, the staff concluded that the summary description of the applicant's actions to address the fatigue analysis of the suppression chamber external piping and penetrations is adequate.

4.6.3.4 Conclusion

The staff reviewed the applicant's TLAA regarding the fatigue analysis of the suppression chamber external piping and penetrations, as summarized in LRA Section 4.6.3, and concluded that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses have been projected to the end of the period of extended operation. In addition, pursuant to 10 CFR 54.21(c)(1)(iii), the staff concluded that the effects of aging on the intended functions will be adequately managed for the period of extended operation. The staff also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation, sufficient to satisfy 10 CFR 54.21(d) requirements.

4.6.4 Drywell-to-Suppression Chamber Vent Line Bellows Fatigue Analysis

4.6.4.1 Summary of Technical Information in the Application

In LRA Section 4.6.4, the applicant discussed the fatigue analysis of the drywell-to-suppression chamber vent line bellows. The applicant stated that the vent line bellows stresses are primarily caused by differential thermal expansion during startup/shutdown and accident conditions. The applicant projected that the number of startup/shutdown cycles in the design will not be exceeded during the period of extended operation; therefore, the applicant concluded that the analysis remains valid for that period.

4.6.4.2 Staff Evaluation

The staff reviewed LRA Section 4.6.4, pursuant to 10 CFR 54.21(c)(1)(i), to verify that the analyses will remain valid for the period of extended operation.

The applicant stated that the drywell-to-suppression chamber vent line bellows stresses are primarily caused by differential thermal expansion of the reactor suppression chamber and drywell during normal startup and shutdown operations. The applicant stated that the design assumes 300 startup/shutdown cycles. As indicated in LRA Section 4.3, the applicant projected fewer than 300 startup/shutdown cycles through the period of extended operation.

4.6.4.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA evaluation of drywell-to-suppression chamber vent line bellows fatigue analysis in LRA Section A3.8. On the basis of its review of the USAR supplement, the staff concluded that the summary description of the applicant's actions to address the drywell-to-suppression chamber vent line bellows fatigue analysis is adequate.

4.6.4.4 Conclusion

The staff reviewed the applicant's TLAA regarding the drywell-to-suppression chamber vent line bellows fatigue analysis, as summarized in LRA Section 4.6.4, and concluded that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses will remain valid for the period of extended operation. The staff also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation, sufficient to satisfy the requirements of 10 CFR 54.21(d).

4.6.5 Primary Containment Process Penetration Bellows Fatigue Analysis

4.6.5.1 Summary of Technical Information in the Application

In LRA Section 4.6.5, the applicant discussed the primary containment process bellows fatigue analysis. Containment pipe penetrations required to accommodate thermal movement have expansion bellows. The applicant stated that these containment process bellows involve piping systems that penetrate the drywell shell and that these bellows were designed for a minimum of 7000 operating cycles. The applicant indicated that the number of expected operating cycles through the period of extended operation is much fewer than 7000; therefore, the applicant concluded that the analysis of the containment process bellows will remain valid for the period of extended operation.

4.6.5.2 Staff Evaluation

The staff reviewed LRA Section 4.6.5, pursuant to 10 CFR 54.21(c)(1)(i), to verify that the analyses will remain valid for the period of extended operation.

The applicant stated that the containment process piping penetration bellows were designed to ASME Code Section III, Class B requirements, which do not require a formal fatigue analysis; however, the criteria for the attached process piping limit the number of full-range bending cycles. As discussed in SER Section 4.3, the applicant indicated that the number of expected operating cycles for the process piping is much fewer than 7000; therefore, the applicant concluded that the analyses of the containment process piping bellows will remain valid for the period of extended operation.

The staff found that the applicant's evaluation of the process piping provides a reasonable upper-bound estimate of the number of full-range thermal bending cycles for the process piping penetration bellows because the evaluation bounds the expected number of TTs, including the number of expected startup/shutdown cycles, for the facility.

4.6.5.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA evaluation of primary containment process penetration bellows fatigue analysis in LRA Section A3.8. On the basis of its review of the USAR supplement, the staff concluded that the summary description of the applicant's actions to address the primary containment process penetration bellows fatigue analysis is adequate.

4.6.5.4 Conclusion

The staff reviewed the applicant's TLAA of the primary containment process penetration bellows fatigue analysis summarized in LRA Section 4.6.5 and concluded that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses will remain valid for the period of extended operation. The staff also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation, sufficient to satisfy the requirements of 10 CFR 54.21(d).

4.7 <u>Environmental Qualification of Electrical Equipment (EQ)</u>

4.7.1 Summary of Technical Information in the Application

The 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," environmental qualification (EQ) program has been identified as a TLAA for the purposes of license renewal. The TLAA of EQ electrical components includes all long-lived, passive electrical components and instrumentation and controls (I&C) components that are important to safety and located in a harsh environment. The harsh environments of the plant are those areas that are subject to environmental effects by a LOCA or a high-energy line break (HELB). The EQ equipment comprises safety-related (SR) and Q-list equipment, nonsafety-related (NSR) equipment whose failure could prevent satisfactory accomplishment of any SR function, and necessary post-accident monitoring equipment.

The applicant's EQ Program manages component thermal, radiation, and cyclic aging through aging evaluations based on 10 CFR 50.49(f) qualification methods. Environmentally qualified equipment must be refurbished, replaced, or have its qualification extended before reaching the aging limits established in the aging evaluation. Aging evaluations for environmentally qualified equipment that specify the qualified life of at least 40 years are considered TLAAs for license renewal.

4.7.2 Staff Evaluation

The staff reviewed LRA Section 4.7, pursuant to 10 CFR 54.21(c)(1)(iii), to verify that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

The results of the electrical equipment EQ in LRA Section 4.7 indicate that the aging effects of electrical equipment EQ identified in the TLAA will be managed during the extended period of operation under 10 CFR 54.21(c)(1)(iii); however, the applicant did not submit information on the attribute of a reanalysis of an aging evaluation to extend the qualification life of such electrical equipment identified in the TLAA. The important attributes of a reanalysis are the analytical methods, the data collection and reduction methods, the underlying assumptions, the acceptance criteria, and the corrective actions. In RAI 4.7-1, dated November 7, 2005, the staff requested that the applicant provide information about the important attributes of reanalysis of an aging evaluation of electrical equipment identified in the TLAA to extend the qualification under 10 CFR 50.49(e).

In its response, by letter dated December 7, 2005, the applicant stated that the reanalysis of an aging evaluation normally extends the qualification by reducing excess conservatism incorporated in the prior evaluation. The staff reviewed this information and found the applicant's response satisfactory; therefore, the staff's concern described in RAI 4.7-1 is resolved.

The staff also reviewed the EQ Program to determine whether it will assure that the electrical components covered under this program will continue to perform their intended function consistently with the CLB for the period of extended operation. The staff's evaluation of the component qualification focused on how the program manages the aging effects through effective incorporation of the following 10 elements—program scope, preventive action, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls, and operating experience.

Program Scope

Pursuant to 10 CFR 50.49, the EQ Program evaluates harsh environments in which electrical equipment important to safety may be required to operate. The applicant stated that an equipment master list, maintained at MNGP, includes SR electrical equipment, NSR equipment whose failure could prevent accomplishment of safety functions, and certain post-accident monitoring equipment. The staff considered the scope of the program acceptable.

Preventive Actions

The ongoing EQ Program ensures that electrical equipment important to safety is capable of performing its intended function in a harsh environment, in accordance with 10 CFR 50.49. Although 10 CFR 50.49 does not require actions that prevent aging effects, EQ Program actions that could be viewed as preventive actions include (1) establishing the component service condition tolerance and aging limits (e.g., qualified life or condition limit) and (2) where applicable, requiring specific installation, inspection, monitoring, or periodic maintenance actions to manage equipment aging effects within the qualification. The staff considered these actions acceptable because 10 CFR 50.49 does not require actions that prevent aging effects.

Parameters Monitored or Inspected

The applicant stated that qualified life is not based on condition or performance monitoring. Pursuant to 10 CFR 50.49(e), the qualification must include and be based on temperature and pressure, humidity, chemical effects, radiation, aging, submergence, synergistic effects, and margins. Pursuant to RG 1.89, Revision 1, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," issued June 1984, monitoring or inspection of certain environmental conditions or component parameters may be used to ensure that the component is within the bounds of its qualification basis or as a means to modify the qualified life. The applicant's EQ Coordinator is responsible for reviewing program data and industry information. Deviations are documented in the Corrective Action Program (CAP) and actions to correct identified issues may include monitoring, inspection, reanalysis, or testing. For example, the EQ Coordinator monitors radiation protection surveys for changes in radiation levels and has initiated a temperature monitoring program in areas containing EQ equipment. The staff considered this monitoring approach appropriate because the program objective is to ensure that the established qualified life of devices is not exceeded.

Detection of Aging Effects

10 CFR 50.49 does not require the detection of aging effects for inservice components. The applicant stated that the CAP, the Preventive Maintenance Program, and the Quality Control Program will identify any aging effects of EQ equipment and initiate corrective action required to maintain equipment qualification. In addition, monitoring and inspection of certain environmental conditions or component parameters may be used to ensure that the component is within the bounds of its qualification basis, or as a means to modify the qualified life. The staff considered the applicant's above programs to detect aging effects acceptable.

Monitoring and Trending

10 CFR 50.49 does not require monitoring and trending of component condition or performance parameters of inservice components to manage the effects of aging. The EQ Program actions that could be viewed as monitoring include monitoring how long qualified components have been installed. Monitoring or inspecting certain environmental, condition, or component parameters may be used to ensure that a component is within its qualification or as a means to modify the qualification. The staff considered this program acceptable because 10 CFR 50.49 does not require monitoring and trending of component condition or performance parameters of inservice components to manage the effects of aging.

Acceptance Criteria

10 CFR 50.49 acceptance criteria require that an inservice EQ component is maintained within its qualification including its (a) established aging limits and (b) continued qualification for the projected accident conditions. 10 CFR 50.49 requires refurbishment, replacement, or requalification before exceeding the aging limits of each installed device. The applicant stated that its program has identified all components subject to the 10 CFR 50.49 acceptance criteria on an SR equipment master list. The EQ Coordinator maintains calculations supporting equipment qualification, which include such information as location, environmental conditions, qualification methods, and acceptance criteria. Before reaching the end of qualified life, affected components are refurbished, requalified, or replaced to ensure continued functionality of the installed components. The staff considered this program acceptable since it is consistent with 10 CFR 50.49 requirements of refurbishment, replacement, or requalification before exceeding the qualified life of each installed device.

Corrective Actions, Confirmation Process, and Administrative Controls

The applicant stated that if an EQ component is found to be outside the bounds of its qualification basis, it implements corrective actions in accordance with the station's CAP. When operational or maintenance activities identify unexpected adverse conditions affecting the environment of a qualified component, it is evaluated and the applicant takes appropriate corrective actions, which may include changes to the qualification bases and conclusions. When an emerging industry aging issue is identified that affects the qualification of an EQ component, it is evaluated and the applicant takes appropriate corrective actions, which may include changes to the qualification bases and conclusions. The staff considered this acceptable because the corrective actions are implemented in accordance with the requirements of Appendix B to 10 CFR Part 50, which ensures the adequacy of corrective actions. SER Section 3.0.4 addresses the evaluations of these program elements.

Operating Experience

The EQ Program includes monitoring and assessment of industry information to assess its impact on EQ components. The applicant stated that the EQ Coordinator is responsible for reviewing the disposition of such information, as well as subsequent assignment of actions to be taken on such information and confirmation that the completion of the actions satisfactorily address potential EQ aging issues. The staff found that the applicant adequately addressed operating experience.

4.7.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA evaluation of electrical equipment EQ in LRA Section A3.9. On the basis of its review of the USAR supplement, the staff concluded that the summary description of the applicant's actions to address electrical equipment EQ is adequate.

4.7.4 Conclusion

The staff concluded that the applicant provided an acceptable demonstration regarding electrical equipment EQ, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions will be adequately managed for the period of extended operation. The staff also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation, sufficient to satisfy the requirements of 10 CFR 54.21(d).

4.8 Stress Relaxation of Rim Holddown Bolts

4.8.1 Summary of Technical Information in the Application

In LRA Section 4.8, the applicant summarized the evaluation of the stress relaxation of rim holddown bolts for the period of extended operation. As described in the SER to BWRVIP-25, plants must consider relaxation of the rim holddown bolts as a TLAA issue. Because MNGP has not installed core plate wedges, the loss of preload must be considered in the TLAA evaluation.

4.8.2 Staff Evaluation

The staff reviewed LRA Section 4.8, 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 LRA states that, for the period of extended operation, the expected preload loss for the rim holddown bolts was assumed to be 19 percent, which bounds the original BWRVIP analysis. With a 19-percent preload loss, the core plate will maintain sufficient preload to prevent sliding under both normal and accident conditions.

In a letter dated June 10, 2005, the applicant provided additional details of the analysis for the rim holddown bolts:

> To more accurately address MNGP for License Renewal, a plant-specific calculation was performed that incorporated the MNGP core plate geometry, an operating temperature of 288 °C (550 °F) and a MNGP fluence calculation that was performed specifically for License Renewal in accordance with guidance provided in Regulatory Guide 1.190, 'Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,' March 2001, (LRA Section 4.2.1). The maximum fluence applicable to the bolts in the highest fluence region of the core plate was determined to be 2.2×10^{19} n/cm^2 at the end of the 60-year plant life. The resultant relaxation was determined to be 8 percent based on GE Design Documents. The analysis assumed that all of the bolts were at this fluence even though many bolts experience a lower fluence depending on their specific location...

In RAI 4.8-2, dated September 28, 2005, the staff requested that the applicant provide the stress relaxation curves, information regarding the material type, the loads used to develop the stress relaxation curves, and show that the axial and bending stresses for the mean and highest loaded holddown bolts will not exceed the ASME allowable stresses.

In its response, dated October 28, 2005, the applicant provided, "The Relaxation of Irradiated Austenitic Steels and Ni-GE Mean Design Curve," based on a model that assumed a stress-linear, primary plus secondary creep law form and was fit to the shown data using stepwise multiple regression.

The rim holddown bolts are Type 304 stainless steel. The data used to develop the curve include several austenitic materials. The applicant analyzed the impact of austenitic material type on stress relaxation from neutron radiation. In its letter, dated October 28, 2005, the applicant stated the following:

> Relaxation of irradiated, structural materials from radiation creep is much less sensitive to 'normal material variations' (e.g., in austenitic stainless steels) than other radiation properties. Radiation segregation and hardening characteristics are similar for all austenitic stainless steels, although some experience presegregation (from annealing). Also, neutron relaxation is among the most consistent and reproducible phenomenon, and little variation is observed in stainless steel (e.g., 304, 316, 321, 347/8, L-grade and nuclear grade). The relaxation behavior of these stainless steels is often used for many different austenitic alloys such as Nitronic 50, Alloy X-750 and Alloy 718.

To support the conclusion that the GE design curves apply to Type 304 stainless steel, the applicant presented stress relaxation data from the BWRVIP-99 report, "Crack Growth Rates in Irradiated Stainless Steels in BWR Internal Components," and from J.P. Foster and Halden. The GE design curve predicts higher relaxation levels (i.e., lower fraction of load remaining) than observed from the Foster and Halden data and is thus conservative compared to these data.

The analysis included the impact of test temperature and neutron flux on stress relaxation:

> More than 80 percent of the tests, shown in Figure 1, were conducted at a temperature of 550 °F and a majority of these were conducted in an operating BWR environment. The other tests, were conducted at either 570 or 600 °F which is expected to produce more relaxation. Since such a large portion of the

data was conducted at typical BWR operating conditions, the data temperature is considered fully representative of the core plate bolts.

While the cumulative fluence information was available as part of the original test reports and the GE Design Curve documentation, the flux conditions were not directly available. Many of the tests were associated with springs which had reached fluences ranging from $8x10^{20}$ to $8x10^{21}$ n/cm^2. Based on a reasonable exposure time, the flux will be expected to range from $7x10^{12}$ to $9x10^{13}$ n/cm^2/s. The fluxes defined for two of the smaller sets of test data were $2.7x10^{14}$ and $2x10^{17}$ n/cm^2/s, respectively. Review of the data over these four orders of magnitude showed no discernable flux dependence; however, the neutron flux levels were at least 100 times higher than that experienced by the core plate bolts. As described above, the temperature data are representative for use in the core plate bolt evaluation. The neutron flux data, however, was measured in specimens subject to fluxes ranging from $1x10^{13}$ to $2x10^{17}$ n/cm^2/s, which is higher than the $8.5x10^{9}$ n/cm^2/s average flux experienced by the core plate bolts themselves. Given the large range of higher flux for which the properties are the same, the impact of the lower flux to which the bolts are exposed is viewed as negligible.

Based on the analysis and supporting data, the staff agreed with the applicant that the GE design curves apply to Type 304 stainless steel used in the core plate bolts.

Based on the GE design curves and a neutron fluence of $2.2x10^{19}$ n/cm^2 (E > 1.0 MeV), the applicant determined that the amount of stress relaxation at the end of the period of extended operation would be 8 percent. This neutron fluence was calculated using a procedure which is in accordance with the guidance in RG 1.190 and corresponds to the maximum fluence applicable to the bolts at the end of the period of extended operation.

The applicant also performed a plant-specific analysis to show that the axial and bending stresses in the core plate holddown bolts, considering the loss of preload (8 percent) at the end of the period of extended operation, will not exceed the ASME Code, Section III allowable Pm (membrane) and Pm + Pb (membrane + bending) stresses. The analysis was based on the assumption that sufficiently high frictional forces, resulting from the preload forces in the bolts and coefficient of static friction test data between the rim and the support surfaces, will be induced between the core plate rim and the shroud support to prevent sliding of the core plate under design-basis loading resulting from maximum horizontal and vertical seismic and accident loads. Under this scenario, no bending stresses are induced in the holddown bolts. The only stresses sustained by the holddown bolts are axial, resulting from the preload and the vertical loading (differential pressure and seismic) on the plate. These stresses were shown to be considerably lower than the ASME Section III allowable Pm stress.

The staff evaluated this analysis and concluded that the postulated coefficient of static friction was not applicable under the reactor fluid operating environment because of other coefficients of friction data indicating that a lower value may be appropriate. A smaller coefficient of static friction permits sliding of the core plate under horizontal acceleration and induces bending stresses in the bolts. This analysis was, therefore, found to be unacceptable.

Inspection and flaw evaluation guidelines of BWR core support plates were previously submitted to the NRC in BWRVIP-25, "BWR Core Plate Inspection and Flaw Evaluation Guidelines," issued December 1996. Appendix A to this report contains a prototypical core plate holddown bolt analysis under representative horizontal and vertical seismic and accident-loading conditions. The analysis was based on a finite element analysis of the core plate and the holddown bolts and did not credit friction between the core rim and the shroud support. This analysis demonstrated that the mean axial and axial + bending stresses in the holddown bolts meet the ASME Section III bolt stress criteria in the report under typical accident loading. The staff reviewed this report and found it acceptable for referencing in LRAs, as stated by letter dated September 6, 2000. Therefore, the staff requested that the applicant perform an analysis, consistent with the methodology used in Appendix A to BWRVIP-25, and demonstrate that the bolt axial and bending stresses meet the ASME Section III Pm and Pm + Pb stress criteria in the report.

By letter dated February 27, 2006, the applicant provided GE technical report GE-NE-0000-0050-5900P, "Comparative Evaluation of the Monticello Core Plate Rim Hold-down Bolts and BWRVIP-25, Appendix A Analysis," issued February 2006. This analysis demonstrates that the MNGP mean axial and bending core plate holddown bolt stresses, considering holddown bolt stress relaxation, were bounded by the analysis approved in BWRVIP-25. However, the horizontal and vertical loads used in the analysis were considerably smaller than those used in the BWRVIP-25 analysis, resulting from the exclusion of certain accident loads, such as SRV hydrodynamic loads, that were considered in the BWRVIP-25 analysis. The staff requested that the applicant provide justification for the exclusion of these loads. In its response, dated March 31, 2006, the applicant stated that all applicable DBA loads specific to the MNGP core plate were applied and provided the horizontal and vertical accelerations at the core plate level. The applicant justified the exclusion of the hydrodynamic loads on the basis that these are not applicable to the MNGP vessel and internals since the MNGP is a BWR/3 Mark I design. The Mark I torus is structurally isolated from the containment by the use of flexible bellows, and the hydrodynamic loads caused by SRV lift and LOCA are, therefore, not transmitted to the containment or the vessel. The staff found this justification reasonable and acceptable.

The BWRVIP-25 analysis is based on a finite element analysis of the core plate and holddown bolts. It was originally performed to help utilities determine a strategy for core plate inspections, wherein conservative geometric conditions and bounding, postulated worst-case scenarios were considered. Because of the similarities in the MNGP and the BWRVIP-25 plates, the applicant/GE used data from the BWRVIP-25 core plate finite element analysis, an analytical procedure, and a comparison to the MNGP specific core plate and loads to extrapolate the BWRVIP-25 analysis to the MNGP core plate and holddown bolts. The applicant showed that the mean core plate bolt axial and axial + bending stresses met the ASME Section III stress criteria in BWRVIP-25. However, the BWRVIP-25 analysis also indicated that not all holddown bolts are uniformly loaded under horizontal and vertical loading. Based on data shown in BWRVIP-25, the staff also determined that the axial + bending stresses in the highest loaded MNGP holddown bolts could exceed the Pm + Pb stress criterion of BWRVIP-25, but by an insignificant margin. However, the applicant's analysis also included bending stresses in the holddown bolts from core plate bowing, which the BWRVIP-25 analysis did not consider. The staff evaluated the applicant's analysis and concluded that it is acceptable because it conforms with accepted structural analysis practice. Therefore, the staff's concern described in RAI 4.8-2 is resolved.

4.8.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA evaluation of stress relaxation of rim holddown bolts in LRA Section A3.6. In its letter, dated April 10, 2006, the applicant provided a revised USAR supplement description which summarized the results of the analysis provided in the GE technical report. On the basis of its review of the USAR supplement description provided in the applicant's letter, dated April 10, 2006, the staff concluded that the summary description of the applicant's actions to address the stress relaxation of rim holddown bolts adequately describes the analysis characterized in the GE technical report.

4.8.4 Conclusion

The staff concluded that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that the stress relaxation of rim holddown bolts analyses have been projected to the end of the period of extended operation. The staff also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation, sufficient to satisfy the requirements of 10 CFR 54.21(d).

4.9 Reactor Building Crane Load Cycles

4.9.1 Summary of Technical Information in the Application

In LRA Section 4.9, the applicant summarized the evaluation of the reactor building crane load cycles for the period of extended operation. The MNGP reactor building crane system consists of an 85-ton bridge crane. The crane is capable of handling the drywell head, reactor vessel head, pool plugs, and spent fuel pool shipping cask. A refueling service platform, with necessary handling and grappling fixtures, services the refueling area and the spent fuel pool. The reactor building crane system has been modified to incorporate redundant safety features which were not a part of the original design. The modification consists of a new trolley with redundant design features and a capacity of 85 tons on the main hook with redundancy features and an auxiliary 5-ton capacity hook. This modification was implemented for handling heavy loads, both during refueling operations and during operations involving the offsite shipment of spent fuel. Such offsite shipments of fuel can take place when the plant is operating or shut down. The redundant crane was installed to reduce the probability of a heavy load drop to the category of an incredible event. NUREG-0612 suggests that cranes should be designed to meet the applicable criteria and guidelines of Chapter 2-1 of American National Standards Institute (ANSI) B30.2-1976, "Overhead and Gantry Cranes," and of Crane Manufacturers Association of America (CMAA)-70, "Specifications for Electric Overhead Traveling Cranes." The reactor building crane, manufactured before the issuance of CMAA-70 and ANSI B30.2-1976, was designed to meet Electric Overhead Crane Institute (EOCI) 61. Since the evaluation used, as a basis, an expected number of load cycles over the 40-year life of the plant, reactor building crane load cycles are a TLAA.

4.9.2 Staff Evaluation

The staff reviewed LRA Section 4.9, pursuant to 10 CFR 54.21(c)(1)(i), to verify that the analyses will remain valid for the period of extended operation.

In LRA Section 4.9, which is related to the reactor building crane load cycles TLAA, the applicant stated that the current analysis of the fatigue life remains valid for the 60-year extended operating period. It is the staff's understanding that this crane will also handle spent fuel pool shipping casks. A refueling service platform with handling and grappling fixtures services the refueling area and the spent fuel pool.

The staff's review of LRA Section 4.9 identified an area for which it needed additional information to complete its evaluation of the applicant's results. The applicant responded to the staff's RAI as discussed below.

In RAI 4.9-1, dated July 20, 2005, the staff indicated concerns regarding the fatigue analysis for the reactor building crane. Therefore, the staff requested that the applicant provide a fatigue analysis associated with lifts of spent fuel casks and explain how the heavy-load fatigue analysis in LRA Section 4.9 governs the TLAA.

In its response, dated August 16, 2005, the applicant stated the following:

> Section 4.9 of the LRA accounted for cycles due to anticipated lifts of spent fuel casks by the addition of heavy lift cycles. The current analysis conservatively assumed 1,120 cycles (for 40 years of operation) due to lifts of reactor building shield blocks and plugs, the reactor vessel head, the drywell vessel head, the steam separator assembly, and the steam dryer assembly. The difference between 1,120 to 2,000 cycles identified in Section 4.9 includes consideration of additional spent fuel cask lifts, as well as additional current design basis lifts attributable to the license renewal period of extended operation.
>
> The reactor building crane is currently being upgraded from 85 tons to 105 tons in anticipation of spent fuel cask duty. Crane calculations are being performed in accordance with CMAA 70-1975, which identifies stress ranges and allowable cycles. Preliminary calculations demonstrate that the maximum stress range for the upgrade design is less than the allowable stress range for the most severe crane classification operating up to 100,000 cycles. The remaining crane components are being designed with a 5:1 safety factor, which assures that the fatigue threshold for 100,000 cycles will not be exceeded. Assuming that offloading of fuel to a spent fuel storage facility must begin with the next refueling outage at a rate equal to fuel replenishments, as well as spent fuel pool offloading due to decommissioning activities, the total number of additional cycles is not expected to exceed 120. This includes the conservative consideration that both cask placement for acceptance of spent fuel and removal of the loaded cask to the spent fuel transfer vehicle are at fully loaded conditions. This results in a total number of cycles, at maximum load for 60 years of operation, of 1,800 out of 70,000 allowable cycles identified in the LRA.
>
> The crane upgrade calculations have not been completed. Upon completion of the modification analysis, an evaluation will be made to determine the effect, if any, on Section 4.9. If the results are not bounded by the current LRA evaluation/disposition, a revised Section 4.9 will be included with the first Annual LRA Supplement required by 10 CFR 54, § 54.21(b).

In its letter, dated February 28, 2006, the applicant verified that the new calculations were completed and were bounded by the original evaluation; therefore, the staff's concern described in RAI 4.9-1 is resolved.

4.9.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA evaluation of reactor building crane load cycles in LRA Section A3.10. On the basis of its review of the USAR supplement, the staff concluded that the summary description of the applicant's actions to address the reactor building crane load cycles is adequate.

4.9.4 Conclusion

The staff concluded that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the reactor building crane load cycles analyses will remain valid for the period of extended operation. The staff also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation, sufficient to satisfy 10 CFR 54.21(d) requirements.

4.10 Fatigue Analyses of HPCI and RCIC Turbine Exhaust Penetrations

4.10.1 Summary of Technical Information in the Application

In LRA Section 4.10, the applicant discussed the evaluation of the high-pressure coolant injection (HPCI) and RCIC turbine exhaust penetrations fatigue analyses. The applicant evaluated these penetrations for the combination of SRV actuations, LOCA loads, and operational testing of the turbines and concluded that the fatigue usage of the HPCI and RCIC turbine exhaust penetrations will remain below 1.0 during the period of extended operation.

4.10.2 Staff Evaluation

The staff reviewed LRA Section 4.10, 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 applicant evaluated the HPCI and RCIC turbine exhaust penetrations for an increased number of SRV cycles resulting from the 1998 power rerate. The applicant combined the fatigue usage from SRV actuations with the LOCA and operating-basis earthquake (OBE) fatigue usage. The resulting 40-year fatigue usage was well below the 1.0 allowable limit. The applicant multiplied the resulting 40-year fatigue usage by 1.5 to estimate the fatigue usage for 60 years of plant operation. The resulting fatigue usage was well below the 1.0 allowable limit.

Based on its review, the staff concluded that the 1.5 factor provides a conservative estimate of the fatigue usage from SRV actuations, simultaneously with a LOCA and OBE, for the period of extended operation.

The applicant evaluated the fatigue usage from operational testing of the HPCI and RCIC turbines separately. The applicant instrumented these nozzles to measure temperatures during the operational tests and calculated the maximum fatigue usage for the RCIC turbine exhaust penetration from these operational tests. The RCIC turbine exhaust penetration had the greatest fatigue usage. The applicant determined that the total fatigue usage will be acceptable, considering more than five RCIC turbine tests per month over the 60-year extended life. Therefore, the applicant concluded that the analyses of the HPCI and RCIC turbine exhaust penetrations will remain valid for the period of extended operation. The staff agreed that the number of HPCI and RCIC turbine tests will average fewer than five per month.

Based on its review of the applicant's analysis, the staff found that the applicant adequately demonstrated that the fatigue usage of the HPCI and RCIC turbine exhaust penetrations will remain within acceptable limits for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii) requirements.

4.10.3 USAR Supplement

The applicant provided a USAR supplement summary description of its TLAA evaluation of fatigue analyses of HPCI and RCIC turbine exhaust penetrations in LRA Section A3.11. On the basis of its review of the USAR supplement, the staff concluded that the summary description of the applicant's actions to address the fatigue analyses of HPCI and RCIC turbine exhaust penetrations is adequate.

4.10.4 Conclusion

The staff reviewed the applicant's TLAA regarding the fatigue analyses of HPCI and RCIC turbine exhaust penetrations, as summarized in LRA Section 4.10, and concluded that the applicant has provided an acceptable demonstration, 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 also concluded that the USAR supplement contains an appropriate summary description of this TLAA evaluation, sufficient to satisfy the requirements of 10 CFR 54.21(d).

4.11 Conclusion for Time-Limited Aging Analyses

The staff reviewed the information in LRA Section 4, "Time-Limited Aging Analyses." On the basis of its review, the staff concluded that the applicant provided an adequate list of TLAAs, as defined in 10 CFR 54.3. Further, the staff concluded that the applicant 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 aging effects 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 USAR supplement for the TLAAs and found that the USAR supplement contains descriptions of the TLAAs sufficient to satisfy the requirements of 10 CFR 54.21(d). In addition, the staff concluded that no plant-specific exemptions are in effect that are based on TLAAs, as required by 10 CFR 54.21(c)(2).

SECTION 5

REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The NRC staff issued its safety evaluation report (SER) related to the renewal of the operating license for the Monticello Nuclear Generating Plant (MNGP) on April 26, 2006. On May 30, 2006 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 staffs evaluation is documented in a final SER that was issued by letter dated July 28, 2006.

During the 535th meeting of the ACRS, September 7, 2006, the ACRS completed its review of the MNGP license renewal application and the NRC staff's SER. The ACRS documented its findings in a letter to the Commission dated September 19, 2006. A copy of this letter is provided on the following pages of this SER Section.

September 19, 2006

The Honorable Dale E. Klein
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 2005-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE MONTICELLO NUCLEAR GENERATING PLANT

Dear Chairman Klein:

During the 535th meeting of the Advisory Committee on Reactor Safeguards, September 7-8, 2006, we completed our review of the license renewal application for the Monticello Nuclear Generating Plant (MNGP) 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 30, 2006. During our review, we had the benefit of discussions with representatives of the NRC staff and the applicant, Nuclear Management Company, LLC (NMC). We also had the benefit of the documents referenced. This report fulfills the requirements of 10 CFR 54.25 that the ACRS review and report on all license renewal applications.

CONCLUSION AND RECOMMENDATION

The programs established and committed to by the applicant to manage age-related degradation provide reasonable assurance that MNGP 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 NMC application for renewal of the operating license for MNGP should be approved.

BACKGROUND AND DISCUSSION

MNGP is a General Electric Boiling Water Reactor-3 (BWR-3) within a Mark-I containment. The current power rating of 1775 MWt includes a 6.3% power uprate that was implemented in 1998. NMC requested renewal of the MNGP operating license for 20 years beyond the current license term, which expires on September 8, 2010.

In the final SER, the staff documented its review of the license renewal application and other information submitted by NMC and obtained during the audits and inspections 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 NMC application is largely consistent with the Generic Aging Lessons Learned (GALL) Report. All deviations from the approaches specified in the GALL Report are documented in the application. The applicant identified the SSCs that fall within the scope of license renewal and performed a comprehensive aging management review for these SSCs. Based on the results of this review, the applicant will implement 36 AMPs for license renewal including existing, enhanced, and new programs. In the SER, the staff concluded that the applicant has appropriately identified the SSCs within the scope of license renewal and that the AMPs described by the applicant are appropriate and sufficient to manage aging of long-lived passive components that are within the scope of license renewal. We concur with this conclusion.

The staff conducted an inspection and an audit. The inspection verified that the scoping and screening methodologies are consistent with the regulations and are adequately reflected in the application. The audit verified the appropriateness of the AMPs and the aging management reviews. Based on the inspection and audit, the staff concluded that these programs are consistent with the descriptions contained in the NMC license renewal application. The staff also concluded that the existing programs, to be credited as AMPs for license renewal, are generally functioning well and that an implementation plan has been established in the applicant's commitment tracking system to ensure timely completion of the license renewal commitments.

During our meetings with the staff and the applicant, we discussed the adequacy of programs proposed by NMC to manage aging of certain components that are a current focus of the staff and the industry, as described below.

Aging of the drywell shell of MNGP will be managed through the use of the ASME Section XI, Subsection IWE Program. We agree with this approach. Even though this Program does not include ultrasonic testing, this approach was chosen by NMC and accepted by the staff because the plant has several design features that prevent water accumulation behind the shell. During each refueling outage, water leakage is monitored from the refueling seal bellows, the drywell air gap drains, and the sand-pocket drains. The refueling seal is within the scope of license renewal. Ultrasonic inspections performed in the past did not identify any degradation.

MNGP has experienced shroud cracking. This cracking was identified through the required licensee inspection process. Periodic inspections of up to 75% of the shroud welds are performed according to the guidelines of the Boiling Water Reactor Vessel and Internals Project (BWRVIP). Previously identified flaws have exhibited no significant crack growth since the introduction of hydrogen water chemistry at MNGP. Aging of the shroud will continue to be managed by using the guidelines in the BWRVIP-76. We find this AMP appropriate.

The MNGP steam dryers are within the scope of license renewal. A 1998 inspection identified an indication that was not structurally significant. A 2001 inspection revealed no change in this indication and no additional indications were identified. A comprehensive inspection conducted in 2005 to examine areas where steam dryer failures had occurred at other plants found new indications on the dryer shell. These indications were evaluated and determined to be acceptable by the applicant. Another inspection is planned for 2007. Aging of the steam dryers will continue to be managed in accordance with the guidelines in the BWRVIP-139 program. We find this AMP appropriate.

The applicant identified the systems and components requiring TLAAs and reevaluated them for 20 more years of operation. Affected TLAAs included those associated with neutron embrittlement, metal fatigue, irradiation-assisted stress corrosion cracking, environmental qualification of electrical equipment, and stress relaxation of hold-down bolts. The staff concluded that the applicant has provided an adequate list of TLAAs. Further, the staff concluded that in all cases the applicant has met the requirements of the license renewal rule by demonstrating that the TLAAs will remain valid for the period of extended operation, or that the TLAAs have been projected to the end of the period of extended operation, or that the aging effects will be adequately managed for the period of extended operation. We concur with the staff that MNGP TLAAs have been properly identified and that criteria supporting 20 more years of operation have been met.

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 MNGP. The programs established and committed to by NMC provide reasonable assurance that MNGP 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 -NMC application for renewal of the operating license for MNGP should be approved.

Sincerely,
/RA/
Graham B. Wallis
Chairman

References:

1) Safety Evaluation Report Related to the License Renewal of the Monticello Nuclear Generating Plant, dated August 2, 2006.
2) Monticello Nuclear Generating Plant - Application for Renewed Operating License, dated March 16, 2005.
3) Audit and Review Report for Plant Aging Management Programs (AMPs) and Aging Management Reviews (AMRs) -Monticello Nuclear Generating Plant, dated October 12, 2005.
4) Monticello Nuclear Generating Plant, Inspection Report 05000263/2006006, dated March 30, 2006.
5) BWR Vessel and Internals Project, BWR Core Shroud Inspection and Flaw Evaluation Guidelines (BWRVIP-76), EPRI Report TR-114232, November 1999.
6) BWR Vessel and Internals Project, Steam Dryer Inspection and Flaw Evaluation Guidelines (BWRVIP-139), EPRI Report TR-1011463, April 2005.

SECTION 6

CONCLUSION

The staff of the U.S. Nuclear Regulatory Commission (NRC or the staff) reviewed the license renewal application (LRA) for the Monticello Nuclear Generating Plant (MNGP), in accordance with the NRC regulations and NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," dated July 2001. Title 10, Section 54.29, "Standards for Issuance of a Renewed License," of the *Code of Federal Regulations* (10 CFR 54.29) provides the standards for issuance of a renewed license.

On the basis of its review, the staff concluded that the applicant adequately identified those systems and components that are within the scope of license renewal, as required by 10 CFR 54.4(a), and those systems and components that are subject to an aging management review, as required by 10 CFR 54.21(a)(1). The staff also concluded that the applicant demonstrated that the aging effects will be adequately managed so that the intended functions will be maintained consistent with the current licensing basis (CLB) for the period of extended operation, as required by 10 CFR 54.21(a)(3). Further, the staff concluded that the applicant demonstrated that (1) the time-limited aging analyses (TLAAs) will remain valid for the period of extended operation, as required by 10 CFR 54.21(c)(1)(i), (2) the TLAAs had 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 aging effects will be adequately managed for the period of extended operation, as required by 10 CFR 54.21(c)(1)(iii). On the basis of its evaluation of the LRA, the staff determined that the requirements of 10 CFR 54.29(a) have been met, that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the CLB, and that any changes made to the plant's CLB in order to comply with this paragraph are in accord with the Act and the Commission's regulations.

The staff notes that any requirements of Subpart A, "National Environmental Policy Act—Regulations Implementing Section 102(2)," of 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," are documented in draft Supplement 26 to NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Regarding Monticello Nuclear Generating Plant Final Report," dated January 23, 2006.