4.0 Time-limited Aging Analyses

4.1 Identification of Time-Limited Aging Analyses

The applicant described its identification of time-limited aging analyses (TLAAs) in Section 4.0 of the McGuire and Catawba LRA. The staff reviewed this section of the LRA to determine if the applicant had identified the TLAAs and demonstrated that they meet one of the criteria required by 10 CFR 54.21(c)(1). The staff also reviewed the LRA to determine if plant-specific exemptions had been identified by the applicant.

4.1.1 Technical Information in the Application

In Section 4.1 of the LRA, the applicant described the requirements for the technical information to be reported in the LRA regarding time limited aging analyses (TLAAs), as stated in 10 CFR 54.21(c). These include a list of TLAAs, as defined in §54.3, "Definitions," and, if applicable, a list of plant-specific exemptions granted pursuant to §50.12 that are based on TLAAs. The applicant also described the following criteria used to identify TLAAs at both McGuire and Catawba as required by 10 CFR 54.3:

- involve systems, structures, and components within the scope of license renewal as delineated in Section 54.4(a)
- consider the effects of aging
- involve time-limited assumptions defined by the current operating term (for example,40 years)
- were determined to be relevant by the applicant in making a safety determination
- involve conclusions or provide the basis for conclusions related to the capability of the system, structure and component to perform its intended functions, as delineated in Section 54.4(b)
- are contained or incorporated by reference in the CLB

The applicant listed the following specific documents that were reviewed to identify potential TLAAs for both plants:

- Duke/NRC licensing correspondence.
- NUREG-0422, as supplemented, SER for McGuire
- NUREG-0954, as supplemented, SER for Catawba
- UFSARs for both McGuire and Catawba
- ITS for both McGuire and Catawba
- Facility Operating Licenses for both McGuire and Catawba

The document set used for the search is contained in the Electronic Licensing Library (ELL). The ELL contains over 30,000 documents and consists of virtually all correspondence between Duke Energy (formerly Duke Power Company) and the NRC (and its predecessor the Atomic Energy Commission). The information developed from the review of plant-specific source documents was reviewed to determine which calculations and analyses meet all six criteria of §54.3. The analyses and calculations that meet all six criteria were identified as either McGuire-specific or Catawba-specific time-limited aging analyses.

As required by §54.21(c)(1), an evaluation of each time-limited aging analyses must be performed to demonstrate that:

(1) the analyses remain valid for the period of extended operation;
(2) the analyses have been projected to the end of the period of extended operation; or
(3) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

In Sections 4.2 through 4.7 of the lRA, the applicant identified the following TLAAs:

- reactor vessel neutron embrittlement, including analyses for upper shelf energy, pressurized thermal shock, and pressure-temperature limits
- metal fatigue, including analyses of ASME Section III Class 1 component fatigue, fatigue environmental effects, and ASME Section III Class 2 and 3 piping fatigue
- environmental qualification of electrical equipment
- containment liner plate, metal containmnets, and penetration fatigue analysis
- reactor coolant pump flywheel fatigue
- leak-before-break analysis
- depletion of nuclear service water pond volume due to runoff

4.1.2 Staff Evaluation

10 CFR 54.21(c) requires the applicant to provide a list of TLAAs as part of the application for the renewal of a license. The staff reviewed the TLAAs identified by the applicant and described in Sections 4.2 through 4.7 of the LRA to verify that they met the six criteria of 10 CFR 54.3. The staff also sought to determine if the applicant had demonstrated that the analyses remain valid for the period of extended operation; the analyses had been projected to the end of the period of extended operation; or the effects of aging on the intended functions will be adequately managed for the period of extended operation, as required by §54.21(c)(1).

4.1.3 Conclusions

The staff reviewed the information provided in LRA Section 4.1 and concludes that the applicant has adequately identified the TLAAs as required by 10 CFR 54.21(c) and that no 10 CFR 50.12 exemptions have been granted on the basis of the TLAA as defined in 10 CFR 54.3.

4.2 Reactor Vessel Neutron Embrittlement

The application includes three TLAAs for evaluation of the reactor vessel (RV) beltline materials, including: (1) calculation of the end-of-extended-life Charpy upper shelf energy value (C_VUSE values) for each beltline material; (2) calculation of the end-of-extended-life reference temperature value (i.e., RT_{PTS} values) for each beltline material; and (3) a calculation of pressure-temperature (P-T) limits. Each analysis has been updated to consider 20 years of additional plant operation at power. The TLAAs take into account the effects of the additional extended-operating-period neutron irradiation on the previous calculated end-of-life C_VUSE , the RT_{PTS}, and P-T limit values for the McGuire and Catawba reactor vessels, and conservatively base the evaluations through 54 EFPY of power operation.

4.2.1 Upper Shelf Energy

Appendix G to 10 CFR Part 50 requires that reactor vessel beltline materials have C_VUSE values in the transverse direction for the base metal and along the weld for the weld material according to the ASME Code, of no less than 75 ft-lb (102 J) initially, and must maintain C_VUSE values throughout the life of the vessel of no less than 50 ft-lb (68 J). However, C_VUSE values below these criteria may be acceptable if it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that the lower values of C_VUSE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials," provides an expanded discussion regarding the calculations of C_VUSE values and describes two methods for determining C_VUSE values for reactor vessel beltline materials, depending on whether a given reactor vessel beltline material is represented in the plant's reactor vessel material surveillance program (i.e., 10 CFR Part 50, Appendix H program).

4.2.1.1 Technical Information in the Application

Section 4.2.1 of the application addressed the requirement that RV beltline materials have a pre-irradiated C_VUSE of not less than 75 ft-lb (102 J) and maintain a C_VUSE of not less than 50 ft-lb (68 J) throughout the life of the vessel, unless it is demonstrated, in a manner approved by the Director of the Office of Nuclear Reactor Regulation, that lower values of C_VUSE will provide margins of safety against fracture that are equivalent to those required by Appendix G of Section XI of the ASME Code. The applicant stated that the C_VUSE value has been calculated through the period of extended operation, using guidance from Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Pressure Vessel Materials." A value of 54 EFPY was used as the end-of-life criterion for the RV. The application contains the information derived from the C_VUSE analysis. It includes a list of all beltline materials, the weight percent copper in the steel, the end-of-life fluence for the reactor vessel located one-quarter from the vessel's inside surface (i.e., 1/4T thickness of the vessel), and the initial and final C_VUSE values. The applicant concludes that the end-of-life C_VUSE results are above the screening criterion of 50 ft-lb (68 J). The applicant states that the calculations have been projected through the period of extended operation and shown to meet the requirements of 10 CFR 54.21(c)(1)(ii).

4.2.1.2 Staff Evaluation

The applicant summarized the end-of-extended operating period upper shelf energy analyses for the McGuire and Catawba reactor vessel beltline materials in Tables 4.2-1 through 4.2-4 of the LRA. Since all of the C_VUSE values are above the 50 ft-lb (68 J) screening criterion, the staff finds that, with respect to C_VUSE, the Duke RVs have sufficient margin to perform their intended function through the end of the period of extended operation.

By letter dated January 28, 2002, the staff requested, in RAI 4.2-1, the applicant to clarify that Tables 4.2-1 through 4.2-4 of the LRA include the results of the TLAAs for upper shelf energies of beltline nozzle plates/forging materials and nozzle weld materials in the McGuire and Catawba vessels.

In its response dated April 15, 2002, the applicant stated that TLAAs for upper shelf energy (USE) of the reactor vessel beltline shell and nozzle materials are addressed in Section 4.2.1

and Tables 4.2-1 through 4.2-4 of the LRA. During its review, the applicant projected that some of the nozzle region locations would have an estimated 54 EFPY fluence greater than 10^{17} neutrons/cm^2. Therefore, in accordance with 10 CFR 50, Appendix H, the applicant performed an analysis of nozzle region locations and confirmed that they are not the most limiting materials with regard to radiation damage. This analysis is based on a review of the certified material test reports which determined bounding material values for the nozzle region materials. This analysis provides the basis for the responses to this RAI and is available for on-site inspection. All nozzle region materials have been evaluated and a bounding value of USE was calculated. Since none of these nozzle region locations are limiting, no changes to the reactor vessel capsule surveillance program are necessary for license renewal.

In its April 15, 2002, response to RAI 4.2-1, the applicant provided the requested information and noted that, during the preparation of the responses to this RAI, Duke identified errors in C_VUSE values for the bounding nozzles materials, as summarized in Tables 4.2-1 through 4.2-4 of the application. The applicant therefore performed revised C_VUSE value calculations for the bounding nozzle base-metal and weld materials and submitted the revised calculations in Table 4.2-1A, which was included in the applicant's response to RAI 4.2-1. Table 4.2-1A provides the updated C_VUSE values for the bounding nozzle region locations and supercedes the C_VUSE values for the nozzle region materials previously provided in Section 4.2.1 of the LRA. The applicant also provided the requested additional unirradiated C_VUSE values and alloying chemistry in Table 4.2-1C to its April 15, 2002, response to RAI 4.2-1.

The staff performed an independent calculation of the end-of-extended life C_VUSE values for the beltline shell and nozzles materials used to fabricate the McGuire and Catawba RVs. The staff confirmed that none of the beltline nozzles materials were represented in the applicant's reactor vessel material surveillance program (i.e., 10 CFR Part 50 Appendix H Program; refer to AMP B.3.26 for a description of this program). For those RV beltline materials that were not represented in the applicant's reactor vessel material surveillance program, the staff applied Regulatory Position 1.2 of Regulatory Guide 1.99, Revision 2, to estimate the percent loss of C_VUSE as a function of copper content and neutron fluence for the beltline materials, as evaluated using the 54 EFPY end-of extended life fluence. For RV materials represented in the applicant's reactor vessel material surveillance program, the staff applied Regulatory Position 2.2 as its bases for estimating the percentage drop in C_VUSE. The staff confirmed that all RV beltline shell and nozzle materials will continue to satisfy the C_VUSE value requirements of 10 CFR Part 50, Appendix G, through the end-of-extended operating lives for the McGuire and Catawba reactors units. The staff therefore concludes that the applicant's TLAA for calculating the USE values of the McGuire and Catawba RV beltline materials is acceptable because it meets the requirements of 10 CFR 54.21(c)(1)(ii) and will ensure that the RV materials will have adequate upper shelf energy levels and fracture toughness through the end-of-extended operating periods for the McGuire and Catawba reactors units.

4.2.2 Pressurized Thermal Shock

10 CFR 50.61 provides the fracture toughness requirements protecting the reactor vessels of pressurized water reactors against the consequences of pressurized thermal shock (PTS). Licensees are required to perform an assessment of the reactor vessel materials' projected values of the PTS reference temperature, RT_{PTS}, through the end of their operating license. If approved for license renewal this would include TLAAs for PTS up through the end-of-extended operating terms for the McGuire and Catawba units. Upon approval of its application for an

extended period of operation for Catawba and McGuire, this period would be 54 EFPYs. The rule requires each licensee to calculate the end-of-life nil ductility temperature value (i.e., RT_{PTS} value) for each material located within the beltline of the reactor pressure vessel. The RT_{PTS} value for each beltline material is the sum of the unirradiated nil-ductility reference temperature (RT_{NDT}) value, a shift in the RT_{NDT} value caused by exposure to high energy neutron irradiation of the material (i.e., @ RT_{NDT} value), and an additional margin value to account for uncertainties (i.e. M value). 10 CFR 50.61 also provides screening criteria against which the calculated RT_{PTS} values are to be evaluated. For reactor vessel beltline base-metal materials (forging or plate materials) and longitudinal (axial) weld materials, the materials are considered to provide adequate protection against PTS events if the calculated RT_{PTS} values are less than orequal to 270 °F; for reactor vessel beltline circumferential weld materials, the materials are considered to provide adequate protection against PTS events if the calculated RT_{PTS} values are less than or equal to 300 °F. Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials," provides an expanded discussion regarding the calculations of RT_{PTS} values and describes two methods for determining RT_{PTS} for reactor vessel materials, depending on whether a given reactor vessel beltline material is represented in the plants reactor vessel material surveillance program (i.e., 10 CFR Part 50, Appendix H program).

4.2.2.1 Technical Information in the Application

Section 4.2.2 of the LRA addresses the 10 CFR 50.61 requirement that the RV be protected against pressurized thermal shock. The applicant states that the screening criteria in 10 CFR 50.61 are 270 °F for plates, forgings, and axial welds, and 300 °F for circumferential welds. According to the regulation, if the calculated RT_{PTS} values for the beltline materials are less than the screening criteria, then the RV is acceptable with respect to risk of failure during postulated thermal shock transients. In this part of the application, the applicant describes the projected values of RT_{PTS} over the period of extended operation (54 EFPY) to demonstrate that the screening criteria are not violated. The applicant states that this analysis has been carried out and that the results do not exceed the screening criteria. The applicant states that the calculations have been projected through the period of extended operation and shown to meet the requirements of 10 CFR 54.21(c)(1)(ii).

4.2.2.2 Staff Evaluation

The applicant provided its end-of-extended operating PTS assessments for the McGuire and Catawba beltline reactor vessel shell materials in Tables 4.2-5 through 4.2-8 of the application, but did not include the PTS assessments of the beltline nozzle and weld materials. By letter dated January 28, 2002, the staff requested, in RAI 4.2-1, the following information regarding the end-of-extended operating period PTS assessments for the McGuire and Catawba beltline reactor vessel shell and nozzle materials:

1. The corresponding pressurized thermal shock time-limited aging analysis (TLAA) assessments for the nozzle plate/forging materials and nozzle weld materials that were analyzed for upper shelf energy adequacy (as provided for in Tables 4.2-1 through 4.2-4 of the LRA); and

2. The unirradiated Charpy impact data, unirradiated initial RT_{NDT} data (i.e., $RT_{NDT(U)}$ data) . . . and alloying chemistry data (especially copper and nickel contents, as well as

phosphorous and sulfur contents) for the beltline nozzle plates/forging materials and nozzle weld materials in the McGuire and Catawba vessels on the respective dockets for the McGuire and Catawba reactor units (i.e., Dockets Nos. 50-369, 50-370, 50-413 and 50-414), and to provide bases for the data being docketed.

In its response to RAI 4.2-1, dated April 15, 2001, the applicant provided the following additional information and data regarding end-of-extended operating period PTS assessments for the McGuire and Catawba beltline reactor vessel shell and nozzle materials:

- Table 4.2-1B, providing revised PTS assessments for the bounding beltline nozzle base metal and weld materials for the McGuire and Catawba reactor vessels.
- Table 4.2-1C, providing selected unirradiated upper shelf energy, unirradiated RT_{NDT}, and alloying chemistry data for the bounding beltline nozzle base metal and weld materials for the McGuire and Catawba reactor vessels.

The staff performed an independent calculation of the RT_{PTS} values for the McGuire and Catawba beltline reactor vessel shell and nozzle materials, as assessed, based on the projected end-of-extended operating term (54 EFPY) neutron fluences for the materials. In reviewing the applicant's description of the PTS analysis, the staff examined the data and results of the analysis, as summarized in Tables 4.2-5 through 4.2-8 of the LRA and in Tables 4.2-1B and 4.2-1C of the applicant's response to RAI 4.2-1. Although the staff's calculated RT_{PTS} values for the RV beltline shell and nozzle materials were not always consistent with the applicant's calculated RT_{PTS} values, both the staff's and the applicant's PTS analyses confirm that the RT_{PTS} values for the McGuire and Catawba beltline materials will remain under the PTS screening criteria of 10 CFR 50.61 through the end-of-the-extended-operating periods for the units. For the McGuire Unit 1 RV, the staff determined that the lower shell plate longitudinal welds 3-442 A and C are the most limiting materials and calculated the end-of-extended operating term RT_{PTS} value for these materials to be 248 °F. For the McGuire Unit 2 RV, the staff determined that lower shell forging 04 is the most limiting material and calculated the end-of-extended operating term RT_{PTS} value for this material to be 152 °F. For the Catawba Unit 2 RV, the staff determined that lower shell forging 04 is the most limiting material and calculated the end-of-extended operating term RT_{PTS} value for this material to be 62 °F. For the Catawba Unit 2 RV, the staff determined that intermediate shell plate B8605-2 is the most limiting material and calculated the end-of-extended operating term neutron fluence for this material to be 133 °F. All of these materials meet the 10 CFR 50.61 screening criteria for longitudinal weld and base metal materials of 270 °F. Based on these considerations, the staff finds the applicant's TLAAs for protecting the McGuire and Catawba vessels against PTS to be acceptable because the staff confirmed that the RT_{PTS} values for all McGuire and Catawba reactor vessel beltline shell and nozzles materials remain below the screening criteria of
10 CFR 50.61. The staff therefore concludes that the applicant's TLAA for calculating the RT_{PTS} values for the McGuire and Catawba RV beltline materials is acceptable because it meets the requirements of 10 CFR 54.21(c)(1)(ii) and will ensure that the RV materials will have sufficient protection against PTS events through the end-of-extended operating periods for the McGuire and Catawba reactors units.

4.2.3 Pressure-Temperature Limits

The requirements in 10 CFR Part 50, Appendix G, are designed to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The staff evaluates the P-T limit curves based on NRC regulations and guidance. Appendix G to 10 CFR Part 50 requires that P-T limit curves be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code. Appendix G to 10 CFR Part 50 also provides minimum temperature requirements that must be considered in the development of the P-T limit curves. SRP Section 5.3.2 provides an acceptable method of determining the P-T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

Operation of the RCS is also limited by the net positive suction curves for the reactor coolant pumps. These curves specify the minimum pressure required to operate the reactor coolant pumps. Therefore, in order to heat up and cool down, the reactor coolant temperature and pressure must be maintained within an operating window established between the Appendix G pressure-temperature limits and the net positive suction curves of the reactor coolant pumps.

4.2.3.1 Technical Information in the Application

In Section 4.2.3 of the LRA, the applicant addresses the requirement in 10 CFR Part 50, Appendix G, that normal operations, including heatup, cooldown, and transient operating conditions, and pressure-test operations of the RV be accomplished within established pressure-temperature limits. These limits are established by calculations that utilize the materials and fluence data obtained through the unit specific reactor surveillance capsule program.

4.2.3.2 Staff Evaluation

The pressure-temperature (P-T) limits are established by calculations that utilize the materials and fluence data obtained through the unit specific reactor surveillance capsule program. Normally, the P-T limits are calculated for several years into the future and remain valid for an established period of time not to exceed the current operating license expiration. The current P-T limit curves for the McGuire units are acceptable through 16 EFPY of power operation; the current P-T limit curves for the Catawba units are acceptable though 15 EFPY of power operation. 10 CFR 50.90 requires licensees to submit new P-T limit curves for operating reactors for review and have the curves approved and implemented into the Technical Specifications for the reactor units prior to the expiration of the most current P-T limits curves approved in the Technical Specifications. The applicant will be required to submit the extended-period-of-operation P-T limit curves for the McGuire and Catawba RVs and have the curves approved against the criteria of 10 CFR Part 50, Appendix G, and implemented into the Technical Specifications prior to operation of the reactors during the extended operating terms for the units. The staff will evaluate the extended-period-of-operation P-T limit curves for the McGuire and Catawba RVs prior to expiration of the 40-year, current-operating-term P-T limit curves for the units. The staff’s review of the extended-period-of-operation P-T limit curves, when submitted, will ensure that the operations of the RCS for the McGuire and Catawba units will be done in a manner that ensures the integrity of the RCS during the extended periods of operation for the McGuire and Catawba units as required by 10 CFR 54.21(c)(1)(ii).

4.2.4 FSAR Supplement

On the basis of the staff's evaluation described above, the summary description for the RCS TLAAs described in the FSAR Supplement (LRA, Appendix A) are acceptable.

4.2.5 Conclusions

The staff has reviewed the TLAAs regarding the maintenance of acceptable Charpy USE levels for the McGuire and Catawba RV materials and the ability of the McGuire and Catawba reactor vessels to resist failure during postulated PTS events. On the basis of this evaluation, the staff concludes that the applicant's TLAAs for Charpy USE and PTS meet the respective requirements of 10 CFR Part 50, Appendix G, and 10 CFR 50.61 for the McGuire and Catawba RV beltline materials as evaluated to the end-of-extended-operating periods for the McGuire and Catawba units, and therefore satisfy the requirements of 10 CFR 54.21(c)(1)(ii) for the period of extended operation. The staff will evaluate the end-of-extended-operating term P-T limit curves for the McGuire and Catawba reactor units upon submittal by the applicant. The staff's review of the extended-period-of-operation P-T limit curves, when submitted, will ensure that the operations of the RCS for the McGuire and Catawba units will be done in a manner that ensures the integrity of the RCS during the extended periods of operation for the McGuire and Catawba units and that the curves, when submitted, will satisfy the requirements of 10 CFR 54.21(c)(1)(ii) for the period of extended operation.

4.3 Metal Fatigue

A metal component subjected to cyclic loads may fail at a load magnitude less than its ultimate load capacity due to metal fatigue, initiating and propagating cracks in the material. The fatigue life of a component is a function of its material, its environment, and the number and magnitude of the applied cyclic loads. Fatigue was a design consideration for plant mechanical components in the McGuire and Catawba facilities and, consequently, fatigue is part of the current licensing basis for these components. The applicant addressed the TLAA evaluations performed to address thermal fatigue analyses of plant mechanical components in Section 4.3 of the LRA. The staff reviewed this section of the LRA to determine whether the applicant has evaluated the TLAA in accordance with the requirements of 10 CFR 54.21(c)(1).

4.3.1 Summary of Technical Information in the Application

The applicant discussed the evaluation of ASME Section III, Class 1 components in Section 4.3.1 of the LRA. The applicant indicated that the Thermal Fatigue Management Program (TFMP) will be used to manage thermal fatigue of these components during the period of extended operation for both McGuire and Catawba. The elements of the TFMP are described in Section 3.3.1.1 of the LRA. The applicant indicated that the scope of the program includes the following components:

- RCS Class 1 components (including piping connected to the RCS falling under the purview of NRC Bulletins 88-08 and 88-11),
- the replacement steam generators (RSG) Class 1 portion and selected non-Class 1 portions of the RSG,

- components falling within the ISI Program that contain flaws that exceed acceptance standards, but were shown to be acceptable using fracture analyses techniques that used an assumed set of thermal transient cycles, and
- four Catawba non-Class 1 heat exchangers designed based on RCS thermal cycle transient limits.

The applicant described the actions taken to address the issue of environmentally assisted fatigue in Section 4.3.1.2 of the LRA for both McGuire and Catawba. The applicant indicated that it will use Method 2 contained in draft EPRI report, "Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application (MRP-47)," to perform the evaluation. The applicant also indicated that the evaluation will address the fatigue sensitive component locations identified in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components."

The applicant described the evaluation of ASME Section III, Class 2 and 3 piping in Section 4.3.2 of the LRA for both McGuire and Catawba. The applicant also indicated that a number of systems designed to the requirements of ANSI B31.1 are in the scope of license renewal. The applicant concluded that the piping analyses of these systems remain valid for the period of extended operation.

4.3.2 Staff Evaluation

Components of the RCS at both McGuire and Catawba were designed to the Class 1 requirements of the ASME Code. The Class 1 requirements contain explicit criteria for the fatigue analysis of components. Consequently, the applicant identified the fatigue analysis of these components as TLAAs. The staff reviewed the applicant's evaluation of the ASME Class 1 RCS components for compliance with the provisions of 10 CFR 54.21(c)(1).

The specific design criterion for ASME Class 1 components involves calculating the 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 resulting from each transient. The design criterion requires that the CUF not exceed 1.0.

The applicant relies on the TFMP to manage the thermal fatigue design basis of Class 1 components during the period of extended operation. Tables 5-2 and 5-49 of the McGuire UFSAR and Table 3-50 of the Catawba UFSAR contain a list of transient design conditions and associated design cycles used for the design of Class 1 components. By letter dated January 28, 2002, the staff requested, in RAI 4.3-1, that the applicant provide the following data:

- the current number of operating cycles and a description of the method used to determine the number and severity of the design transients from the plant operating history; and
- the number of operating cycles estimated for 60 years of plant operation and a description of the method used to estimate the number of cycles at 60 years.

In its response dated April 15, 2002, the applicant indicated that plant operating conditions are continuously monitored for plant conditions that meet the definition of a transient monitored by the TFMP. The applicant further indicated that the parameters associated with the number and severity of these transients is entered into a database. The applicant provided the current number of cycles for each transient at each unit in Table 4.3-1 of its response. The applicant's response indicated that the projected number of transients would not exceed the number assumed in the design during the period of extended operation. The applicant also identified new transients associated with the McGuire replacement steam generators that will be added to the TFMP. These new transients are also identified in Table 4.3-1.

The applicant also identified the design transients listed in Tables 5-2 and 5-49 of the of the McGuire UFSAR and Table 3-50 of the Catawba UFSAR that are not tracked by the TFMP. The applicant indicated that the estimated design cycles associated with loading and unloading at 5% of full power were based on the assumption of load-follow operation, whereas the plant is operated in the base-load mode. The staff agrees that the number of design cycles listed in the UFSAR tables for these transients are conservative based on the information presented in NUREG/CR-6260 for Westinghouse plants. The applicant also indicated that the step load increase and decrease of 10 percent of full power causes insignificant fatigue and is not counted. This transient was not identified as a major contributor to fatigue usage for Westinghouse plants in NUREG/CR-6260. The staff notes that, although this transient is monitored at the Turkey Point, Surry and North Anna facilities, the responses to staff RAIs regarding the LRAs for these facilities indicates that the number of these design transients is expected to be far less than design number for the period of extended operation. On the basis of monitoring at other facilities and the information presented in NUREG/CR-6260, the staff finds the applicant's statement, that this transient causes insignificant fatigue, a reasonable justification for why the step load increase and decrease of 10 percent of full power is not counted in the TFMP.

The Catawba UFSAR lists a large number of design cycles for charging and letdown flow changes. The applicant's response indicates that these transients cause insignificant fatigue and are not counted. NUREG/CR-6260 contains a discussion of these transients for the newer vintage Westinghouse plant. The discussion indicates that these transients are not normally counted at PWRs, although some PWRs have reported that the actual cycles of these transients are less than the numbers assumed in the design calculations. However, the NUREG/CR-6260 evaluation indicates the fatigue usage at the charging nozzle for these transients is significant when the reactor water environment is considered. The charging nozzle is one of the locations that the applicant will assess for fatigue environmental effects. The assessment for fatigue environmental effects is discussed later in this section of this SER.

The Westinghouse Owners Group (WOG) issued Topical Report WCAP-14575-A, "Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components," to address aging management of the RCS piping. Renewal applicant action item 8 of the accompanying staff SE requests that a license renewal applicant perform an additional fatigue evaluation or propose an AMP to address components labeled I-M and I-RA in Tables 3-2 through 3-16 of WCAP-14575. By letter dated January 28, 2002, the staff requested, in RAI 4.3-3, that the applicant discuss how the TFMP addresses the components labeled I-M and I-RA in Tables 3-2 through 3-16 of WCAP1475-A. In its response dated April 15, 2002, the applicant indicated that WCAP-14575-A had not been considered in the LRA. However, The applicant indicated that it had reviewed WCAP-14575-A in order to respond to the RAI. The applicant indicated that the TFMP manages the thermal fatigue design basis for the components identified in WCAP-14575-A. The applicant's TFMP requires corrective actions to be initiated if the number of cycles exceeds the number assumed in the design. The staff concludes that the components identified in WCAP-14575-A will be adequately addressed by the applicant's TFMP.

The WOG issued the generic Topical Report WCAP-14574-A, License Renewal Evaluation: Aging Management Evaluation for Pressurizers," to address aging management of pressurizers. Renewal applicant action item 1 of the accompanying staff SE requests that a license renewal applicant demonstrate that the pressurizer sub-component CUFs remain below 1.0 for the period of extended operation. Table 2-10 of WCAP 14574-A indicates that the ASME Section III Class 1 fatigue CUF criterion could be exceeded at several pressurizer sub-component locations during the period of extended operation. WCAP-14574-A also identified recent unanticipated transients that were not considered in the original ASME Section III Class 1 fatigue analyses. By letter dated January 28, 2002, the staff requested, in RAI 4.3-4, that the applicant provide the following information:

- An evaluation to confirm that the additional transients discussed in WCAP-14574-A, not considered in the original design, have been addressed at McGuire and Catawba;
- A list of the ASME Section III Class 1 CLB CUFs for the applicable sub-components of the McGuire and Catawba pressurizers specified in Table 2-10 of WCAP-14574-A and the corresponding CUFs for the extended period of operation; and
- A discussion of the impact of the environmental fatigue correlations provided in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue on Fatigue Design Curves of Austenitic Stainless Steels," on the above results.

In its response dated April 15, 2002, the applicant indicated that WCAP-14574-A had not been considered in the LRA. However, the applicant indicated that it had reviewed WCAP-14574-A in order to respond to the RAI. Regarding the first issue, the applicant indicated that modified operating procedures had been implemented at McGuire and Catawba to mitigate the effects of insurge/outsurge. In addition, historical plant instrument data was analyzed to determine the insurge/outsurge history both before and after modification of the operating procedures. The applicant indicated that an analysis including these events found that the design CUFs of all components will remain less than 1.0. By letter dated July 9, 2002, the applicant provided the CUFs for the sub-components listed in Table 2-10 of WCAP-14574-A but did not discuss the impact of the environmental fatigue correlations on these sub-components. Pending completion of the staff's review of the information provided and assessment of the impact of the environmental correlations for these sub-components, this issue is characterized as open item 4.3-1.

The WOG has issued the generic Topical Report WCAP-14577, Revision 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals," to address aging management of the reactor vessel internals. Renewal Applicant Action Item 11 of the accompanying staff SE indicates that the fatigue TLAA of the reactor vessel internals should be addressed on a plant specific basis. In the LRA, the applicant indicates that the TFMP will assure that component fatigue analyses will remain within their design values for the period of extended operation. By letter dated January 28, 2002, the staff requested, in RAI 4.3-2, that the applicant list the transients that contribute to the fatigue usage for each component listed in Table 3-3 of WCAP-14577, Revision 1-A and discuss how the TFMP monitors these transients.

In its response dated April 15, 2002, the applicant indicated that WCAP-14577, Revision 1-A had not been considered in the LRA. The applicant stated that no TLAAs were identified for the McGuire or Catawba reactor internals, and that the reactor vessel internals were designed to ASME Section III, Class 2 criteria which specified no time or cycle dependent requirements for the internals. The applicant did indicate that the rod cluster guide tube pins at McGuire and Catawba were replaced, and the replacement pins were analyzed for fatigue considering a sixty-year design life. The applicant further indicated that the transients that contribute to the fatigue usage are included in the TFMP. The staff considers the applicant's response acceptable.

The applicant's TFMP tracks transients and cycles of RCS components that have explicit design transient cycles to assure that these components stay within their design basis. Generic Safety Issue (GSI)-166, Adequacy of the Fatigue Life of Metal Components," raised concerns regarding the conservatism of the fatigue curves used in the design of the RCS components. Although GSI-166 was resolved for the current 40-year design life of operating components, the staff identified GSI-190, "Fatigue Evaluation of Metal Components for 60-year Plant Life," to address license renewal. The NRC closed GSI-190 in December, 1999, concluding:

> 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 breaks as plants continue to operate. Thus, the staff concludes that, consistent with existing requirements in 10 CFR 54.21, licensees should address the effects of coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.

Section 4.3.1.2 of the LRA discusses the applicant's evaluation of the impact of the reactor water environment on the fatigue life of components. The discussion indicates that the applicant's evaluation will use method 2 contained in draft EPRI Report MRP-47. The applicant provided a discussion of its proposed implementation of the EPRI Report MRP-47 guidelines. The applicant's proposed evaluation will include a sample of 6 to 10 locations which will be selected for assessment. The sample will include the fatigue sensitive component locations identified in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components."

The staff is currently reviewing EPRI Report MRP-47 and has not yet endorsed the guidelines presented in the report. Consequently, by letter dated January 28, 2002, the staff requested, in RAI 4.3-5, that the applicant provide the following additional information regarding the evaluation of reactor water environmental effects:

- Confirmation that the environmental fatigue correlations contained in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue on Fatigue Design Curves of Austenitic Stainless Steels," will be used in the evaluation.

- Design basis usage factors for each of the six component locations listed in NUREG/CR-6260.

- Detailed technical evaluation which demonstrates that the proposed inspections provide an adequate basis for detecting fatigue cracking before such cracking leads to through wall cracking or pipe failure. The detailed technical evaluation was required to be sufficiently conservative to address all uncertainties associated with the technical evaluation (e.g., fatigue crack initiation and detection, fatigue crack size, and fatigue crack growth rate considering environmental factors). As an alternate to the detailed technical evaluation, a commitment was needed to monitor the fatigue usage, including environmental effects, during the period of extended operation, and to take corrective actions, as approved by the staff, if the usage was projected to exceed one. The detailed technical evaluation was required because the applicant had indicated that ASME Section XI flaw evaluation and inspection procedures could be used as an alternate method to manage environmental fatigue. The NRC staff indicated that it has not endorsed a procedure on a generic basis which allows for ASME Section XI inspections in lieu of meeting the fatigue usage criteria.

- Additional data and evaluations which demonstrate that (1) there is sufficient margin in the procedure to account for material variability and experimental data scatter, size effects, surface finish effects and loading history; and (2) environmental effects and surface effects are not independent effects. As an alternative, the applicant's procedure should be revised to eliminate the Z factor. This information was needed because the applicant's procedure indicated that the environmental factor would be adjusted by a Z factor to take credit for moderate environmental effects in the existing ASME fatigue curves. The staff considers the use of the Z factor an open issue regarding implementation of the EPRI procedure.

In its response dated April 15, 2002, the applicant confirmed that the environmental fatigue correlations in NUREG/CR-6583 and NUREG/CR-5704 will be used in the evaluations. Although the applicant indicated that NUREG/CR-6260 locations applicable to McGuire and Catawba correspond to those identified for a newer Westinghouse plant, the applicant did not provide the design usage factors for these locations in its RAI response. However, the applicant did provide, in subsequent electronic correspondence dated May 23, 2002, a table of CUFs for newer-vintage Westinghouse plant locations identified in NUREG/CR-6260. This table was attached as an enclosure to a June 4, 2002, conference call summary, summarized by memorandum dated June 19, 2002 (ML021700621). By letter dated July 9, 2002, the applicant provided these data in official correspondence. The staff's review of these data is ongoing. Additionally, the staff requests the applicant to provide design stresses and fatigue usage factors associated with the Catawba charging system flow changes discussed previously in this SER. Pending the staff's receipt of information pertaining to the Catawba charging flow changes and completion of the staff's review of the environmental impact on the fatigue usage for plant locations identified in NUREG/CR-6260, this issue is characterized as open item 4.3-2.

The applicant agreed not to use the flaw tolerance/inspection procedures specified in Note 1 unless such procedures have been accepted by the NRC. In addition, The applicant agreed to revise the procedure specified in LRA Section 4.3.1.2 to set Z equal to 1.0. The staff finds these commitments acceptable.

ASME Section III, Class 2 and 3 piping design criteria requires that a reduction factor be applied to the allowable bending stress range if the number of full range thermal cycles exceeds 7,000. ANSI B31.1 contains the same requirement. The applicant indicates that two locations at McGuire and Catawba could reach the 7,000 cycle limit during the period of extended operation. By letter dated January 28, 2002, the staff requested, in RAI 4.3-7, that the applicant identify these locations and indicate how the number of expected cycles was determined. The staff also requested that the applicant describe the re-evaluation that was performed to demonstrate that these locations will be acceptable for the period of extended operation.

In its response dated April 15, 2002, the applicant indicated that the starting air compressor discharge piping in the Diesel Generator Starting Air System at McGuire and Catawba is expected to exceed 7,000 cycles during the period of extended operation because of the frequent cycling of the air compressor. The applicant's response indicated that a portion of the drain piping in the Main Steam System at McGuire was projected to exceed 7,000 cycles during the period of extended operation due to significant thermal cycling during startup. In addition, the pressurizer liquid sample piping at Catawba was frequently used to sample boron. The applicant indicated that the stresses in these piping systems were within Code limits after conservative stress range reduction factors were applied. The staff finds the response acceptable because the applicant indicated that the piping systems will continue to meet acceptable Code limits during the period of extended operation.

The LRA does not address the issue of underclad cracks. The WOG submitted for staff review a topical report WCAP-15338, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants (MUHP-6110)" by letter dated March 1, 2001. WCAP-15338 indicates that underclad cracks are confined to forging materials, SA 508 Class 2 and 3. WCAP-15338 also indicates that underclad cracks were observed in SA 508 Class 3 nozzles clad with multiple-layer, strip electrode, submerged-arc welding processes where preheating and post-heating were applied to the first layer but not to the subsequent layers. By letter dated January 28, 2002, the staff requested, in RAI 4.3-6, that the applicant provide additional information regarding the susceptibility of the McGuire/Catawba vessel forgings to underclad cracking. Subsequently, the staff identified the following information in Catawba UFSAR Section 5.3.1.4:

> **<u>Section 5.3.1.4, "Special Controls for Ferretic and Austenitic Stainless Steels," page 5.3-2</u>**
> **Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Allow Steel Components (5/73)"**
>
> <u>Discussion</u>
>
> Westinghouse practices achieve the same purpose as Regulatory Guide 1.43 by requiring qualification of any high head input process, such as the submerged-arc wide-strip welding process and the submerged-arc 6-wire process used on ASME SA-508, Class 2, material, with a performance test as described in Regulatory Position 2 of the guide. No qualifications are required by the regulatory guide for ASME SA-533 material and equivalent chemistry for forging grade ASME SA-508, Class 3, material. The fabricator monitors and records the weld parameters to verify agreement with the parameters established by the procedure qualification as stated in Regulatory Position C.3.

Since RG 1.43 contains guidance for control of stainless steel cladding of low-alloy steel components, the staff concluded that underclad cracking was not a concern for Catawba 1 and 2. The staff was unable to verify that the same controls were in effect for the McGuire units. In its response dated April 15, 2002, the applicant did not provide adequate justification for the staff to conclude that all the McGuire reactor vessel forgings are not susceptible to underclad cracks.

By letter dated June 26, 2002, the staff provided a list of potential open items to the applicant and requested that the applicant provide written responses to resolve those open or confirmatory items that it considered reconcilable. RAI 4.3-6 was characterized as open for the McGuire units. In its response dated July 9, 2002, the applicant provided information that was excerpted from a letter dated May 12, 1980, in which the NRC requested information on the McGuire reactor vessel nozzle base metal material, clad process type, heat input, and manufacturer or subcontractor who fabricated the vessel and applied the nozzle cladding. The applicant also provided the following excerpt from a letter dated July 17, 1980, which transmitted the NRC's safety evaluation of information subsequently provided by Duke in a letter dated June 6, 2002:

> We have determined that the McGuire Unit No. 2 reactor vessel was fabricated by Rotterdam-Nuclear of the Netherlands using procedures for welding and pre- and post-clad heat treatments that increase the potential for underclad cracking. For this reason, we require that augmented ultrasonic examination for underclad cracking be performed on the McGuire unit No. 2 reactor vessel nozzles prior to issuance of an operating license. The inspections should be conducted using techniques that have been designed to detect underclad cracks. These techniques previously have been used at Sequoyah 1, North Anna 2 and Salem 2. The McGuire Unit No. 1 vessel was fabricated by Combustion Engineering using welding heat treat practices expected not to cause underclad cracking. Therefore, we do not require that augmented preservice inspections be performed on the Unit No. 1 vessel. In the future augmented ultrasonic examinations will be required for a reactor vessel whose nozzles were clad in the U. S., but only as part of a program to verify that cladding heat treatments used by U. S. manufacturers do not result in underclad cracking.

Based upon this excerpt from the staff's safety evaluation regarding the reactor vessel nozzles at McGuire, the staff concludes that the applicant need not address this issue for McGuire 1. However, underclad cracking remains a concern for McGuire 2. The applicant is relying upon ultrasonic inspection for resolution of this issue. However, the staff believes that ultrasonic inspection is not effective at detecting defects of the size generated by this phenomenon. Therefore, this issue can be resolved for McGuire 2 only by analysis. For this reason, this issue is characterized as open item 4.3-3 and applies to McGuire 2 only.

As discussed previously, the applicant relies on the TFMP to manage the thermal fatigue design basis to assure that the analyses remain valid for the period of extended operation. The staff review of the TFMP focused on how the program manages fatigue through effective incorporation of the following 10 elements: program scope, preventive or mitigative actions, parameters monitored or inspected, detection of aging effects, monitoring and trending, corrective actions, confirmation process, administrative controls and operating experience.

[Program Scope] The scope of the program includes the reactor coolant pressure boundary Class 1 components including piping connected to the RCS addressed by NRC Bulletins 88-08 and 88-11, replacement steam generators, components evaluated using fracture mechanics analyses, and four Catawba heat exchangers. The staff considers the scope of the program, which includes components with analyses that explicitly addressed thermal fatigue transient limits, acceptable.

[Preventive and Mitigative Actions] The applicant indicates that the TFMP assures the thermal fatigue design basis remains valid for the period of extended operation. The TFMP accomplishes this monitoring and tracking transients used in the fatigue analyses of components. The staff did not identify a need for any further actions.

[Parameters Inspected or Monitored] The program monitors the transients used in the analyses of the components. The staff considers this monitoring appropriate because the program objective is to ensure the analyses remain valid for the period of extended operation.

[Detection of Aging Effects] The program monitors the number of design transients used in the fatigue analysis of components to provide assurance that the fatigue analyses of record remain valid during the period of extended operation. The staff did not identify a need for any further actions.

[Monitoring and Trending] The program monitors the number of design transients used in the fatigue analysis of components. The program also monitors the pressure and temperature profiles of these transients. The monitored values are compared to design values. The staff considers this monitoring appropriate because the program objective is to ensure the analyses remain valid for the period of extended operation.

[Acceptance Criteria] The acceptance criteria are the number of cycles of each transient assumed in the design analyses and the temperature and pressure profiles for each transient. The staff considers this criteria acceptable because the program objective is to ensure the analyses remain valid for the period of extended operation.

[Corrective Actions and Confirmation Process] The applicant indicates that, if the number of transient cycles approaches the assumed bases for the plant design, further analysis will be performed to account for the magnitude of these cycles. The applicant also indicates that the corrective action program is triggered if the temperature or pressure profiles exceed the design limits. The staff's evaluation of the corrective action program and confirmation process is documented is Section 3.0.4 of this SER.

[Administrative Controls] The applicant indicates that implementation procedures are reviewed, approved and maintained as controlled documents in accordance with the procedure control process. The staff's evaluation of the administrative controls is documented in Section 3.0.4 of this SER.

[Operating Experience] The applicant indicates that thermal fatigue transients have been tracked since operation began at both McGuire and Catawba. The staff identified open item 4.3-1 regarding the applicant's response to issues identified in Topical Report WCAP-14574-A. Pending the completion of the staff's review of this issue, the staff is unable to conclude that operating experience has been adequately considered in the program.

4.3.3 FSAR Supplement

The applicant provided a McGuire FSAR Supplement for Section 3.9.2 and a Catawba FSAR Supplement for Section 3.9.3 which indicates that stress range reduction factors were used in the evaluation of ASME Class 2 and 3 piping systems. The applicant also provided a McGuire FSAR Supplement for Section 5.2.1 and a Catawba FSAR Supplement for Section 3.9.1 to indicate that Catawba TFMP will continue to manage thermal fatigue into the period of extended operation. However, the applicant did not describe its commitment to evaluate the effects of the environment on fatigue of reactor coolant system pressure boundary components, and the applicant did not provide a description of its TFMP. Because the these items should be described in a revised FSAR Supplement, this issue is characterized as open item 4.3-4.

4.3.4 Conclusions

On the basis of its evaluation of McGuire and Catawba components the applicant concluded that, the fatigue analysis of ASME Section III, Class 2 and 3 piping will remain valid for the period of extended operation. The applicant also has a TFMP to maintain a record of the transients used in the fatigue analyses of ASME Section III, Class 1 components and other components where thermal fatigue limits were explicitly addressed at McGuire and Catawba, and to ensure that the process will continue during the period of extended operation. The TFMP will provide assurance that the fatigue design of these components remains valid for the period of extended operation.

On the basis of it review, with the exception of open items 4.3-1, 4.3-2 and 4.3-3, the staff concludes the applicant's actions and commitments satisfy the requirements of 10 CFR 54.21(c)(1).

4.4 Environmental Qualification of Electrical Equipment

7The aging (or qualified life) analysis for electrical components included as part of the EQ program (required by 10 CFR 50.49) that involve time-limited assumptions (as defined by the current operating term for the McGuire and Catawba plants, i.e., 40 years) meet the 10 CFR 54.3 definition for TLAAs and are thus considered TLAAs for license renewal. The EQ program [together with other plant programs/processes] have been evaluated [pursuant to 10 CFR 54.21(c)(1)(iii)] to determine that they will adequately manage the effects of aging on the intended function(s) of electrical components for the period of extended operation.

The applicant, in the following four sections of their LRA, described the technical bases and justification for why the McGuire and Catawba EQ Program (together with other plant programs/processes) adequately manages the effects of aging on the intended function(s) of electrical components for the period of extended operation: Section 4.4, "Environmental Qualification (EQ) of Electric Equipment"; Section 4.4.1, "McGuire and Catawba Environmental Qualification Program Background"; Section 4.4.2, "McGuire and Catawba EQ Component Reanalysis Attributes"; and Section 4.4.3, McGuire and Catawba Environmental Qualification Program Evaluation and Technical Basis." The staff reviewed these sections of the LRA to determine whether the applicant has demonstrated that the effects of aging on the intended function(s) of electrical components will be adequately managed (through the McGuire and Catawba EQ Program, together with other plant programs/processes) during the period of extended operation as required by 10 CFR 54.21(c)(1)(iii).

4.4.1 Technical Information in the Application

The McGuire and Catawba EQ Program meet the requirements of 10 CFR 50.49. Section 50.49 defines the scope of components to be included, requires the preparation and maintenance of a list of in-scope components, and requires the preparation and maintenance of a qualification file that includes component performance specifications, electrical characteristics and the environmental conditions to which the components could be subjected. Section 50.49(e)(5) contains provisions for aging that require, in part, consideration of all significant types of aging degradation that can affect component functional capability. Section 50.49(e) also requires replacement or refurbishment of components not qualified for the current license term prior to the end of designated life, unless additional life is established through ongoing qualification. Section 50.49(f) establishes four methods of demonstrating qualification for aging and accident conditions. Sections 50.49(k) and (l) permit different qualification criteria to apply based on plant and component vintage. Compliance with 10 CFR 50.49 provides reasonable assurance that the component can perform its intended functions during accident conditions after experiencing the effects of inservice aging.

The McGuire and Catawba EQ Program manages component thermal, radiation and cyclical aging, as applicable, through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, EQ components not qualified for the current license term are to be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation.

Under 10 CFR 54.21(c)(1)(iii), the McGuire and Catawba EQ Program, which implements the requirements of 10 CFR 50.49, is viewed as an aging management program for license renewal. Important attributes for the reanalysis of an aging evaluation include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria and corrective actions (if acceptance criteria are not met).

The reanalysis of an aging evaluation is normally performed to extend the qualification by reducing excess conservatism incorporated in the prior evaluation. Reanalysis of an aging evaluation to extend the qualification of a component is performed pursuant to 10 CFR 50.49(e) as part of the McGuire and Catawba EQ Program. While a component life limiting condition may be due to thermal, radiation or cyclical aging, the vast majority of component aging limits are based on thermal conditions. Conservatism may exist in aging evaluation parameters, such as the assumed ambient temperature of the component, an unrealistically low activation energy, or in the application of a component (de-energized versus energized). The reanalysis of an aging evaluation is documented according to McGuire and Catawba quality assurance program requirements, which requires the verification of assumptions and conclusions. As already noted, important attributes of a reanalysis include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met). These attributes are discussed below.

Analytical Methods: The McGuire and Catawba EQ Program uses the same analytical models in the reanalysis of an aging evaluation as those previously applied during the prior evaluation. The Arrhenius methodology is an acceptable thermal model for performing a thermal aging evaluation. The analytical method used for a radiation aging evaluation is to demonstrate qualification for the total integrated dose (that is, normal radiation dose for the projected installed life plus accident radiation dose). For license renewal, one acceptable method of establishing the 60 year normal radiation dose is to multiply the 40 year normal radiation dose by 1.5 (that is, 60 years/40 years). The result is added to the accident radiation dose to obtain the total integrated dose for the component. For cyclical aging, a similar approach may be used. Other models may be justified on a case-by-case basis.

Data Collection & Reduction Methods: Reducing excess conservatism in the component service conditions (for example, temperature, radiation, cycles) used in the prior aging evaluation is the chief method used for a reanalysis per the McGuire and Catawba EQ Program. Temperature data used in an aging evaluation should be conservative and based on plant design temperatures or on actual plant temperature data. When used, plant temperature data can be obtained in several ways including monitors used for technical specification compliance, other installed monitors, measurements made by plant operators during rounds, and temperature sensors on large motors (while the motor is not running). A representative number of temperature measurements are conservatively evaluated to establish the temperatures used in an aging evaluation. Plant temperature data may be used in an aging evaluation in different ways, such as (a) directly applying the plant temperature data in the evaluation or (b) using the plant temperature data to demonstrate conservatism when using plant design temperatures for an evaluation. Any changes to material activation energy values as part of a reanalysis are to be justified on a plant-specific basis. Similar methods of reducing excess conservatism in the component service conditions used in prior aging evaluations can be used for radiation and cyclical aging.

Underlying Assumptions: McGuire and Catawba EQ Program component aging evaluations contain sufficient conservatism to account for most environmental changes occurring due to plant modifications and events. When unexpected adverse conditions are identified during operational or maintenance activities that affect the normal operating environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions.

Acceptance Criteria & Corrective Action: Under the McGuire and Catawba EQ Program, the reanalysis of an aging evaluation could extend the qualification of the component. If the qualification cannot be extended by reanalysis, the component must be refurbished, replaced, or requalified prior to exceeding the period for which the current qualification remains valid. A reanalysis is to be performed in a timely manner (that is, sufficient time is available to refurbish, replace, or requalify the component if the reanalysis is unsuccessful).

In addition to these important attributes for reanalysis of the aging evaluation, the McGuire and Catawba EQ Program includes the following attributes:

[Scope] The McGuire and Catawba EQ Program includes certain electrical components that are important to safety and could be exposed to harsh environment accident conditions, as defined in 10 CFR 50.49.

[Preventive Actions] Section 50.49 does not require actions that prevent aging effects. McGuire and Catawba EQ Program actions that could be viewed as preventive actions include (a) establishing the component service condition tolerance and aging limits (for example, qualified life or condition limit); and (b) where applicable, requiring specific installation, inspection, monitoring or periodic maintenance actions to maintain component aging effects within the bounds of the qualification basis.

[Parameters Monitored or Inspected] The qualified life of a component in the McGuire and Catawba EQ Program is not based on condition or performance monitoring. However, pursuant to Regulatory Guide 1.89, Rev. 1, such monitoring programs are an acceptable basis to modify a qualified life through reanalysis. 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.

[Detection of Aging Effects] Section 50.49 does not require the detection of aging effects for in-service components. As implemented by the McGuire and Catawba EQ Program, 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.

[Monitoring and Trending] Section 50.49 does not require monitoring and trending of component condition or performance parameters of in-service components to manage the effects of aging. McGuire and Catawba EQ Program actions that could be viewed as monitoring include monitoring how long qualified components have been installed. Monitoring or inspection of certain environmental, condition, or component parameters may be used to ensure that a component is within the bounds of its qualification basis or as a means to modify the qualification.

[Acceptance Criteria] Section 50.49 acceptance criteria, as implemented by the McGuire and Catawba EQ Program, are that an inservice EQ component is maintained within the bounds of its qualification basis, including (a) its established qualified life and (b) continued qualification for the projected accident conditions. Section 50.49 requires refurbishment, replacement, or requalification prior to exceeding the qualified life of each installed device. When monitoring is used to modify a component qualified life, plant-specific acceptance criteria are established based on applicable 10 CFR 50.49(f) qualification methods.

[Corrective Action & Confirmation Process] If a component in the McGuire and Catawba EQ Program is found to be outside the bounds of its qualification basis, corrective actions are implemented in accordance with the station's corrective action program. When unexpected adverse conditions are identified during operational or maintenance activities that affect the environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, 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, the affected component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions. Confirmatory actions, as needed, are implemented as part of the McGuire and Catawba corrective action program, pursuant to 10 CFR 50, Appendix B.

[Administrative Controls] The McGuire and Catawba EQ Program is implemented through the use of station policy, directives, and procedures. The McGuire and Catawba EQ Program will continue to comply with 10 CFR 50.49 throughout the renewal period, including development and maintenance of qualification documentation demonstrating reasonable assurance that a component can perform required functions during harsh accident conditions. McGuire and Catawba EQ Program documents identify the applicable environmental conditions for the component locations. McGuire and Catawba EQ Program qualification files are maintained at McGuire and Catawba in an auditable form for the duration of the installed life of the component. McGuire and Catawba EQ Program documentation is controlled under the station's quality assurance program.

[Operating Experience] EQ programs include consideration of operating experience to modify qualification bases and conclusions, including qualified life. Compliance with 10 CFR 50.49 provides reasonable assurance that components can perform their intended functions during accident conditions after experiencing the effects of inservice aging.

Based on the above described attributes for reanalysis of the aging evaluation and EQ program, the applicant concluded that the McGuire and Catawba EQ Program has been demonstrated to be capable of programmatically managing the qualified lives of the components falling within the scope of the program for license renewal. The continued implementation of the McGuire and Catawba EQ Program provides reasonable assurance that the aging effects will be managed and that components falling within the scope of the EQ Program will continue to perform their intended functions for the period of extended operation. This result meets the requirement of 10 CFR 54.21(c)(1)(iii).

4.4.2 Staff Evaluation

The staff reviewed the information in Sections 4.4, 4.4.1, 4.4.2, and 4.4.3 of the LRA to determine whether the applicant has demonstrated that the effects of aging on the intended function(s) of electrical components will be adequately managed [through their existing EQ program (together with other plant programs/processes) during the period of extended operation as required by 10 CFR 54.21(c)(1)(iii).

The applicant is required to have an EQ program that meets the requirements of 10 CFR 50.49. The staff, therefore, agrees with the applicant's conclusion that their EQ program {together with other plant programs/processes} will adequately manage the effects of aging on the intended function(s) of electrical components for the period of extended operation. The staff therefore concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that their EQ program (together with other plant programs/processes) will adequately manage the effects of aging on the intended functions and can be considered an acceptable aging management program for license renewal.

<u>Generic Safety Issue (GSI) -168, Environmental Qualification of Electrical Equipment</u>

This GSI was developed to address environmental qualification of electrical equipment. By letter from C. Grimes (NRC staff) to D. Walters (NEI), dated June 2, 1998, the staff issued the following guidance to the industry:

- GSI-168 issues have not been identified to a point that a license renewal applicant can be reasonably expected to address these issues, specifically at this time; and
- An acceptable approach is to provide a technical rationale demonstrating that the CLB for EQ will be maintained in the period of extended operation.

For the purpose of license renewal, there are three options for addressing issues associated with a GSI, as discussed in the statement of considerations (SOC) accompanying the final rule, 60 FR 22,484, May 8, 1995:

1) If the issue is resolved before the renewal application is submitted, the applicant can incorporate the resolution into the LRA.

2) An applicant can submit a technical rationale that demonstrates that the CLB will be maintained until some later point in the period of extended operation, at which time one or more reasonable options would be available to adequately manage the effects of aging.
3) An applicant can develop a plant-specific aging management program that incorporates a resolution to the aging issue.

The applicant did not provide information in Section 4.4 to address the GSI-168 options. In electronic correspondence from the applicant, dated June 17, 2002 (ADAMS ML Accession number not yet available), Duke provided the following account of GSI-168 as it applies to McGuire and Catawba:

> As discussed in SECY-93-049, the staff reviewed significant license renewal issues and found that several were related to environmental qualification (EQ). A key aspect of these issues was whether the licensing bases should be reassessed or enhanced in connection with license renewal, and whether this reassessment should be extended to the current license term. In late 1993, the Commissioners instructed the staff that the current EQ licensing basis must be used in the license renewal period and that any EQ concerns identified by the staff during the review of EQ for license renewal should be evaluated for the effect on current licenses, independent of license renewal.
>
> The NRC Staff's EQ Task Action Plan (EQ-TAP) was initiated to address the adequacy of current EQ practices. Upon completion of the EQ-TAP review, the focus of Staff concerns was limited to issues related to the adequacy of accelerated aging practices in existing qualifications, and the lack of a "feedback mechanism" in EQ programs (i.e., programmatic requirements to determine the current condition of EQ equipment so that it can be evaluated against the assumptions and parameters for qualification). The EQ-TAP was subsequently closed and six remaining open issues were incorporated into GSI 168 for management tracking purposes. The EQ-TAP review did not identify any generic safety issues related to these six open issues.
>
> NRC guidance for addressing GSI 168 for license renewal is contained in a June 1998 letter to NEI. In this letter, the NRC states:
>
> With respect to addressing GSI 168 for license renewal, until completion of an ongoing research program and staff evaluations, the potential issues associated with GSI 168 and their scope have not been defined to the point that a license renewal applicant can reasonably be expected to address them at this time. Therefore, an acceptable approach described in the SOC is to provide a technical rationale demonstrating that the current licensing basis for EQ pursuant to 10 CFR 50.49 will be maintained in the period of extended operation. Although the SOC also indicates that an applicant should provide a brief description of one or more reasonable options that would be available to adequately manage the effects of aging, the staff does not expect an applicant to provide the options at this time.
>
> Environmental qualification evaluations of electrical equipment are identified as time-limited aging analyses for McGuire and Catawba. The McGuire and Catawba EQ program evaluations contained in Section 4.4 of the Application are considered to be the technical rationale that the current licensing basis will be maintained during the period of extended operation. Consistent with the above NRC guidance, no additional information is required to address GSI 168 in a renewal application at this time.

By letter dated July 9, 2002, the applicant provided this same response in official correspondence. Therefore, the staff finds that the applicant has submitted, in accordance with the SOC, a technical rationale that demonstrates that the CLB will be maintained until some later point in the period of extended operation, at which time one or more reasonable options would be available to adequately manage the effects of aging. However, the staff requests that the applicant also indicate that it would monitor updates to NUREG-0933, "A Prioritization of

Generic Safety Issues," for revisions to GSI-168 during the review of its application, or that it would supplement its license renewal application if the issues associated with GSI-168 become defined such that providing the options or pursuing one of the other approaches described in the SOC becomes feasible. Pending the staff's receipt of this information, this issue is characterized as confirmatory item 4.4-1.

4.4.3 FSAR Supplement

The applicant, in the LRA, Appendix A, pages A.1-5 and A.2-7, states that the existing EQ process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This statement is consistent with the conclusion that plant EQ programs, which implement the requirements of 10 CFR 50.49, are viewed as acceptable aging management programs for license renewal under 10 CFR 54.21(c)(1)(iii). This statement thus provides a summary description of the programs and activities for the evaluation of TLAA for the period of extended operation for electrical components, meets the requirements of 10 CFR 54.21(d), and is considered acceptable.

4.4.4 Conclusions

The staff has reviewed the information in Sections 4.4, 4.4.1, 4.4.2, and 4.4.4 of the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated that the effects of aging on the intended function(s) of electrical components (that meet the definition for TLAA as defined in 10 CFR 54.3) will be adequately managed during the period of extended operation as required by 10 CFR 54.21(c)(1)(iii). The staff concludes that the FSAR supplement contains a summary description of the programs and activities for the evaluation of TLAA for the period of extended operation as required by 10 CFR 54.21(d). As discussed in 4.4.2, the applicant provided information that describes how issues associated with GSI-168 are being addressed, but the staff requests that the applicant also indicate that it would monitor updates to NUREG-0933, "A Prioritization of Generic Safety Issues," for revisions to GSI-168 during the review of its application, or that it would supplement its license renewal application if the issues associated with GSI-168 become defined such that providing the options or pursuing one of the other approaches described in the SOC becomes feasible. Pending the staff's receipt of this statement in official correspondence, this issue is characterized as confirmatory item 4.4-1.

4.5 Concrete Containment Tendon Prestress

4.5.1 Summary of the Technical Information in the Application

The applicant stated that this topic is not applicable to either the McGuire or the Catawba Nuclear Station Ice Condenser containments. Ice condenser containments do not use prestressed tendons.

4.5.2 Staff Evaluation

The staff concurs with the applicant that this topic is not applicable, and prestress of the concrete containment tendons at the McGuire and Catawba plants is therefore not a TLAA.

4.5.3 Conclusion

The staff finds the applicant's statement that this topic is not applicable to the McGuire and Catawba Nuclear plants, acceptable.

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

4.6.1 Technical Information in the Application

The applicant stated that McGuire and Catawba have ice-condenser metal containments, and therefore do not have containment liner plates, like pre-stressed concrete containments. The topic of fatigue analysis for containment liner plates is therefore not applicable to these plants.

The McGuire and Catawba ice condenser containments are steel containment vessels (SCVs), described in Section 2.4 of the LRA. The Design Code of Record for the McGuire SCV is the ASME "Boiler & Pressure Vessel Code," Section III, Subsection B, 1968 Edition, including all addenda and code cases through the Summer of 1970. The Code of Record for Catawba is the "ASME Boiler & Pressure Vessel Code," Section III. Subsection NE, 1971 Edition, including all addenda through the summer of 1972.

The SCV contain piping through-wall hot and cold mechanical penetration assemblies. Typical hot penetration assemblies are shown in the McGuire and Catawba UFSARs. The hot penetrations consist of the process line and flued head, the guard pipe and the expansion bellows. The bellows are designed to accommodate process line thermal expansions, and displacements between the SCV and the Reactor Building due to cyclic thermal expansion, seismic movements and containment test conditions, and to act as barriers against the release of fission products during design basis events. Fatigue is a progressive failure of a structural part under repeated, cycling or fluctuating loads. Because of the bellows design, the bellows absorb the cyclic piping loads that could cause fatigue and are not transferred to the SCV. Therefore, no fatigue analysis was required for the SCV, and containment fatigue is not a TLAA for either McGuire or Catawba.

All bellows expansion joints are of two-ply construction with a wire mesh between plys for testability of the bellows and the bellows welds to the piping. The McGuire bellows were manufactured, installed and examined in accordance with paragraph NC-3649 of the ASME Code, Section III, 1971 Edition. The design requirements are contained in McGuire engineering documents. As part of the design, the Code required the manufacturer to consider combined stresses due to pressure and relative displacement due to thermal expansion. The cyclic life data for the bellows was based on actual tests, where bellows designs similar to those installed were cycled to failure. A search of the applicant's engineering records did not locate any manufacturer's records for a fatigue calculation on the original design of the McGuire bellows. During later modifications at McGuire, the bellows manufacturer reviewed the design for revised feedwater penetration movements, and determined that these were good for over 32,000 cycles, considerably in excess of the number of cycles that the bellows would see under normal operating conditions.

For Catawba, the bellows assemblies were manufactured, installed and examined in accordance with paragraph NC-3649 of the ASME Code, Section III, 1974 Edition. The design requirements for these bellows are contained in Catawba engineering documents. The manufacturer has provided calculations to the applicant for the cyclic life evaluation of the penetrations. These cyclic life values were used by the manufacturer to demonstrate that the design met the Code requirements.

For McGuire and Catawba, the applicant stated that the fatigue analysis of the bellows was determined not to be relevant in making any safety determination. On this basis the fatigue of bellows is not a TLAA because Criterion 4 of 10 CFR 54.3 definition of a TLAA was not met. However, the aging effect which could result from cyclic fatigue, cracking, has been identified as an aging effect for the bellows, requiring management for the period of extended operation. Local leak rate testing has been identified as the general program that includes managing of cracking of the bellows. Local leak rate testing is discussed as part of Appendix B.3.8 Containment Leak Rate Testing Program, of the LRA.

4.6.2 Staff Evaluation

By letter dated January 28, 2002, the staff requested, in RAI 4.6.1, the applicant to provide a detailed justification for determining that a fatigue TLAA was not required for the SCV for loadings resulting from operating transients, peak containment internal pressure resulting from the design basis LOCA, design basis safe shutdown earthquake (SSE) and leakage rate testing, in addition to the loading resulting from the transient expansions of the bellows. In its response dated March 11, 2002, the applicant stated that the penetration bellows are provided to absorb the loads associated with thermal expansion during operational transients as well as loads induced during the containment leak testing. Peak containment internal pressure resulting from the design basis LOCA at the design basis SSE are one time occurrences and not cyclic loads that could cause fatigue failure. The SCV is, therefore, not subjected to cyclic loading and as a result, no fatigue analysis was necessary or performed. The staff finds the response to the RAI acceptable and considers the issue resolved.

Operating experience with containment bellows at both McGuire and Catawba, as reported in Appendix B-3.8 of the Application, indicates that leaks were detected during containment leakage tests within 20 years of the start of plant operations. During a conference call between the staff and the applicant on November 20, 2001, summarized by memorandum dated January 10, 2002, the applicant stated that 20 leaking bellows at McGuire and 3 leaking bellows at Catawba were identified during testing. However, these bellows were not replaced as long as leakage did not exceed TS surveillance acceptance criteria. Since the bellows were not replaced, root cause evaluations were not performed to determine the cause (fatigue or SCC) of the leakage. Therefore, the applicant indicated that bellows leakage during the tests could not be attributed definitively to cracking by fatigue and that some other cause may be responsible. By letter dated January 28, 2002, the staff requested, in RAI 4.6-2, that the applicant provide the root cause of the cracking (leakage) since the vendors of the bellows performed cyclic fatigue life evaluations and stated that the life of the bellows is well beyond what the bellows would experience during 40 years of normal plant operation.

In its response to RAI 4.6-2, dated March 11, 2002, the applicant stated that since leakage of the bellows is not attributed to cyclic fatigue, the vendor-analyzed cyclic life remains valid for the period of extended operation. The applicant stated that the leakage during the tests could be attributed to transgranular stress-corrosion cracking from contact with a chlorine environment and other causes such as manufacturing process defects, improper installation, and damage incurred during construction or maintenance activities. The potential leakage that could result from any one of these causes is managed by the Containment Leak Rate Testing Program.

This program is identified in Table 3.5-1 of the LRA as a program for managing bellows cracking which would manifest itself during the leakage testing. The staff finds the response to this RAI acceptable and considers this issue resolved.

During the conference call on November 20, 2001, the applicant stated that the calculations and analyses for bellows were not considered relevant in making a safety determination, and that aging of these components would be managed by an aging management program. By letter dated January 28, 2002, the staff requested, in RAI 4.6-3, the applicant to clarify this statement. In its response dated March 11, 2002, the applicant stated that a cyclic analysis of the bellows had been originally performed, but the number of cycles to failure was too large to preclude any safety judgement based on this number. Since this analysis was not used as the basis for any safety judgement, the analysis does not meet Criterion 4 of 10 CFR 54.3 for the definition of a TLAA as defined in 10 CFR 54.3. Because the function of the bellows is within the scope of license renewal, and leaks have been observed at both McGuire and Catawba, cracking has been identified as an aging effect for bellows in Table 3.5-1 of the Application. Aging of penetration bellows will therefore be managed under the containment leak rate testing program, discussed in Appendix B.3.8 of the LRA. The staff's evaluation of the AMP is documented in Section 3.0.3.4 of this SER. Since the containment leak rate testing program will reveal leakage (cracks) caused by both fatigue and SCC, the staff finds this response acceptable and considers this issue resolved.

4.6.3 FSAR Supplement

The applicant has not provided a supplement to the FSAR, since no new information regarding Section 4.6 was provided in the Application.

4.6.4 Conclusion

On the basis of its review, and the responses to the staff requests for additional information, the staff concludes that the applicant has provided adequate information and reasonable assurance to demonstrate that, pursuant to 10 CFR 54.21(c)(iii), the effects of aging of the containment penetration bellows will be adequately managed for the period of extended operation.

4.7 Other Plant-specific TLAAs

4.7.1 Reactor Coolant Pump Flywheel Fatigue

4.7.1.1 Technical Information in the Application

The applicant has addressed the TLAA related to fatigue of the RCP flywheel in Section 4.7.1 of the LRA. The reactor coolant pump motors at McGuire and Catawba are of the same design. The reactor coolant pump motors are large, vertical, squirrel cage, induction motors. The motors have flywheels to increase rotational inertia, thus prolonging pump coastdown and assuring a more gradual loss of main coolant flow to the core in the event that pump power is lost. The flywheel is mounted on the upper end of the rotor, below the upper radial bearing and inside the motor frame. The aging effect of concern is fatigue crack initiation in the flywheel bore keyway from stresses due to starting the motor. Therefore, this topic is considered as a TLAA for license renewal based on the criteria contained in 10 CFR 54.3.

4.7.1.2 Staff Evaluation

The licensee estimates that the existing analysis is valid for the period of extended operation, meeting the requirements of 10 CFR 54.21(c)(1).

To estimate the magnitude of fatigue crack growth during plant life, an initial radial crack length of 10 percent of the distance through the flywheel (from the keyway to the flywheel outer radius) was conservatively assumed. The analysis assumed 6000 cycles of pump starts and stops for a 60-year plant life. Reaching 6000 starts in 60 years would require a pump start on average every 3.7 days. Since a pump start normally occurs every 200 to 300 days, on average, the design of the reactor coolant pump flywheels is conservative. In addition, crack growth from postulated flaws in each flywheel is only a few mils [Reference 4.8-1]. The staff concurs with the applicant's assessment and the assumptions made in arriving at the above estimate of pump starts.

4.7.1.2 FSAR Supplement

The staff has reviewed the changes in the FSAR Supplement to existing Section 3.5.2.1 of the UFSAR, provided in Appendices A-1 and A-2 of the LRA for McGuire and Catawba, respectively, and has confirmed that these changes are appropriate because they reflect the validity of the analysis for the period of extended operation.

4.7.1.3 Conclusion

Because the applicant has demonstrated that the existing analysis for the reactor coolant pump flywheel is valid for the extended period of operation, the staff concludes that the applicant has provided an acceptable TLAA involving components of the RCP flywheel, as defined in 10 CFR 54.21(c)(1)(i).

4.7.2 Leak-Before-Break

The applicant's leak-before-break analysis is provided in Section 4.7.2 of the LRA.

4.7.2.1 Technical Information in the Application

The successful application of LBB to the McGuire reactor coolant system primary loop piping is described in WCAP-10585, "Technical Basis for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for McGuire Units 1 and 2. "Likewise, the successful application of LBB to the Catawba reactor coolant system primary loop piping is described in WCAP-10546, "Technical Basis for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Catawba Units 1 and 2." These reports provide the technical basis for evaluating postulated flaw growth in the main reactor coolant system piping under normal plus faulted loading conditions.

The applicant stated that there are two considerations for the LBB analysis. The first analysis consideration is that the material properties of the cast austenitic stainless steel can change over time. Cast austenitic stainless steels used in the reactor coolant system are subject to thermal aging during service. This thermal aging causes an elevation in the yield strength of the material and a degradation of the fracture toughness, the degree of degradation being a function of the level of ferrite in the material. Thermal aging in these stainless steels will continue until a saturation or fully aged point is reached.

NRC-approved WCAP-10456, "The Effects of Thermal Aging on the Structural Integrity of Cast Stainless Steel Piping for Westinghouse Nuclear Steam Supply Systems," presented a detailed study of the effects of thermal aging on piping integrity. This report concluded that the thermal aging process does not significantly change the failure characteristics of the cast stainless steel piping. WCAP-10585 (McGuire) and WCAP-10546 (Catawba) used the findings of this report to make the determination that the material properties in WCAP-10456 were bounding for McGuire and Catawba. Fully aged, lower bounding data was used in performing the LBB evaluation. Additionally, during the license renewal review, the lower bound data in WCAP-10456 was compared to the lower bound data in NUREG 6177, "Assessment of Thermal Embrittlement of Cast Stainless Steels," and found to be comparable. Therefore, because the original analysis supporting LBB relied on fully aged stainless steel material properties, the analysis does not have a material property time-dependency that requires further evaluation for license renewal.

The second analysis consideration is the accumulation of actual fatigue transient cycles over time that could invalidate the fatigue flaw growth analysis that was done as part of the original LBB analysis. A review of the accumulation of the applicable fatigue transient cycles is considered to meet the TLAA definition. This review was done within the scope of the thermal fatigue management program. The applicant stated that the continued implementation of the thermal fatigue management program provides reasonable assurance that thermal fatigue will be managed for the Class I components such that they will continue to perform their intended function(s) for the period of extended operation.

4.7.2.2 Staff Evaluation

In the LRA regarding LBB, the applicant intended to demonstrate through qualitative assessment that the plant-specific thermal fatigue management program is capable of programmatically managing the assumptions, including the fatigue cycles, in the existing LBB analyses for the period of extended operation. The staff confirmed that the LBB applications for the primary loop piping were approved by the NRC on April 7, 1987, for Catawba 1; on April 23, 1985, for Catawba 2; and on May 5, 1986, for McGuire 1 and 2. The LBB analyses, which provided technical bases for these approved LBB applications, considered the thermal aging of the cast austenitic stainless steel material of the piping, and assuming 40 years of operation. Since the primary loop piping contains cast stainless steel material, the LBB application is a TLAA for both plants.

The thermal aging of the cast stainless steel material has been identified as an issue to be reevaluated. This reevaluation revealed that the original LBB analyses had employed the thermal aging properties documented in WCAP-10456, "The Effects of Thermal Aging on the Structural Integrity of Cast Stainless Steel Piping for Westinghouse Nuclear Steam Supply Systems," which bounded the aging material data for Catawba and McGuire. In addition, the applicant performed a comparison of the material aging information in WCAP-10456 with the more recent information in NUREG-6177, and found that the WCAP-10456 toughness data, after long-term aging considering fluence, time, operating temperature, chemical composition, and ferrite content, were bounding. The staff has examined the information in the above-mentioned documents and agreed with the applicant's conclusion that fully-aged, lower bounding material property was used in the original LBB analyses. Hence, the properties for the cast stainless steel piping material are acceptable because they will not degrade below the fully-aged properties in the extended period of operation.

For the rest of the primary loop piping materials, instead of revising the original analyses by taking into account the fatigue transient cycles for the period of extended operation, the applicant relies on the plant-specific thermal fatigue management program to ensure that the accumulation of the applicable fatigue transient cycles over time would not invalidate the fatigue flaw growth analysis that was performed as part of the original LBB analyses. With this program in place, which calls for constant review of the accumulation of applicable fatigue transient cycles, the applicant concluded that "the continued implementation of the thermal fatigue management program provides reasonable assurance that thermal fatigue will be managed for the Class 1 components such that they will continue to perform their intended function(s) for the period of extended operation." The staff has reviewed the thermal fatigue management program and determined that the three monitoring actions of the program are adequate to monitor the applicable set of transients and their limits, and to count the actual thermal cycle transients to ensure that it is within the allowable limits of the defined transients.

In the event that the pressure and temperature profile for a specific transient is outside the parameters for the defined transient set, or the actual cycle count for a transient set is approaching or exceeding the cycle limit assumed in the original LBB analyses, the applicant proposed to take corrective actions such as conducting ISI activities, implementing plant modifications, and performing revised analyses. The staff considers these measures appropriate and agrees with the applicant's conclusion that this TLAA is in accordance with 10 CFR 54.21(c)(1)(ii), and the continued implementation of the thermal fatigue management program provides reasonable assurance that thermal fatigue will be managed for the primary loop piping and components such that it will continue to perform its intended function for the period of extended operation.

4.7.2.3 Conclusion

The properties for the cast stainless steel piping material are acceptable because they will not degrade below the fully-aged properties in the extended period of operation. Furthermore, the thermal fatigue management program is adequate to ensure that allowable limits are maintained. The applicant has proposed to take appropriate corrective actions if the pressure and temperature profile for a specific transient is outside the parameters for the defined transient set, or the actual cycle count for a transient set is approaching or exceeding the cycle limit. Therefore, the staff concludes that the applicant has provided an acceptable TLAA regarding LBB and meets 10 CFR 54.21 (c)(i)(ii).

4.7.3 Depletion of Nuclear Service Water Pond Volume Due to Run-Off

The depletion of Nuclear Service Water Pond Volume due to run-off time-limited aging analysis is not applicable at McGuire. The drainage area serving the McGuire Nuclear Service Water Ponds is such that the run-off and resulting sedimentation are negligible. The volume of the McGuire Nuclear Service Water Pond has been previously reviewed and accepted by the NRC in the initial McGuire Safety Evaluation Report, Section 4.2 [Reference 4.8-2].

The depletion of Nuclear Service Water Pond Volume due to run-off TLAA is applicable to Catawba, which is provided in Section 4.7.3 of the LRA.

4.7.3.1 Technical Information in the Application

The standby nuclear service water (SNSW) pond is a nuclear safety-related impoundment constructed by placing a dam across a small cove of Lake Wylie. Because of the design of the SNSW pond, an analysis was performed to predict the total loss of volume in the pond due to sedimentation during the 40-year plant life. This analysis is described in the Catawba UFSAR, Section 2.4.8 [Reference 4.8-3] and the Catawba SER, Section 2.4.4.2 [Reference 4.8-4]. The analysis estimated that the SNSW pond volume would be depleted by about 10 acre-feet of sediment during the 40-year plant life.

Because all of the criteria contained in 10 CFR 54.3 have been met, the sedimentation of the SNSW pond, over time, is a time-limited aging analysis for Catawba Nuclear Station. TLAA demonstration option, (iii), which states that the effects of aging will be adequately managed for the period of extended operation, is chosen to manage the SNSW pond sedimentation TLAA. The Standby Nuclear Service Water Pond Volume Program manages the volume of water in the pond.

Catawba TS 3.7.9.1 requires that the water level of the SNSW pond remain greater than or equal to 571 feet mean sea level. This requirement ensures that a sufficient volume of water is available to allow the nuclear service water system to operate for at least 30 days following the design basis LOCA. The SNSW pond’s level is monitored and makeup water is provided should the pond level drop to 571.5 feet. TS 3.7.9 requires immediate makeup to restore the pond level or the station is shutdown. The minimum allowable pond level includes margin to account for evaporation and the use of SNSW pond water for fire protection, assured auxiliary feedwater, assured component cooling makeup, and assured fuel pool makeup for a full 30 days after a postulated accident [Reference 4.8-3, Section 9.2.5.4].

Catawba UFSAR Figure 9-54 contains the area volume curves which are used in the thermal analysis for the ultimate heat sink. The UFSAR also includes a commitment that soundings will be taken around the SNSW intake structure at five-year intervals to assure that sediment deposits will not adversely affect the operation of the nuclear service water system. Although an earlier calculation for the volume of the SNSW pond was documented, more recent calculations have been performed which validate the volume of water in the SNSW Pond.

4.7.3.2 Staff Evaluation

The applicant has chosen to utilize TLAA demonstration option (iii). The staff evaluation of the TLAA, therefore, focused on how the SNSW Pond Volume Program manages the aging effect of pond volume depletion through effective incorporation of the following 10 elements: program scope, preventive or mitigative actions, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls, and operating experience.

It is noted that corrective actions, confirmation process, and administrative controls for license renewal are in accordance with the site-controlled quality assurance program pursuant to 10 CFR, Part 50, Appendix B, and covers all structures and components subject to an aging management review. The staff's evaluation of the applicant's corrective actions, confirmation process and administrative controls is documented in Section 3.0.4 of this safety evaluation report. This program satisfies the elements of corrective actions, confirmation process and administrative controls. The remaining seven elements are discussed below.

[Scope] The scope of the Standby Nuclear Service Water Ponds Volume Program includes the volume of water in the SNSW pond. The staff finds the scope of the program acceptable, because this is the only commodity in which the aging effect is to be managed.

[Preventive or Mitigative Actions] No actions are taken as part of this program to prevent aging effects or mitigate aging degradation, and the staff has not identified the need for any.

[Parameters Monitored or Inspected] The volume of water in the pond is the only parameter that is monitored. The Standby Nuclear Service Water Pond Volume Program requires a topographic survey of the ponds to determine toe topography of the bottom of the SNSW pond. Calculations are then performed using the survey data to determine the volume of water within the SNSW pond. This is acceptable to the staff because this parameter provides an effective means of managing the aging effect of water depletion.

[Detection of Aging Effects] The applicant stated that no actions are taken as part of this program to detect aging effects, and the application is silent in regard to the remedial action which the applicant will take in case a future survey of the topography of the bottom of the pond indicates a reduction in the volume of water due to the buildup of sediment. By letter dated January 28, 2002, the staff requested, in RAI 4.7.3-1, the applicant to clarify this aspect of the SNSW pond volume program. In its response dated March 11, 2002, the applicant stated that, in the event that a future survey of the topography of the bottom of the SNSW pond indicates a reduction in the volume of water due to the buildup of sediment, remedial actions may include, but not be limited to:

- Enlargement of the pond by excavation
- Raising the required Technical Specification elevation
- Dredging of the pond
- Modification of the pond to raise the surface elevation

The staff finds these remedial actions acceptable because they provide an effective means of managing the aging effect due to sedimentation. With the closure of this RAI concern, the staff finds the detection of aging effects acceptable.

[Monitoring and Trending] The design parameter (volume of water within the SNSW pond) is validated using the SNSW pond volume program. Conventional methods of surveying and volume calculation are used. A contour map with a known scale is developed as a result of the survey. Areas within each contour at different elevations are determined. Using the contour intervals and the area contour interval, volumes are computed for each contour elevation. The computed (surface) areas and the volume of water (below the specified pond surface elevations) at each contour elevation are compared to the areas and volumes in Figure 9-54 in the Catawba UFSAR to ensure that an adequate volume of water is available.

The SNSW pond volume program is performed once every three years, and is documented and retained in sufficient detail to permit adequate confirmation of the results. The staff finds the monitoring and trending acceptable, because the monitoring frequencies will permit an effective management of the aging effects, and because monitoring is performed by utilizing reliable and conventional surveying methods.

[Acceptance Criteria] The acceptance criteria are contained in the area-volume curve shown in Catawba UFSAR, Figure 9-54. Calculated areas and volumes are compared to the criteria in Figure 9-54. The staff finds the acceptance criteria to be adequate and acceptable, because the applicant has used conservative and reasonable margins to estimate the volume of water in the SNSW pond.

4.7.3.3 UFSAR Supplements

The changes to the McGuire and Catawba UFSARs related to this TLAA are provided in the UFSAR Supplements in Appendices A-1 and A-2 of the LRA. The staff reviewed the changes documented therein and finds them appropriate and acceptable.

4.7.3.4 Conclusion

On the basis of the review described above, the staff concludes that the applicant has provided adequate information and reasonable assurance to demonstrate that, in accordance to 10 CFR 54.21 (c)(1)(iii), the SNSW Pond Volume Program will adequately manage the aging effect during the extended period of operation.

4.8 References:

4.8-1 WCAP-14535A, November 1996, “Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination,” Section 4.3.1, Westinghouse Electric Corporation.

4.8-2 NUREG-0422, “Safety Evaluation Report Related to the Operation of the McGuire Nuclear Station, Units 1 and 2,” March 1978, as supplemented, U.S. Nuclear Regulatory Commission, Docket Nos. 50-369 and 50-370.

4.8-3 Catawba Nuclear Station Updated Final Safety Analysis Report, as revised.

4.8-4 NUREG-0954, “Safety Evaluation Report Related to the Operation of the Catawba Nuclear Station, Units 1 and 2,” February 1983, as supplemented, U.S. Nuclear Regulatory Commission, Docket Nos. 50-413 and 50-414.