4. TIME-LIMITED AGING ANALYSES

4.1 Identification of Time-Limited Aging Analyses

This section addresses the identification of time-limited aging analyses (TLAAs). The applicant provided a list of TLAAs in LRA Sections 4.2 through 4.7. The staff's review of the TLAAs can be found in Sections 4.2 through 4.7 of this safety evaluation report (SER).

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

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

4.1.1 Summary of Technical Information in the Application

The applicant evaluated calculations for Ginna against the six criteria specified in 10 CFR 54.3 to identify the TLAAs. The applicant indicated that calculations that meet the six criteria were identified by searching the current licensing basis (CLB), which includes the updated final safety analysis report (UFSAR), design basis documents, previous license renewal applications, technical specifications, NUREG-1800,and Nuclear Energy Institute (NEI) 95-10. The applicant listed the following TLAAs in Table 4.1-1 of the LRA:

- reactor vessel neutron embrittlement, including analyses for upper-shelf energy, pressurized thermal shock, and pressure/ temperature limits
- metal fatigue, including ASME Class 1, reactor vessel underclad cracking, ANSI B31.1, accumulator check valves, reactor vessel nozzle-to-vessel weld defect, and pressurizer fracture mechanics analysis
- environmental qualification of electrical equipment
- containment concrete tendon prestress
- containment liner plate and penetration fatigue
- containment liner stress
- containment tendon fatigue
- containment liner anchorage fatigue
- containment tendon bellows fatigue
- crane load cycle limit
- reactor coolant pump flywheel
- thermal aging embrittlement of cast austenitic stainless steel

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

4.1.2 Staff Evaluation

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

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

(1) involve systems, structures, and components within the scope of license renewal, as delineated in 10 CFR 54.4(a)
(2) consider the effects of aging
(3) involve time-limited assumptions defined by the current operating term, for example, 40 years
(4) determined to be relevant by the applicant in making a safety determination
(5) involve conclusions, or provide the basis for conclusions, related to the capability of the system, structure, and component to perform its intended functions, as delineated in 10 CFR 54.4(b)
(6) contained or incorporated by reference in the current licensing basis

The applicant listed the TLAAs applicable to Ginna in Table 4.1-1 of the LRA. Tables 4.1-2 and 4.1-3 in NUREG-1800 identify potential TLAAs determined from the review of other license renewal applications (LRAs). In request for additional information (RAI) 4.1-1, the staff requested that the applicant discuss whether there were any calculations or analyses at Ginna that addressed the topics listed in Tables 4.1-2 and 4.1-3 of NUREG-1800 but were not included in Table 4.1-1 of the LRA.

In its RAI response dated May 23, 2003, the applicant indicated that the following topics listed in NUREG-1800 as applicable to pressurized-water reactor (PWR) facilities were not included in Table 4.1-1 of the LRA:

- metal corrosion allowance
- inservice local metal containment corrosion allowance
- high energy line break analysis based on cumulative usage factor
- fatigue analysis for the main steam supply lines to the auxiliary feedwater pump
- fatigue analysis of the polar crane

The applicant stated that a metal containment corrosion allowance is not applicable to Ginna because it has a concrete containment, and that high energy line breaks were not postulated based on fatigue usage at Ginna. The staff review of the Ginna UFSAR did not identify TLAAs associated with these topics. Therefore, the staff finds the applicant's response acceptable.

The applicant indicated that no specific fatigue analysis was performed for the turbine driven auxiliary feedwater steam supply lines; however, the B31.1 analysis of the these lines is addressed in LRA Subsection 4.3.2. The applicant also indicated that Ginna did not have a polar crane, and that other cranes are addressed in LRA Section 4.7.5. The staff finds the applicant's response reasonable because the applicant has identified crane load cycle limits as TLAAs, and the applicant has performed a TLAA assessment of American National Standards Institute (ANSI) B31.1 piping.

The applicant also indicated that metal corrosion was used in supplier calculations, but was not incorporated in the CLB. The staff requested that the applicant describe the basis for the corrosion allowance used in the vendor calculations. The applicant indicated that the corrosion allowance was based on consideration of 40 years of operation, but that no specific TLAA evaluation was performed. Instead, the applicant indicated that a one-time inspection is planned to assess the loss of material due to corrosion. The applicant indicated that this inspection would be completed prior to the period of extended operation. The staff finds that the applicant's proposed one-time inspection is an acceptable method to assess the loss of material due to corrosion. A discussion of the One-Time Inspection Program is discussed in Section 3.0.3.7 of this SER.

4.1.3 Conclusions

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

4.2 Reactor Vessel Neutron Embrittlement

The applicant has identified three analyses affected by irradiation embrittlement that have been identified as TLAAs. These analyses are discussed in Sections 4.2.1 through 4.2.3 of the LRA and include the following:

(1) reactor vessel upper-shelf energy
(2) pressurized thermal shock
(3) pressure/temperatures curves

Neutron embrittlement is a significant aging mechanism for all ferritic materials that have a neutron fluence of greater than 10^{17} neutron per squared centimeters (E>1 one million electron volts (MeV)). The relevant calculations use predictions of the cumulative damage to the reactor vessel (RV) from neutron embrittlement, and were originally based on the 40-year expected life of the plant. The reactor pressure vessel (RPV) contains the core fuel assemblies and is made of thick steel plates that are welded together. Neutrons from the fuel in the reactor irradiate the vessel as the reactor is operated and change the material properties of the steel. The most pronounced and significant changes occur in the material property known as fracture toughness. Fracture toughness is a measure of the resistance to crack extension in response to stresses. A reduction in this material property due to irradiation is referred to as embrittlement. The largest amount of embrittlement usually occurs at the section of the vessel's wall closest to the reactor fuel, otherwise referred to as the vessel's beltline. The rate at which the vessel's steel embrittles also depends on its chemical composition. The amounts of two elements in the steel, specifically copper and nickel, are the most important chemical elements in determining how sensitive the steel is to neutron irradiation.

4.2.1 Reactor Vessel Upper–Shelf Energy

The US Nuclear Regulatory Commission (NRC) regulations that provide screening criteria for the "upper-shelf energy" (USE) is in 10 CFR 50, Appendix G. Appendix G to 10 CFR Part 50 requires that reactor vessel beltline materials have Charpy USE values in the transverse direction for the base metal and along the weld for the weld material, according to the American Society of Mechanical Engineers (ASME) Code, of no less than 75 foot-pounds (ft-lb) (102 joules(J)) initially, and must maintain Charpy USE values throughout the life of the vessel of no less than 50 ft-lb (68 J). However, in accordance with paragraph IV.A.1.a., Charpy USE values below these criteria may be acceptable if it is demonstrated, in a manner approved by the Director, Office of Nuclear Reactor Regulation, that the lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," provides an expanded discussion regarding the calculations of Charpy USE values and describes two methods for determining Charpy USE values for RV beltline materials, depending on whether or not a given RV beltline material is represented in the plant's reactor Vessel Material Surveillance Program (i.e., 10 CFR Part 50, Appendix H program). If surveillance data are not available, the Charpy USE is determined in accordance with position 1.2 in RG 1.99, Revision 2. If two or more surveillance data are available, the Charpy USE should be determined in accordance with position 2.2 in RG 1.99, Revision 2. These methods refer to Figure 2 in RG 1.99, Revision 2, which indicates that the percentage drop in Charpy USE is dependent upon the amounts of copper and the neutron fluence. Because the analyses performed in accordance with Appendix G to 10 CFR Part 50 are based on a flaw with a depth equal to one-quarter of the vessel wall thickness (1/4T), the neutron fluence used in the Charpy USE analysis is the neutron fluence at the 1/4T depth location.

4.2.1.1 Technical Information the Application

The applicant indicates that calculations have shown that the vessel beltline Charpy USE for the limiting weld will be less than 50 ft-lb based on RG 1.99, Revision 2. Consequently, a low upper-shelf fracture mechanics analysis has been performed to evaluate the weld material for ASME Levels A, B, C, and D service loadings, based on the acceptance criteria of the ASME Code, Section XI, Appendix K, "Assessment of Reactor Vessels With Lower Upper Shelf Energy Charpy Impact Energy Levels." The analysis demonstrates that the limiting RV beltline weld satisfies the ASME Code requirements of Appendix K for ductile flaw extensions and tensile stability using projected low upper-shelf Charpy impact energy levels for the weld material at 52 effective full-power years (EFPYs), which corresponds to the EFPY at the end of the extended period. ASME Code, Section XI, Appendix K contains a methodology and criteria acceptable to the staff for satisfying the requirement in paragraph IV.A.1.a. to demonstrate that materials with Charpy USE values below 50 ft-lb provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. Therefore, by performing an analysis in accordance with ASME Code, Section XI, Appendix K, the applicant has satisfied the requirements in paragraph IV.A.1.a of Appendix G, 10 CFR Part 50 for 52 EFPY.

4.2.1.2 Staff Evaluation

In response to request for additional information (RAI) RAI 4.2.1-1, the applicant provided, in an April 11, 2003 letter, its low upper-shelf fracture mechanics analysis of the limiting weld in the Ginna RV. The analysis is described in Framatone Report BAW-2425, Revision 1, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessel of R.E. Ginna for Extended Life Through 54 Effective Full Power Years," June 2002, which is Enclosure 1 in the April 11, 2003, letter. The analysis concludes that the SA-847 circumferential weld in the Ginna RPV satisfies ASME Code Section XI, Appendix K for Levels A, B, C, and D Service Loadings for 54 EFPY (60 years of operation at 90 percent capacity factor). This analysis satisfies the requirements of 10 CFR 54.21(c)(1)(ii) for RV USE for 54 EFPY. Because the applicant indicates that 52 EFPY, corresponds to the end of the period of extended operation, 54 EFPY, represents 2 EFPY, beyond the period of extended operation.

To confirm that the applicant's analysis satisfied the criteria in ASME Code Section XI, Appendix K, the staff performed an independent analysis using the methodologies and models specified in RG 1.161, "Evaluation of Reactor Pressure Vessels With Charpy Upper-Shelf Energy Less Than 50 ft-lb;" NUREG/CR-5729, "Multivariable Modeling of Pressure Vessel and Piping J-R Data;" and ASME Code Section XI, Appendix K. The NRC staff confirmed the applicant's conclusion that the Ginna RV would have margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code through the period of extended operation.

4.2.2 Pressurized Thermal Shock

Section 50.61 of Title 10 of the *Code of Federal Regulations* provides the fracture toughness requirements protecting the reactor vessels of PWRs against the consequences of pressurized thermal shock (PTS). Licensees are required to perform an assessment of the RV materials' projected values of the reference temperature for PTS, RT_{PTS}, through the end of their operating license. The rule requires each licensee to calculate the end-of-life RT_{PTS} value for each material located within the beltline of the RPV. The RT_{PTS} value for each beltline material is the sum of the unirradiated reference nil ductility transition temperature (RT_{NDT}) value, a shift in the RT_{NDT} value caused by exposure to high energy neutron irradiation of the material (i.e., DRT_{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 values are to be evaluated.

RV beltline base metal materials (forging or plate materials) and longitudinal (axial) weld materials are considered to provide adequate protection against PTS events if the calculated RT_{PTS} values are less than or equal to 270 °F. RV beltline circumferential weld materials are considered to provide adequate protection against PTS events if the calculated RT_{PTS} values are less than or equal to 300 °F. RG 1.99, Revision 2, provides an expanded discussion regarding the calculations of the shift in the RT_{NDT} value caused by exposure to high energy neutron irradiation and the margin value to account for uncertainties. In this RG, the shift in the RT_{NDT} value caused by exposure to high energy neutron irradiation is the product of a chemistry factor and a fluence factor. The fluence factor is dependent upon the neutron fluence and the chemistry factor may be determined from surveillance material or from the tables in the RG. If the RV beltline material is not represented by surveillance material, its chemistry factor and the shift in the RT_{NDT} value caused by exposure to high energy neutron irradiation may be determined using the methodology documented in position 1.1 and the tables in this RG. The chemistry factor determined from the tables in the RG depends upon the amount of copper and nickel in the beltline. If the RV beltline material is represented by surveillance material, its chemistry factor may be determined from the surveillance data using the methodology documented in position 2.1 of RG 1.99, Revision 2. Section 50.61of Title 10 of the *Code of Federal Regulations* contains methods of determining RT_{NDT} values equivalent to RG 1.99, Revision 2.

4.2.2.1 Technical Information in the Application

The applicant indicates that it has completed the projected RT_{PTS} calculation using generic data although plant-specific data were available, because generic data proved to be more conservative. Therefore, RG 1.99, Revision 2, position 1.1 was used to calculate the RT_{PTS} for 52 EFPY. The licensee calculations of RT_{PTS} values for the Ginna RV beltline materials will remain below the 10 CFR 50.61 PTS screening criteria through 52 EFPY, the end of the period of extended operation.

4.2.2.2 Staff Evaluation

In the license renewal submittal, the licensee provided RT_{PTS} analyses for the materials in the Ginna RV beltline. Table 4.2-1 in the submittal provides the chemistry factor and the predicted RT_{PTS} value through 52 EFPY for each forging and weld in the Ginna RV beltline. In addition, the applicant has projected that one weld is to be 11°F below the PTS screening limit at the end of the period of extended operation.

To verify the predicted RT_{PTS} value through 52 EFPY, the staff requested that the applicant describe the analysis performed to determine the neutron fluence. In response to RAI 4.2.2-1, the applicant provided its analysis to determine the neutron fluence for the pressure vessel. The analysis is described in Westinghouse Commercial Atomic Power report WCAP-15885, Revision 0, "R.E. Ginna Heatup and Cooldown Limit Curves for Normal Operation," July 2002, which was enclosed in the April 11, 2003 letter. Section 3.0 of this WCAP contains the calculational methodology and the validation of the calculated results with comparison to surveillance capsule measured data.

The methodology is based on the DORT 3.1 discrete ordinates neutron transport code, using the BUGLE-96 cross sections derived from the ENDF/B-VI cross section library. A three dimensional solution was constructed based on $\varphi(r,\theta)$ and $\varphi(r,z)$ solutions. The $\varphi(r,\theta)$ solution used 170 and 67 radial and azimuthal intervals, respectively, for the representation of the 1/8 core. The $\varphi(r,z)$ used 153 and 90 intervals, respectively. Both meshes are considered adequate. Source distributions were obtained from cycle loading reports which is the usual practice. In general, the methodology satisfies the guidance in RG 1.190; therefore, it is acceptable.

This section of the WCAP also compares the plant-specific dosimetry measurements to the corresponding calculations. The comparisons indicate an excellent mean value although the uncertainties are higher than normal but well within the guidance of RG 1.190. Therefore, they are acceptable. The staff concludes that the methodology and the validation for the Ginna 32 and 54 EFPY fluence values are acceptable and may be used to determine the impact of neutron fluence on RV materials in the beltline region (the region adjacent to the reactor core).

This methodology has not been qualified for calculations above or below the beltline region. However, for regions above or below the core, the expected uncertainties would be higher than those in the beltline region, but no greater than a factor of two. As a result of extending the license, the applicant determined that the weld between the intermediate shell and the nozzle shell would receive neutron fluence greater than 10^{18} n/cm^2 (E>1 MeV). This weld is 10 inches above the top of the core. Because the methodology has not been qualified for determining the neutron fluence for this weld, the staff requested that the applicant perform a RT_{PTS} calculation assuming a factor of 2 increase in the neutron fluence.

This analysis is contained in Attachment 4 to a June 10, 2003, letter. In the RT_{PTS} calculation contained in this attachment, the applicant compared the RT_{PTS} values for the intermediate shell to the nozzle weld (identified as the SA-1101 weld) to the intermediate shell to the lower shell weld (identified as the SA-847 weld). The SA-1101 and SA-847 welds are circumferentially oriented, because the Ginna RPV shell segments were fabricated from forgings. The neutron fluence at the inside surface for the SA-847 weld was 5.01 x 10^{19} n/cm^2 and the neutron fluence for the SA-1101 weld was 0.396 x 10^{19} n/cm^2. The RT_{PTS} value for the SA-1101 weld was 268.2°F when using the Table 1 chemistry factor (based on the amounts of copper and nickel in the weld). The RT_{PTS} value for the SA-847 weld were 270.6°F when utilizing surveillance data to calculate the chemistry factor and 282.5°F when utilizing the Table 1 in RG 1.99, Revision 2 to calculate the chemistry factor. The chemistry factors in Table 1 of RG 1.99, Revision 2 are based on the amount of copper and nickel in the weld and are the same as those specified in Table 1 of 10 CFR 50.61. Because the RT_{PTS} value for the SA-1101 weld is less than the value for the SA-847 weld and since the value for the SA 1101 weld was calculated using a neutron fluence two times greater than the mean value using the applicant's neutron fluence methodology, the SA-847 weld will not be limiting throughout the period of extended operation.

In the June 10, 2003, letter, the applicant changed its method of determining the RT_{PTS} value for the limiting weld, SA-847, from one that was based on the chemistry factor from Table 1 in RG 1.99, Revision 2 and 10 CFR 50.61 to one that was based on the use of the Ginna surveillance data. Section 50.61of Title 10 of the *Code of Federal Regulations* identifies two methods of determining the chemistry factor and RT_{PTS} value—one method based on the amount of copper and nickel in the weld and one based on the use of surveillance data. As specified in 10 CFR 50.61(c)(2)(ii)(A), the surveillance data deemed credible, according to the criteria of 10 CFR 50.61(c)(2)(i), must be used to determine the material-specific chemistry factor. The applicant chose to utilize surveillance data in determining the chemistry factor, but has not demonstrated that the data satisfy the credibility criteria of 10 CFR 50.61 (c)(2)(i). The chemistry factor identified in the June 10, 2003 letter is 161.9 °F. The chemistry factor identified for this weld in the Reactor Vessel Integrity Data Base (RVID) is 158.7°F, which is based on the surveillance data. Although the difference in the chemistry factor calculated by the applicant and that in the RVID is small, the staff would like to review the surveillance data and methodology utilized by the applicant to determine the chemistry factor and to confirm that the results satisfy 10 CFR 50.61. The applicant is to provide the surveillance data, the detailed calculations for determining the chemistry factor from the surveillance data, and the analysis that demonstrates that the surveillance data meet the credibility criteria in 10 CFR 50.61. In addition, this analysis differs from that identified in UFSAR Section A3.1.2. The applicant is also requested to provide an update to this UFSAR Section. This is Open Item 4.2.2-1.

4.2.3 Plant Heatup/Cooldown (Pressure/Temperature) Curves

4.2.3.1 Technical Information in the Application

The current pressure/temperature (P/T) analyses are valid beyond the current operating license period but not to the end of the period of extended operation. The technical specifications will continue to be updated as required by Appendices G and H of 10 CFR Part 50, or as operational needs dictate. This will assure that operational limits remain valid for current and projected cumulative neutron fluence levels.

4.2.3.2 Staff Evaluation

The technical specifications will continue to be updated as required by either Appendices G or H of 10 CFR Part 50, or as operational needs dictate. This will assure that operational limits remain valid for current and projected cumulative neutron fluence levels. Because the technical specifications will continue to be updated, additional analysis at this time is not required.

4.2.4 UFSAR Supplement

On the basis of the staff's evaluation described above, pending resolution of Open Item 4.2.2-1, the summary description for the reactor coolant system TLAA for reactor vessel USE, PTS, and P/T limits described in the UFSAR Supplement (LRA, Appendix A) provides an adequate description of this TLAA, as required by 10 CFR 54.21.

4.2.5 Conclusions

The staff has reviewed the TLAAs regarding the maintenance of acceptable reactor vessel USE for the Ginna RV materials and the ability of the Ginna RV to resist failure during postulated PTS events. On the basis of this evaluation, pending resolution of Open Item 4.2.2-1, 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 Ginna RV beltline materials as evaluated to the end of extended operating periods. Therefore, they satisfy the requirements of 10 CFR 54.21(c)(1)(ii) for 60 years of operation. The staff will evaluate the end-of-extended-operating term P/T limit curves for Ginna 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 reactor coolant system for Ginna will be done in a manner that ensures the integrity of the reactor coolant system during the extended periods of operation and that the curves, when submitted, will satisfy the requirements of 10 CFR 54.21(c)(1)(ii) for the period of extended operation. Because evaluation of P/T limit curves are performed when the limits are updated through the technical specification process, they need not be evaluated now; however, they will be evaluated as part of the plant's technical specifications The technical specification process provides a process for managing P/T limits, in accordance with 10 CFR 54.21(c)(1)(iii).

4.3 Metal Fatigue

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

4.3.1 Summary of Technical Information in the Application

4.3.1.1 ASME Boiler and Pressure Vessel Code, Section III, Class 1

The applicant discussed the design requirements for components of the reactor coolant system in Section 4.3.1 of the LRA. The reactor vessel, steam generators, reactor coolant pumps, and pressurizer were designed to the ASME Boiler and Pressure Vessel Code, Section III requirements for Class 1 components. The applicant stated that the reactor vessel internals (RVI) were designed in accordance with Westinghouse criteria which were later incorporated in the ASME Code. These components were subjected to design transient cycles intended to bound the thermal and pressure cycles for a 40-year operating life.

The applicant stated that, based on review of the frequency and severity of actual operating transients, it projects that the original 40-year transient set will remain bounding for 60 years of plant operation. Therefore, the applicant concluded that the fatigue analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i). The applicant indicated that, prior to the expiration of the current operating license, a Fatigue Monitoring Program will be implemented as a confirmatory program. The Fatigue Monitoring Program is discussed in Section B.3.2 of the LRA.

4.3.1.2 ANSI B31.1 Piping

The applicant discussed the evaluation of United States of America Standards (USAS) ANSI B31.1 components in Section 4.3.2 of the LRA. USAS B31.1 specifies that a stress reduction factor be applied to the allowable thermal bending stress range if the number of full range cycles exceeds 7000. The applicant indicated that most piping systems within the scope of license renewal are only subject to occasional cyclic operation and, consequently, the analyses will remain valid during the period of extended operation. However, the applicant did indicate that the nuclear steam by supply system (NSSS) sampling system could exceed the 7000 cyclic limit during the period of extended operation. The applicant committed to complete an engineering analysis of the NSSS sampling system prior to the period of extended operation.

4.3.1.3 Reactor Vessel Underclad Cracking

The applicant indicate that WCAP-15338 is bounding for all Westinghouse plants and, the analysis has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii). The underclad cracking TLAA is described in the UFSAR supplement.

4.3.1.4 Accumulator Check Valves

The applicant discussed the fatigue evaluation of the accumulator check valves in Section 4.3.4 of the LRA. The applicant stated that fatigue analyses were performed on the swing check valves. The applicant indicated that the number of design transients used in the analyses bound the number of plant transient limits. Therefore, the applicant concluded that the fatigue analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.3.1.5 Reactor Vessel Nozzle-to-Vessel Weld Defect

During the first inservice inspection (ISI) of the reactor vessel in 1970, a flaw indication was discovered by ultrasonic testing (UT) in a primary nozzle-to-vessel weld at Nozzle N2B. It was determined at that time that the size of the flaw was in excess of the size permissible by the acceptance criteria in ASME Code Section XI, 1974 Edition. Consequently, the flaw was reevaluated in accordance with ASME Code Section XI, Appendix G, requirements for acceptance by evaluation and found to be acceptable. A review of the original construction radiographs confirmed the presence of slag at the same location as the flaw indication. The same flaw indication was found by UT inspection during the second ISI in 1989.

The flaw was reexamined after the second ISI finding in 1989 using a 15° focused beam search unit. This reexamination revealed that there are two separate flaws instead of one single flaw. The licensee asserts that the dimensions of the two separate flaws each meet the criteria for acceptance by examination in ASME Section XI, 1974 Edition with 1975 Addenda. Furthermore, the licensee states that a fracture mechanics analysis was performed and confirmed that the indication was acceptable by evaluation according to the requirements of ASME Section XI, Appendix G.

4.3.1.6 Pressurizer Fracture Mechanics Analysis

Preservice UT examination of the pressurizer detected a “defect-like” indication in the lower shell-to-head circumferential weld (C-3). The indication was reported as a linear reflector with the approximate dimensions of 11.5" length x 0.5" width and embedded partially in the circumferential weldment and the base metal of the pressurizer shell.

Fracture mechanics analysis was performed by Westinghouse and concluded that the “defect” would not cause failure of the pressurizer during the design life (40 years) of the component. The indication has been subject to six UT examinations since the preservice examination. The most recent UT examination (as well as two other examinations) characterized the indication as consisting of several intermittent, low-amplitude indications located in the center third of the weld thickness. The most recent examination used both automated and manual UT examinations. These indications were also evaluated and found to meet the acceptance criteria by examination in ASME Code, Section XI, 1995 Edition with 1996 Addenda. Because it has been demonstrated that the initial indication is actually a number of small, discrete indications which meet the ASME Code, Section XI acceptance criteria by examination, the fracture mechanics analysis is no longer applicable or relevant.

4.3.1.7 Environmentally Assisted Fatigue Evaluation

The applicant described the actions taken to address the issue of environmentally assisted fatigue in Section 4.3.7 of the LRA. The applicant described its evaluation of the following fatigue sensitive component locations:

- reactor vessel lower shell and lower head (lower shell at the core support pads)
- reactor vessel inlet and outlet nozzles
- pressurizer surge line (including hot leg and pressurizer nozzles)
- reactor coolant piping charging system nozzle
- reactor coolant piping safety injection nozzle
- residual heat removal system Class 1 piping

Based on its evaluation, the applicant indicated that all locations were found acceptable for the period of extended operation, with the exception of the pressurizer surge line. The applicant indicated that the pressurizer surge line will require further evaluation prior to the period of extended operation.

4.3.2 Staff Evaluation

4.3.2.1 ASME Boiler and Pressure Vessel Code, Section III, Class 1

As stated in the previous section, components of the reactor coolant system at Ginna were designed to the Class 1 requirements of the ASME Code. These requirements contain explicit criteria for the fatigue analysis of components. Consequently, the applicant identified the fatigue analyses of these components as TLAAs. The staff reviewed the applicant's evaluation of the RCS components for compliance with the provisions of 10 CFR 54.21(c)(1).

The specific design criterion for fatigue analysis of ASME Class 1 components involves calculating the cumulative usage factor (CUF). The fatigue damage in the component caused by each transient depends on the magnitude of the resulting stresses. The CUF sums the fatigue damage resulting from each transient pair. The design criterion requires that the CUF not exceed 1.0. The applicant indicated that review of the Ginna plant operating history suggests that the number of cycles and severity of the transients assumed in the design of these components envelopes the expected transients during the period of extended operation. In RAI 4.3.1-1, the staff requested that the applicant provide the following information for each of the transients reviewed:

- the current number of operating cycles and a description of the method used to determine the number of the design transients from the plant operating history

- 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

- a comparison of the design transients with the transients monitored by the Fatigue Monitoring Program described in Section B.3.2 of the LRA, identification of any transients listed in the LRA that are not monitored by the Fatigue Monitoring Program, and an explanation of why it is not necessary to monitor these transients

The applicant's May 13, 2003, response indicated that the number of design transient cycles for the first 19 years of operation was documented in a report issued in 1989. The number of design transient cycles was updated in a 1995 report and subsequently updated on an annual basis. The applicant provided the cycle counts from these reports in Table 1 of the response. The design transients provided in Table 1 include the design transients listed in UFSAR Table 5.1-4. The applicant used two methods to estimate the number of cycles expected for 60 years of plant operation. The first method was a linear projection based on the first 30 years of plant operation. The second method used a weighted average based on the expectation that the more recent plant operation provided a better estimate for the future plant operation. Both methods indicated that the number of transients is not expected to exceed the number of design cycles for 60 years of plant operation.

Table 2 of the applicant's May 13, 2003, response provided a detailed list of transients tracked by the Fatigue Monitoring Program. The list includes the design transients detailed in UFSAR Table 5.1-4. The applicant indicated that plant loading and unloading at 5 percent of full power per minute are not counted due to the small rate of accumulation in the first 20 years of operation. The staff agrees that the number of design cycles listed in UFSAR Table 5.1-4 for these transients is conservative based on the information presented in NUREG/CR-6260 for an older vintage Westinghouse plant. The applicant also identified several design transients for residual heat removal, safety injection, charging, and pressurizer operations that are not listed in UFSAR Table 5.1-4. These transients affect the fatigue sensitive components identified in NUREG/CR-6260 that the applicant evaluated for environmental fatigue. Table 2 of the applicant's May 13, 2003, response indicates that it will use the automated fatigue monitoring software, "FatiguePro," to estimate the number of these cycles for 60 years of plant operation once FatiguePro has been operating for 2 years. The staff compared the transients listed in Table 2 with the transients identified in NUREG/CR-6260 that are significant contributors to the fatigue usage of the safety injection nozzle, the charging nozzle, and the residual heat removal system components. As discussed later, the applicant will perform detailed stress monitoring of critical pressurizer components using the FatiguePro software. On the basis of the review described above, the staff concludes that the applicant has provided sufficient information to assure that the thermal transients that are significant contributors to the design fatigue usage of RCS components will be monitored by the Fatigue Monitoring Program.

The Westinghouse Owners Group (WOG) issued Topical Report WCAP-14577, Revision 1-A, "Aging Management for Reactor Internals," to address the aging management of the RVI. The staff review of WCAP-14577, Revision 1-A identified a number of issues that should be addressed on a plant-specific basis. Renewal applicant action item 11 indicates that the fatigue TLAA of the RVI should be addressed on a plant-specific basis. Table 3.2.0-2 of the LRA provides the applicant's response to renewal applicant action item 11. The applicant stated that fatigue of the RVI is addressed in Section 4.3 of the LRA. Section 4.3 of the LRA indicates that the RVI were designed in accordance with Westinghouse criteria which were later incorporated into the ASME Code. In RAI 4.3.1-2, the staff requested that the applicant discuss the transients that contribute to the fatigue usage for each component listed in Table 3-3 of WCAP-14577, Revision 1-A and detail how these transients were evaluated during the transient review described in RAI 4.3.1-1.

The applicant's May 13, 2003, response indicated that Westinghouse had performed an evaluation of the structural integrity of the RVI in 1995 in support of a proposed reduction in Tavg after installation of the replacement steam generators. The applicant indicated that fatigue usage factors were calculated for a number of the RVI components which included the components identified in Table 3-3 of WCAP 14577, Revision 1-A. The applicant listed the transients used in the analysis of the RVI. The applicant stated that the number of transients used in the analysis bound the number expected for 60 years of plant operation. In addition, the applicant indicated that the transients will be tracked by the Fatigue Monitoring Program. The staff finds that the applicant has adequately addressed applicant action item 11 in WCAP 14577, Revision 1-A by assuring that the thermal transients that are significant contributors to the design fatigue usage of RVI components will be monitored by the Fatigue Monitoring Program.

The 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 safety evaluation (SE) requests that a license renewal applicant perform an additional fatigue evaluation or propose an aging management program to address components labeled I-M and I-RA in Tables 3-2 through 3-16 of WCAP-14575. In Table 3.2.0-1 of the LRA, the applicant indicates that an automated cycle monitoring program has been implemented at Ginna, and that this program will monitor the fatigue-sensitive locations. Table 3.2.0-1 refers to the discussion of the fatigue monitoring methodology in Section 4.0 of the LRA. The staff review of the applicant's Fatigue Monitoring Program, as discussed above, found that the program monitors the design transients that are significant contributors to the fatigue usage of RCS components. The staff finds the applicant has adequately addressed applicant action item 8 in WCAP 14575-A by assuring that the thermal transients that are significant contributors to the design fatigue usage of RCS components will be monitored by the Fatigue Monitoring Program.

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 3.3.1.1-1 of the accompanying staff SE requests that a license renewal applicant demonstrate that the pressurizer subcomponent CUFs remain below 1.0 for the period of extended operation, including insurge/outsurge transients discussed in WCAP-14574-A and considering the effects of the reactor coolant environment. In Table 3.2.0-3 of the LRA, the applicant states that FatiguePro will be used to monitor the fatigue usage of the fatigue-sensitive pressurizer locations (spray nozzle, surge nozzle, upper shell, and heater well penetration). The applicant also indicated that the effects of the reactor coolant environment would be included in the evaluation of these subcomponents. The applicant subsequently completed the evaluation of these subcomponents and found that the CUFs are not expected to exceed 1.0 during the period of extended operation. The staff's review of the applicant's evaluation of the pressurizer heater penetration and surge line nozzle is contained in Section 4.3.2.7 of this SER. The staff finds that the applicant has adequately addressed applicant action item 3.3.1.1-1 in WCAP 14574-A by evaluating the fatigue-sensitive pressurizer subcomponents for insurge/outsurge transients by considering the effects of the reactor coolant environment, and by assuring that the thermal transients that are significant contributors to the design fatigue usage of RCS components will be monitored by the Fatigue Monitoring Program.

On the basis of its projection of the number of design transients, the applicant has concluded that the existing fatigue analyses of the RCS components remain valid for the extended period of operation. The applicant uses its Fatigue Monitoring Program to provide assurance that the number of design cycles will not be exceeded during the period of extended operation. The staff finds that the applicant's Fatigue Monitoring Program provides an acceptable program for monitoring the fatigue usage of RCS components in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

The applicant's UFSAR Supplement for metal fatigue of ASME Class 1 components is provided in Section A3.3.1 of the LRA. The staff finds that the UFSAR Supplement provides an adequate description of the fatigue TLAA of ASME Class 1 components to satisfy 10 CFR 54.21(d).

4.3.2.2 ANSI B31.1 Piping

ANSI B31.1 requires that a reduction factor be applied to the allowable bending stress range if the number of full range thermal cycles exceeds 7000. The applicant stated that the number of design transient cycles was found to bound the number of transient cycles expected for 60 years of plant operation, except for the NSSS sampling system. The applicant indicated that the ANSI B31.1 limit of 7000 equivalent full range cycles may be exceeded during the period of extended operation for the NSSS sampling system and that an engineering evaluation will be performed prior to the period of extended operation. The applicant further indicated that the effects of fatigue may be managed by an inspection program if the results of the engineering evaluation are not acceptable. In RAI 4.3.2-1, the staff requested that the applicant provide additional clarification regarding the proposed options for addressing the NSSS sampling system. The staff also requested that the applicant describe the existing qualification of the NSSS sampling system and provide the maximum calculated thermal stress range for affected portions of the system.

The applicant's June 10, 2003, response indicated that the engineering evaluation of the affected portions of the NSSS sampling had been completed. The applicant reported that the maximum calculated thermal stress range in the piping is 4,660 pounds per square inch (psi) which is less than the ANSI B31.1 stress limit specified for 100,000 or more cycles. The applicant concluded that the NSSS sampling system is acceptable for the period of extended operation. The staff agrees with the applicant's conclusion because ANSI B31.1 does not limit the number of cycles at the calculated stress range.

The applicant's UFSAR Supplement for metal fatigue of ANSI B31.1 components is provided in Section A3.3.3 of the LRA. The applicant should update the UFSAR Supplement summary to include the TLAA evaluation of the NSSS sampling system as described above. This is Confirmatory Item 4.3-1.

4.3.2.3 Reactor Vessel Underclad Cracking

Underclad cracks were first discovered in October 1970 during examination of the Atucha reactor vessel. They have been reported to exist only in SA-508, Class 2 reactor vessel forgings manufactured to a coarse grain practice and clad by high heat input submerged arc processes. The underclad cracking issue was first addressed by Westinghouse Topical Report WCAP-7733 which justified the continued operation of Westinghouse plants for 32 EFPY. Subsequently, Westinghouse submitted WCAP-15338 which extended the analysis to justify operation of Westinghouse plants for 60 years of plant operation. The staff review of WCAP-15338 is contained in a September 25, 2002, letter to R.A. Newton (Westinghouse Owners Group) and concluded that LRAs should include the following two action items:

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

(2) As required by 10 CFR 54.21(d), an UFSAR Supplement for the facility must contain a summary description of the programs and activities for managing the effects of aging and the evaluation of the TLAA for the period of extended operation. Those applicants for

license renewal referencing the WCAP-15338 report for the RPV components shall ensure that the evaluation of the TLAA is summarily described in the UFSAR Supplement.

The NRC SER for WCAP-15338 requires the applicant to verify that its plant is bounded by the report. In response to RAI 4.3.3-1 in a May 23, 2003 letter, the applicant indicated that the analysis of the number of Ginna cycles and transients had been accomplished and that they were within the assumed bounds of WCAP-15338. Because the number of Ginna cycles and transients are within the assumed bounds of WCAP-15338, the conclusions of WCAP-15338 are applicable to Ginna and the amount of flaw growth resulting from 60 years of operation will not result in the loss of integrity of the Ginna RPV. Therefore, this analysis satisfies the requirements of 10 CFR 54.21(c)(1)(ii).

On the basis of the staff's evaluation described above, the summary description for the reactor coolant system TLAA for reactor vessel unclad cracking described in the UFSAR Supplement (LRA, Appendix A) provides an adequate description of this TLAA, as required by 10 CFR 54.21(d).

4.3.2.4 Accumulator Check Valves

The applicant stated that fatigue analyses were performed on the swing check valves. The applicant reviewed the design report and concluded that the number of design transients used in the analyses bound the number of plant transients monitored by plant procedures, and that the analysis remains valid for the period of extended operation. In addition, the applicant indicated that the design transients will be monitored by plant procedures. As discussed previously, the staff found that the Fatigue Monitoring Program monitors the design transients that are significant contributors to the fatigue usage of RCS components. On the basis of the applicant's evaluation, as confirmed by the Fatigue Monitoring Program, the staff finds that fatigue of the accumulator check valves has been adequately evaluated for the period of extended operation in accordance with the requirements of 10 CFR 54.21 (c)(1)(i).

The applicant's UFSAR Supplement for metal fatigue of the accumulator check valves is provided in Section A3.3.4 of the LRA. The staff finds that the UFSAR supplement provides an adequate description of the fatigue TLAA of the accumulator check valves to satisfy 10 CFR 54.21(d).

4.3.2.5 Reactor Vessel Nozzle-to-Vessel Weld Defect

The NRC performed a safety evaluation on the reactor vessel nozzle-to-vessel weld defect and concluded that the flaw probably results from an embedded volumetric reflector from the fabrication process that has remained unchanged since construction. However, the licensee's assertion that no further evaluation is warranted is inaccurate and resulted in RAI 4.3.5-1 which reads as follows–

> In Section 4.3.5 it is stated that using a 15^{o} focused beam search unit, the indication was resolved into two separate indications which met the criteria for acceptance by examination in ASME Section XI, 1974, with Summer 1995 Addenda. However, according to the Staff Evaluation section of the referenced document, USNRC Letter Johson to Mecredy, "Ginna Flaw Indication in the Reactor Vessel Inlet Nozzle Weld—1989 Reactor Vessel Examination (TAC No. 71906)," July 7, 1989, "The staff's evaluation determined that the licensee's final dimensions of 4.94" x .48" is a realistic representation of the actual flaw size. If the flaw length were assumed constant, a reduction of .036"

> in the depth dimension (.480"—.44") would result in a flaw indication that meets the ASME Section XI acceptance standard." Consequently, according to the staff SER, the dimensions of the flaw are not within ASME Section XI acceptance standards. Therefore a fatigue analysis for the extended period of operation for this flaw is a TLAA and its results must be provided in accordance with 10 CFR 54.21(c) and must be described in the UFSAR supplement.

In a June 10, 2003, letter, the applicant provided a clarification to RAI 4.3.5-1. In this letter, the applicant provided a fracture mechanics analysis for the flaw based on the design transients for 40 years of operation and compared the number of design transients used in the fatigue crack growth analysis to that projected for the period of extended operation. The analysis is described in an April 26, 1989, letter from Structural Integrity Associates. The analysis considered the impact of neutron irradiation and fatigue crack growth. The flaw indication is 57 inches from the top of the core. At this distance, the impact of neutron irradiation was determined to be insignificant. Because the flaw is located 57 inches from the top of the core and because there is significant attenuation of neutron fluence above the core, the staff agrees with the conclusion that the impact of neutron irradiation is insignificant. The staff's July 7, 1989, safety evaluation indicates that the 1989 inspection results and fracture mechanics analysis demonstrates that the structural integrity of the RPV will be maintained during the service with the flaw indication in the vessel. To demonstrate that the fatigue crack growth for 40 years of operation is applicable for the period of extended operation, the applicant compared the number of design transients used in the fatigue crack growth analysis to that projected for the period of extended operation. The applicant indicates that the number of design transients used in the fatigue crack growth analysis is bounding for the period of extended operation. Therefore, the analysis satisfies the requirements of 10 CFR 54.21(c)(1)(ii).

In a letter dated July 31, 2003, and supplemented in a letter dated August 1, 2003, the applicant provided a UFSAR Supplement for this TLAA. The UFSAR Supplement provides an adequate description of this TLAA, as required by 10 CFR 54.21(d).

4.3.2.6 Pressurizer Fracture Mechanics Analysis

The most recent UT examination showed that the "defect" was not just a single defect but rather a number of small discrete indications which meet ASME Code, Section XI criteria. Therefore the staff has concluded that fracture mechanics analysis is no longer required. Because the fracture mechanics analysis is no longer required, the analysis does not meet the criteria for a TLAA and no additional analysis is required.

On the basis of the staff's evaluation described above, the summary description for this TLAA is not required.

4.3.2.7 Environmentally Assisted Fatigue Evaluation

The applicant indicated that the Fatigue Monitoring Program will be implemented as a confirmatory program prior to the period of extended operation to assure that design cycle limits are not exceeded. The applicant's Fatigue Monitoring Program 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, and concluded the following:

> The results of the probabilistic analyses, along with the sensitivity studies performed, the iterations with industry (NEI and EPRI), and the different approaches available to the licensees to manage the effects of aging, lead to the conclusion that no generic regulatory action is required, and that GSI-190 is closed. This conclusion is based primarily on the negligible calculated increases in core damage frequency in going from 40 to 60 year lives. However, the calculations supporting resolution of this issue, which included consideration of environmental effects, and the nature of age-related degradation indicate the potential for an increase in the frequency of pipe 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.

The applicant evaluated the component locations listed in NUREG/CR-6260 that are applicable to an older-vintage Westinghouse plant for the effect of the environment on the fatigue life of the components. The applicant also indicated that the later 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,” were considered in the evaluation. The applicant applied environmental correction factors to the calculated fatigue usage factor at those component locations with specific fatigue calculations. The applicant stated that USAS B31.1 was the design code for three of the component locations and, consequently, detailed fatigue calculations were not available for these locations. The applicant indicated that the component analysis results reported in NUREG/CR-6260 were used to develop Ginna-specific environmental fatigue calculations. The applicant stated that all locations were found acceptable for the period of extended operation, with the exception of the pressurizer surge line. In RAI 4.3.7-1, the staff requested that the applicant provide the following information for each of the six component locations listed in NUREG/CR-6260:

- For those locations with existing fatigue analyses, provide the results of the fatigue usage factor calculation, including the calculated environmental multiplier (F_{en}). Show how F_{en} was calculated.

- For the USAS B31.1 locations discussed in Section 4.3.7.3 of the LRA, describe the fatigue usage factor calculation and provide the calculated fatigue usage factor. Include a detailed comparison of the Ginna Station components with the components listed in NUREG/CR-6260 and discuss the significance of the differences. This comparison should also include any differences in the thermal sleeve designs. In addition, provide a comparison of the design transients used in the analysis of the NUREG/CR-6260 components with the anticipated transients for the Ginna Station components.

The applicant’s June 10, 2003, response indicated that a plant-specific fatigue analysis was performed for all six component locations. The analyses concluded that acceptable fatigue usage existed at all six locations.

The design fatigue usage factors for the Ginna reactor vessel are provided in Table 5.3-8 of the UFSAR. The applicant multiplied the design usage factors for the RPV lower shell and the RPV inlet/outlet nozzles by the maximum environmental factor for low alloy steel components. The resulting CUFs were below 1.0. The staff notes that the environmental multipliers as a function of temperature shown in the applicant's response are incorrect. The environmental factor should be a constant value for low oxygen environments. However, because the applicant used the highest calculated environmental factor, the applicant's evaluation of these locations is conservative. The staff finds the applicant's evaluation of the effect of the environment on the RPV shell and the RPV inlet/outlet nozzles acceptable.

The staff notes that the highest reported CUF in Table 5.3-8 of the UFSAR for the RV nozzles is at the safety injection nozzle. This location was not identified in NUREG/CR-6260 as a high fatigue usage location for an older vintage Westinghouse PWR. The facility evaluated in NUREG/CR-6260 was a three loop Westinghouse PWR that did not have a safety injection nozzle located on the reactor vessel. The staff requested that the applicant provide additional discussion regarding this nozzle. In its July 16, 2003, response the applicant indicated that the design usage factor for the nozzle is much less than the usage factor reported in Table 5.3-8 of the UFSAR. The applicant provided additional information regarding the safety injection nozzle in a letter dated August 8, 2003. The applicant compared the Ginna nozzle to a similar nozzle at the Point Peach facility. Point Beach is a two loop Westinghouse PWR designed to standard Westinghouse transients. The Point Beach design usage factor is also much lower than the value reported in Table 5.3-8 of the UFSAR. The applicant multiplied the Point Beach design usage factor by the maximum environmental factor. The resulting usage factor was less than 1.0. On the basis of the applicant's comparison of the Ginna safety injection nozzle to a similar nozzle at the Point Beach facility, the staff finds the applicant has performed an acceptable evaluation of environmental effects for this nozzle. The applicant should correct the design usage factor for the safety injection nozzle provided in UFSAR Table 5.3-8.

The applicant calculated the design usage factor for the safety injection nozzle and the residual heat removal line tee connection. The applicant multiplied these usage factors by the maximum environmental factor for stainless steel components. The resulting CUFs were below 1.0. This is consistent with results presented in NUREG/CR-6260 for the safety injection nozzle and the residual heat removal line tee connection. On the basis of comparison of the applicant's results with the results presented in NUREG/CR-6260, the staff finds the applicant's evaluation of these locations acceptable.

The applicant used actual plant data for the loss of letdown flow transient to calculate the fatigue usage for the charging nozzle. The applicant indicated that the loss of letdown flow transient is the most significant contributor to the fatigue usage at this nozzle. Loss of letdown transients were also identified in NUREG/CR-6260 as the most significant contributors to fatigue usage of the charging nozzle. The use of actual transient data resulted in a relatively low CUF for the charging nozzle. The applicant provided further discussion of fatigue usage of the charging nozzle in a supplemental response dated July 16, 2003. The applicant indicated that the fatigue usage for the charging nozzle is based on plant-specific geometry and transients, and that the calculated Ginna usage factor is comparable to the usage factor shown in NUREG/CR-6260. The staff agrees with the applicant that the calculated Ginna charging nozzle usage factor is reasonable in comparison to the usage factor reported in NUREG/CR-6260. The staff also notes that the resulting Ginna charging nozzle CUF is less than 1.0. Therefore, the staff finds the applicant's evaluation of the charging nozzle acceptable.

The remaining locations evaluated by the applicant were the pressurizer lower head and surge line. As discussed in Section 4.3.2.1 of this SER, the applicant stated that detailed fatigue monitoring would be used to monitor the fatigue-sensitive pressurizer locations, including the environmental effects. The applicant's June 10, 2003, response provided additional details regarding its evaluation of the pressurizer heater penetration weld, surge line nozzle, and pressurizer lower head components. The applicant developed finite element models for the pressurizer surge leg nozzle and the RCS hot leg nozzle for use with the Fatigue Monitoring Program. The applicant uses FatiguePro to compute the incremental fatigue usage at these component locations for known plant transients using measured plant data. The applicant indicated that the fatigue usage at these component locations is primarily due to the temperature differentials that occur during plant heatup and cooldown cycles. The applicant used historical data from actual plant heatup and cooldown events to obtain the temperature differentials for early plant operations. The applicant performed an analysis using FatiguePro to estimate the fatigue usage resulting from these temperature differentials.

The applicant's evaluation of the surge leg nozzle and the RCS hot leg nozzle indicates that the CUF is not expected to exceed 1.0 during the period of extended operation. In addition, the applicant indicated that an evaluation of pressurizer heater penetration found that the CUF is not expected to exceed 1.0 during the period of extended operation. The applicant's evaluation is based on number of heatup and cooldown cycles specified in UFSAR Table 5.1-4. The staff finds the applicant's evaluation of these component locations reasonable. The applicant's Fatigue Monitoring Program will provide assurance that the number of design cycles will not be exceeded during the period of extended operation.

The applicant evaluated the effects of the reactor water environment on the fatigue sensitive locations at Ginna and concluded that the resulting fatigue usage will be acceptable for the period of extended operation. The applicant uses its Fatigue Monitoring Program to provide assurance that the number of design cycles used in the evaluations will not be exceeded in the period of extended operation. The staff finds that the applicant's Fatigue Monitoring Program provides an acceptable program for monitoring the environmental fatigue usage of fatigue-sensitive locations in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

The applicant's UFSAR Supplement for metal fatigue of environmentally assisted fatigue is provided in Section A3.3.5 of the LRA. The applicant provided an additional discussion of the pressurizer surge line in Section A3.3.6 of the LRA. The applicant should update the UFSAR supplement summary to include a description of the completed environmental fatigue evaluation of the pressurizer surge line as described above. This is Confirmatory Item 4.3-2.

4.3.3 Conclusions

On the basis of its review, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1), that, for the metal fatigue TLAA, the effects of aging on the intended functions will be adequately managed during the period of extended operation. Therefore, pending resolution of confirmatory item 4.3-1, the staff has reasonable assurance that the safety margins established and maintained during the current operating term will be maintained during the period of extended operation, as required by 10 CFR 54.21(c)(1).

4.4 Environmental Qualification of Electrical Equipment

The Ginna 10 CFR 50.49, Environmental Qualification Program has been identified as a TLAA for the purpose of license renewal. The applicant is considering for review as a TLAA only the environmental qualification (EQ) packages which indicate a qualified life of greater than 40 years. Equipment qualification packages that indicate a qualified life of less than 40 years are not considered TLAAs and will not be considered as such in the context of license renewal. The applicant stated that many of the EQ analyses may have been adequate under existing conditions and conform to 10 CFR 54.21(c)(1)(i); however, the applicant chose to be conservative and performed a confirmatory evaluation to verify that the assumptions in the existing analysis were adequate for the period of extended operation.

4.4.1 Summary of Technical Information in the Application

In LRA Section 4.4, the applicant describes its TLAA evaluation for EQ components. The applicant stated that many of the EQ analyses may have been adequate under existing conditions and conform to 10 CFR 54.21(c)(1)(i). However, the applicant was conservative and performed a confirmatory evaluation to verify that the assumptions in the existing analysis were adequate for the period of extended operation. The TLAAs for EQ components were evaluated in accordance with 10 CFR 54.21(c)(1).

On October 15, 2002, representatives of the applicant met with the staff to review a sample of the EQ calculations.

The following calculations were reviewed:

Item	Licensee Item No	
1.	4.4.1.4	Valcor Solenoid Operated Valve Models V526-6042-3 and V526-6042-17
2.	4.4.5.10	General PVC Insulated and Jacketed Control Cable
3.	4.4.5.14	Conax Core Exit Thermocouple Connector/Cable Assemblies
4.	4.4.5.3	Conax Electric Conductor Seal Assembly
5.	4.4.3.2	Westinghouse/WX32714 Electrical Penetration Assembly
6.	4.4.5.1	Kerite 600V HTK Insulated and, FR Jacketed Power Cable
7.	4.4.2.1	Westinghouse Containment Recirculation Fan Motor
8.	4.4.6.5	AMP Nuclear PIDG Window Butt Splices and Terminals
9.	4.4.7.2	Conax RTD Model 7DB9-10000
10.	4.4.4.4	Raychem WCSF-N Nuclear Low Voltage Tubing
11.	4.4.5.16	Brand-Rex XLPE Insulated and CSPE Jacketed Cable
12.	4.4.2.2	Limitorque Actuators, Outside Containment
13.	4.4.8	Victoreen High Range Radiation Monitor
14.	4.4.1.2	Valcor Solenoid Operated Valve Model V526-5440-2
15.	4.4.2.4	Limitorque Motor Operated Valve Model SMB-00-15 with Reliance Motor

The sample calculations represent certain equipment types including solenoid operated valves, electric motors, electrical penetration assemblies, heat shrink tubing, wire and cable, electrical connectors, resistance temperature detectors, and high range radiation monitors.

The applicant discussed items mostly falling under 10 CFR 54.21(c)(1)(ii), but also provided examples of the use of 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii). The Arrhenius methodology was used to perform the thermal aging evaluation, and the temperature data used in the evaluation were based on plant design temperature or on actual plant temperature data.

The applicant stated that, due to the date of installation of certain equipment, reanalysis was not required for license renewal purposes for equipment whose previous thermal and radiation analyses support a qualified life of 40 years.

The applicant also confirmed that there are no plans to extend the qualification of certain equipment which is expected to be replaced at the end of its qualified life. For the equipment that is not expected to be replaced at the end of its qualified life, the applicant will perform a reanalysis provided in NUREG-1800, Table 4.4-1.

4.4.2 Staff Evaluation

In accordance with 10 CFR 54.29, the staff reviewed the Ginna LRA, Section 4.4, “Environmental Qualification of Electric Equipment,” and Appendix B, Section B 3.1, “Environmental Qualification Program,” to determine if the applicant had demonstrated compliance with the requirement set forth in 10 CFR 54.21(c)(1) for EQ components. The staff also reviewed the following EQ guidance documents (as applicable), RG-1.97, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident,” Revision 3, Information and Enforcement Bulletin 79-01B, “Environmental Qualification of Class 1E Equipment;” NUREG-0588 (Category II), “Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment,” and the NRC guidance for addressing generic safety issue GSI-168 for license renewal as contained in a letter to NEI dated June 2, 1998.

In the LRA Appendix A, UFSAR Supplement, Section A3.4, “Environmental Qualification of Electric Equipment,” the applicant states that the EQ equipment is identified in the Ginna Station EQ Master List. Only the equipment qualification packages which indicate a qualified life of greater than 40 years were reviewed by the applicant as a TLAA. Equipment qualification packages that indicate a qualified life of less than 40 years are not TLAAs, as defined in 10 CFR 54.3, and therefore need not be discussed in the context of license renewal. The EQ re-analysis has been performed to verify extension of EQ qualification to 60 years. The reanalysis attributes and methodology descriptions used for EQ TLAAs include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions. These attributes are discussed in the following paragraphs.

<u>Analytical Methods</u>. The applicant uses the same analytical models as those in previous evaluations, and the Arrhenius methodology used by the applicant is the thermal model accepted by the staff for performing its current thermal aging analyses. The analytical method used by the applicant for radiation aging analysis demonstrates qualification for the total integrated dose, that is, the normal radiation dose for the projected installed life plus the accident radiation dose. The staff accepted this approach for the current operating term of 40 years.

For license renewal, the applicant states that it will establish the 60-year normal radiation dose by multiplying the 40-year normal radiation dose by 1.5 (60 years/40 years = 1.5). The result is added to the accident radiation dose to obtain the total integrated dose for each applicable component. For cyclical aging, a similar approach will be used. The staff determined that the applicant's approach for thermal, radiation, and cyclical aging is consistent with its CLB, and that it can be effective in determining the added aging for the period of extended operation.

Data Collection and Reduction Methods. The applicant states that reduction of excess conservatism in component service conditions (e.g., temperature, radiation, cycles) that was used in its prior aging analyses is the primary method that will be used for reevaluating the qualified life for the period of extended operation. The temperature used in an aging evaluation should be conservative and should be based on plant design temperature or actual plant temperature data. When used, temperature data can be obtained in several ways, including monitors, measurements made by plant operators during rounds, and temperature sensors on large motors. Plant temperature data may be used in an aging evaluation in different ways, such as (a) directly in the evaluation, or (b) to demonstrate conservatism when using plant design temperature for an evaluation.

Changes to material activation energy values as part of a reevaluation are to be justified on a component/materials-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.

The staff reviewed the applicant's data collection approach and found it to be conservative and bounding. The elimination of excessive conservatism is consistent with 10 CFR 50.49. In addition, the reduction method described by the applicant is also acceptable to the staff. The staff also agrees that changes to material activation energy values need to be determined on a component/material-specific basis.

Underlying Assumptions. The applicant states that EQ 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 and Corrective Action. The applicant states that under its Environmental Qualification Program, the reevaluation of an aging analysis could extend the qualified life of a component. If the qualified life of a component cannot be extended by reevaluation, the component must be refurbished, replaced, or requalified prior to exceeding the period for which the current qualification remains valid, in conformance with 10 CFR 50.49. Reevaluations must be performed in a timely manner, that is, the reevaluation must be completed with sufficient time available to refurbish, replace, or requalify the component prior to exceeding its qualified life if the reevaluation is unsuccessful.

4.4.3 Conclusions

On the basis of the review described above, the staff has determined that there is reasonable assurance that the applicant has adequately identified the TLAA for EQ components as defined in 10 CFR 54.3. In addition, the staff finds that, in combination with the staff's review of the Environmental Qualification Program, as documented in Appendix B of the LRA the applicant has demonstrated that the effects of aging on the intended functions of these components will be adequately managed during the period of extended operations, as required by 10 CFR 54.21(c)(1).

The staff has also reviewed the revised UFSAR Supplement for this TLAA and concludes that it provides an adequate summary description of the TLAA to satisfy 10 CFR 54.21(d).

4.5 Concrete Containment Tendon Prestress

The prestressing tendons in prestressed concrete containments lose their prestressing forces with time due to creep and shrinkage of concrete and relaxation of the prestressing steel. During the design phase, engineers estimate these losses to arrive at the end of operating life, normally 40 years. The operating experiences with the trend of prestressing forces indicate that the prestressing tendons lose their prestressing forces at a rate higher than predicted due to sustained high temperature. Thus, it is necessary to perform TLAAs for the period of extended operation. The adequacy of the prestressing forces in prestressed concrete containments is reviewed for the period of extended operation.

4.5.1 Summary of Information in the Application

The applicant briefly described the basic characteristics of the containment in Section 4.5 of the LRA. The applicant stated that the prestressing force of containment tendons may decrease over time due to creep, shrinkage, and elastic shortening of the concrete and stress relaxation of the prestressing tendon wires. The applicant indicated that prestressing tendon integrity is monitored and confirmed by the ASME Section XI, Subsection IWE/IWL Inservice Inspection Program.

The applicant provided the following description of its containment tendon prestress TLAA:

> An analysis was performed to evaluate the trend in the loss of prestress for each of the 160 tendons at Ginna Station. A review of the historical lift-off force measurements for the tendons was conducted. It was appropriate to review the results as two separate groups, i.e., the 23 tendons which were retensioned in 1969, and the 137 tendons which were retensioned in 1980. Of the 23 tendons that were retensioned in 1969, eleven have been tested during the surveillances since 1980. Of the 137 tendons that were retensioned in 1980, forty-seven have been tested during the subsequent surveillances. The number of tendons sampled during the surveillance tests exceeds the requirements of Regulatory Guide 1.35.
>
> Using the guidance in RG 1.35.1, tolerance bands were calculated and the lift-off forces measured during surveillance tests were expressed in terms of margins. It was concluded that the group of 23 tendons originally retensioned in 1969 should be retensioned as documented in the Evaluation of Loss of Prestress in Containment Tendons TLAA. These tendons have exhibited loss of prestress as determined during previous surveillance tests. Retensioning should preclude further loss of prestress.

Based on the analysis performed, the applicant concluded that retensioning the group of 23 tendons in 2005 will provide additional assurance that the minimum design tendon prestress force will be maintained through the period of extended operation.

4.5.2 Staff Evaluation

In addition to the review of Section 4.5 of the LRA, the staff reviewed the relevant information in Sections 2.4.1, 3.6, B2.1.3, B3.3, and Section 4.7.4 of the LRA. In the earlier years (1979—1980), the applicant had experienced lower prestressing forces than the minimum required value (MRV) estimated at 40 years. In order to understand the present state of the prestressing forces in the Ginna containment, the staff requested that the applicant provide the following additional information in RAI 4.5-1:

a) For the 137 tendons which were retensioned in 1980, the applicant is requested to provide the predicted lower limit line, MRV expected in 2005 and at 60 years (if not retensioned in 2005), a trend line for this group of tendons, and prestressing force values as points above and below the trend line measured during prior inspections.

b) The applicant is requested to provide the same information for the remaining 23 tendons for the inspections performed after 1969 retensioning.

c) In the operating experience element of Section B3.3 of the LRA, it indicates that 23 tendons out of 160 tendons were retensioned 1000 hours after initial prestressing. It is not clear if the 23 tendons retensioned after initial prestressing were parts of the randomly selected tendons in the subsequent surveillance of tendons performed as per Regulatory Guide 1.35, or IWL-2520. The applicant is requested to provide information regarding the trending of prestressing forces in these 23 tendons.

In response, the applicant provided the following information:

> a) Based on the available test information for this set of 137 tendons that were retensioned in 1980, the below graph shows the trend line from 1980 projected through 2005, and out to 2029. The graph also includes a constant line at 636 kips, which is the minimum design prestress force.

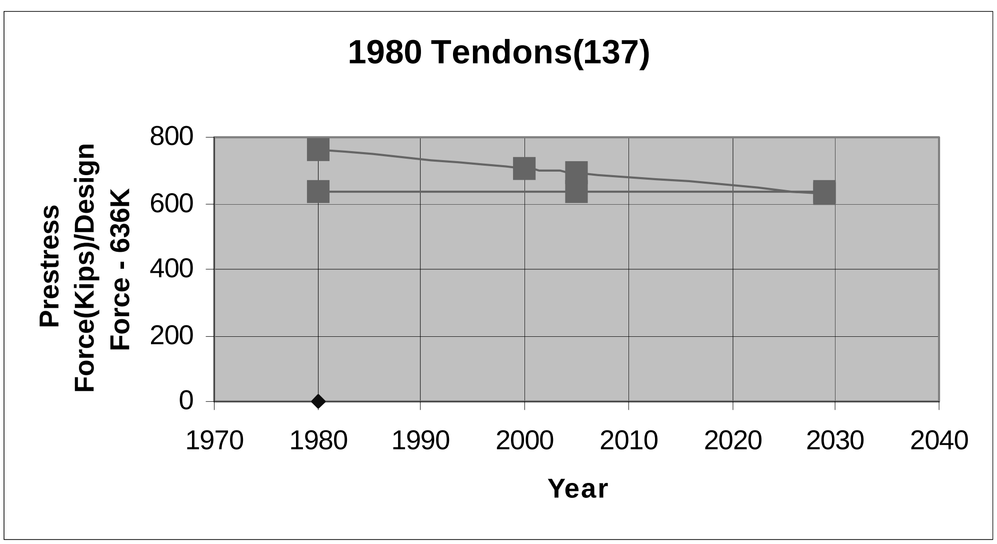
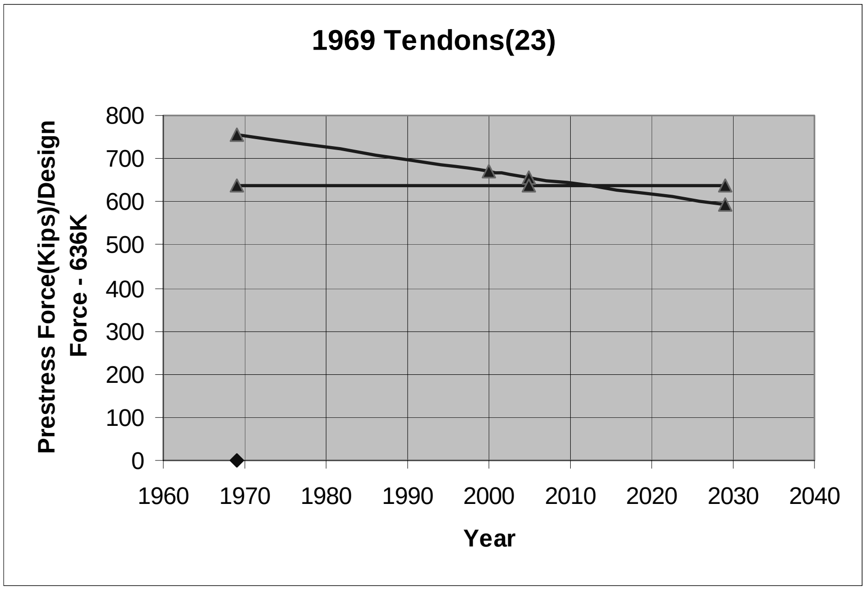


Figure 1

b) Based on the available test information for this set of 23 tendons that were retensioned in 1969, the below graph shows the trend line from 1969 projected through 2005, and out to 2029. The graph also includes a constant line at 636 kips, which is the minimum design prestress force.



c) Following the initial installation and retensioning of those 23 tendons in 1969, they were then included into the total population of tendons for the structure (160), which were subsequently randomly sampled and tested per Regulatory Guide 1.35 through the year 2000. The next scheduled tendon surveillance will be performed in the year 2005 and will be tested per the requirements of IWL-2520.

Since their retensioning in 1969, there have been 22 lift off tests performed on that population of 23 tendons.

The staff notes that in constructing the trend lines, it appears that the analyst has averaged the prestressing forces measured during each inspection. In Information Notice 99-10, the staff discouraged the averaging method. The regression analysis is more representative when each measured value is independently considered and the individual measured values are plotted on both sides of the trend line. The applicant was requested to show the individual measured values obtained during each inspection.

In the applicant's response dated July 16, 2003, the applicant provided the trend line based on the regression analysis of the measured tendon forces for the inspections performed after 1980 retensioning. On the basis of its review, the staff finds the process used in the trending analysis acceptable as it takes into account the contribution of all measured tendon forces in the regression analysis to arrive at the trend line. The staff considers the issue relative to RAI 4.5-1 to be resolved.

The applicant provided a qualitative description regarding the prestressing forces in the Ginna containment in Section A4.1 of the LRA (UFSAR Supplement). The staff believes that the applicant should, at a minimum, provide target prestressing forces that will be maintained at 40 years and 60 years. In RAI 4.5-2, the staff requested that the applicant supplement the present description in A4.1 with the basic quantitative description.

The applicant provided the following response to the staff request:

> The required minimum design prestress force for the containment structure of Ginna Station is 636 Kips. Based on data compiled for all tendons since 1969, the graphs presented in response to F-RAI 4.5.1 a) and 4.5.1 b) above, show the projected values for the 1969 and 1980 populations of tendons out to years 2005 and 2029. The prestress values for those two groups of tendons will provide a 4% and 8% margin above the minimum design force in 2005, and would fall below the minimum force if projected out to 2029. The tendons in the 1969 population will be retensioned in the 2005 surveillance and those in the 1980 population will be done in subsequent surveillances.

On the basis of the description provided in LRA Section A4.1, as supplemented by the above response, the staff considers the actions the applicant committed to in Section A4.1 of the LRA adequate to ensure the adequacy of prestressing forces in the containment during the period of extended operation.

4.5.3 Conclusions

On the basis of its review, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that for the concrete containment tendon prestress TLAA, the effects of aging on the intended functions will be adequately managed during the period of extended operation. The staff also concludes that the UFSAR Supplement contains an appropriate summary description of the concrete containment tendon prestress TLAA evaluation for the period of extended operation, as reflected in the license condition, to satisfy 10 CFR 54.21(d). Therefore the staff concludes that the safety margins established and maintained during the current operating term will be maintained during the period of extended operation, as required by 10 CFR 54.21(c)(1).

4.6 Containment Liner Plate and Penetration Fatigue

The interior surface of the concrete containment structure is lined with welded thin metallic plates to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment, as required by 10 CFR Part 50. At all penetrations, the liner plate is thickened to reduce stress concentrations.

Fatigue of the liner plates may be considered in the design based on an assumed number of loading cycles for the current operating term. The cyclic loads include reactor building interior temperature variation during the heatup and cooldown of the reactor coolant system, a loss-of-coolant accident, annual outdoor temperature variations, thermal loads due to high energy containment penetration piping lines (such as steam and feedwater lines), seismic loads, and pressurization due to periodic Type A integrated leak rate tests.

4.6.1 Summary of Technical Information in the Application

The applicant stated that the containment liner, liner penetrations, and liner steel components of the Ginna Station containment structure comply with the ASME Code Section III–1965 for pressure boundary and the American Institute of Steel Construction (AISC) Code for structural steel. The containment liner and penetrations were designed as Class B vessels. The Winter

1965 Addenda of ASME Section III, Subsection B, N-1314(a) requires that the containment vessel satisfy the provisions of Subsection A, N-415.1, “Vessels Not Requiring Analysis for Cyclic Operation,” in order for Subsection B rules to be applicable.

The applicant stated that a fatigue analysis of the containment penetrations at Ginna was not required, in accordance with the ASME Code, Section III 1965, N-415.1, provided the specified operation and service loading of the vessel or component meets the six conditions addressed in the following types of mechanical and thermal loads:

(1) atmospheric to operating pressure cycles
(2) normal service pressure fluctuations
(3) temperature difference–startup and shutdown
(4) temperature difference–normal service
(5) temperature difference–dissimilar materials
(6) mechanical loads

The pressure boundary components analyzed include the liner adjacent to the penetration, the penetration sleeve, and the annular plate connecting the pressure piping to the sleeve. The applicant stated that an analysis was performed which verified that each of these six conditions was satisfied for the period of extended operation, and therefore demonstrated that liner and penetrations comply with the ASME Section III, 1965, Code rules for fatigue through the period of extended operation. The applicant concluded that the fatigue analyses have been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

4.6.2 Staff Evaluation

In RAI 4.6-1, the staff requested that the applicant provide a list of the design transients and corresponding cycles that were specified in the design of the containment liner penetrations. In response, the applicant indicated that the design cycles for the containment liner penetrations were based on the design cycles listed in UFSAR Table 5.4-1. These design transients are discussed in the applicant’s response to RAI 4.3.1-1.

The applicant indicated, in response to RAI 4.3.1-1, that the projected cycles for all transients over a 60-year period are significantly less that the total cycles given in UFSAR Table 5.1-4 over a 40-year period. However, for conservatism, the total number of design cycles from Table 5.1-4, excluding startup and shutdown, was used for the 60-year period. The applicant’s response to RAI 4.3.1-1 also indicated that, based on operating experience, the total number of heatup/cooldown pressure cycles over a 60-year period is projected to be less than 120. The applicant used 120 heatup/cooldown cycles for the evaluation instead of the 200 cycles specified in UFSAR Table 5.1-4.

The applicant summarized the six conditions and the corresponding design cycles as follows:

(1) atmospheric to operating pressure cycles	120
(2) normal service pressure fluctuations	17,145
(3) temperature difference–startup and shutdown	120
(4) temperature difference–normal service	17,145
(5) temperature difference–dissimilar materials	1,000.000

(6) mechanical loads 10

The staff finds these transients and their cycles acceptable because they were shown to conform with cyclic transients prescribed in Table 5.1-4 of the UFSAR, and because they are consistent with cycles provided in response to RAI 4.3.1-1.

In RAI 4.6-2, the staff requested that the applicant demonstrate that, for the penetration sleeve and annular plate connecting pressure piping to the sleeve, the six conditions stated in ASME Section III, Subsection A, N-415.1, 1965, will be satisfied for the period of extended operation.

In response, the applicant stated that the pressure boundary components evaluated in the calculation included the liner adjacent to the penetration, the penetration sleeve, and the annular plate. The liner and all penetration sleeves are made of carbon steel. Most of the annular plates are also carbon steel, and the rest are made of stainless steel. Because the allowable alternating stress intensity, Sa, for stainless steel at any specific number of cycles is always greater than that allowed for carbon steel, as shown in ASME Code Figures N-415(A) and N-415(B), the design fatigue curve for carbon steel was used in all calculations.

- Condition 1, “Atmospheric to Operating Pressure Cycles,” requires that the projected number of heatup/cooldown pressure cycles be less than the allowable cycles corresponding on the material fatigue curve to an alternating stress value Sa equal to 3Sm, where Sm is the design stress intensity for the material at temperature. The pressure boundary components evaluated included the liner adjacent to the penetration, the penetration sleeve, and the annular plate connecting the pressure piping to the sleeve. For these components, the applicant determined the allowable cycles from the applicable design fatigue curve for carbon steel. On this basis, the applicant showed that Condition 1 was met because the projected number of heatup/cooldown cycles (120), was below 1500 cycles, the allowable number of cycles corresponding to 3Sm.

- Condition 2, “Normal Service Pressure Fluctuations,” requires that the full range of the pressure fluctuations that occur during normal operation not exceed an allowable full range pressure fluctuation equal to the quantity design pressure x (Sa/3Sm), where Sa is determined from the applicable design fatigue curve at temperature for the total specified number of significant pressure fluctuations. Based on the total number of significant pressure fluctuations of 17,145 and a design pressure of 60 psi, the applicant determined the allowable full range pressure fluctuation to be 26 psi. The pressure fluctuations in the containment during normal operation are ordinarily atmospheric fluctuations and therefore negligible compared to the allowable full range pressure fluctuation. The applicant has therefore demonstrated that Condition 2 is satisfied

- Condition 3, “Temperature Difference–Startup and Shutdown” requires that the temperature difference in degrees F between any two adjacent points of the component during normal operation, and during startup and shutdown, not exceed an allowable temperature difference equal to Sa/2Eα, where Sa is the value obtained from the applicable design fatigue curve for the specified number of startup shutdown cycles, and E and α are the modulus of elasticity and the instantaneous coefficient of thermal expansion, respectively, at the mean value of the temperatures at the two points.

The applicant stated that the maximum temperature difference between any two points, and the maximum temperature difference between adjacent points, occurs at the main steam line penetration, where an annular plate joins the penetration sleeve. The mean temperature of the insulated steam line is 530 °F, and the containment ambient temperature is 80 °F. The maximum temperature difference at this location is therefore 450 °F. Based on an assumed 120 full startup/shutdown cycles over 60 years, the allowable temperature difference was calculated as 462 °F. The maximum temperature difference is smaller than the allowable temperature difference, and, therefore, this condition was shown to be satisfied.

- Condition 4, "Temperature Difference–Normal Service," requires that the temperature difference in degrees F between any two adjacent points of the component during normal operation not exceed an allowable temperature difference equal to $Sa/2E\alpha$, where Sa is the value obtained from the applicable design fatigue curve for the total number of significant temperature difference fluctuations, and E and α are the modulus of elasticity and instantaneous coefficient of thermal expansion, respectively, at the mean value of the temperatures at the two points.

  The applicant stated in UFSAR Section 3.2.2.1.5 that during normal operation, the temperature in the main steam line fluctuates between 514 °F and 547 °F. The containment inside temperature fluctuates approximately ± 20 °F from the mean. Conservatively assuming that these two temperature fluctuations occur totally out-of-phase results in a maximum fluctuation range of 73 °F, and assuming 17,145 significant temperature difference fluctuations, the allowable temperature difference was calculated to be 77 °F. The maximum temperature fluctuation is smaller than the allowable temperature difference, and, therefore, this condition was shown to be satisfied.

- Condition 5, "Temperature Difference–Dissimilar Materials," requires that for components fabricated from materials of differing modulii of elasticity and/or coefficients of thermal expansion, the total range of temperature fluctuations experienced by the component during normal operation shall not exceed the allowable temperature fluctuation $Sa/[2(E_1\alpha_1-E_2\alpha_2)]$, where Sa is the value obtained from the applicable design fatigue curve for the total specified number of significant temperature fluctuations. A fluctuation shall be considered significant if its total excursion exceeds the quantity $S/[2(E_1\alpha_1-E_2\alpha_2)]$, where S is the value obtained from the applicable design fatigue curve for 10^6 cycles.

  The applicant stated that the only dissimilar material interface in a penetration occurs at the junction of a carbon steel sleeve and an austenitic stainless steel annular plate. Based on the material properties of the two metals at operating temperature, a temperature fluctuation was determined to be significant if it exceeds 81 °F. During normal operations, the temperature fluctuations at the junction of a penetration sleeve and an annular plate are less than this value. Therefore, there are no significant temperature fluctuations and this condition is shown to be satisfied.

- Condition 6, "Mechanical Loads," requires that the specified full range of mechanical loads, excluding pressure but including pipe reactions, shall not result in load stresses whose range exceeds the Sa value from the applicable design fatigue curve for the total specified number of significant load fluctuations.

The applicant stated that the only mechanical loads acting on the penetrations are dead load and, pressure and seismic loads, and the corresponding piping reactions. Only seismic loads need be considered since dead loads are not cyclic. The applicant did a bounding calculation. The number of maximum stress cycles that was considered during an safe-shutdown earthquake event was 10. At 10 cycles, the corresponding value of Sa equals 550 ksi. The largest possible elastic peak stress, calculated as the product of the highest allowable primary plus secondary stress intensity 3Sm for all materials considered in the penetrations, and the largest stress intensification factor specified in N-415.3, of value 5, was determined as 350 ksi. This peak stress is significantly less than the Sa, and, therefore, this condition is shown to be satisfied.

The staff has evaluated the responses to the RAIs and concludes that the applicant has provided a reasonable demonstration that the conditions for exclusion of a Section III fatigue analysis of the containment liner and liner penetrations have been met, in accordance with the provisions of ASME Section III, Subsection A, N-415.1, 1965.

In RAI 4.6-3, the staff requested that the applicant indicate whether the hot piping penetration assemblies contain bellows, and to provide the justification for not identifying fatigue of the bellows as TLAAs in accordance with 10 CFR 54.3. In response, the applicant stated that the hot and cold mechanical penetrations at Ginna Station contain bellows, which, by design, do not perform a containment isolation function. The only functions of these bellows is to accommodate lateral and axial pipe movements. Therefore, the applicant concluded that fatigue of these bellows is not a TLAA as defined in 10 CFR 54.3. The staff agrees with the applicant's conclusion.

4.6.3 Conclusions

On the basis of its review, the staff concludes that the applicant has demonstrated that the containment liner penetrations fatigue TLAA has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii). The staff also concludes that the UFSAR Supplement contains an appropriate summary description of the containment liner plate and penetrations fatigue TLAA evaluation for the period of extended operation, as reflected in the license condition, to satisfy 10 CFR 54.21(d). Therefore the staff has reasonable assurance that the safety margins established and maintained in the containment liner plate and penetrations fatigue TLAA during the current operating term will be maintained during the period of extended operation, as required by 10 CFR 54.21(c)(1).

4.7 Other Plant-Specific Time-Limited Aging Analyses

There are certain plant-specific safety analyses that may have been based on an explicitly assumed 40-year plant life and may, therefore, be TLAAs. Pursuant to 10 CFR 54.21(c)(1), a license renewal applicant is required to evaluate TLAAs, as defined in 10 CFR 54.3. License renewal reviews focus on the period of extended operation.

The applicant has identified seven additional TLAAs for license renewal:

(1) containment liner stress

(2) containment tendon fatigue
(3) containment liner anchorage fatigue
(4) containment tendon bellows fatigue
(5) crane load cycle limit
(6) reactor coolant pump flywheel
(7) thermal aging of cast austenitic stainless steel

The staff's evaluation of these TLAAs is provided below.

4.7.1 Containment Liner Stress

4.7.1.1 Summary of Technical Information in the Application

The containment liner is fabricated from carbon steel plate conforming to ASTM A442-60T Grade 60 with a minimum yield stress of 32,000 psi, and a buckling stress of 16,600 psi at operating conditions. The liner plate thickness is 1/4" for the base and 3/8" for the cylinder and dome. The applicant stated that the liner meridional stress due to initial prestressing was calculated as 4500 psi in compression. As a result of concrete strain due to shrinkage and creep, the liner stress was projected to increase to 14,100 psi at the end of 40 years. The creep and shrinkage strain that will occur at the end of the 60-year plant life has been evaluated, and the resulting compressive liner stress due to both time-dependent and non-time-dependent loads was determined to be 14,870 psi. The liner stress has thus been determined to be less than the liner buckling stress of 16,600 psi for the period of extended operation.

4.7.1.2 Staff Evaluation

Based on the values provided by the applicant, the staff finds the applicant's evaluation of the additional compressive stress induced by concrete shrinkage and creep into the liner reasonable and acceptable. The staff also concludes that, based on the information discussed above, the applicant has performed an acceptable TLAA to demonstrate that adequate margin against liner buckling will be maintained for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

4.7.1.3 Conclusions

On the basis of its review, the staff concludes that the applicant has demonstrated that the containment liner stress TLAA has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii). The staff also concludes that the UFSAR Supplement contains an appropriate summary description of the containment liner stress TLAA evaluation for the period of extended operation, as reflected in the license condition, to satisfy 10 CFR 54.21(d). Therefore the staff has reasonable assurance that the safety margins established and maintained in the containment liner stress TLAA during the current operating term will be maintained during the period of extended operation, as required by 10 CFR 54.21(c)(1).

4.7.2 Containment Tendon Fatigue

4.7.2.1 Summary of Information in the Application

The applicant provided the following description of its containment tendon fatigue TLAA:

> A discussion of seismic considerations for tendons is provided in the Ginna UFSAR. Fatigue tests were conducted on tendon wire materials in 1960 by an independent testing lab. The tests indicated that the tendons were capable of withstanding over 2 million cycles at stress levels between 135 and 158 ksi. The test results were used to conclude that dynamic loads, considering especially pulsating loads resulting from an earthquake, do not jeopardize buttonhead anchorage. This discussion may not meet the definition of a TLAA as described in 10 CFR 54.3, however it has been included for conservatism.

Furthermore, the applicant asserts that the tendons were tested to 2 million cycles, which exceeds, by many orders of magnitude, the total cycles that could accumulate through multiple seismic events over 60 years.

Based on the qualitative analysis, the applicant concluded that the seismic fatigue evaluation remains valid through the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.7.2.2 Staff Evaluation

Although the applicant indicated that tendon fatigue may not meet the definition of a TLAA as described in 10 CFR 54.3, the applicant performed the TLAA evaluation to be conservative. The cyclic loading testing is a part of the prequalification testing of the tendon anchorage hardware to ensure its integrity under normal fluctuating loads and occasional vibratory loads (due to design basis seismic loads) induced in the hardware. The prequalification tests are marginally relevant to the TLAA as the tests were not performed on the aged hardware components. These prequalification tests are required for all prestressing tendon systems utilized in the containments of nuclear power plants. Recognizing the requirements of prequalification tests, and that the TLAA performed for the prestressing tendon force in Section 4.5 of this LRA, provides necessary information regarding the time-limited aging effects on prestressing tendon systems, the staff does not require a TLAA for tendon fatigue.

The applicant has summarized the TLAA supporting activities for the containment tendon fatigue in the UFSAR Supplement provided in Section A3.5.1 of the LRA. The description is identical to that provided in Section 4.7.4 of the LRA.

4.7.2.3 Conclusions

On the basis of the TLAA review of Section 4.7.2, “Containment Tendon Fatigue,” of the LRA, the staff concludes that the analysis reinforces the staff’s conclusion in Section 4.5 of this SER that the containment tendons system will perform its intended function during the extended period of operation.

4.7.3 Containment Liner Anchorage Fatigue

4.7.3.1 Summary of Technical Information in the Application

The applicant stated that a fatigue analysis of the fillet weld attaching the channel anchors to the containment liner was performed as part of the original design. The allowable fatigue stress of the attachment weld was set equal to the stress caused by static loading. This stress equals 13,600 psi and corresponds to 100,000 allowable stress cycles. The applicant also indicated that this analysis may not meet the definition of a TLAA as described in 10 CFR 54.3.

4.7.3.2 Staff Evaluation

In RAI 4.7.3-1, the staff requested that the applicant provide the design transients and corresponding cycles which generated the static stress of 13,600 psi in the fillet weld. The applicant stated that the fillet weld attaching the channel anchors to the liner was designed for 100,000 full stress cycles, which corresponds to 13,600 psi on the design fatigue curve for carbon steel. The only cyclic loads on the fillet weld are those caused by the temperature and pressure fluctuations in the containment. The static stress value bounds the stresses due to these fluctuations.

In RAI 4.7.3-2, the staff requested that the applicant provide the design code to which the fatigue analysis of the fillet welds was performed. In response, the applicant stated that the fillet welds were designed to AISC Specification, Section 1.7.2 and other non-ASME structural codes. The staff finds this acceptable because it conforms with industry practice for this vintage plant.

In RAI 4.7.3-3, the staff requested that the applicant provide justification for not applying a fatigue strength reduction factor to the static stress for determining the allowable cycles for the fillet weld. In response, the applicant stated that the static stress was taken as the allowable fatigue stress in the original containment design, in accordance with the design codes used to design the channel anchors, therefore a fatigue strength reduction factor was not applicable. This stress corresponds to 100,000 cycles, which correlates to more than 4 full stress cycles per day for 60 years. The pressure and temperature fluctuations in the containment are not of sufficient magnitude to cause four full cycles of design basis stress at the liner anchorage weld every day. The original fatigue analysis therefore remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i). The staff agrees with the applicant's assessment that the number of full pressure and temperature fluctuations will be less than 100,000 cycles for 60 years of plant operation.

In RAI 4.7.3-4, the staff requested the applicant to clarify why it thought this fatigue analysis may not meet the definition of a TLAA, as described in 10 CFR 54.3. In response, the applicant stated that this analysis was located in vendor calculations, and it is not apparent that it was submitted to the NRC and made part of the Ginna CLB. Therefore, the sixth element for defining a TLAA may not have been met. The applicant indicated that it took a conservative approach when it evaluated this issue as a TLAA.

4.7.3.3 Conclusions

On the basis of its review, the staff concludes that the applicant has demonstrated that the containment liner anchorage fatigue analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i). The staff also concludes that the UFSAR Supplement contains an appropriate summary description of the containment liner anchorage fatigue evaluation for the period of extended operation to satisfy 10 CFR 54.21(d). Therefore the staff has reasonable assurance that the safety margins established and maintained during the current operating term in the containment liner anchorage fatigue analysis will be maintained during the period of extended operation, as required by 10 CFR 54.21(c)(1).

4.7.4 Containment Tendon Bellows Fatigue

4.7.4.1 Summary of Information in the Application

The applicant provided the following description of the TLAA related to the fatigue consideration of the containment tendon bellows:

> The allowable radial and vertical displacements of the containment stainless steel tendon bellows are given in the UFSAR and are limited to two cycles per year for the 40-year life of the plant. This limits the total number of allowable displacement cycles to 80. Since the completion of construction, displacements at the tendon bellows have occurred due to pressure testing and temperature changes in the cylindrical shell wall due to summer/winter conditions and reactor shutdown during refueling outages. [Furthermore]–this discussion may not meet the definition of a TLAA as described in 10 CFR 54.3, however it has been included for conservatism.

The applicant also stated, "assuming that 80 full cycles of allowable displacement results in a fatigue usage factor of 1.0, the actual fatigue usage factor over a 60-year period has been calculated to be much less than 0.01." Therefore, the applicant concluded that the original fatigue analysis remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.7.4.2 Staff Evaluation

The Ginna containment is a unique containment, for which the hinge formation of the containment shell has been deliberately designed. As part of the design, to alleviate any extraneous bending and shear forces, the applicant has provided stainless steel bellows in the tendon sheathing. The applicant is prudent in performing TLAA of the bellows for the period of extended operation.

In RAI 4.7.4-1, the staff requested the applicant to provide the UFSAR reference regarding the allowable radial and vertical displacements of the containment stainless steel tendon bellows and calculations showing the fatigue usage factor less than 0.01.

In response, the applicant provided the following information.

> The containment stainless steel tendon bellows were designed to the requirements of USAS B31.1 Code for pressure piping. Figure 3.8-18 is referenced in UFSAR Section 3.8.1.4.4.1. Figure 3.8-18 contains the following information:
>
> (1) Movements:
> Case (1)–From undeflected position vertically downward 0.14 inches
> Case (2)–From above position vertically upward 0.10 inches and simultaneous laterally 0.16 inches
> (2) Fatigue: Two cycles per year
> (3) Working Pressure: 60 psig
> Test Pressure: Hydrostatic at 150% of working pressure
> Pneumatic at 125 percent of Working Pressure

(4) Maximum working temperature: 160 °F
(5) Standard Specification: ASA B31.1 Code for Pressure Piping
(6) Test two random assemblies for specified movements

Based on this information, the total number of thermal cycles including those accumulated during the period of extended operation would not exceed the allowable number of cycles (7000) for a stress range reduction factor of 1.0 in the USAS B31.1 pressure piping code.

A calculation of CUF for the tendon bellows for the period of extended operation was also performed as a TLAA in DA-CE-2002-016-07. This calculation is presented below:

The allowable radial and vertical displacements of the containment stainless steel tendon bellows are given in UFSAR Figure 3.8-18 and are limited to two cycles per year for the 40 year life of the plant. This limits the total number of allowable displacement cycles to 80. Assume that 80 cycles of allowable displacement results in a fatigue usage factor of 1.0.

From ASME Code Figure N-415(B) for stainless steel, the allowable alternating stress amplitude, Sa, for 80 cycles is 260 ksi. Assume that this is the peak stress produced in the bellows due to the combined effect of design basis vertical and radial displacements of 0.10" and 0.16" given in UFSAR Figure 3.8-18, and that this stress is directly proportional to the absolute sum of the displacements. Thus, Sa = 260 ksi is produced at an absolute displacement of 0.26".

The absolute magnitude of the combined vertical and radial displacements that actually occur is 0.030" + 0.014" = 0.044". This displacement corresponds to a bellows stress level of (0.044/0.26)(260) = 44 ksi.

The alternating stress intensity amplitude of 44 ksi corresponds to 40,000 cycles. The fatigue usage factor is 144/40,000 = 0.004.

Therefore, the structural integrity of the tendon bellows will be maintained through the period of extended operation.

The applicant was requested to provide the bases for (1) 0.030 in (vertical) displacement, (2) 0.0014 in (radial) displacement, and (3) 144 cycles used in the final fatigue usage factor calculations.

In Attachment 3 of the applicant’s letter dated July 16, 2003, the applicant provided a detailed analysis of how it arrived at the vertical and horizontal displacement as well as a basis for using 144 cycles in the tendon bellows fatigue analysis. The analysis is based on the observed displacements of the containment wall during the structural integrity tests (SITs) performed in 1969 and 1996. In calculating the number of cycles of the bellows movements in 60 years, the applicant considered 4 SITs, 20 Integrity Leakage Rate Tests, 60 seasonal temperature variation cycles, and 60 shutdown/startup cycles. The staff considers the method of calculating the fatigues usage factor for the tendon bellows acceptable, as it utilizes the actual observations in establishing the bellows movements, and conservative ways of calculating the number of cycles. The staff considers the issue relative to RAI 4.7.4-1 to be resolved and that the fatigue analysis remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

The applicant has summarized the TLAA supporting activities for the concrete containment tendon force in Section A3.5.2. The description is similar to that provided in Section 4.7.4 of the LRA, except that it provides the fatigue usage factor of 0.004 for the tendon bellows. The staff considers the description adequate for the UFSAR Supplement.

4.7.4.3 Conclusions

On the basis of the review of Section 4.7.4, “Containment Tendon Bellows Fatigue,” of the LRA, the staff concludes that there is a reasonable assurance that the containment tendons bellows will perform their intended function during the extended period of operation, provided their aging degradation is managed or monitored; and that the fatigue analysis remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i). The staff also concludes that the UFSAR Supplement contains an appropriate summary description of the containment tendon bellows fatigue TLAA evaluation for the period of extended operation, as reflected in the license condition, to satisfy 10 CFR 54.21(d).

4.7.5 Crane Cycle Load Limits

4.7.5.1 Summary of Technical Information in the Application

The applicant states that each of the crane estimated cycle numbers were compared to the design load cycles. They are all well below the upper design loading cycle limit. In addition, the average percent of the rated load lifted was well below the 50 percent level, relative to the design load cycles, as set forth in the design criteria.

Because the number of operating load cycles for the cranes will be less than the design cycles and the average percent of rated load lifted is less than 50 percent for the design load cycles, the applicant contends that the crane designs will remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

4.7.5.2 Staff Evaluation

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

In RAI 4.7.5-1, the staff requested the applicant to provide the estimated number of load cycles and the assumptions used in the estimation. The applicant was also requested to provide the upper design loading cycle limit for each crane within the scope of license renewal.

In its response dated May 28, 2003, the applicant provided the requested data which was extracted from the applicant’s design analysis document, DA-2002-016-03, “Containment Time-Limited Aging Analyses For License Renewal Calculation of Crane Load Cycles and Fatigue Evaluation.” The data indicate that most loads are significantly below the capacity of cranes. Where the loads are near or comparable to the rated capacity of the cranes, the number of cycles is significantly below the design loading cycles, reducing fatigue loading on the cranes.

The staff finds the applicant’s response acceptable because the data provided by the applicant support its contention that the crane designs will remain valid for the period of extended operation.

Section 4.7.5 of the LRA states that the average percent of the rated load lifted is less than 50 percent for the design load cycles. In RAI 4.7.5-2, the staff requested the applicant to provide assurance that the percentage will not change during the period of extended operation.

In its response dated May 28, 2003, the applicant stated that the percentages are based on the types of loads the cranes are designed to lift versus their capacity. The applicant indicated that any projected load lifts above 50 percent capacity have a frequency that is only a small percentage of the design loading cycles. The applicant also indicated that the crane duty is not expected to change between now and 2029, and that any significant changes would be subject to engineering analysis. The staff finds the applicant's response satisfactory and acceptable because the data provided by the applicant support this assessment.

4.7.5.3 Conclusions

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

4.7.6 Reactor Coolant Pump Flywheel

The function of the reactor coolant pump (RCP) in the reactor coolant system of a PWR plant is to maintain an adequate cooling flow rate by circulating a large volume of primary coolant water at high temperature and pressure through the reactor coolant system. A concern about overspeed of the RCP and its potential for failure led to the issuance of RG 1.14 in 1971. The regulatory position of RG 1.14 concerning ISI calls for an in-place ultrasonic volumetric examination of the areas of higher stress concentration at the bore and keyway at approximately 3-year intervals and a surface examination of all exposed surfaces and complete ultrasonic volumetric examination at approximately 10-year intervals. The flywheel inspection schedule is to coincide with the individual plant's ISI schedule as required by Section XI of the ASME Code.

In January 1996, Westinghouse submitted WCAP-14535, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination." This report, which provides engineering analysis based on fracture mechanics, is intended to eliminate RCP flywheel ISI requirements for all operating Westinghouse plants and some Babcock and Wilcox Plants. However, the NRC safety evaluation of WCAP-14535 stated, "The staff believes that even for flywheels meeting all the design criteria of RG 1.14, as modified in this SER, inspections should not be completely eliminated." The NRC safety evaluation went on to say–

> ...the staff finds the following acceptable:
>
> (1) Licensees who plan to submit a plant-specific application of this topical report for flywheels made of SA 533 B material need to confirm that their flywheels are made of SA 533 B material. Further, licensees having Group-15 flywheels need to demonstrate that material properties of their

A516 material is equivalent to SA 533 B material, and its reference temperature, RT_{NDT}, is less than 30 °F.

(2) Licensees who plan to submit a plant-specific application of this topical report for their flywheels not made of SA 533 B or A516 material need to either demonstrate that their flywheel material properties are bounded by those of SA 533 B material, or provide the minimum specified ultimate tensile stress, S_u, the fracture toughness, K_{1c}, and the reference temperature, RT_{NDT}, for that material. For the latter, the licensees should employ these material properties, and use the methodology in the topical report, as extended in the two responses to the staff's RAI, to provide an assessment to justify a change in inspection schedule for their plants.

(3) Licensees meeting either (1) or (2) above should either conduct a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or conduct a surface examination (MT and/or PT) of exposed surfaces defined by the volume of the disassembled flywheels once every 10 years. The staff considers this 10-year inspection requirement not burdensome when the flywheel inspection is conducted during scheduled ISI inspection or RCP motor maintenance. This would provide an appropriate level of defense in depth.

4.7.6.1 Technical Information in the Application

Westinghouse Topical Report WCAP-14535 presents an evaluation of the probability of failure over an extended operating period of 60 years. This report demonstrates that the flywheel design has a high structural reliability with very high flaw tolerance and negligible flaw crack extension over a 60-year service life. Based on WCAP-14535A, and in accordance with NRC recommendations, Robinson Gas and Electric Corporation (RG&E) requested and received a relief request from the NRC allowing it to revise the ISI frequency of flywheel examination to once every 10 years.

4.7.6.2 Staff Evaluation

By letter dated March 18, 1997, RG&E submitted for staff review its assessment of the plant-specific applicability of WCAP-14535 for Ginna. In Ginna's submittal, the licensee confirmed that its flywheels are made of SA 533 B material, and they do not belong to either group 10 or Group 15 flywheels by WCAP-14535 definitions. Therefore, the plant-specific applicability of WCAP-14535 to RG&E had been established, and the 10-year inspection requirement with details specified in the NRC safety evaluation of WCAP-14535 were acceptable. Furthermore, fatigue crack growth calculations from WCAP-14535 showed that for 60 years of operation, crack growth from large postulated flaws in each of the flywheel groups is only a few mils. Therefore, the flywheel inspections completed prior to service are sufficient to ensure their integrity during service.

4.7.6.3 UFSAR Supplement

On the basis of the staff's evaluation described above, the summary description for the RCP flywheel TLAA in the UFSAR Supplement (LRA, Appendix A), as modified in a May 23, 2003, letter, provides an adequate description of this TLAA, as required by 10 CFR 54.21(d).

4.7.6.4 Conclusions

The staff has reviewed the TLAA on RCP flywheels. On the basis of this evaluation, the staff concludes that the inservice inspection program requirements for the RCP flywheels at RG&E

will continue to ensure that the effects of aging on the intended functions will be adequately managed for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii).

4.7.7 Thermal Aging of Cast Austenitic Stainless Steel

4.7.7.1 Summary of Technical Information in the Application

The applicant described the performance of TLAAs for thermal aging of cast austenitic stainless steel (CASS) for the period of extended operation in Section 4.7.7 of the LRA. Two TLAAs were performed which were documented respectively in WCAP-15837, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the R.E. Ginna Nuclear Power Plant for the License Renewal Program," April 2002, and WCAP-15873, "A Demonstration of the Applicability of ASME Code Case N-481 to the Primary Loop Casings of R.E. Ginna Nuclear Power Plant for the License Renewal Program," April 2002.

In WCAP-15837, a leak before break (LBB) analysis was performed to demonstrate that leaks from through-wall cracks in RCS piping would be detected by plant monitoring systems before the cracks become unstable. The analysis considered the reduction of fracture toughness in CASS elbows due to thermal aging and assessed the crack stability in the reactor coolant piping for the period of extended operation. The results of the analysis showed that a significant margin exists between detectable flaw sizes and critical flaw sizes.

In WCAP-15873, a fracture mechanics analysis (flaw tolerance) was performed for the CASS RCP casings according to the requirements of ASME Code Case N-481 for the period of extended operation. The results of the analysis showed that the stability criteria will be met with the fracture toughness of the pump casing materials in a fully aged condition. Therefore, for the inservice inspection of CASS pump casings at the Ginna station, the alternative visual examinations as delineated in Code Case N-481 can be performed in lieu of the volumetric examinations required by ASME Code, Section XI.

4.7.7.2 Staff Evaluation

Thermal aging refers to the gradual change in the microstructure and properties of a susceptible material due to its exposure to elevated temperature for an extended period of time. Thermal aging may result in a reduction of the fracture toughness of a susceptible material such as CASS, since the thermal aging embrittlement effect (loss of fracture toughness) is a time-dependent phenomena. The associated aging effect requires a TLAA to ensure that it will be adequately managed through the extended period of operation.

To support its review of Ginna's LRA, the staff requested additional information from the applicant. In RAI 4.7.7-1, the staff requested the applicant to confirm whether the two Westinghouse reports (WCAP-15837 and WCAP-15873) referenced in Section 4.7.7 have been submitted to the NRC for review and approval. If these reports have not been reviewed and approved by the NRC, the staff requested the applicant to submit the reports in support of Ginna's LRA. In response to the staff's RAI, the applicant submitted two copies each of the Proprietary Class 2 Westinghouse Topical Reports WCAP-15873, May 2002, and WCAP-

15837, April 2002. In addition, the nonproprietary versions of these topical reports dated May 2003, were also submitted.

The Westinghouse Topical Report WCAP-15837 provides a plant-specific LBB analysis for RCS piping at the Ginna station through the period of extended operation (60 years). The staff has completed the review of WCAP-15837 and has resolved all concerns regarding the number of years for which the fatigue crack growth analysis was performed and the CASS material degradation due to aging. The NRC staff's review confirmed that the fatigue crack growth analysis in WCAP-15837 was based on thermal transients of 60 years, and is, therefore, appropriate through the expiration of the extended operating period for Ginna. For the concern about the thermal aging of RCS primary loop piping and components made from CASS, the staff has verified that the applicant considered appropriate, fully-aged toughness for CASS. Based on the above evaluation, the staff agrees with the applicant's conclusion that this TLAA is in accordance with 10 CFR 54.21(c)(1)(ii), and the LBB application for the primary loop piping and components is acceptable for the period of extended operation.

The purpose of the flaw tolerance analysis, as documented in Westinghouse Topical Report WCAP-15873, is to support the application of Code Case N-481 for the ISI examination of the RCP casings at the Ginna station. Code Case N-481 allows the performance of visual examination of cast austenitic pump casing in lieu of volumetric examination.

In a response to RAI B2.1.34-1, the applicant stated that the Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program is consistent with the guidelines in AMP XI.M12 of NUREG-1801. AMP XI.M12 states that the existing ASME Section XI requirements, including the alternative requirements of ASME Code Case N-481 for pump casings, are adequate for all pump casings and valve bodies. The program element for detection of aging effects also states that for pump casings and valve bodies and “not susceptible” piping, no additional inspection or evaluations are required to demonstrate that the material has adequate fracture toughness.

The staff notes that the applicant's ASME Code, Section XI, Subsections IWB, IWC, and IWD Inservice Inspection program is required to be updated by the applicant and reviewed by the staff every 10 year interval. The acceptability of using Code Case N-481 as an alternative requirement for the ISI of pump casing will be evaluated by the staff during the review of the applicant's ISI program, which is submitted for NRC approval every 10-year interval. Therefore, it is more appropriate for the staff to review the applicant's fracture mechanics analysis, during the staff's review of the applicant's ISI program. Based on the consideration discussed above, the staff has determined that there is no need for the staff to review the applicant's fracture mechanics analysis, as documented in WCAP-15873, to support the use of Code Case N-481 for ISI of pump casing for the applicant's LRA as would otherwise be mandated by 10 CFR 54.21(c)(1).

The staff also reviewed the UFSAR Supplement to determine whether it provides an adequate description of the program. The staff finds the subject supplement acceptable with the exception that the second paragraph of Section A3.2 pertaining to the discussion of the Westinghouse Topical Report WCAP-15873 should be deleted, because the staff will not review this report for the applicant's LRA as discussed above.

4.7.7.3 Conclusions

On the basis of its review, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), and that for the TLAA on thermal aging of CASS RCS components, the analysis remains valid through the period of extended operation. The staff also concludes that the UFSAR Supplement contains an adequate summary description (pending editorial changes as stated above) of the TLAA on thermal aging of CASS RCS components for the period of extended operation, as required by 10 CFR 54.21(d). Therefore, the staff has reasonable assurance that the safety margins established and maintained during the current operating term for the primary reactor coolant loop piping will be maintained through the period of extended operation for the Ginna Station.

4.8 Evaluation Findings

The staff has reviewed the information in Section 4 of the LRA. On the basis of its review, pending satisfactory resolution of Open Item 4.2.2-1 and Confirmatory Items 4.3-1 and 4.3-2, the staff concludes that the applicant has provided an adequate list of TLAAs, as defined in 10 CFR 54.3. Further, the staff concludes that the applicant has demonstrated that the TLAAs (1) will remain valid for the period of extended operation, as required by 10 CFR 54.21(c)(1)(i), (2) have been projected to the end of the period of extended operation, as required by 10 CFR 54.21(c)(1)(ii), or (3) the aging effects will be adequately managed for the period of extended operation, as required by 10 CFR 54.21(c)(1)(iii). In addition, the staff concludes that there are no plant-specific exemptions in effect that are based on TLAAs, as required by 10 CFR 54.21(c)(2). On this basis, the staff has reasonable assurance that the aging effects associated with the SCs subject to TLAAs will perform their intended functions in accordance with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3).