18 UPDATED FINAL SAFETY ANALYSIS REPORT SUPPLEMENT FOR LICENSE RENEWAL

18.1 INTRODUCTION

On July 30, 2002 the Rochester Gas and Electric Corporation (RG&E) submitted an application to the Nuclear Regulatory Commission (NRC) to renew the operating license for the R.E. Ginna Nuclear Power Plant for an additional 20 years beyond the original expiration date of September 18, 2009 (*Reference 1*). This application was submitted in accordance with the applicable requirements of 10 CFR 54. Between the date of receiving the License Renewal Application (LRA) and January 2004, the NRC reviewed the Ginna LRA, issued requests for additional information (RAIs), reviewed Ginna's response to the RAIs, and finally, issued a Safety Evaluation Report (SER) on March 3, 2004 (*Reference 11*). This SER concluded that the requirements of 10 CFR 54.29(a) have been met.

18.1.1 RENEWED FACILITY OPERATING LICENSE

A renewed operating license was issued to R.E. Ginna Nuclear Power Plant by NRC letter dated May 19, 2004. The renewed operating license is effective from the date of issuance through September 18, 2029.

Technical Specification Amendment 115 was issued on April 1, 2014 which approved the transfer of the license for R. E. Ginna Nuclear Power Plant (Ginna) held by Constellation Energy Nuclear Group, LLC (CENG) to R. E. Ginna Nuclear Power Plant, LLC, (Ginna LLC). Ginna LLC is currently owned by the Exelon Generating Company, LLC (*Reference 2*).

18.1.2 UPDATED FINAL SAFETY ANALYSIS REPORT SUPPLEMENT

Pursuant to the requirements of 10 CFR 54.21(d), this UFSAR Supplement, Chapter 18, is being added to the UFSAR. The purpose of this supplement is to describe certain future activities to be completed prior to the period of extended operation (September 18, 2009 through September 18, 2029.) This supplement also contains a summary description of the programs and activities for managing the effects of aging and the evaluation of time-limited aging analysis (TLAA) for the period of extended operation.

18.1.3 STRUCTURES, SYSTEMS, OR COMPONENTS ADDED SUBSEQUENT TO RENEWED LICENSE

For holders of a renewed license, 10 CFR 54.37(b) requires that newly identified Structures, Systems, or Components (SSCs) be included in the FSAR update required by 10 CFR 50.71(e). The FSAR update must describe how the effects of aging will be managed such that the intended function(s) identified per 10 CFR 54.4 will be effectively maintained during the period of extended operation. Renewal licensees are obligated to comply with 10 CFR 54.37(b) as a condition of license renewal.

Newly identified SSCs are SSCs that were installed in the plant prior to receiving the renewed license (i.e. prior to May 19, 2004) but were not identified during the License Renewal process as meeting the scoping criteria of 10 CFR 54.4. They are listed below:

- During the period of September 2007 to December 2008, a review was conducted of all SSCs added to Ginna's Configuration Management Information System (CMIS) since the original analysis was performed. Using the same methodology described in the R. E. Ginna Nuclear Power Plant LR Application (LRA) that was submitted on July 30, 2002, a total of 81 "newly identified" SSCs were found. *Reference 13* documents this review. Existing Aging Management Programs and aging management strategies have been invoked to adequately detect and manage the aging effects of these "newly identified" SSCs consistent with aging management table line items already identified and included in Ginna's LRA.

- During the period of October 2017 to March 2019, three newly identified SSCs (TE-404A, TE-408A, and TE-410-A-2) were discovered as a result of a periodic 10 CFR 54.37(b) review. These Reactor Coolant System RTDs were installed in the plant prior to license renewal and were added to the scope of the 10 CFR 50.49 Environmental Qualification (EQ) Program after the renewed license was issued. These SSCs are not subject to an Aging Management Review based on the screening criteria of 10 CFR 54.21(a)(1)(ii) because, being classified as EQ, they are subject to the EQ Program's Time Limited Aging Analysis (TLAA) and to replacement.

REFERENCES FOR SECTION 18.1

1. NRC Order from R. V. Guzman, NRC, to H. B. Barron, CENG, "Order Superseding October 9, 2009, Order Approving the Transfer of Renewed Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant (TAC No. ME0445), Letter No. WPLNRC-1002215, dated October 30, 2009.

2. Letter from Nadiyah S. Morgen, NRC, to Mary G. Korsnick and Bryan P. Wright, Constellation Energy Nuclear Group: R.E. Ginna Nuclear Power Plant - Issuance of Amendment to Conform the Renewed Facility Operating License to Reflect the Direct Transfer of Operating Authority (TAC No. MF2588).

18.2 PROGRAMS THAT MANAGE THE EFFECTS OF AGING

This section provides summaries of the programs and activities credited for managing the effects of aging, in alphabetical order.

Each Program is based on meeting the requirements of the ten elements of an aging management program as described by NUREG 1801 Revision 0.

The Ginna Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B and is consistent with the summary in Section A.2 of NUREG-1800, Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants, published in July 2001. The Quality Assurance Program includes the elements of corrective action, confirmation process, and administrative controls, and is applicable to the safety-related and non-safety-related structures, systems, and components that are within the scope of license renewal.

18.2.1 AGING MANAGEMENT PROGRAMS

The description of the Ginna Aging Management Programs are consistent with their status as configured to apply to the period of extended operation.

18.2.1.1 Aboveground Carbon Steel Tanks

The purpose of this program is to assess and monitor the condition of all accessible surfaces of aboveground carbon steel storage tanks. The programs provide for periodic system walk-downs and inspections to monitor the condition of selected above ground carbon steel storage tanks, including an assessment of tank surfaces protected by paints or coatings, although the coatings themselves are not credited to perform a preventive function. For inaccessible surfaces such as concrete foundation interfaces, an inspection of the tank bottom wall thickness is performed.

18.2.1.2 ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection

The program consists of periodic volumetric, surface, and/or visual examinations and leakage tests of all Class 1, 2, and 3 pressure-retaining components to identify evidence of degradation. Surface and visual examinations of integral attachments are also performed. This pro- gram is in accordance with the Inservice Inspection (ISI) Plan. The program also provides for evaluation of inspection results and appropriate corrective actions. The pressurizer manway stainless steel insert will receive a visual and surface examination as part of Ginna's ISI Pro- gram to detect potential stress-corrosion cracking.

18.2.1.3 ASME Section XI, Subsections IWE & IWL Inservice Inspection

The program consists of periodic visual inspection of concrete surfaces for reinforced and prestressed concrete containments, and periodic visual inspection and sample tendon testing of unbonded post-tensioning systems for prestressed concrete containments, for evidence of degradation, assessment of damage and corrective actions. Measured tendon lift-off forces are compared to predicted tendon forces calculated in accordance with Regulatory Guide 1.35. The ASME Section XI, Subsection IWE program consists of periodic visual, surface, and volumetric inspection of pressure retaining components of steel and concrete containments for evidence of degradation. The first interval program also provides for assessment of damage and appropriate corrective actions. This program is in accordance with ASME Section XI, Subsections IWE and IWL, 1992 edition including 1992 addenda. Future 10-year Inspection Interval Requirements will be in accordance with the Containment Inservice Inspection (CISI) Plan.

18.2.1.4 ASME Section XI, Subsection IWF Inservice Inspection

This program consists of periodic visual examinations of component supports for evidence of degradation. The program provides for evaluation of inspection results and appropriate corrective actions. This program is in accordance with the Inservice Inspection (ISI) Plan.

18.2.1.5 Bolting Integrity

The purpose of this program is to verify that a representative sample of all the various types of bolting within the scope of license renewal will be inspected during the period of extended operation. The program consists of periodic inspections of pressure retaining bolting as delineated in NUREG-1339, and other industry recommendations in EPRI NP-5679 (with exceptions noted in NUREG-1339) for safety related bolting, and EPRI TR-104213 for pressure retaining and structural bolting. The program provides for periodic inspection of closure bolting for indication of loss of preload, cracking, and loss of material.

18.2.1.6 Boric Acid Corrosion

The program consists of: (1) visual inspection of external surfaces that are potentially exposed to borated water leakage, (2) timely discovery of leak path and removal of the boric acid residues, (3) assessment of any damage, and (4) follow-up inspections to assure effectiveness of corrective actions. The program scope includes RCS components in accordance with Generic Letter 88-05 as well as non-RCS mechanical, electrical and structural components susceptible to boric acid corrosion which are potentially exposed to borated water leaks.

18.2.1.7 Buried Piping and Tanks Inspection

The purpose of this program is to ensure that buried piping and tanks within the scope of license renewal are periodically monitored and assessed and maintained in an acceptable condition. The Program includes provisions for: (1) preventive measures to mitigate corrosion, and (2) periodic inspections to manage the effects of corrosion on the pressure retaining capacity of buried carbon steel piping and tanks during inspections of opportunity. Preventive measures are in accordance with standard industry practice for maintaining external coatings and wrappings. Buried piping and tanks are inspected visually for any evidence of degradation when they are uncovered for any reason.

18.2.1.8 Closed-Cycle (Component) Cooling Water System

The program includes: 1) preventive measures to minimize corrosion by maintaining corrosion inhibitor concentrations within specified limits, 2) surveillance tests and inspections, and 3) nondestructive evaluations of internal surfaces of system components. Evaluations to verify the effectiveness of water chemistry controls are based on the guidelines of EPRI TR-107396 for closed-cycle cooling water systems.

18.2.1.9 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

The program requires that cables and connections in accessible areas exposed to adverse localized environments caused by heat, radiation, or moisture are inspected on a periodic basis. Visual inspections for cable and connector jacket surface anomalies such as embrittlement, discoloration, cracking, and surface contamination are performed at least once every ten years.

18.2.1.10 Electrical Cables Not Subject to 10CFR50.49 Environmental Qualification Requirements Used In Instrument Circuits

Electrical cables used in circuits with sensitive, low-level signals, such as radiation monitoring and nuclear instrumentation, are periodically (at least once every 10 years) tested for insulation resistance to identify changes that could have an adverse impact on circuit operation.

Combined with the program to perform visual inspections of accessible portions of these circuits, as described in *Section 18.2.1.9*, sufficient indication of the need for corrective action is provided.

18.2.1.11 Fire Protection

The program includes inspection of fire barriers and functional testing of fire pumps. The Fire Protection Program requires periodic visual inspections of fire barrier penetration seals, fire barrier walls, ceilings, and floors, and periodic visual inspections and functional tests of fire rated doors and dampers to ensure that their operability is maintained. The Fire Protection Program requires that the diesel driven fire pump be periodically tested to ensure that the fuel supply line can perform the intended function. The program also includes periodic inspection and testing of the halon fire suppression system.

18.2.1.12 Fire Water System

The program consists of inspections and functional tests of fire suppression components such as sprinklers, hydrants, valves and piping. Periodic full flow flush tests and system performance tests are conducted to prevent corrosion due to silting and biofouling of components. In addition, the system is normally maintained at required operating pressure and is monitored such that loss of system pressure is immediately detected and corrective actions initiated. Internal portions of the fire water system are visually inspected when disassembled for maintenance. Volumetric NDE inspections using appropriate techniques are performed to detect wall loss and fouling. Replacement or representative sample testing of sprinklers with a service life of 50 years is specified.

18.2.1.13 Flow-Accelerated Corrosion

The program consists of: (1) conducting appropriate analyses to determine critical locations susceptible to FAC, (2) conducting baseline inspections to determine the extent of thinning at these locations, and (3) performing follow-up inspections to confirm predicted degradation rates. Corrective actions such as repair or replacement are evaluated based on inspection results and predicted rates of wall loss. The program implements the EPRI guidelines in the Nuclear Safety Analysis Center (NSAC) 202L and utilizes the CHECWORKS predictive code.

18.2.1.14 Fuel Oil Chemistry

The program consists of a combination of surveillance and maintenance activities. Monitoring and control of fuel oil contamination in accordance with the guidelines in ASTM Standards D975, D1796, D2709, D4057 and D2276 (or its successor) maintains fuel oil quality. Exposure to fuel oil contaminants such as water and microbiological organisms is minimized by periodic cleaning/draining of storage tanks and verifying the quality of new fuel oil before introduction into the tanks.

18.2.1.15 Inaccessible Medium-Voltage Cables Not Subject to 10CFR50.49 Environmental Qualification Requirements

In-scope, medium-voltage cables exposed to significant moisture (exposures of more than a few days at a time) and significant voltage (subject to system voltage more than 25% of the time) are tested at least once every 10 years to provide an indication of the condition of the conductor insulation. The specific type of test performed will be determined prior to the initial test, and is to be a proven test for detecting deterioration of the insulation system due to wetting, such as insulation resistance, polarization index, dissipation factor, and time domain reflectometry.

18.2.1.16 Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems

The program evaluates the effectiveness of testing and monitoring activities as well as the effects of past and future usage on the structural reliability of the cranes, hoists and lifting devices that were evaluated in Ginna Station's response to NUREG-0612. The number and magnitude of lifts made by the hoist or crane are also reviewed. Rails and girders are visually inspected on a periodic basis for evidence of degradation. Functional tests are also performed to assure structural integrity.

18.2.1.17 One-Time Inspection

The intent of this program is to verify the effectiveness of existing aging management programs by confirming the absence of an aging effect or verifying that the aging effect is developing so slowly that the intended function is expected to be maintained through the period of extended operation. The program methodology includes selection of appropriate inspection techniques and sample size to ensure that the specified age-related degradation will be dis- covered in a timely manner. The program provides for evaluation of inspection results and appropriate corrective actions. Locations judged to be potentially susceptible to thermal fatigue will be included in the sample population of small bore piping to be examined by appropriate volumetric technique.

18.2.1.18 Open-Cycle Cooling (Service) Water System

The program is based on implementation of the recommendations of Generic Letter 89-13 to ensure that the effects of aging on open cycle cooling water system components will be man- aged for the extended period of operation. The program includes surveillance and control techniques to manage aging effects caused by biofouling, corrosion, erosion, protective coating failures, and silting in the open cycle cooling water system or structures and components serviced by the open cycle cooling water system.

18.2.1.19 Periodic Surveillance and Preventive Maintenance

The Periodic Surveillance and Preventive Maintenance (PSPM) Program specifies tests, inspections, and surveillances and assures they are completed in accordance with appropriate frequencies. Together with the evaluation of results, the program assists site personnel in maintaining proper equipment and structural conditions to ensure necessary reliability. This program also provides for replacement or refurbishment of certain components on a specified frequency, based on operational experience.

A subset of the PSPM Program, the Periodic Surveillance and Preventive Maintenance Aging Management Program (PSPMAMP), provides for the generation of repetitive tasks and schedules for activities performed by other License Renewal Aging Management programs. It also provides for inspections, testing, and examinations of select long-lived passive equipment which is not included in the scope of another License Renewal Aging Management Program. These activities are to be performed on specified frequencies to identify evidence of age-related degradation such as corrosion, cracking, wear, or fouling. This PSPMAMP also provides for evaluation of inspection/test/examination results, and the initiation of appropriate corrective actions as needed to provide reasonable assurance of no loss of intended functions.

18.2.1.20 Nickel-Alloy Nozzles and Penetrations Inspection

The Nickel-Alloy Nozzles and Penetration Inspection Program includes: 1) susceptibility assessment of head components (including alloy 690TT subcomponents) to primary water stress corrosion cracking (PWSCC), 2) monitoring and control of reactor coolant water chemistry to mitigate PWSCC, and 3) inservice inspection (ISI) of reactor vessel head penetrations and bottom-mounted instrument tube penetrations, in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWB, Table IWB 2500-1 (1995 edition through the 1996 addenda). The program provides for evaluation of inspection results and appropriate corrective actions. Ginna LLC will, in concert with industry initiatives, develop a nickel-alloy nozzle and penetration inspection program.

18.2.1.21 Reactor Vessel Internals

The Reactor Vessel Internals Program credits inspections prescribed by the ASME Code Section XI In-Service Inspection Program (see Section 18.2.1.2), the Water Chemistry Control Program (see Section 18.2.1.28), and includes additional inspections based on MRP-227 Materials Reliability Program: Internals Inspection Program to the NRC in February 2009 (Reference 14). In order to align with Category A facilities, as specified in NRC Regulatory Issue Summary 2011-07, Ginna withdrew this submittal in

September 2011 (Reference 15). A revised Reactor Vessel Inspection Program was submitted in September 2012 (Reference 16) to be in accordance with MRP-227-A.

18.2.1.22 Reactor Vessel Surveillance

The program provides for periodic testing of metallurgical surveillance samples to monitor the progress of neutron embrittlement of reactor pressure vessel materials as a function of neutron fluence in accordance with Regulatory Guide 1.99, Rev. 2. The Reactor Vessel Surveillance program applies to P/T Limit Curves, Upper Shelf Energy, Pressurized Thermal Shock, and LTOP setpoints.

18.2.1.23 Spent Fuel Pool Neutron Absorber Monitoring

The program monitors long-term performance of borated stainless steel (BSS) panels, credited as a neutron absorber in portions of the spent fuel pool (soluble boron is credited in the rest of the pool). Borated stainless steel surveillance coupons are periodically removed and examined to evaluate coupon thickness and weight loss. The program provides for evaluation of inspection results and appropriate corrective actions.

18.2.1.24 Steam Generator Integrity

The program incorporates the guidance of NEI 97-06 and EPRI TR-107569 for managing aging of the secondary side of the steam generators, as well as the tubes. The effects of aging are managed by a balance of prevention, inspection, assessment, repair, and leakage monitoring measures. Plant Technical Specifications assure timely assessment of tube integrity and compliance with primary to secondary leakage limits.

18.2.1.25 Structures Monitoring

The Structures Monitoring Program consists of periodic inspection and monitoring of the condition of structures and structural elements as well as selected non-safety component sup- ports to ensure that aging degradation will be detected and corrected prior to loss of **Systems** intended function. The program is implemented in accordance with 10 CFR 50.65, NUMARC 93-01, Rev. 2, and Regulatory Guide 1.160, Rev. 2. The program provides for evaluation of inspection results and appropriate corrective actions.

18.2.1.26 Monitoring

The program identifies the evidence of age-related degradation on normally accessible exterior surfaces of piping, components and equipment in systems which are within the scope of license renewal. As part of the implementation of 10 CFR 50.65 (Maintenance Rule), specific guidelines for assessing the material condition of systems, structures, and components during system engineer walkdowns were developed. The effects of aging are managed through visual inspection and monitoring of external surfaces for leakage and evidence of material degradation, such as corrosion, cracking, degradation of coatings, sealants and caulking, deformation, debris and corrosion product buildup. The program provides for evaluation of inspection results and appropriate corrective actions.

18.2.1.27 Thimble Tubes Inspection

The program manages the integrity of the incore neutron monitoring thimble tubes and the guide tubes which serve as a portion of the reactor coolant pressure boundary. The program provides for periodic eddy current inspections to detect cracking due to stress corrosion cracking (SCC) of the thimble tubes and guide tubes, thimble tube wall thinning due to wear caused by flow-induced vibration, as well as preventive maintenance activities such as flushing, cleaning, and replacement. Cracking due to SCC is detected in portions of the thimble tubes and guide tubes exposed to the reactor coolant environment at temperatures >140°F.

Thimble tube wear is detected at locations where the thimble tubes pass through the lower core plate. In addition, the fillet welds joining the guide tubes to the bottom-mounted instrument penetrations are visually inspected periodically by a VT-1 technique. The program pro- vides for evaluation of inspection results and appropriate corrective actions.

18.2.1.28 Water Chemistry Control

The program mitigates the effects of aging by controlling the internal environment of components in the primary, borated, and secondary water systems. Chemical species known to accelerate corrosion (e.g., chloride, fluoride, and sulfate) are controlled within specified limits. The program implements the guidelines in EPRI TR-105714 for primary water chemistry, and TR-102134 for secondary water chemistry. The program provides for assessment and trending of water chemistry and implements corrective action strategies.

18.3 EVALUATION OF TIME-LIMITED AGING ANALYSES

As part of a License Renewal Application, 10 CFR 54.21(c) requires that an evaluation of time-limited aging analyses (TLAAs) for the period of extended operation be provided. The following TLAAs have been identified and evaluated to meet this requirement.

18.3.1 REACTOR VESSEL NEUTRON EMBRITTLEMENT

The following analyses affected by neutron irradiation caused embrittlement that have been identified as TLAAs:

- Upper shelf energy *Section 18.3.1.1*
- Pressurized thermal shock *Section 18.3.1.2*
- RCS pressure-temperature operating limits *Section 18.3.1.3*

18.3.1.1 Upper Shelf Energy

The Charpy upper shelf energy (USE) is associated with the determination of acceptable Reactor Vessel toughness during operation. 10 CFR Part 50 Appendix G requires that the reactor vessel beltline materials must have a USE of no less than 50 ft-lb) throughout the life of the reactor vessel, unless otherwise approved by the NRC. If the USE of a reactor vessel beltline material is predicted to not meet Appendix G requirements, then licensees must submit an analysis that demonstrates an equivalent margin of safety at least three years prior to the time the material is predicted to not meet those requirements.

In the event that the 50 ft-lb requirement cannot be satisfied as stated in 10 CFR 50 Appendix G, or by alternative procedures acceptable to the NRC, the reactor vessel may continue to operate provided requirement IV.A.1. of Appendix G is satisfied. This requirement states that an analysis must conservatively demonstrate, making appropriate allowances for uncertainties, the existence of equivalent margins of safety for continued operation. Procedures for the analysis are provided in NUREG-0744. Acceptance criteria are included in ASME Section XI, Appendix K.

The upper shelf energy of the limiting circumferential beltline weld in the Ginna reactor vessel is expected to decrease below 50 ft-lbs during the period of extended operation. In order to demonstrate equivalent margins of safety for continued operation, a low upper-shelf tough- ness fracture mechanics analysis has been performed (BAW-2425, Rev.1) to evaluate the limiting circumferential beltline weld (SA-847) for ASME Levels A, B, C, and D Service Loadings. The analysis demonstrates that the limiting beltline weld satisfies the Appendix K requirements for ductile flaw extensions and tensile stability using projected low upper-shelf Charpy impact energy levels for the weld material based on a bounding reactor vessel fluence corresponding to the end of the extended period of plant operation (2029).

18.3.1.2 Pressurized Thermal Shock

The PTS rule, 10 CFR 50.61 provides screening criteria that are a measure of a limiting level of reactor vessel material embrittlement beyond which operation may not continue without further plant-specific evaluation. The pressurized thermal shock screening criteria are given in terms of reference temperature RT_{PTS}. The screening criteria are 270°F for plates and axial welds, and 300°F for circumferential welds.

The results of the revised PTS analysis for the limiting material have been reviewed for compliance with 10CFR50.61. The methodology used in PTS analysis is based on the projected neutron fluence at the end of the period of extended operation. For the intermediate and lower shell forgings, the analysis was based on Regulatory Guide 1.99, Revision 2, Position 1.1, which does not rely on plant-specific surveillance data to calculate ΔRT_{PTS}. For the circumferential weld, the analysis was based on Regulatory Guide 1.99, Revision 2, Position 2.1, which does relay on plant-specific surveillance date to calculate ΔRT_{PTS}. The analysis associated with PTS has been projected to the end of the period of extended operation, in accordance with 10CFR54.21(c)(1)(ii) and found to be acceptable.

18.3.1.3 Pressure-Temperature Limits

10 CFR Part 50 Appendix G requires that the reactor pressure vessel (RPV) be maintained within established pressure-temperature (P-T) limits including during heatup and cooldown. These limits specify the maximum allowable pressure as a function of reactor coolant temperature. As the reactor pressure vessel becomes embrittled and its fracture toughness is reduced, the allowable pressure (given the required minimum temperature) is reduced.

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced delta RT_{NDT} and adding a margin.

The reactor vessel neutron fluence values corresponding to the end of the period of extended operation and the reactor vessel beltline material properties have been calculated consistent with Regulatory Guide 1.190. The revised fluence values have been used to determine the limiting value of RT_{NDT} using the methods of Regulatory Guide 1.99. The limiting value of RT_{NDT} was used to calculate reactor coolant system (RCS) pressure-temperature (P-T) operating limits that are valid through the end of the period of extended operation. Consistent with NUREG-1800 section 4.2.2.1.3.3, it is not necessary to implement P-T limits to carry the reactor vessel through 60 years at the time of application. The updated limits will be contained in a pressure-temperature limit report (PTLR) or in the Technical Specification (TS) prior to the period of extended operation. The analysis associated with P-T operating limits has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

18.3.2 THERMAL AGING EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL (CASS)

The potential exists for a loss of fracture toughness due to thermal aging of cast austenitic stainless steel (CASS) components. An evaluation of the susceptibility of CASS components at Ginna Station was made, based on the casting method, molybdenum content, and percent ferrite. It was determined that the CASS RCS elbows were susceptible to a loss of fracture toughness due to thermal aging. A plant-specific flaw tolerance evaluation was conducted, and documented 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. The evaluation concluded that adequate fracture toughness exists for the RCS loop, including the cast elbows, for the period of extended operation (60 years).

A separate evaluation was made for the reactor coolant pump casings. In WCAP-15873, "A Demonstration of the Applicability of ASME Code Case N-481 to the Primary Loop Pump Casings of R.E. Ginna Nuclear Power Plant for the License Renewal Program," May 2002, it was concluded that the primary loop pump casings are qualified to item (d) of ASME Code Case N-481 for the period of extended operation (60 years).

The evaluation associated with thermal aging embrittlement has been found to be acceptable to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

18.3.3 METAL FATIGUE

The following issues are considered separately under the TLAA for Metal Fatigue:

- ASME Boiler and Pressure Vessel Code, Section III, Class 1; *Section 18.3.3.1*
- Reactor Vessel Underclad Cracking *Section 18.3.3.2*
- ANSI B31.1 Piping *Section 18.3.3.3*
- Accumulator Check Valves *Section 18.3.3.4*
- Environmentally Assisted Fatigue *Section 18.3.3.5*
- Reactor Vessel Nozzle-to-Vessel Weld Defect; *Section 18.3.3.6*
- Pressurizer Fracture Mechanics Analysis *Section 18.3.3.7*

Fatigue is the gradual deterioration of a material that is subjected to repeated cyclic loads. Components have been designed or evaluated for fatigue according to the requirements of applicable codes.

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

The reactor vessel, pressurizer, steam generators, and reactor coolant pumps have been designed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class 1. The reactor vessel internals were designed according to Westinghouse criteria which were later incorporated into ASME Boiler and Pressure Vessel Code. Design codes for the above components are identified in UFSAR *Table 5.2-3*. The ASME Boiler and Pressure Vessel Code, Section III, Class 1 requires a design analysis to address fatigue and establish limits such that initiation of fatigue cracks is precluded.

Fatigue usage factors for critical locations in the NSSS components were determined using design cycles that were specified in the plant design process. These design cycles were intended to be conservative and bounding for all foreseeable plant operational conditions. The design cycles were subsequently utilized in the design stress reports for various NSSS components satisfying ASME fatigue usage design requirements, and became part of the plant Technical Specifications.

Experience has shown that actual plant operation is often very conservatively represented by these design cycles. The use of actual operating history data allows the quantification of these conservatisms in the existing fatigue analyses. To demonstrate that the Class 1 component fatigue analyses remain valid for the period of extended operation, the design cycles applicable to the Class 1 components were assembled. The actual frequency of occurrence for the design basis cycles was determined and compared to the design cycle set. The severity of the actual plant transients was compared to the severity of the design cycles. This comparison was performed in order to demonstrate that on an event-by-event basis the design cycle pro- files envelope actual plant operation. In addition, a review of the applicable administrative and operating procedures was performed to verify the effectiveness of the current design cycle counting program.

This review concluded that the existing design cycles and cycle frequencies are conservative and bounding for the period of extended operation. The analyses associated with verifying the structural integrity of the reactor vessel, reactor vessel internals, pressurizer, steam genera- tors, and reactor coolant pumps have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i). A confirmatory Fatigue Monitoring Program (described in Appendix B of the LRA) has also been implemented at Ginna Station to provide additional assurance that the fatigue analyses remain valid during the period of extended operation.

18.3.3.2 Reactor Vessel Underclad Cracking

Underclad cracking has been reported in the low alloy base metal heat-affected zone (HAZ) beneath the austenitic stainless steel weld overlay that is deposited to protect the ferritic material from corrosion.

A re-evaluation (WCAP 15338) of the generic Westinghouse fracture mechanics evaluation (WCAP 7733) concerning the underclad cracking issue has been performed for 60 years of plant operation. It was concluded that "underclad cracks are of no concern to the structural integrity of the vessel for continued plant operation, even through 60 years of operation." WCAP 15338 is bounding for all Westinghouse plants.

The analysis associated with reactor vessel underclad crack growth has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

18.3.3.3 ANSI B31.1 Piping

Design requirements in ANSI B31.1 assume a stress range reduction factor to provide conservatism in the piping design to account for fatigue due to thermal cyclic operation. This reduction factor is 1.0 provided the number of anticipated cycles is limited to 7000 equivalent full temperature cycles. This represents a condition where a piping system would have to be cycled approximately once every 3 days over the extended plant life of 60 years. Considering this limit, a review of the ANSI B31.1 piping within the scope of license renewal was per- formed in order to identify those systems that operate at elevated temperature and to establish their cyclic operating practices. Under current plant operating practices, piping systems within the scope of license renewal are only occasionally subject to cyclic operation. Typically these systems are subject to continuous steady-state operation and vary operating temperatures only during plant heatup and cooldown, during plant transients, or during periodic testing.

The results of the evaluation for ANSI B31.1 piping systems demonstrated that the number of assumed thermal cycles will not be exceeded in 60 years of plant operation except for the Nuclear Sampling System. For all systems except the Nuclear Sampling System, it has been determined that operation can be projected to the end of the period of license renewal, in accordance with 19 CFR 54.21 (c)(1)(ii). For the sampling system, a detailed engineering analysis was performed which showed that the maximum thermal stress developed during heatups and cooldowns was less than the code allowable stress range for 100,000 or more operational cycles. The existing nuclear steam supply system (NSSS) sampling is thus acceptable for the period of extended operation, in accordance with 10CFR54.21(c)(1)(ii).

18.3.3.4 Accumulator Check Valves

Fatigue of components is recognized as time dependent and therefore the analysis was reviewed for fatigue related to these valves. Fatigue failure is based upon the criteria of the cumulative usage factor (CUF). An analysis was performed on the accumulator check valves at Ginna Station. The analysis concludes that the maximum CUF is 0.74 based on specified load conditions.

Plant transients were reviewed to confirm transient limits and total transient counts to date. The load condition occurrences used in the above analysis bound the transient limits monitored by plant procedures. In accordance with 10 CFR 54.21(c)(1)(i), the existing CUF calculations remain valid because the number of assumed transients would not be exceeded during the period of extended operation.

18.3.3.5 Environmentally Assisted Fatigue

Generic Safety Issue (GSI)-190, Fatigue Evaluation of Metal Components for 60 Year Plant Life, identifies a concern of the NRC staff about the potential effects of reactor water environments on reactor coolant system component fatigue life during the period of extended operation. GSI-190, which was closed in December 1999, has concluded that environmental effects have a negligible impact on core damage frequency, and as such, no generic regulatory action is required. However, as part of the closure of GSI-190, the NRC has concluded that licensees who apply for license renewal should address the effects of coolant environment on component fatigue life as part of their aging management programs.

Fatigue-sensitive component locations were evaluated in NUREG/CR-6260 for the older vintage Westinghouse plant. These locations were:

1. Reactor vessel shell and lower head (lower shell at the core support pads)
2. Reactor vessel inlet and outlet nozzles
3. Pressurizer surge line (including hot leg and pressurizer nozzles)

4. Reactor coolant piping charging system nozzle
5. Reactor coolant piping safety injection nozzle
6. Residual Heat Removal system Class 1 piping

Environmental fatigue calculations have been performed for Ginna for those component locations included in NUREG/CR-6260 using the appropriate environmental life correction factor formulae contained in NUREG/CR-6583 for carbon/low alloy steel material, or NUREG/CR- 5704 for stainless steel material, as appropriate.

Based on these results, all component locations were determined to be acceptable for the period of extended operation.

18.3.3.6 Reactor Vessel Nozzle-to-Weld Defect

In 1979, during the first-interval ISI of the reactor vessel, a flaw indication was discovered by UT examination in a primary inlet nozzel-to-vessel weld (Nozzle N2B). The size of this indication was determined to exceed the size permitted by the acceptance criteria for the examination method in ASME Section XI, 1974 Edition. The flaw was again sized as unacceptable in 1989. Fracture mechanics analyses performed in 1979 and 19789 concluded that the only stresses of significance acting across the flaw are those due to vessel pressurization and weld residual stress, and that the flaw satisfied the ASME section XI Code criteria for acceptance by evaluation. In addition, the analyses concluded that:

- Irradiation effects from the core are negligible at the flaw location.
- The applied stress intensity (K) for the embedded flaw with a through-wall dimension of 0.48 inches and a length of 4.94 inches is calculated as 7351 psi/in due to pressure loading and weld residual stress.
- The above K value provides a margin of 27.2 against an upper shelf reference K (K_{IR}) of 200,000 psi/in, compared to the required Section XI margin of 3.16.
- Predicted fatigue crack growth, even for 1200 full cycles of vessel pressurization, is insignificant.

It has been determined that the design basis transient set for Ginna Station remains bounding for the period of extended operation. The number of design cycles of heatups and cooldowns, and therefore vessel pressurizations, is 200. Since irradiation effects at the flaw location are negligible and fatigue crack growth is insignificant even for 1200 cycles of pressurization, the flaw will remain stable and of no structural significance for the period of extended operation.

Since the number of pressurizations during the period of extended operation are less than that analyzed in the fatigue crack growth analysis, the amount of flaw growth for the period of extended operation is bounded by the fracture mechanics analysis performed in 1989.

18.3.3.7 Pressurizer Fracture Mechanics Analysis

During the preservice UT examination of the pressurizer, a "defect-like" indication was reported in the lower shell-to-head circumferential weld (C-3). The indication was reported as a linear reflector approximately 11 1/2" long X 1/2" width embedded partially in the circumferential weldment and the base metal of the pressurizer shell. Based on a fracture mechanics analysis performed by Westinghouse, it was concluded that the "defect" would not cause failure of the pressurizer during the design life (40 years) of the component. The analysis was based on several conservative assumptions, including transposing the defect from the embedded position to the internal surface of the pressurizer wall.

This indication was subsequently examined by UT in 1971, 1972, 1974, 1980, 1991, and 2002 during ASME Section XI inservice inspections. The examinations in 1974, 1980, and 1991 characterized the indication as consisting of several intermittent, low-amplitude indications located in the center 1/3 of the weld thickness. These indications were evaluated and found to meet the acceptance criteria by examination of ASME Code, Section XI. The most recent inspection was performed using both automated and manual UT examinations. Intermittent, low-amplitude indications were recorded in the center 1/3 of the weld thickness. These indications were also evaluated and found to meet the acceptance criteria by examination in ASME Code, Section XI, 1995 Edition (1996 Addenda).

Thus, even though this issue was evaluated as a TLAA, 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. Thus, the fracture mechanics analysis is no longer applicable or relevant, but included for completeness.

18.3.4 ENVIRONMENTAL QUALIFICATION OF ELECTRIC EQUIPMENT

10 CFR 50.49, Environmental qualification of electric equipment important to safety for nuclear power plants, requires that selected electrical equipment that is relied upon to remain functional during and following a design basis event be environmentally qualified to perform its intended function. Equipment within the scope of the EQ rule has been identified in accordance with 10 CFR 50.49 paragraph (d) and are listed in the Ginna Station EQ Master List. Only the equipment qualification packages which indicate a qualified life of greater than 40 years were reviewed as a Time-Limited Aging Analysis (TLAA). Equipment qualification packages that indicate a qualified life of less than 40 years are not a TLAA as defined in 10 CFR 54.3 and therefore need not be discussed in the context of license renewal. To establish reasonable assurance that the safety related electrical equipment will perform its safety function when exposed to postulated harsh environmental conditions, licensees are required to develop an environmental qualification program. The program must demonstrate that the safety related electrical equipment required to perform the various safety related functions, identified in 10 CFR 50.49, are qualified to perform as intended. The program must maintain the environmental qualification of the equipment for its installed life. Periodic replacement and/or refurbishment of equipment are performed in order to maintain the qualified life of the device. The qualified life of an equipment type is that period of time the equipment is installed, under normal and abnormal plant operating conditions (thermal and radiation expo- sure), and still be expected to perform its intended function following a postulated design basis event. The qualified life of an equipment type is determined using the ambient environ- mental conditions to which it is exposed for the predicted installation period as well as any internal heat rise and cyclic stresses.

EQ reanalyses have been performed to verify extension of EQ qualification to 60 years for most equipment, and shown to be acceptable per 10 CFR 54.21(c)(1)(i) or (c)(1)(ii). Calculations, for all EQ components that will not be replaced at the end of qualified life, have been completed and shown to be acceptable through the period of extended operation.

18.3.5 CONCRETE CONTAINMENT TENDON PRESTRESS

The Ginna Station containment structure is post-tensioned by 160 vertical tendons. The design for the containment provides for prestressing the concrete in the cylinder walls in the longitudinal direction with a sufficient compressive force to ensure that upon application of the design load combinations there will be no tensile stresses in the concrete due to membrane forces.

The prestressing forces of containment tendons decrease over time due to creep and shrinkage of concrete, and stress relaxation of the prestressing steel wires.

One hundred and thirty seven tendons were retensioned in 1979. The remaining twenty three tendons, which had been retensioned in 1969, were not included in the 1979 retensioning activity. Review of tendon surveillance lift-off data indicates that prestressing forces will remain at acceptable levels through the period of extended operation for all tendons except the group of twenty three which were retensioned in 1969. These tendons will be retensioned prior to the end of the current license period. Technical Specifications require that the results of the surveillance be compared with predicted values to verify that prestressing forces are maintained above the minimum design prestress levels.

Based on this review, the program will adequately manage loss of prestress in containment tendons during the extended period of operation in accordance with 10 CFR 54.21(c)(1)(iii).

18.3.5.1 Containment Tendon Fatigue

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.

The tendon fatigue test results indicate that the fatigue limit of the tendon wires exceeds, by many orders of magnitude, the total number of cycles that could accumulate through multiple seismic events over 60 years. The seismic fatigue evaluation remains valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

18.3.5.2 Containment Tendon Bellows Fatigue

The allowable radial and vertical displacements of the containment stainless steel tendon bellows are limited to two cycles per year for the 40-year life of the plant. This limits the total number of allowable bellows 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 seasonal variations and reactor shutdown during refueling outages.

The fatigue usage factor of the tendon bellows has been calculated to be .004 over a 60-year period. Therefore, the structural integrity of the tendon bellows will be maintained through the period of extended operation. Thus, the original fatigue analysis remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

18.3.6 CONTAINMENT LINER PLATE AND PENETRATION FATIGUE

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 AISC Code for structural steel. The containment liner and penetrations, including the equipment and personnel hatch 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 that Subsection B rules be applicable.

ASME Section III, N-415.1 states that a fatigue analysis is not required, and it may be assumed that the peak stress intensity limit has been satisfied for a vessel or component by compliance with the applicable requirements for materials, design, fabrication, testing, and inspection, provided the service loading of the vessel or component meets all of six (6) conditions. An analysis was performed which verifies that each of the six conditions are satisfied for the period of extended operation. This analysis demonstrates that the liner and penetrations comply with the ASME Section III - 1965 Code Rules for fatigue through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

18.3.6.1 Containment Liner Anchorage Fatigue

A fatigue analysis of the fillet weld attaching the channel anchors to the liner was performed as part of the original containment 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 stress cycles.

A total of 100,000 stress cycles corresponds to more than 4 full stress cycles per day for 60 years. Fluctuations of temperature and pressure in containment on a daily basis are not significant enough in magnitude to cause four cycles of design basis stress at the liner anchorage weld each day. Therefore, the original fatigue analysis remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

18.3.7 CONTAINMENT LINER STRESS

The containment liner is carbon steel plate conforming to ASTM A442-60T Grade 60 with a minimum yield of 32,000 psi, and a buckling stress of 16,600 psi at operating conditions. The liner plate thickness is 1/4 in. for the base and 3/8 in. for the cylinder and dome. The liner stresses (meridional directions) were calculated to be 4500 psi compression based upon a pre-stress force of 0.70 fs. The concrete strain due to creep and shrinkage was established as being 320 x 10^{-6} in/in. This increases the liner stress to 14,100 psi at the end of 40 years.

The creep and shrinkage strain occurring over a 60-year plant life was calculated, and the resulting compressive liner stress due to both time-dependent and non-time dependent loads is determined to be 14,870 psi. This liner stress is less than the liner buckling stress of 16,600 psi and therefore the analysis has been projected to the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

18.3.8 OTHER TIME-LIMITED AGING ANALYSES

18.3.8.1 Crane Load Cycle Limit

The estimated number of load cycles for each crane was compared to the number of design load cycles. The comparison demonstrated that all estimated load cycle combinations were well below the upper design loading cycle limit. In addition, the average percent of the rated load lifted was well below 50% of the limit as set forth in the design criteria. Since the number of operating load cycles for the cranes will be fewer than the design cycles and the aver- age percent of rated load lifted is less than 50% for the design load cycles, the crane load cycle limits will remain valid for the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

18.3.8.2 Reactor Coolant Pump (RCP) Flywheel

During normal operation, the reactor coolant pump (RCP) flywheel possesses sufficient kinetic energy to produce high energy missiles in the event of failure. The aging effect of concern is fatigue flaw growth in the flywheel bore keyway. In accordance with the ISI program, Ginna Station performs an examination once every 20 years (see UFSAR Section 5.4.1.2.5). The method of examination includes either an ultrasonic examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius, or an ultrasonic and surface examination of exposed surfaces defined by the volume of the disassembled flywheel.

18.3.9 EXEMPTIONS

The requirements of 10 CFR 54.21(c) stipulate that the application for a renewed license should include a list of plant-specific exemptions granted pursuant to 10 CFR 50.12 and that are based on time-limited aging analyses, as defined in 10 CFR 54.3. Each active 10 CFR 50.12 exemption has been reviewed to determine whether the exemption is based on a time-limited aging analysis. No plant-specific exemptions granted pursuant to 10 CFR 50.12 and based on a time-limited aging analysis as defined in 10 CFR 54.3 have been identified.

18.4 TLAA SUPPORTING ACTIVITIES

18.4.1 CONCRETE CONTAINMENT TENDON PRESTRESS

The prestressing forces generated by the containment wall tendons diminish over time due to stress relaxation of the steel tendon wires and shrinkage and creep of the surrounding concrete. The aging management program developed to monitor the prestressing tendon forces ensures that, by periodic surveillance lift-off tests, the trend lines of the measured prestressing forces meet the requirements of 10 CFR 50.55a(b)(2)(viii)(B). If the trend lines cross the predicted lower limits, corrective action such as retensioning will be taken.

18.4.2 ENVIRONMENTAL QUALIFICATION PROGRAM

Equipment environmental qualification has been reviewed and one of the following options was used for those components re-analyzed:

- The original environmental qualification qualified life has been shown to remain valid for the period of extended operation.
- The environmental qualification has been projected to the end of the period of extended operation. Reanalysis addresses attributes of analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions.

Calculations for all EQ components that will not be replaced at the end of qualified life have been completed, and shown to be acceptable through the period of extended operation.

18.4.3 FATIGUE MONITORING PROGRAM

The program is a confirmatory program that monitors loading cycles due to thermal and pressure transients at selected locations on critical reactor coolant system components. The pro- gram provides means for evaluating transients using either a stress based or cycle based methodology. The program provides an analytical basis for confirming that the number of cycles established by the analysis of record will not be exceeded before the end of the period of extended operation.

The effects of the reactor coolant environment on component fatigue life are considered by evaluation of a sample of critical components that include, as a minimum, those components selected in NUREG/CR-6260 using the appropriate environmental fatigue correction factors. Formulas for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels.

The EPRI FatiguePro software program was customized to monitor fatigue-critical locations in the surge line and pressurizer lower head in the Ginna plant. An analysis was performed based on available template sets of real plant data to determine the incremental fatigue usage factor for known plant transients, including the effects of "insurge/outsurge" and environmentally-assisted fatigue (EAF). Cumulative usage factor for the operating life of the plant were computed based on the results of real plant data, and expected future usage was computed using projections of expected plant cycles.

The technical approach is summarized as follows:

- The flow rate in the surge line was computed based on a mass balance approach, using the incoming spray demand and the rate of change of the pressurizer water level, taking into account temperature effects.
- A 2-dimensional model was created to take into account (a) the advance and time delay of colder water from the hot leg into the surge line and lower head of the pressurizer, and (b) the heat transfer between the fluid and metal.
- Finite element models (including thermal sleeves in the pressurizer surge nozzle and hot leg RCS surge nozzle) were created to compute stress responses to step changes in temperature at various zones in the pressurizer. Stresses could then be computed based on the calculate fluid temperatures at the various zones in the pressurizer and surge line.
- The stress history was used to compute fatigue usage in FatiguePro.

Real plant data from various heatup/cooldown cycles since 1996 were analyzed to compute incremental fatigue usage for a heatup/cooldown cycle. The location with the highest fatigue usage in the pressurizer bottom head was determined to be the heater tube-to-lower head (penetration) weld. For the heater penetration location, the primary stress transient is not due to insurge and outsurge, but rather the general thermal expansion stress that arises from the global heatup and cooldown of the pressurizer. This location is a stainless steel weld to the tube and clad very close to the low alloy steel pressurizer shell. A high steady state dissimilar metal thermal expansion stress is established during the heatup and is relaxed during the cooldown. It is of a magnitude that overwhelms the small stress additions coming from insurges and outsurges of fluid. The next most fatigue sensitive location is the pressurizer surge nozzle. This location is affected most by insurges and outsurges, having essentially no steady state stress. This location has a much smaller stress concentration effect than the heater weld.

The cumulative usage factors for the heater penetration, pressurizer surge nozzle and surge line nozzle-to-RCS hot leg connection were calculate and the results are as follows:

PRESSURIZER HEATER PENETRATION

Material:	Type 316 Stainless Steel
Usage Factor (60 years[a]):	0.048
Maximum Environmental Factor:	15.35
Usage Factor (60 years):	**$CUF_{60env} = 0.74$**

PRESSURIZER SURGE NOZZLE

Material:	SA 376 Type 316 Stainless Steel

PRESSURIZER SURGE NOZZLE

Usage Factor (60 years[a]):	6.276×10^{-7}
Maximum Environmental Factor:	15.35
Usage Factor (60 years):	$\mathbf{CUF_{60env} = 9.633 \times 10^{-6}}$

RCS HOT LEG SURGE NOZZLE

Material:	SA 376 Type 316 Stainless Steel
Usage Factor (60 years[a]):	0.0132
Maximum Environmental Factor:	15.35
Usage Factor (60 years):	$\mathbf{CUF_{60env} = 0.2022}$

a. note: Fatigue usage factor was calculated based on 200 heatup and cooldown cycles.

REFERENCES FOR SECTION 18

1. Letter from Robert C. Mecredy, RG&E, to Jack Cushing, NRC, "Application for Renewed Operating License, R.E. Ginna Nuclear Power Plant, Docket No. 50-244," dated July 30, 2002.

2. Letter from Robert C. Mecredy, RG&E, to Russell Arrighi, NRC, "Supplemental Response to LRA Request for Additional Information (RAI)," with the addition of the TLAA of RCP Flywheel, UFSAR Supplement Section18.2.1.23, dated May 23, 2003.

3. Letter from Robert C. Mecredy, RG&E, to Russell Arrighi, NRC, "Annual LRA Supplement," with revisions to UFSAR Supplement Sections: 18.2.1.1, 18.2.1.5, 18.2.1.7, 18.2.1.18, 18.2.1.20, 18.2.1.22, 18.3.3, and 18.3.3.7; dated July 30, 2003.

4. Letter from Robert C. Mecredy, RG&E, to Russell Arrighi, NRC, "Supplemental Information Regarding July 30, 2003 Correspondence," with revision to UFSAR Supplement Section18.3.3.7; dated August 1, 2003.

5. Letter from Robert C. Mecredy, RG&E, to Russell Arrighi, NRC, "Supplemental Information Regarding July 30, 2003 Correspondence," with revision to UFSAR Supplement Section18.3.3.6; dated August 1, 2003.

6. Letter from Robert C. Mecredy, RG&E, to Russell Arrighi, NRC, "RAI 3.7-3 Response," with revision to UFSAR Supplement Section18.2.1.10, dated August 6, 2003.

7. Letter from Robert C. Mecredy, RG&E, to Russell Arrighi, NRC, "C-RAI 4.3.7-1(a) Section B2.1.14 Updates," with revision to UFSAR Table 5.3-8; dated 8/8/2003.

8. Letter from Robert C. Mecredy, RG&E, to Russell Arrighi, NRC, "Open and Confirmatory Item Responses," with revision to UFSAR Supplement Sections 18.3.3.3, 18.3.3.5, and 18.4.3; dated September 16, 2003.

9. Letter from Robert C. Mecredy, RG&E, to Russell Arrighi, NRC, "Open and Confirmatory Item Responses to Ginna SER with Open Items," with revision to UFSAR Supplement Section18.3.1.2, dated December 9, 2003.

10. Letter from Robert C. Mecredy, RG&E, to Russell Arrighi, NRC, "Draft SER Comments," with addition of the Inaccessible Medium-Voltage Cables Not Subject to 10CFR50.49 Environmental Qualification Requirements, UFSAR Supplement Section 18.3.5, and changes to UFSAR Supplement Sections: 18.2.1.2, 18.2.1.14, 18.2.1.17, 18.2.1.20, 18.2.1.21, 18.2.1.27, 18.3.4, and 18.4.2; dated December 19, 2003.

11. Letter from Pao-Tsin Kuo, NRC, to Robert C. Mecredy, RG&E, "License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant," dated March 3, 2004.

12. Letter from David A. Holm, Ginna Station, to Document Control Desk (NRC), "Change in Submittal Date for the Augmented Reactor Vessel Internals Inspection Program Document," dated July 31, 2007.

13. Letter from George W. Herrick, Volian Enterprises, to Jay Wells, Ginna Station, "Documentation of Newly Identified Structures, Systems, and Components for License Renewal", dated January 13, 2009.

14. Letter from Joseph E. Pacher, Ginna LLC, to Document Control Desk (NRC), "License Renewal Aging Management Reactor Vessel Internals Program," dated February 27, 2009.

15. Letter from Thomas G. Mogren, Constellation Energy Group R.E. Ginna Nuclear Power Plant, LLC, to Document Control Desk (NRC), "R.E. Ginna Nuclear Power Plant, License Renewal Aging Management Withdraw Reactor Vessel Internal Program Document and Commit to Submit to Revised Reactor Vessel Internals Program Document in Accordance with RIS 2011-07," Dated September 13, 2011.

16. Letter from Thomas G. Mogren, Constellation Energy Nuclear Group, LLC EDF Group R.E. Ginna Nuclear Power Plant, LLC, to Document Control Desk (NRC), "R.E. Ginna - License Renewal Aging Management, Submit Revised Reactor Vessel Internals Program Document in Accordance with RIS 2011-07," Dated September 28, 2012.