APPENDIX A

UPDATED FINAL SAFETY ANALYSIS REPORT SUPPLEMENT

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A.0 INTRODUCTION

This appendix provides the information to be submitted in an Updated Final Safety Analysis Report Supplement as required by 10 CFR 54.21(d) for the James A. FitzPatrick Nuclear Power Plant (JAFNPP) License Renewal Application (LRA). The LRA contains the technical information required by 10 CFR 54.21(a) and (c). Appendix B of the JAFNPP LRA provides descriptions of the programs and activities that manage the effects of aging for the period of extended operation. Section 4 of the LRA documents the evaluations of time-limited aging analyses for the period of extended operation. Appendix B and Section 4 have been used to prepare the program and activity descriptions for the JAFNPP Updated Final Safety Analysis Report (UFSAR) Supplement information in this appendix.

This appendix is divided into two parts. The first part identifies changes to the existing sections of the UFSAR related to license renewal. The second part provides new information to be incorporated into the UFSAR. The information presented in both parts will be incorporated into the UFSAR following issuance of the renewed operating license. Upon inclusion of the UFSAR Supplement in the JAFNPP UFSAR, future changes to the descriptions of the programs and activities will be made in accordance with 10 CFR 50.59.

A.1 CHANGES TO EXISTING UFSAR INFORMATION

This section identifies changes to existing sections of the UFSAR that reflect a renewed operating license. Proposed text deletions are indicated by a strike-through and proposed text additions are indicated by underline.

A.1.1 UFSAR Chapter 3 Changes

Section 3.3.6 Inspection and Testing

(9th paragraph)

The BWR vibration acceptance criteria establish allowable sensor motions for continuous cyclic operation of the reactor for ~~a period of 40 years or~~ approximately 10^{10} cycles. The durations of the vibration tests were sufficiently long to assure that these acceptance criteria will not be violated for normal steady state of transient modes of plant operation.

A.1.2 UFSAR Chapter 4 Changes

Section 4.2.2 - Power Generation Design Bases

2. Reactor vessel design lifetime <u>was</u>~~is~~ originally forty years. <u>Evaluation of time limited aging analyses and management of aging effects have extended the life of the reactor vessel through the period of extended operation.</u>

Section 4.2.5.1 - Reactor Vessel

(1st paragraph)

The reactor vessel is a vertical, cylindrical pressure vessel with hemispherical heads of welded construction (Figure 3.3-1). The reactor vessel ~~is~~<u>was originally</u> designed and *fabricated for a service life of 40 years, based upon the specified design and operating* conditions. The limitations on the occurrences of design transient conditions for 60 years are specified in Table 4.2-3.

(5th paragraph)

According to a report from General Electric Co., which addressed the Reactor Vessel Material Surveillance Program, the net ~~end of forty years life~~ effect of a neutron fluence of 2 x 10^{18} nvt (1 Me V) on submerged metal arc welds made with copper coated electrodes will result in a 25°F variation in transition temperature.

Section 4.2.6 - Safety Evaluation

(4th paragraph)

To produce brittle fracture at or below the NDTT, a stress of 5000 to 8000 psi is considered necessary. Therefore, during operation when pressure is dependent upon temperature, brittle failure of the vessel is not considered possible until the neutron fluence of the reactor vessel reaches a value of the order of 10^{20} nvt. This value is a factor of more than 30~~50~~ times the maximum neutron fluence conservatively calculated during the lifetime of this plant.

Section 4.2.7 - Safety Evaluation

(5th paragraph)

NRC Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials", Revision 2, May 1988, provides the basis for the reactor vessel material surveillance analysis which accounts for irradiation embrittlement effects in the reactor vessel core region, or beltline. The best estimate fluence for the peak locations in the lower shell and the lower intermediate shell after 54~~32~~ effective full power years (EFPY) or 60~~40~~ years of power operation at 90~~80~~% capacity factor are expected to be 2.7~~1.61~~ x 10^{18} n/cm^2 and 3.1~~1.81~~ x 10^{18} n/cm^2 respectively at the vessel ID.

Section 4.3.4 - Reactor Recirculation System Description

(17th paragraph)

The original design objective for the recirculation pump casing ~~is~~was a useful life of 40 years, accounting for corrosion, erosion, and material fatigue. Management of corrosion, erosion, and fatigue for the period of extended operation is expected to result in a useful life of at least 60 years.

Section 4.6.3 - Main Steam Line Isolation Valve Description

(18th paragraph)

The isolation valve is designed to pass saturated steam at 1250 psig and 575°F with a moisture content of approximately 0.25 percent, oxygen content of 30 ppm, and a hydrogen content of 4 ppm. The original design objective for the valve was~~is~~ a minimum of 40 years service at the specified operating conditions. The original estimated operation cycles per year was~~is~~ 100 cycles during the first year and 50 cycles per year thereafter. The design basis cycles (2050 total) will not be exceeded during the extended period of operation. In addition to minimum wall thickness required by applicable codes, a corrosion allowance of 0.120 in minimum~~,~~ is added~~to provide for 40~~

~~years service~~. Management of corrosion for the period of extended operation is expected to result in a useful life of at least 60 years.

(19th paragraph)

Design specification ambient operating conditions for the valves and accessories are 135°F normal, 150°F maximum, at 100 percent relative humidity, in a radiation field of 10.0 R per hr gamma and 25 Rad per hr neutron plus gamma, continuous for design life. The inboard valves are not exposed to these design radiation levels continuously and the outboard valves are in much less severe ambient conditions. Aging management of the thermal and radiation aging effects will extend the useful life of the valve through the period of extended operation.

Section 4.10.3.4 - Leakage Detection System

(Drywell Continuous Atmosphere Radioactivity Monitoring, 2nd paragraph)

The drywell continuous atmosphere radioactivity monitor is not considered to be an adequate system for monitoring Reactor Coolant System leakage during initial period of plant operation when coolant activity is low. Systems of this nature have a minimum sensitivity on the order of 1×10^{-11} μ Ci/cc. If the plant has just begun operation, coolant activities for all isotopes are so low that the system is virtually useless. Therefore, during this period of plant life which is relatively a minor percentage when compared to the ~~40 year~~ life of the plant, other systems such as pump seal leakoff flow detection, drywell equipment and floor drain sump fillup rate detection, drywell local area temperature detection, relief valve discharge temperature detection, and drywell pressure detection, would be the primary means of detecting leakage.

A.1.3 UFSAR Chapter 7 Changes

Section 7.1.10 - The Design Criteria for Radiation Effects on Materials and Components

(3rd paragraph)

The electrical power and control cabling for safety system equipment which must function in a radiation environment has been tested under simulated post-accident radiation environment. The cabling has been irradiated with a Co-60 source to a dose of at least 2×10^7 rads which is in excess of that which safety system cabling inside the Primary Containment would experience during the original 40 years normal operations plus that which would be experienced according to the assumptions stated in Chapter 14. Certified test results indicate that the power and control cabling of the safety systems is capable of satisfactory performance in a BWR primary containment environment and elsewhere since the same quality of cable is used throughout the plant. Effects of aging will be adequately managed to assure the cabling remains qualified throughout the period of extended operation per 10 CFR 50.49.

(4th paragraph)

The individual components and lubricants of electric motor operators have been reviewed by the manufacturer for their ability to withstand the design basis radiation environment; i.e., that experienced during the original 40 years of normal operation plus that radiation which would be experienced resulting from a fission product release into the primary containment according to the assumptions stated in Chapter 14, during that portion of a LOCA in which valve operation would be required. The manufacturer's review indicates that the Limitorque operators are capable of proper operation after irradiation in excess of the design basis radiation environment. Effects of aging will be adequately managed to assure the electric motor operators remain qualified throughout the period of extended operation per 10 CFR 50.49.

Section 7.2.3.10 - Wiring

(1st paragraph)

Wiring and cables for Reactor Protection System instrumentation are selected to avoid deterioration due to temperature and humidity during the design life of the plant. Cables and connectors used inside the primary containment are designed for continuous operation at an ambient temperature of 150°F, a relative humidity of 90 percent, and a cumulative radiation dose of 5.5×10^{7} rads ~~over a 40 year span~~. Cables and connectors in the drywell are designed for the following short term conditions during and after the design basis accident:

A.1.4 UFSAR Chapter 12 Changes

Section 12.4.5 - Tornado Loads

(10th paragraph)

The structural design conditions consider the probability of a tornado striking the site and causing a telephone pole missile to penetrate these doors on the following basis. The probability of a tornado striking this plant site is estimated to be in the order of ~~3.12 x~~ 10^{-3} in a reactor lifetime ~~of 40 years~~. However the probability of a missile traveling (a) in a near horizontal position without bouncing, (b) with its maximum cross-sectional area exposed to the full wind force, and (c) striking the area of exterior doors instead of the remaining concrete structure, with such a trajectory to permit penetration of the inner pair of doors and thus entering the secondary containment, causes the probability of a missile penetrating the access lock doors to decrease to a value of ~~2.25 x~~ 10^{-9}. It is concluded from the foregoing that design against missile penetration is unwarranted.

A.1.5 UFSAR Chapter 16 Changes

Section 16.2.2.2 – Allowable Limits

(4th paragraph)

The term SF min is defined as the minimum safety factor on load or deflection and is related to the event probability by the following equation:

$$SF\ min = \frac{9}{3 - \log_{10} P_{\underline{60}\cancel{40}}}$$

where: $10^{-1} > P_{\underline{60}\cancel{40}} \geq 10^{-5}$

For event probabilities smaller than 10^{-5} or greater than 10^{-1}, the following apply:

Event Probability	Min. Safety Factor
$10^{-1} > P_{\underline{60}\cancel{40}} \geq 10^{-6}$	1.125
$1.0 > P_{\underline{60}\cancel{40}} \geq 10^{-1}$	2.25

These expressions show the probabilistic significance of the classical safety factor concept as applied to reactor safety. The SFmin values corresponding to the event probabilities are summarized in Table 16.2-2.

Section 16.2.3.2 - Reactor Vessel

(2nd paragraph)

Stress analysis requirements and load combinations for the reactor vessel have been evaluated for the cyclic conditions expected throughout plant~~the 40-year~~ life, with the conclusion that ASME code limits are satisfied.

Section 16.2.3.3 - Reactor Vessel Internals

Cumulative Fatigue Usage

U(Allowable) = 1.0

U(Calculated) = 0.~~65~~98

Table 16.2-1 - Loading Condition Probabilities

TABLE 16.2-1

LOADING CONDITION PROBABILITIES

Upset (likely)	$1.0 > P > 10^{-1}$
Emergency (low probability)	$10^{-1} > P > 10^{-3}$
Faulted (extremely low probability)	$10^{-3} > P > 10^{-6}$

where P = 60~~40~~ year event encounter probability

Table 16.2-7 - (sheet 41 of 45) Recirculation Loop Piping

Criteria

1. The sum of the longitudinal stresses due to pressure, dead weight, and inertia effects of design earthquake must be less than 1.8 times the hot allowable stress. The probability of this load occurrence during the 60~~40~~ yr plant life is 10^{-3} and SF = 1.5.

2. The sum of the longitudinal stresses due to maximum pressure, dead weight, and inertia effects of operating earthquake must be less than 1.5 times the hot allowable stress. The probability of this load occurring during the 60~~40~~ yr plant life is 10^{-2} and SF = 1.8.

3. The sum of the longitudinal stresses due to maximum pressure, dead weight, and inertia effects of design earthquake must be less than 2.0 times the hot allowable stress. The probability of this load combination occurring during the 60~~40~~ yr plant life is .~~25 x~~ 10^{-3} and SF = 1.5~~1.36~~.

Section 16.5.2.1.1 Wall Thickness

(3rd paragraph)

Adequate allowance is made in the pipe wall thickness for corrosion and erosion according to individual systems requirements~~ for a design life of 40 years~~.

Section 16.9.3.17.6 Reactor Pressure Vessel Fracture (RPV) Toughness

(2nd paragraph)

An assessment of the impact of power uprate on the RPV is detailed in Section 3.3.1 of Reference 5. The current design basis for end of life (EOL) fluence for FitzPatrick is 54~~32~~ effective full power years (EFPY) based upon 60~~40~~ years of power operation at 90~~80~~ percent capacity factor. ~~Based upon future operation, FitzPatrick will not reach 32 EFPY, therefore previous EOL evaluations are still valid.~~ An equivalent margins analysis was prepared to demonstrate compliance with Appendix G, 10 CFR Part 50. The analyses in the GE Topical Report NEDO-32205-A, Revision 1, February 1994, for the James A. FitzPatrick Nuclear Power Plant, have been updated for License Renewal~~are applicable to the FitzPatrick reactor vessel~~. The FitzPatrick reactor vessel will maintain margins of safety against fracture equivalent to those required by 10 CFR Part 50, Appendix G and the ASME code. The GE equivalent margins analyses are applicable with power uprate.

(10th paragraph)

The current pressure-temperature curves were developed from General Electric Report, "Plant FitzPatrick RPV Surveillance Materials Testing and Analysis of 120° Capsule at 13.4 EFPY," Revision 1, (GE-NE-B1100732-01), dated February 1998 with Errata and Addenda sheets dated June 17, 1999 and December 3, 1999. These curves are valid through 32 EFPY. This report was prepared after power uprate and specifically accounts for the increased fluence~~ in estimation of end-of-life vessel properties~~.

~~The current design basis for end of life (EOL) fluence for FitzPatrick is 32 EFPY based upon 40 years of power operation at 80 percent capacity factor. The predicted fluence for 32 EFPY was used to determine that vessel material properties meet the upper shelf energy (USE) requirements of 10 CFR Part 50. At the end of cycle 12, FitzPatrick operated for less than 13.5 EFPY. Future operation of FitzPatrick will not reach 32 EFPY, therefore previous EOL evaluations are still valid.~~ The current estimate for EOL conservatively assumes 90 percent capacity factor for 60 years of operation (54 EFPY). Pressure-temperature curves will be updated prior to reaching 32 EFPY, considering any surveillance capsule data that has been collected by that time. ~~1.05 EFPY for each future year of operation (due to uprate), 100 percent capacity factor, and a 45 day refueling outage every two years. The NEDO-32205-A, Revision 1, is still applicable.~~

~~Existing P-T curves, see Figure 16.5-1, are valid and conservative for FitzPatrick power uprate implementation. Ongoing plant-specific analyses bound the effects of power uprate and demonstrate compliance with 10 CFR Part 50, Appendix G.~~

A.2 NEW UFSAR SECTION

The following information will be integrated into the UFSAR to document aging management programs and activities credited in the JAFNPP license renewal review and time-limited aging analyses evaluated for the period of extended operation. References to other sections are to UFSAR sections, not to sections in the LRA.

A.2.0 Supplement for Renewed Operating License

The James A. FitzPatrick Nuclear Power Plant license renewal application (Reference A.2-1) and information in subsequent related correspondence provided sufficient basis for the NRC to make the findings required by 10 CFR 54.29 (Final Safety Evaluation Report) (Reference A.2-2). As required by 10 CFR 54.21(d), this UFSAR supplement contains a summary description of the programs and activities for managing the effects of aging (Section A.2.1) and a description of the evaluation of time-limited aging analyses for the period of extended operation (Section A.2.2). The period of extended operation is the 20 years after the expiration date of the original operating license.

A.2.1 Aging Management Programs and Activities

The integrated plant assessment for license renewal identified aging management programs necessary to provide reasonable assurance that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis (CLB) for the period of extended operation. This section describes the aging management programs and activities required during the period of extended operation. All aging management programs will be implemented prior to entering the period of extended operation.

JAFNPP quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B. The Entergy Quality Assurance Program applies to safety-related structures and components. Corrective actions and administrative (document) control for both safety-related and nonsafety-related structures and components are accomplished per the existing JAFNPP corrective action program and document control program and are applicable to all aging management programs and activities that will be required during the period of extended operation. The confirmation process is part of the corrective action program and includes reviews to assure that proposed actions are adequate, tracking and reporting of open corrective actions, and review of corrective action effectiveness. Any follow-up inspection required by the confirmation process is documented in accordance with the corrective action program.

A.2.1.1 Buried Piping and Tanks Inspection Program

The Buried Piping and Tanks Inspection Program includes (a) preventive measures to mitigate corrosion and (b) inspections to manage the effects of corrosion on the pressure-retaining capability of buried carbon steel, copper alloy, gray cast iron, and stainless steel components. Preventive measures are in accordance with standard industry practice for maintaining external coatings and wrappings. Buried components are inspected when excavated during maintenance. If trending within the corrective action program identifies susceptible locations, the areas with a history of corrosion problems are evaluated for the need for additional inspection, alternate coating, or replacement.

A focused inspection will be performed within the first 10 years of the period of extended operation, unless an opportunistic inspection (or an inspection via a method that allows assessment of pipe condition without excavation) occurs within this ten-year period.

A.2.1.2 BWR CRD Return Line Nozzle Program

Under the BWR CRD Return Line Nozzle Program, JAFNPP has cut and capped the CRD return line nozzle to mitigate cracking and continues inservice inspection (ISI) examinations to monitor the effects of crack initiation and growth on the intended function of the control rod drive return line nozzle and cap. ISI examinations include ultrasonic inspections of the nozzle-to-vessel weld, nozzle inside radius section, and the dissimilar metal weld overlay at the nozzle.

A.2.1.3 BWR Feedwater Nozzle Program

Under the BWR Feedwater Nozzle Program, JAFNPP has removed all identified feedwater blend radii flaws, removed feedwater nozzle cladding, and installed a double piston ring, triple thermal sleeve sparger to mitigate cracking. This program continues enhanced inservice inspection (ISI) of the feedwater nozzles in accordance with the requirements of ASME Section XI, Subsection IWB and the recommendation of General Electric (GE) NE-523-A71-0594 to monitor the effects of cracking on the intended function of the feedwater nozzles.

A.2.1.4 BWR Penetrations Program

The BWR Penetrations Program includes (a) inspection and flaw evaluation in conformance with the guidelines of staff-approved boiling water reactor vessel and internals project (BWRVIP) documents BWRVIP-27-A and BWRVIP-49-A and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-130 to ensure the long-term integrity of vessel penetrations and nozzles.

A.2.1.5 BWR Stress Corrosion Cracking Program

The BWR Stress Corrosion Cracking Program includes (a) preventive measures to mitigate intergranular stress corrosion cracking (IGSCC), and (b) inspection and flaw evaluation to monitor IGSCC and its effects on reactor coolant pressure boundary components made of stainless steel or CASS.

JAFNPP has taken actions to prevent IGSCC and will continue to use materials resistant to IGSCC for component replacements and repairs following the recommendations delineated in NUREG-0313, Generic Letter 88-01, and the staff-approved BWRVIP-75-A report. Inspection of piping identified in NRC Generic Letter 88-01 to detect and size cracks is performed in accordance with the staff positions on schedule, method, personnel qualification and sample expansion included in the generic letter and the staff-approved BWRVIP-75-A report.

A.2.1.6 BWR Vessel ID Attachment Welds Program

The BWR Vessel ID Attachment Welds Program includes (a) inspection and flaw evaluation in accordance with the guidelines of staff-approved BWR Vessel and Internals Project (BWRVIP) BWRVIP-48-A, and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-130 to ensure the long-term integrity and safe operation of reactor vessel inside diameter (ID) attachment welds and support pads.

A.2.1.7 BWR Vessel Internals Program

The BWR Vessel Internals Program includes (a) inspection, flaw evaluation, and repair in conformance with the applicable, staff-approved BWR Vessel and Internals Project (BWRVIP) documents, and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-130 to ensure the long-term integrity of vessel internals components.

A.2.1.8 Containment Leak Rate Program

As described in 10 CFR 50, Appendix J, containment leak rate tests are required to assure that (a) leakage through primary reactor containment and systems and components penetrating primary containment shall not exceed allowable values specified in technical specifications or associated bases and (b) periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of containment, and systems and components penetrating primary containment. Corrective actions are taken if leakage rates exceed acceptance criteria.

A.2.1.9 Diesel Fuel Monitoring Program

The Diesel Fuel Monitoring Program entails sampling to ensure that adequate diesel fuel quality is maintained to prevent loss of material in fuel systems. Exposure to fuel oil contaminants such as water and microbiological organisms is minimized by periodic sampling and analysis, draining and cleaning of tanks, and by verifying the quality of new oil before its introduction into the storage tanks.

A.2.1.10 Environmental Qualification (EQ) of Electric Components Program

The JAFNPP EQ of Electric Components Program manages the effects of thermal, radiation, and cyclic aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, EQ components exceeding their qualification are refurbished, replaced, or their qualification extended prior to reaching the aging limits established in the evaluations. Aging evaluations for EQ components are considered time-limited aging analyses (TLAAs) for license renewal.

A.2.1.11 External Surfaces Monitoring Program

The External Surfaces Monitoring Program entails inspections of external surfaces of components subject to aging management review. The program is also credited with managing loss of material from internal surfaces, for situations in which internal and external material and environment combinations are the same such that external surface condition is representative of internal surface condition.

Surfaces that are inaccessible during plant operations are inspected during refueling outages. Surfaces are inspected at frequencies to provide reasonable assurance that effect of aging will be managed such that applicable components will perform their intended function during the period of extended operation.

A.2.1.12 Fatigue Monitoring Program

In order not to exceed design limits on fatigue usage, the Fatigue Monitoring Program tracks the number of critical transients for selected reactor coolant system components. The program ensures the validity of analyses that explicitly assumed a fixed number of fatigue transients by assuring that the actual effective number of transients does not exceed the assumed limit.

The transient cycles tracked by this program are referenced in Section 4.3.

A.2.1.13 Fire Protection Program

The Fire Protection Program includes a fire barrier inspection and a diesel-driven fire pump inspection. The fire barrier inspection requires periodic visual inspection of fire barrier penetration seals, fire dampers and frames, fire barrier walls, ceilings, and floors, and periodic visual inspection and functional tests of fire rated doors to ensure that their operability is maintained. The diesel-driven fire pump inspection requires that the pump and its driver be periodically tested and inspected to ensure that diesel engine sub-systems, including the fuel supply line, can perform their intended functions.

A.2.1.14 Fire Water System Program

The Fire Water System Program applies to water-based fire protection systems that consist of sprinklers, nozzles, fittings, valves, hydrants, hose stations, standpipes, and aboveground and underground piping and components that are tested in accordance with applicable National Fire Protection Association (NFPA) codes and standards. Such testing assures functionality of systems. Also, many of these systems are normally maintained at required operating pressure and monitored such that leakage resulting in loss of system pressure is immediately detected and corrective actions initiated.

In addition, wall thickness evaluations of fire protection piping are periodically performed on system components using non-intrusive techniques (e.g., volumetric testing) to identify evidence of loss of material due to corrosion.

A sample of sprinkler heads will be inspected using the guidance of NFPA 25 (2002 Edition) Section 5.3.1.1.1, which states, "Where sprinklers have been in place for 50 years, they shall be replaced or representative samples from one or more sample areas shall be submitted to a recognized testing laboratory for field service testing." This sampling will be repeated every 10 years after initial field service testing.

A.2.1.15 Flow-Accelerated Corrosion Program

The Flow-Accelerated Corrosion Program applies to safety-related and nonsafety-related carbon steel components in systems containing high-energy fluids carrying two-phase or single-phase high-energy fluid > 2% of plant operating time.

The program, based on EPRI recommendations for an effective flow-accelerated corrosion program, predicts, detects, and monitors FAC in plant piping and other pressure retaining components. This program includes (a) an evaluation to determine critical locations, (b) initial operational inspections to determine the extent of thinning at these locations, and (c) follow-up inspections to confirm predictions. The program specifies repair or replacement of components as necessary.

A.2.1.16 Heat Exchanger Monitoring Program

The Heat Exchanger Monitoring Program inspects heat exchangers for degradation. If degradation is found, then an evaluation is performed to evaluate its effects on the heat exchanger's design functions including its ability to withstand a seismic event.

Representative tubes within the population of heat exchangers are eddy current tested at a frequency determined by internal and external operating experience to ensure that effects of aging are identified prior to loss of intended function. Along with each eddy current test, visual inspections are performed on accessible heat exchanger heads, covers and tube sheets to monitor surface condition for indications of loss of material. The population of heat exchangers includes the HPCI turbine lube oil coolers and gland seal condensers, and EDG lube oil heat exchangers.

A.2.1.17 *Inservice Inspection - Containment Inservice Inspection (CII) Program*

The Containment Inservice Inspection Program outlines the requirements for the inspection of Class MC pressure-retaining components (primary containment) and their integral attachments in accordance with the requirements of 10 CFR 50.55a and the ASME Boiler and Pressure Vessel Code, 1992 Edition with no Addenda, Section XI, Subsection IWE examination Category E-A, Item No. E1.11, and 1998 Edition with no Addenda, Section XI, Subsection IWE Examination Category E-A, Item No. E1.10.

The primary inspection method for the primary containment and its integral attachments is visual examination. Visual examinations are performed either directly or remotely with illumination and resolution suitable for the local environment to assess general conditions that may affect either the containment structural integrity or leak tightness of the pressure retaining component. The program includes augmented ultrasonic exams to measure wall thickness of the containment drywell structure.

A.2.1.18 Inservice Inspection – Inservice Inspection (ISI) Program

The ISI Program is based on ASME Inspection Program B (Section XI, IWA-2432), which has 10-year inspection intervals. Every 10 years the program is updated to the latest ASME Section XI code edition and addendum approved in 10 CFR 50.55a. On September 28, 1997, JAFNPP entered the third ISI interval. The code edition and addenda used for the third interval is the 1989 Edition with no Addenda.

The program consists of periodic volumetric, surface, and visual examination of components and their supports for assessment, signs of degradation, flaw evaluation, and corrective actions.

A.2.1.19 Metal-Enclosed Bus Inspection Program

Under the Metal-Enclosed Bus Inspection Program, internal portions of the non-segregated phase bus T2Y and T3Y components are inspected for cracks, corrosion, foreign debris, excessive dust buildup, and evidence of water intrusion. Bus insulation is inspected for signs of embrittlement, cracking, melting, swelling, or discoloration, which may indicate overheating or aging degradation. Internal bus supports are inspected for structural integrity and signs of cracks. Since bolted connections are covered with heat shrink tape or insulating boots per manufacturer's recommendations, a sample of accessible bolted connections is visually inspected for insulation material surface anomalies. Enclosure assemblies are visually inspected for evidence of loss of material and, where applicable, enclosure assembly elastomers are visually inspected and manually flexed to manage cracking and change in material properties.

These inspections are performed at least once every five years.

A.2.1.20 Non-EQ Instrumentation Circuits Test Review Program

Under the Non-EQ Instrumentation Circuits Test Review Program, calibration or surveillance results for non-EQ electrical cables in circuits with sensitive, high voltage, low-level signals; (i.e., neutron flux monitoring instrumentation); are reviewed. Most neutron flux monitoring system cables and connections are calibrated as part of the instrumentation loop calibration at the normal calibration frequency, which provides sufficient indication of the need for corrective actions based on acceptance criteria related to instrumentation loop performance. The review of calibration results is performed once every 10 years.

For neutron flux monitoring system cables that are disconnected during instrument calibrations, testing is performed at least once every 10 years using a proven method for detecting deterioration for the insulation system (such as insulation resistance tests, or time domain reflectometry).

A.2.1.21 Non-EQ Insulated Cables and Connections Program

The Non-EQ Insulated Cables and Connections Program provides reasonable assurance that intended functions of insulated cables and connections exposed to adverse localized environments caused by heat, radiation and moisture can be maintained consistent with the current licensing basis through the period of extended operation. An adverse localized environment is significantly more severe than the specified service condition for the insulated cable or connection.

A representative sample of accessible insulated cables and connections in adverse localized environments is visually inspected at least once every 10 years for cable and connection jacket surface anomalies such as embrittlement, discoloration, cracking or surface contamination.

A.2.1.22 Oil Analysis Program

The Oil Analysis Program maintains oil systems free of contaminants (primarily water and particulates) thereby preserving an environment that is not conducive to loss of material, cracking, or fouling. Activities include sampling and analysis of lubricating oil for detrimental contaminants, water, and particulates.

Sampling frequencies are based on vendor recommendations, accessibility during plant operation, equipment importance to plant operation, and previous test results.

A.2.1.23 One-Time Inspection Program

The elements of the One-Time Inspection Program include (a) determination of the sample size based on an assessment of materials of fabrication, environment, plausible aging effects, and operating experience; (b) identification of the inspection locations in the system or component based on the aging effect; (c) determination of the examination technique, including acceptance criteria that would be effective in managing the aging effect for which the component is examined; and (d) evaluation of the need for follow-up examinations to monitor the progression of any aging degradation.

A one-time inspection activity is used to verify the effectiveness of the water chemistry control programs by confirming that unacceptable cracking, loss of material, and fouling is not occurring on components within systems covered by water chemistry control programs [Sections A.2.1.32, A.2.1.33, and A.2.1.34].

One-time inspection activities on

- internal surfaces of HPCI system components containing untreated air,
- carbon steel and cast iron plant drain components exposed to indoor air,
- internal surfaces of carbon steel components in the EDG system containing untreated air,

- internal surfaces of stainless steel and aluminum components in the radioactive waste system containing raw water,
- internal surfaces of stainless steel and copper alloy components in the raw water treatment system containing raw water,
- internal surfaces of copper alloy components in the plumbing, sanitary and lab system and the city water system containing raw water,
- small bore piping in the reactor coolant system and associated systems that form the reactor coolant pressure boundary,
- internal surfaces of carbon steel scram accumulators,
- reactor vessel flange leakoff line, and
- main steam flow restrictors

are used to confirm that loss of material, cracking, and reduction of fracture toughness, as applicable, are not occurring or are so insignificant that an aging management program is not warranted.

When evidence of an aging effect is revealed by a one-time inspection, routine evaluation of the inspection results will identify appropriate corrective actions.

A.2.1.24 Periodic Surveillance and Preventive Maintenance Program

The Periodic Surveillance and Preventive Maintenance Program includes periodic inspections and tests that manage aging effects not managed by other aging management programs. The preventive maintenance and surveillance testing activities are generally implemented through repetitive tasks or routine monitoring of plant operations.

Periodic inspections using visual or other non-destructive examination techniques verify that the following components are capable of performing their intended function.

- battery racks "A" & "B" carbon steel framing
- reactor building cranes, crane rails and girders
- equipment access lock doors
- refueling platform carbon steel components
- reactor track bay inner & outer doors carbon steel components
- drywell equipment hatch (16X-1A, 16X-1B) and drywell personnel hatch (16X-2A) carbon steel components
- elastomer seals for equipment lock doors at reactor track bay inner & outer doors
- main steam relief valve tailpipes in the waterline region of the torus
- carbon steel portion of T quenchers in the waterline region of the torus
- HPCI, RCIC, and core spray piping listed as susceptible to erosion
- loop seal piping and valves on demister drain piping off each filter train and at drain piping downstream of the SGT fans
- piping (including loop seals) and valves in the vent piping and from the stack analyzer sample chambers (including loop seal)
- piping downstream of the SGT fans between the drain and the outlet of the stack sump

- piping, valves and flow elements in the discharge piping from the steam packing exhauster and the condenser air removal pumps to the SGT discharge piping to the stack
- external surfaces of coils for CAD heat exchangers 27E-1A/B, 27NV-A/B, 27PBC-1A/B
- internal surfaces of EDG air intake components - aftercoolers (fins), flexible duct connection
- internal surfaces of EDG exhaust and air start subsystem components
- HVAC duct flexible connections
- air handling units 70AHU-3A & B, 70AHU-12A & B, 70AHU-19A, B
- heat exchanger portions of the control and relay room chillers 70RWC-2A and 70RWC-2B
- floor drain components that provide a drain path for fire suppression water from floor drains to the floor drain collection tank or to the yard drain system
- internal surfaces of security generator exhaust gas components
- external surfaces of security generator radiator heat exchanger coils
- internal surfaces of carbon steel components in the radwaste system
- internal surfaces of carbon steel and copper alloy components in the circulating water system
- internal surfaces of carbon steel components in the turbine building closed loop cooling system
- internal surfaces of carbon steel components in the raw water treatment system
- internal surfaces of carbon steel components in the contaminated equipment drain system
- internal surfaces of carbon steel and stainless steel components used in chemical treatment in the service water system
- internal surfaces of carbon steel pump casings in the turbine building ventilation system
- external surfaces of copper alloy tube for administration building ventilation and cooling system unit coolers 72UC-12A & B, 72UC-25, 72UC-26, 72UC-35
- internal surfaces of carbon steel components BFP-255, WSC-250 - 260, STR-253 in the plumbing, sanitary and lab system
- internal surfaces of carbon steel components WSC-7A - 7C, WSC-8, WSC-37, WSC-40 in the city water system

A.2.1.25 Reactor Head Closure Studs Program

The Reactor Head Closure Studs Program includes inservice inspection (ISI) in conformance with the requirements of the ASME Code, Section XI, Subsection IWB, and preventive measures (e.g. rust inhibitors, stable lubricants, appropriate materials) to mitigate cracking and loss of material of reactor head closure studs, nuts, washers, and bushings.

A.2.1.26 Reactor Vessel Surveillance Program

JAFNPP is a participant in the BWR vessel and internals project (BWRVIP) Integrated Surveillance Program (ISP). The Reactor Vessel Surveillance Program monitors changes in the fracture toughness properties of ferritic materials in the reactor pressure vessel (RPV) beltline region. As BWRVIP-ISP capsule test reports become available for RPV materials representative of JAFNPP, the actual shift in the reference temperature for nil-ductility transition of the vessel material may be updated. In accordance with 10 CFR 50 Appendices G and H, JAFNPP reviews relevant test reports to assure compliance with fracture toughness requirements and P-T limits.

BWRVIP-116, "BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Implementation for License Renewal," describes the design and implementation of the ISP during the period of extended operation. BWRVIP-116 identifies additional capsules, their withdrawal schedule, and contingencies to ensure that the requirements of 10 CFR 50 Appendix H are met for the period of extended operation.

A.2.1.27 Selective Leaching Program

The Selective Leaching Program ensures the integrity of components made of cast iron, bronze, brass, and other alloys exposed to raw water, treated water, soil, or other environments that may lead to selective leaching. The program includes a one-time visual inspection and hardness measurement of selected components that may be susceptible to selective leaching to determine whether loss of material due to selective leaching is occurring, and whether the process will affect the ability of the components to perform their intended function for the period of extended operation.

A.2.1.28 Service Water Integrity Program

The Service Water Integrity Program relies on implementation of the recommendations of NRC GL 89-13 to ensure that the effects of aging on the service water systems (SWS) will be managed for the period of extended operation. The SWS includes the normal service water (NSW), emergency service water (ESW), and residual heat removal service water (RHRSW). The program includes component inspections for erosion, corrosion, and blockage and performance monitoring to verify the heat transfer capability of the safety-related heat exchangers cooled by SW. Chemical treatment using biocides and chlorine and periodic cleaning and flushing of redundant or infrequently used loops are the methods used to control or prevent fouling within the heat exchangers and loss of material in SW components.

A.2.1.29 Structures Monitoring - Masonry Wall Program

The objective of the Masonry Wall Program is to manage aging effects so that the evaluation basis established for each masonry wall within the scope of license renewal remains valid through the period of extended operation.

The program includes all masonry walls identified as performing intended functions in accordance with 10 CFR 54.4. Included components are the 10 CFR 50.48-required masonry walls, radiation shielding masonry walls, and masonry walls with the potential to affect safety-related components.

Masonry walls are visually examined at a frequency selected to ensure there is no loss of intended function between inspections.

A.2.1.30 Structures Monitoring - Structures Monitoring Program

Structures monitoring is in accordance with 10 CFR 50.65 (Maintenance Rule) as addressed in Regulatory Guide 1.160 and NUMARC 93-01. Periodic inspections are used to monitor the condition of structures and structural components to ensure there is no loss of structure or structural component intended function.

A.2.1.31 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel

The purpose of the Thermal Aging and Neutron Irradiation Embrittlement of CASS Program is to assure that reduction of fracture toughness due to thermal aging and reduction of fracture toughness due to radiation embrittlement will not result in loss of intended function during the period of extended operation. This program evaluates CASS components in the reactor vessel internals and requires non-destructive examinations as appropriate.

A.2.1.32 Water Chemistry Control – Auxiliary Systems Program

The purpose of the Water Chemistry Control – Auxiliary Systems Program is to manage loss of material for components exposed to treated water.

Program activities include sampling, analysis, and replacement of coolant for control room and relay room chilled water system, security generator jacket cooling water, auxiliary boiler heating water, decay heat removal cooling water, and the stator cooling water system to minimize component exposure to aggressive environments.

The One-Time Inspection Program for Water Chemistry utilizes inspections or non-destructive evaluations of representative samples to verify that the Water Chemistry Control - Auxiliary Systems Program has been effective at managing loss of material.

A.2.1.33 Water Chemistry Control – BWR Program

The objective of the Water Chemistry Control – BWR Program is to manage aging effects caused by corrosion and cracking mechanisms. The program relies on monitoring and control of water chemistry based on EPRI Report 1008192 (BWRVIP-130). BWRVIP-130 has three sets of guidelines: one for primary water, one for condensate and feedwater, and one for control rod drive (CRD) mechanism cooling water. EPRI guidelines in BWRVIP-130 also include recommendations for controlling water chemistry in the torus, condensate storage tank, demineralized water storage tanks, and spent fuel pool.

The Water Chemistry Control – BWR Program optimizes primary water chemistry to minimize the potential for loss of material and cracking. This is accomplished by limiting the levels of contaminants in the RCS that could cause loss of material and cracking. Additionally, JAFNPP has instituted hydrogen water chemistry (HWC) and noble metal chemical addition (NMCA) to limit the potential for intergranular SCC (IGSCC) through the reduction of dissolved oxygen in the treated water.

The One-Time Inspection Program for Water Chemistry utilizes inspections or non-destructive evaluations of representative samples to verify that the Water Chemistry Control – BWR Program has been effective at managing loss of material.

A.2.1.34 Water Chemistry Control – Closed Cooling Water Program

The Water Chemistry Control – Closed Cooling Water Program includes preventive measures that manage loss of material, cracking, and fouling for components in closed cooling water systems (jacket cooling water subsystem for the emergency diesel generator, reactor building closed loop cooling, and turbine building closed loop cooling). These chemistry activities provide for monitoring and controlling closed cooling water chemistry using JAFNPP procedures and processes based on EPRI guidance for closed cooling water chemistry.

The One-Time Inspection Program for Water Chemistry utilizes inspections or non-destructive evaluations of representative samples to verify that the Water Chemistry Control - Closed Cooling Water Program has been effective at managing loss of material.

A.2.1.35 Bolting Integrity Program

The Bolting Integrity Program relies on recommendations for a comprehensive bolting integrity program, as delineated in NUREG-1339, and industry recommendations, as delineated in the Electric Power Research Institute (EPRI) NP-5769, with the exceptions noted in NUREG-1339 for safety-related bolting. The program relies on industry recommendations for comprehensive bolting maintenance, as delineated in EPRI TR-104213 for pressure retaining bolting and structural bolting.

A.2.2 Evaluation of Time-Limited Aging Analyses

In accordance with 10 CFR 54.21(c), an application for a renewed license requires an evaluation of time-limited aging analyses (TLAA) for the period of extended operation. The following TLAA have been identified and evaluated to meet this requirement.

A.2.2.1 Reactor Vessel Neutron Embrittlement

The reactor vessel neutron embrittlement TLAA has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii). Fifty-four EFPY will be the effective full power years at the end of the period of extended operation assuming an average capacity factor of 90% for 60 years.

A.2.2.1.1 Reactor Vessel Fluence

Calculated fluence is based on a time-limited assumption defined by the operating term. As such, fluence is the time-limited assumption for the time-limited aging analyses that evaluate reactor vessel embrittlement.

The existing 32 EFPY fluence is based on a General Electric analysis of measured fluence from the JAFNPP surveillance flux wires (Reference A.2-8). These fluence values were further extrapolated to 54 EFPY to obtain peak plate ID fluences with 1/4 T values derived using RG 1.99 formula and conservative wall thicknesses.

A.2.2.1.2 Pressure-Temperature Limits

The P-T limits were derived from calculations made in accordance with the guidance of ASME Appendix G, as modified by Code Cases N-588 and N-640, ASTM Standards, 10 CFR 50 Appendices G and H, Regulatory Guide 1.99 Revision 2, and Generic Letter 88-11.

Pressure-temperature limits are valid through 32 EFPY. The fact that the projected maximum RT_{NDT} is well below the 200°F suggested in Section 3 of Regulatory Guide 1.99, gives confidence that P-T curves will provide acceptable operating area through 54 EFPY. The BWRVIP Integrated Surveillance Program (BWRVIP Reports 86-A, 102, 116 and 135) will be used to adjust projected RT_{NDT} values as additional surveillance capsule results are collected. JAFNPP will submit additional P-T curves prior to the period of extended operation.

A.2.2.1.3 Charpy Upper-Shelf Energy

The predictions for percent drop in C_VUSE at 54 EFPY are based on chemistry data and unirradiated C_VUSE data submitted to the NRC in the JAFNPP response to GL 92 01, and 1/4 T fluence values.

The 54 EFPY C_VUSE values were calculated using Regulatory Guide 1.99, Position 1, Figure 2; specifically, the formula for the lines was used to calculate the percent drop in C_VUSE.

All C_VUSE values are predicted to remain well above the requirement of 50 ft-lbs during the period of extended operation. As such, this TLAA has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.2.2.1.4 Adjusted Reference Temperature

JAFNPP has projected values for RT_{NDT} and adjusted reference temperature (ART) at 54 EFPY using the methodology of Regulatory Guide 1.99. These values were calculated using the chemistry data, margin values, initial RT_{NDT} values, and chemistry factors (CFs) contained in the JAFNPP response to GL 92-01 and other licensing correspondence (Reference A.2-6). New fluence factors (FFs) were calculated using the expression in Regulatory Guide 1.99, Revision 2, Equation 2 using 54 EFPY fluence values.

The RT_{NDT} TLAA has been projected through the period of extended operation, with acceptable results, in accordance with 10 CFR 54.21(c)(1)(ii).

A.2.2.1.5 Reactor Vessel Circumferential Weld Inspection Relief

Relief from reactor vessel circumferential weld examination requirements under Generic Letter 98-05 is based on assessments indicating an acceptable probability of failure per reactor operating year. The analysis is based on reactor vessel metallurgical conditions as well as flaw indication sizes and frequencies of occurrence that are expected at the end of a licensed operating period.

JAFNPP received NRC approval for this relief for the remainder of the original 40-year license term (Reference A.2-3). The basis for this relief request is an analysis that satisfied the limiting conditional failure probability for the circumferential welds at the expiration of the current license, based on the NRC SERs for BWRVIP-05 (Reference A.2-5) and BWRVIP-74 (Reference A.2-7) and the extent of neutron embrittlement.

The JAFNPP reactor pressure vessel circumferential weld parameters at 54 EFPY will remain within the NRC's (64 EFPY) bounding CEOG parameters from the BWRVIP-05 SER. Although a conditional failure probability has not been calculated, the fact that the JAFNPP values at the end of license are less than the 64 EFPY value provided by the NRC leads to the conclusion that the JAFNPP RPV conditional failure probability is bounded by the NRC analysis. As such, the conditional probability of failure for circumferential welds remains below that stated in the NRC's Final Safety Evaluation of BWRVIP-05. Therefore, this analysis has been projected through the period of extended operation per 10 CFR 54.21 (c)(1)(ii).

A.2.2.1.6 Reactor Vessel Axial Weld Failure Probability

The BWRVIP recommendations for inspection of reactor vessel shell welds are based on generic analyses supporting an NRC SER (References A.2-4, A.2-5). The generic-plant axial weld failure rate is no more than 5×10^{-6} per reactor year as calculated in the BWRVIP-74 SER (Reference A.2-7). BWRVIP-05 showed that this axial weld failure rate is orders of magnitude greater than the 40 year end-of-life circumferential weld failure probability, and used this analysis to justify relief from inspection of the circumferential welds as described above.

The BWRVIP-74 SER states it is acceptable to show that the mean RT_{NDT} of the limiting beltline axial weld at the end of the period of extended operation is less than the limiting value given in the SERs for BWRVIP-74 and BWRVIP-05. The projected 54 EFPY mean RT_{NDT} values for JAFNPP are less than the limiting 64 EFPY RT_{NDT} in the analysis performed by the NRC staff (Table 2.6-5 of the BWRVIP-05 SER). As such, this TLAA has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.2.2.2 Metal Fatigue

A.2.2.2.1 Class 1 Metal Fatigue

Class 1 components evaluated for fatigue and flaw growth include the reactor pressure vessel (RPV) and appurtenances, certain reactor vessel internals, the reactor recirculation system (RRS), and the reactor coolant system (RCS) pressure boundary. The JAFNPP Class 1 systems include components within the ASME Section XI, IWB inspection boundary.

The design of the reactor vessel internals is in accordance with the intent of ASME Section III. A review of the design basis documents reveals that fatigue analyses were performed and determined the most significant fatigue loading occurs in the jet pump-shroud-shroud support area of the internals. The location of the maximum fatigue usage is at the ID of the jet pump diffuser adapter at the thin end of the tapered transition section. Additionally, a fatigue evaluation was performed on the tie rod assemblies installed as part of the core shroud repair. The maximum CUF values identified have been projected to the end of the period of extended operation and remain less than 1.0.

The JAFNPP fatigue monitoring program will assure that the allowed number of transient cycles is not exceeded. The program requires corrective action if transient cycle limits are approached. Consequently, the TLAA (fatigue analyses) based on those transients will remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i) or are projected through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.2.2.2.2 Non-Class 1 Metal Fatigue

For non-Class 1 components identified as subject to cracking due to fatigue, a review of system operating characteristics was conducted to determine the approximate frequency of any significant thermal cycling. If the number of equivalent full temperature cycles is below the limit used for the original design (usually 7000 cycles), the component is suitable for extended operation. If the number of equivalent full temperature cycles exceeds the limit, evaluation of the individual stress calculations require evaluation. No components were identified with projected cycles exceeding 7000. Therefore, the TLAA for non-Class 1 piping and components remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(i).

A.2.2.2.3 Environmental Effects on Fatigue

The effects of reactor water environment on fatigue were evaluated for license renewal. Projected cumulative usage factors (CUFs) were calculated for the limiting locations identified in NUREG/CR-6260. For the locations with CUFs less than 1.0, the TLAA has been projected through the period of extended operation per 10 CFR 54.21(c)(1)(ii). Several locations may exceed a CUF of 1.0 with consideration of environmental effects during the period of extended operation. For these locations, prior to the period of extended operation, JAFNPP will (1) refine the fatigue analysis to lower the predicted CUF to less than 1.0; (2) manage fatigue at the affected locations with an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC); or (3) repair or replace the affected locations. Depending on the option chosen, which may vary by component, this TLAA will be projected through the period of extended operation or the effects of environmentally assisted fatigue will be managed per 10 CFR 54.21(c)(1)(iii).

A.2.2.3 Environmental Qualification of Electrical Components

The JAFNPP EQ Program implements the requirements of 10 CFR 50.49 (as further defined by the Division of Operating Reactors Guidelines, NUREG-0588, and Reg. Guide 1.89). The program requires action before individual components exceed their qualified life. In accordance with 10 CFR 54.21(c)(1)(iii), implementation of the EQ Program provides reasonable assurance that the effects of aging on components associated with EQ TLAAs will be adequately managed such that the intended functions can be maintained for the period of extended operation.

A.2.2.4 Fatigue of Primary Containment, Attached Piping, and Components

In conjunction with the Mark I Containment Long-Term Program, the torus and attached piping systems were analyzed for fatigue due to mechanical loadings as well as thermal and anchor motion. This analysis was based on assumptions of the number of SRV actuations, operating basis earthquakes, and accident conditions during the life of the plant.

The analysis considered all BWR plants which utilize the Mark I containment design. The analysis concluded that for all plants and piping systems considered, the fatigue usage factor for an assumed 40-year plant life was less than 0.5. Extending plant life by an additional 20 years would produce a usage factor below 0.75. Since this is less than 1.0, the fatigue criteria are satisfied. This TLAA has been projected through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.2.2.5 Recirculation Valve Fatigue Evaluation

The recirculation isolation valves are evaluated for 30 cycles of normal pressurization followed by blowdown and 270 cycles of normal pressurization followed by normal depressurization.

This number of cycles evaluated exceeds the value allowed as part of the Fatigue Monitoring Program, so the transients suggested will not be exceeded. Thus this TLAA will remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.2.2.6 Leak Before Break

The leak detection system is designed to detect and in some cases automatically isolate a leak before it becomes a break. The crack growth analysis supporting these leak detection systems is a TLAA. Prior to the period of extended operation, JAFNPP will revise the leak detection system supporting documentation such that either (1) it does not include a TLAA, or (2) the existing TLAA is projected through the period of extended operation.

A.2.2.7 Core Plate

The loss of preload and cracking of the core plate rim hold-down bolts is a TLAA per the NRC SER for BWRVIP-25. Appendix B to BWRVIP-25 projected this calculation to 60 years, showing that the core hold down bolts at JAFNPP will retain at least 81% of their preload through the period of extended operation. Preload of the core plate holddown bolts is required to prevent lateral motion of the core plate for those plants that have not installed core plate wedges (including JAFNPP). A plant-specific calculation is required to determine minimum bolting requirements to prevent core plate motion. JAFNPP commits to perform a plant-specific calculation prior to the period of extended operation unless core plate wedges are installed during the remainder of the current licensing term. Thus the loss of core plate hold down bolt preload will be projected for the period of extended operation.

A.2.2.8 Shroud Support

The fatigue analysis of the shroud support is considered TLAA. The shroud support is included in the 60-year fatigue analysis and shows a CUF of 0.9. This analysis remains valid for the period of extended operation per 10 CFR 54.21(c)(1)(i).

A.2.2.9 Lower Plenum

The fatigue analysis of the lower plenum pressure boundary components is considered a TLAA. The bottom head, shroud support, and CRD penetrations in the lower plenum are included in the 60-year fatigue analysis. Values for CUF are 0.03, 0.90, and 0.0234 respectively. This analysis remains valid for the period of extended operation per 10 CFR 54.21(c)(1)(i).

A.2.3 References

A.2-1 [JAFNPP License Renewal Application—later]

A.2-2 [(NRC SER for JAFNPP License Renewal—later]

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A.2-4 Lainas, G. C. (NRC), to C. Terry (Niagara Mohawk Power Company, BWRVIP Chairman), BWRVIP-05 SER (Final), Final Safety Evaluation of the BWRVIP Vessel and Internals Project BWRVIP-05 Report (TAC No. M93925), letter dated July 28, 1998.

A.2-5 Lainas, G. C. (NRC), to C. Terry (BWRVIP), Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC No. M93925), letter dated July 28, 1998.

A.2-6 Josiger, W. A. (NYPA), to USNRC Document Control Desk, "James A. Fitzpatrick Nuclear Power Plant, Docket No. 50-333, Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity," letter JPN-94-041 dated August 10, 1994.

A.2-7 Grimes, C. I. (NRC), to C. Terry (BWRVIP Chairman), Acceptance for referencing of EPRI Proprietary Report TR-113596, BWR Vessel and Internals Project, BWR Reactor Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74) and Appendix A, Demonstration of Compliance with the Technical Information requirements of the License Renewal Rule (10CRF54.21), letter dated October 18, 2001.

A.2-8 GE-NE-B1100732-01, Revision 1, February 1998, Plant Fitzpatrick RPV Surveillance Materials Testing and Analysis of 120° Capsule at 13.4 EFPY