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 10CFR54.21(d) for the Arkansas Nuclear One, Unit 2 (ANO-2) License Renewal Application (LRA). The LRA contains the technical information required by 10CFR54.21(a) and (c). Appendix B of the ANO-2 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. These LRA sections have been used to prepare the program and activity descriptions for the ANO-2 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 ANO-2 UFSAR, future changes to the descriptions of the programs and activities will be made in accordance with 10CFR50.59.

A.1 CHANGES TO EXISTING UFSAR INFORMATION

This section identifies changes to existing sections of the UFSAR that reflect a renewed operating license. In the proposed changes to the existing UFSAR, reference to Section “A.2” of the UFSAR indicates the chapter which incorporates the new UFSAR section in A.2 of this appendix. Proposed text deletions are indicated by a strikethrough and proposed text additions are indicated by double underline.

A.1.1 UFSAR Chapter 1 Change

Section 1.2.2.1.2 – Second paragraph

The reactor vessel and its closure head are fabricated from SA-533 Grade B steel clad with stainless steel. ~~The vessel and its internals are designed so that the integrated neutron fluence (greater than 1.0 Mev) at the vessel wall will be less than 3.64E19 nvt over a 40-year period.~~ Fluence and fracture toughness of the reactor vessel are discussed in Section 4.3.3.3 and Section 5.2.4.

A.1.2 UFSAR Chapter 3 Changes

Section 3.6.1 – Second main paragraph and list

Breaks in the Shutdown Cooling (SDC) line from the reactor coolant pipe nozzle to the first isolation valve have the same consequences as a LOCA, as such breaks have been postulated in this section of the SDC line. Pipe breaks were not postulated in the SDC piping downstream of the isolation valve 2CV-5086-2. The shutdown cooling mode is a manually initiated operation and is initiated only under strict administrative controls when the reactor coolant temperature and pressure are below 300°F and 300 psia. The system exceeds 200°F and 275 psig less than two percent of the system operating time. This is based on the following conservative assumptions:

- Refueling occurs once ~~a year~~ every 18 months and the SDC System is in operation for an average of 21 days (500 hours) each refueling period over the ~~40-year~~ plant life.
- In addition, 20 shutdowns to cold shutdown condition were assumed to occur over the life of the plant, each lasting four days or 96 hours.
- During each shutdown, the system is assumed to operate over 200 °F and 275 psig for six hours. This assumes an average cooldown rate of 25 °F per hour while maximum cooldown rate is 75 °F per hour. The SDC System is not used to heat up the Reactor Coolant System (RCS) after shutdown.

Section 3.8.1.6.4 – New paragraph at end of section

The evaluation of tendon prestress for the period of extended operation associated with license renewal is discussed in Section [A.2.2.4.]

A.1.3 UFSAR Chapter 4 Changes

Section 4.2.2.1.2 – First paragraph

The pressure boundary portions of the in-core instrumentation support system are designed to Section III of the ASME Boiler and Pressure Vessel Code. The supports for the pressure tubing are designed to withstand design loadings plus earthquake forces. All components and materials are consistent with ~~40-year plant life,~~ a 10-year surveillance cycle and a 5-year mean-time-between instrument replacements. The guide tubes are designed to provide smooth and snag-free paths for easy insertion and removal of the instrument assemblies.

Section 4.3.2.9 – Delete first paragraph and modify second paragaph

~~The design of the reactor internals and of the water annulus between the active core and vessel wall is such that for reactor operation at the full power rating and an 80 percent capacity factor the vessel fluence greater than one MeV at the vessel wall will not exceed 3.47 x 10 19 n/cm² over the 40-year design life of the vessel. The calculated exposure includes a 10 percent uncertainty factor.~~

The maximum fast neutron fluxes greater than 1 MeV used in the determination of vessel fluence incident on the vessel ID and shroud ID are as shown in Table 4.3-8. Fluence aspects, in regards to the reactor vessel, are discussed in Section 4.3.3.3 and Section 5.2.4. The fluxes are based on a time averaged equilibrium cycle radial power distribution and an axial power distribution with a peak to average of 1.20. ~~The calculation assumed a thermal power of 2,900 MWt.~~

Section 4.3.3.3 – First paragraph

The vessel fluence was calculated based on the methods described in reference 141. These methods use the two dimensional discrete ordinates transport theory code, DORT along with cross-sections, geometry definition and other modeling techniques consistent with the requirements of Regulatory Guide 1.190 (March 2001). DORT is used to calculate the neutron flux distribution, radially and azimuthally along the core mid-plane. A separate axial flux distribution is determined as a function of radial position from the center of the core. These two distributions are combined to produce a synthesized three-dimensional flux distribution for the vessel. The fluence is determined by integrating the actual operating history using the flux distribution applicable to each reload core through the end of cycle 14. Integration ~~to 32 EFPY~~ beyond cycle 14 is performed based on conservatively extrapolated 18 month core designs. The integration assumes full power operation of 2815 MWt for cycles 1-15 with a power increase to 3026 MWt occurring at the beginning of cycle 16. The ~~resulting~~ results of the vessel fluence ~~will not exceed 3.791x10^{19} n/cm^{2} at 32 EFPY or 5.58x10^{19} n/cm^{2} at 48 EFPY (reference 142)~~ calculations are discussed in Section 5.2.4.

A.1.4 UFSAR Chapter 5 Changes

Section 5.2.1.5 – Second paragraph

The following design transients are used in the fatigue analyses required by the applicable codes. The basis for ~~the~~ each transient~~s~~ is indicated and the number of occurrences assumed is to provide a system/component design which will not be limited by expected cyclic operation over the life of the plant. ~~The number of occurrences selected for design purposes meets or exceeds the expected number.~~ The design life is not dependent on years of service but on fatigue cycles. A renewed operating license extends the license term from 40 years to 60 years for ANO-2. The actual numbers of transient occurrences were extrapolated to 60 years. The extrapolated numbers of occurrences do not exceed the numbers of occurrences selected for the original design. System integrity is further assured by using conservative methods of predicting the range of pressure and temperature for the transients. The list of transients is intended to include normal plant power range operation, controller or instrument failure, equipment malfunction, and operator errors which result in reactor trips. An explanatory discussion of each transient is also given. The applicable operating condition category as designated by the ASME Code Section III is also indicated in each case. The system and component service and test limits (operating condition category) have been established in accordance with design specification. Service limit designations Level A (normal conditions) and Level B (upset conditions) transients are based on nominal plant operating conditions exclusive of instrument uncertainties. Consistent with this, non-safety grade plant control systems (such as the Pressurizer Pressure Control System, Pressurizer Level Control System, Steam Dump & Bypass Control System, and Feedwater Control System) are assumed operational. Level C (emergency) and Level D (faulted) transients include the effect of response characteristics of the reactor protective and control systems. Additionally, the transients represent conservative estimates for design purposes only and may not be accurate representations of actual transients or actual procedures.

Section 5.2.4.1.4 – First paragraph and new Chapter 5 reference

For 32 EFPY, ~~T~~the highest predicted ~~End of Life (EOL)~~ adjusted reference temperature (ART) for the materials in the reactor vessel beltline is 113°F. This value is based on Regulatory Guide 1.99, Revision 2 and includes a shift of 101°F for an inside surface fluence of 3.791 x 10^{19} n/cm^2 at 32 EFPY. For 48 EFPY, the highest predicted adjusted reference temperature (ART) for the materials in the reactor vessel beltline is 117°F. This value is based on Regulatory Guide 1.99, Revision 2 and includes a shift of 105.1°F for an inside surface fluence of 5.277 x 10^{19} n/cm^2 at 48 EFPY. (Ref. 24) Testing results of the unirradiated baseline samples from the reactor vessel surveillance program are in Table 5.2-16.

24 Framatome-ANP Calculation 32-5024086-00, ANO-2 Reactor Vessel Adjusted RT/NDT Values at 48 EFPY, 2/13/03

Section 5.2.4.2 – Second, third and fourth paragraphs

In order to take into consideration irradiation effects, the design of the reactor internals and of the water annulus between the active core and vessel wall is such that, for reactor operation at the maximum expected output, and an 80 percent plant capacity factor, the vessel fluence (E>1 Mev) at the inner (wetted) clad surface will not exceed 3.791 x 10^{19} n/cm^2 for 32 EFPY nor 5.277 x 10^{19} n/cm^2 for 48 EFPY. The values of 32 EFPY and 48 EFPY are based on full-power operation for 40 years and 60 years, respectively, assuming an 80 percent reactor capacity factor.

The fluence analysis is discussed in section 4.3.3.3.

The reactor vessel beltline region consists of steel having controlled residual element content. ~~By proper control of~~ Based on the copper and nickel content in A-533B material, the maximum predicted ~~EOL~~ ART for the predicted fluence is calculated to be 113°F at 32 EFPY and 117°F at 48 EFPY per Regulatory Guide 1.99, Revision 2. As of May 1988, the NRC issued Regulatory Guide 1.99, Revision 2 as an acceptable approach for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light water cooled reactor vessels. Previous shift predictions were based on Figure 5.2-28 (developed from the references given in Table 5.2-10).

Section 5.2.4.3 – Last paragraph

Since the neutron spectra and flux measured at the samples and reactor vessel inside radius should be nearly identical, the measured reference transition temperature shift for a sample can be applied to the adjacent section of the reactor vessel for later stages in plant life equivalent to the difference in calculated flux magnitude. The maximum exposure of the reactor vessel will be obtained from the measured sample exposure by application of the calculated azimuthal neutron flux variation. The maximum integrated fast neutron (E > 1 Mev) exposure of the reactor vessel including tolerance is computed to be 3.791 x 10^{19} n/cm^2 for ~~40 years~~ 32 EFPY (5.277 x 10^{19} n/cm^2 for 48 EFPY) assuming operation at 2815 MWt through Cycle 15 and 3026 MWt for Cycle 16 to end of life and 80 percent load factor. The predicted RT_{NDT} shift for ~~an integrated fast neutron (E > 1 Mev) exposure of 3.791 x 10^{19} n/cm^2 is~~ 32 EFPY and 48 EFPY are ~~113°F and 117°F~~101°F and 105°F, respectively, using the methods of Regulatory Guide 1.99, Revision 2. The actual shift in RT_{NDT} will be established periodically during plant operation by testing of reactor vessel material samples which are irradiated cumulatively by securing them near the inside wall of the reactor vessel as described in Section 5.2.4.4.2 and shown in Figure 5.2-29. To compensate for any increase in RT_{NDT} caused by irradiation, limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown. During the first two years of reactor operation, conservatively high fluence of 1.8 X 1018 n/cm2 is assumed which corresponds to 2,900 MWt and an 80 percent load factor. The corresponding RT_{NDT} shift is 20°F, based on the curve shown in Figure 5.2-28. Thus, for this interval, the upper limit to the RT_{NDT} is (initial + shift) or 0°F + 20 °F = 20 °F. As of May 1988, the NRC issued Regulatory Guide 1.99, Revision 2 as a means of predicting the shift in RT_{NDT} due to irradiation embrittlement. This guide describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light water cooled reactor vessels.

Section 5.2.4.3.2 – Add new Section 5.2.4.3.2 as follows, and renumber existing sections accordingly

Section 5.2.4.3.2 48 EFPY P/T Limits

Adjusted reference temperatures (ARTs) were calculated for all ANO-2 beltline materials out to 48 EFPY following Regulatory Position 1.1 of Regulatory Guide 1.99, Revision 2. The initial material properties given in Tables 5.2-20 and 5.2-20a, and a calculated peak vessel inside surface neutron fluence of 5.277 x 10^{19} n/cm^2 (E>1MeV) were used to compute the ART at the 1/4t and 3/4t locations. The limiting (highest) ART value is 117°F at the 1/4t and 104°F at the 3/4t for plate C-8010-1 corresponding to 48 EFPY operation.

The pressure/temperature limit lines in Technical Specification (TS) Figures 3.4-2A, 3.4-2B, and 3.4-2C remain based on 32 EFPY of operation. Revised curves for extended operation will be prepared and submitted prior to reaching 32 EFPY. For license renewal, the pressure temperature curves were evaluated to 48 EFPY using the methods described in Section 5.2.4.3.1, and were acceptable for operation. See Section [A.2.2.1.3] for further discussion.

New Section 5.2.4.3.4 (Current Section 5.2.4.3.3) – Seventh, tenth, sixteenth, third from last and last paragraphs

Based on predictions of percentage decrease in C_VUSE values of beltline materials due to irradiation through ~~End of Life (EOL) at~~ 32 EFPY (1/4t fluence of 2.293E+19 n/cm 2) using Figure 2 in Reg. Guide 1.99, Revision 2, it is expected that C_VUSE values will be maintained well above the 50 ft-lbs minimum required. As shown in Tables 5.2-22a, b and c, the lowest C_VUSE value conservatively predicted for all beltline materials at ~~EOL~~ 32 EFPY is 54.6 ft-lbs (weld 2-203 A) and 54 ft-lbs (weld 2-203 A) at 48 EFPY.

~~Based on an extrapolated flux incorporating power uprate conditions, the projected end-of-life (32 EFPY) peak fast fluence of the Arkansas Nuclear One Unit 2 reactor vessel beltline region clad surface is 3.791 x 10^19 n/cm^2 (E > 1.0 MeV).~~

The ART for the reactor vessel beltline region materials are calculated in accordance with Regulatory Guide 1.99, Revision 2. The ART is calculated by adding the initial RT_{NDT}, the predicted radiation-induced RT_{NDT}, and a margin term to cover the uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence, and the calculational procedures. The predicted radiation induced RT_{NDT} is calculated using the respective reactor vessel beltline materials copper and nickel contents and the neutron fluence applicable to 32 EFPY and 48 EFPY including an estimated increase in flux due to a proposed power uprate. The 1/4t and 3/4t wall locations for each beltline material are determined by adding the thickness of the cladding to the distance into the base metal at the 1/4t and 3/4t locations. The 1/4t and 3/4t ART results for the ANO-2 reactor vessel beltline region materials applicable to 32 EFPY are presented in the Framatome Report BAW-2405 (Ref. 22). The 1/4t and 3/4t ART results for the ANO-2 reactor vessel beltline region materials applicable to 48 EFPY were calculated to support the ANO-2 license renewal. Based on these results, the controlling beltline material for the ANO-2 reactor vessel is the lower shell plate C-8010-1. The applicability of 32 EFPY is also consistent with the removal schedule for the next capsule as shown in ANO-2 SAR Table 5.2-12.

The dosimetry of the W-104 capsule was located in the reactor for a total irradiation time of 5726 effective full power days (EFPDs) for cycles 1-14. The rated thermal power for the fourteen cycles was 2815 MWt. The fluence on the center of the capsule must be estimated in order to allow for analysis of the Charpy and tensile specimens. As a result, the entire cycle 1-14 analysis results in a maximum capsule fluence of 2.937 X 10^{19} n/cm 2 . Based on an extrapolated flux incorporating power uprate conditions, the projected ~~end-of-life (~~32 EFPY~~)~~ peak fast fluence of the ANO-2 reactor vessel beltline region clad surface was determined to be 3.791 X 10^{19} n/cm^2 .

A pressurized thermal shock (PTS) evaluation for the ANO-2 reactor vessel beltline materials was performed for 32 EFPY in accordance with 10CFR50.61. The results of the PTS evaluation are shown in Table 4-1 of the BAW-2405 report (ref. 22). These results demonstrate that the ANO-2 reactor vessel beltline materials will not exceed the PTS screening criteria before 32 EFPY. The controlling beltline material for the ANO-2 reactor vessel with respect to PTS is the lower shell plate C-8010-1, with a RT_{PTS} value of 118.8 °F that is well below the PTS screening criterion of 270 °F. The PTS evaluation for ANO-2 was extended to 48 EFPY to support the ANO-2 license renewal. The 48 EFPY results demonstrate that the ANO-2 reactor vessel beltline materials will not exceed the PTS screening criteria before 48 EFPY. The controlling beltline material for the ANO-2 reactor vessel with respect to PTS remains the lower shell plate C-8010-1, with a RT_{PTS} value of 122.6°F that is well below the PTS screening criterion of 270°F.

Section 5.3.6 – Second and third paragraphs

Concerns have been expressed regarding postulated pressurized thermal shock transients involving rapid overcooling of the reactor vessel with subsequent repressurization (NUREG-0737, Item II.K.2.13). The Unit 2 reactor vessel is a later generation vessel with controlled residual element weld and plate material.

For such vessels, ~~previous~~ analysis indicates that no operator action is required to mitigate the consequences of a steam line break over 40 years of plant operation. This issue was resolved as documented in the Safety Evaluation Report issued June 5, 1984 (0CNA068413). Pressurized thermal shock of the ANO-2 reactor vessel has been evaluated for license renewal. All beltline material was demonstrated to be below the 10CFR61 screening criteria at 48 EFPY. Details of reactor vessel irradiation and EOL properties are in Section 5.2.4.

Section 5.5.2.3.2 –First paragraph

Several corrosion mechanisms have been identified in operating nuclear power plant steam generators that can result in unacceptable tube degradation. The ANO-2 steam generator design addresses these degradation mechanisms and provides a design that is very resistant to tube corrosion. Given that appropriate water chemistry is maintained, the steam generators are designed for a cumulative operating service of 40 years. The ANO-2 replacement steam generators (RSGs) were installed in 2000, thus their 40 year design life extends to 2040. License renewal extends the ANO-2 operating license to 2038, prior to the end of the design life of the RSGs.

Table 5.2-12 – Add new table footnote

Table 5.2-12

CAPSULE REMOVAL SCHEDULE

Specimen	Removal Interval
1.	1.69 EFPY (End of Cycle 2)
2.	~15.7 EFPY (End of Cycle 14)
3.	30 EFPY
4.	Standby [a]
5.	Standby [a]
6.	Standby [a]

Prior to changing removal intervals, NRC approval is required per 10CFR50, Appendix H, "Reactor Vessel Surveillance Specimen Withdrawal Schedules."

(a) If required, Capsules 4, 5, or 6 will be repositioned to ensure that peak fluence is obtained prior to 60 years.

Table 5.2-22 – Renumber the current two copies of Table 5.2-22 as 5.2-22a and 5.2-22b. Add the following new Table 5.2-22-c.

Table 5.2-22-c
Predictions of Minimum Charpy Upper Shelf Energy (CvUSE) for RPV Beltline Materials to 48 EFPY
(ref SAR Section 5.2.4.5.3)

RPV Beltline Material	Material ID	% Cu (note 1)	48 EFPY 1/4t Fluence(n/cm^2) (note 1)	Unirradiated CvUSE Value (ft-lbs) (note 1)	%Decrease Predicted in CvUSE (notes 1, 2)	Predicted 48 EFPY CvUSE Value (ft-lbs) (notes 1, 2)
Intermediate Shell Plate	C-8009-1	0.098	3.188E+19	95	24.7	71
	C-8009-2	0.085	3.188E+19	92	23.0	71
	C-8009-3	0.096	3.188E+19	87	24.5	66
Lower Shell Plate	C-8010-1	0.085	3.192E+19	90	23.0	69
	C-8010-2	0.083	3.192E+19	94	22.8	72
	C-8010-3	0.080	3.192E+19	98	22.4	76
Intermediate Shell Longitudinal Weld	2-203 A	0.046	3.015E+19	71	24.2	54
	2-203 B	0.046	2.327E+19	71	22.7	55
	2-203 C	0.046	2.327E+19	71	22.7	55
Lower Shell Longitudinal Weld	3-203 A	0.046	3.020E+19	79	24.2	60
	3-203 B	0.046	2.331E+19	79	22.7	61
	3-203 C	0.046	2.331E+19	79	22.7	61
Intermediiate to Lower Shell Horizontal (Girth) Weld	9-203	0.045	3.187E+19	95	24.3	72

Notes:

1. From Framatome-ANP Calculation 32-5024494-01, “Charpy Upper-Shelf Energy for ANO-2”, July, 2003 (Ref. 27)
2. The measured percent decrease in CvUSE for the surveillance base metal plate and weld metal are within reasonable agreement with the values predicted using Regulatory Guide 1.99, Revision 2.

A.1.5 UFSAR Chapter 9 Changes

Table 9.3-8 – Heading of second column on each of three pages of table

Cycles ~~in 40 Years~~

A.2 NEW UFSAR SECTION

The following information will be integrated into the UFSAR to document aging management programs and activities credited in the ANO-2 license renewal review and time-limited aging analyses evaluated for the period of extended operation.

A.2.0 Supplement for Renewed Operating License

EOI prepared a license renewal application for Arkansas Nuclear One, Unit 2 (Reference A.2.3.1). The application and information provided in additional correspondence provided sufficient basis for the NRC to make the findings required by 10CFR54.29 (Final Safety Evaluation Report) (Reference A.2.3.2). As required by 10CFR54.21(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 that will be required during the period of extended operation.

A.2.1.1 ALLOY 600 AGING MANAGEMENT PROGRAM

This program will manage aging effects of alloy 600/690 items and alloy 52/152 and 82/182 welds in the reactor coolant system that are not addressed by the Reactor Vessel Head Penetration Inspection Program, Section A.2.1.21, and the Steam Generator Integrity Program, Section A.2.1.26. This program will detect cracking from primary water stress corrosion cracking (PWSCC) by using the examination and inspection requirements specified in ASME Section XI. The Alloy 600 Aging Management Program will be initiated prior to the period of extended operation.

A.2.1.2 BOLTING AND TORQUING ACTIVITIES

The Bolting and Torquing Activities Program manages the loss of mechanical closure integrity for bolted connections and bolted closures in high temperature systems and in applications subject to significant vibration. The program relies on recommendations for a comprehensive bolting integrity program, as delineated in the Electric Power Research Institute EPRI NP-5067, Good Bolting Practices. This program also relies on industry recommendations for comprehensive bolting maintenance, as delineated in the EPRI TR-104213, Bolted Joint Maintenance & Applications Guide.

A.2.1.3 BORIC ACID CORROSION PREVENTION PROGRAM

The Boric Acid Corrosion Prevention Program relies on implementation of recommendations of NRC Generic Letter (GL) 88-05 to monitor the condition of ferritic steel and electrical components on which borated water may leak. The program will detect borated water leakage by periodic visual inspection of borated water containing systems for deposits of boric acid crystals and the presence of moisture. This program will manage loss of material, loss of mechanical closure integrity, and corrosion of connector surfaces.

A.2.1.4 BURIED PIPING INSPECTION PROGRAM

The Buried Piping Inspection Program will include preventive measures to mitigate corrosion and periodic inspection to manage the effects of corrosion on buried carbon steel piping. Preventive measures will be in accordance with standard industry practice for maintaining external coatings and wrappings. Buried pipes will be inspected when they are excavated during maintenance. The Buried Piping Inspection Program will be initiated prior to the period of extended operation.

A.2.1.5 CAST AUSTENITIC STAINLESS STEEL (CASS) EVALUATION PROGRAM

The Cast Austenitic Stainless Steel (CASS) Evaluation Program will augment the inspection of reactor coolant system components in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI. The CASS Evaluation Program will manage the effects of loss of fracture toughness in reactor coolant system CASS components susceptible to thermal aging embrittlement using additional inspections and a component-specific flaw tolerance evaluation. This program will not include reactor vessel internals CASS components which are evaluated and inspected as part of the Reactor Vessel Internals Program (Section A.2.1.23 and Section A.2.1.24). The CASS Evaluation Program will be initiated prior to the period of extended operation.

A.2.1.6 CONTAINMENT LEAK RATE PROGRAM

As described in 10CFR Part 50, Appendix J, containment leak rate tests are required to assure that (a) leakage through the primary reactor containment and systems and components penetrating primary containment do not exceed allowable values 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 the containment. This program manages loss of material and cracking for equipment constituting the containment pressure boundary.

A.2.1.7 DIESEL FUEL MONITORING PROGRAM

The Diesel Fuel Monitoring Program ensures that adequate diesel fuel quality is maintained to prevent plugging of filters, fouling of injectors, and corrosion of fuel systems. This program manages aging effects on the internal surfaces of diesel fuel tanks and piping within the scope of license renewal. The program monitors fuel oil quality and the levels of water and microbiological organisms in the fuel oil. Visual inspections of tanks drained for cleaning ensures that significant degradation is not occurring. This program manages the loss of material and cracking for fuel oil system components.

A.2.1.8 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIC COMPONENTS PROGRAM

The EQ Program manages component thermal, radiation, and cyclical aging of electrical equipment important to safety as required by 10CFR50.49. The EQ Program manages aging effects through the use of aging evaluations based on 10 CFR50.49(f) qualification methods. As required by 10CFR50.49, EQ components not qualified for the license term are to be refurbished, replaced, or have their qualification extended prior to reaching aging limits.

A.2.1.9 FATIGUE MONITORING PROGRAM

The Fatigue Monitoring Program tracks the number of critical thermal and pressure transients for selected reactor coolant system (RCS) components in order not to exceed the design limit on fatigue usage. The program ensures the validity of analyses containing explicit cycle count assumptions. The components managed by this program are those shown to be acceptable by analyses that explicitly addressed thermal and pressure fatigue transient limits.

A.2.1.10 FIRE PROTECTION PROGRAM

The Fire Protection Program includes fire barrier inspections and a diesel-driven fire pump inspection. The fire barrier inspections entail periodic visual inspection of fire barrier penetration seals, fire barrier walls, ceilings, and floors, and periodic visual inspection and functional tests of fire rated doors. The diesel-driven fire pump inspection requires that the pump be periodically tested to ensure that the fuel supply line can perform its intended function.

A.2.1.11 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, water storage tanks, and aboveground and underground piping and components that are tested in accordance with the applicable National Fire Protection Association (NFPA) codes and standards. Such testing assures the minimum functionality of the systems. 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. A sample of sprinkler heads will be inspected using the guidance of NFPA 25.

A.2.1.12 FLOW-ACCELERATED CORROSION PROGRAM

The Flow-Accelerated Corrosion Program manages loss of material due to flow-accelerated corrosion. This program includes (a) an analysis to determine critical locations, (b) limited baseline inspections to determine the extent of thinning at these locations, and (c) follow-up inspections to confirm the predictions, or component repair or replacement as necessary.

A.2.1.13 HEAT EXCHANGER MONITORING PROGRAM

The Heat Exchanger Monitoring Program will manage loss of material and cracking, as applicable, on heat exchangers in various systems. The Heat Exchanger Monitoring Program will inspect heat exchangers for degradation using non-destructive examinations, such as eddy-current inspections and visual inspections. If degradation is found, then an evaluation will be performed to determine its effects on the heat exchanger's design functions. The Heat Exchanger Monitoring Program will be initiated prior to the period of extended operation.

A.2.1.14 INSERVICE INSPECTION - CONTAINMENT INSERVICE INSPECTION PROGRAM

The Containment Inservice Inspection Program implements the applicable requirements of ASME Section XI, Subsections IWE and IWL as modified by 10 CFR50.55a. Every 10 years the containment inservice inspection program for ANO-2 is updated to the latest ASME Section XI code edition and addendum approved by the Nuclear Regulatory Commission in 10CFR50.

A.2.1.15 INSERVICE INSPECTION - INSERVICE INSPECTION PROGRAM

The Inservice Inspection Program implements the applicable requirements of ASME Section XI, Subsections IWB, IWC, IWD and IWF, and other requirements specified in 10CFR50.55a with approved NRC alternatives and relief requests,. Every 10 years the Inservice Inspection Program for ANO-2 is updated to the latest ASME Section XI code edition and addendum approved by the Nuclear Regulatory Commission in 10CFR50.

A.2.1.16 NON-EQ INACCESSIBLE MEDIUM-VOLTAGE CABLE PROGRAM

The Non-EQ Inaccessible Medium-voltage Cable Program will apply to inaccessible (e.g., in conduit or direct buried) medium-voltage cables within the scope of license renewal that are exposed to significant moisture simultaneously with applied voltage. In this aging management program, periodic actions will be taken to prevent cables from being exposed to significant moisture. In-scope medium-voltage cables exposed to significant moisture and voltage will be tested 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. The Non-EQ Inaccessible Medium-voltage Cable Program will be initiated prior to the period of extended operation.

A.2.1.17 NON-EQ INSULATED CABLES AND CONNECTIONS PROGRAM

The Non-EQ Insulated Cables and Connections Program will apply to accessible (i.e., able to be approached and viewed easily) insulated cables and connections installed in structures within the scope of license renewal and prone to adverse localized environments. An adverse localized environment is significantly more severe than the specified service condition for the insulated cable or connection. The program will visually inspect a representative sample of accessible insulated cables and connections for cable and connection jacket surface anomalies. The Non-EQ Insulated Cables and Connections Program will be initiated prior to the period of extended operation.

A.2.1.18 OIL ANALYSIS PROGRAM

The Oil Analysis Program ensures the oil environment in mechanical systems in the scope of license renewal is maintained to the required quality. By monitoring oil quality, 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.

A.2.1.19 PERIODIC SURVEILLANCE AND PREVENTIVE MAINTENANCE PROGRAM

The Periodic Surveillance and Preventive Maintenance Program consists of periodic inspections and tests that are relied on to manage aging effects that are not managed by other aging management programs. Preventive maintenance and surveillance testing activities provide for periodic component inspections and testing to detect various aging effects applicable to those components included in the program for license renewal.

A.2.1.20 PRESSURIZER EXAMINATIONS PROGRAM

The Pressurizer Examinations Program will use volumetric examinations required by ASME Section XI to manage cracking of the stainless steel and nickel-based alloy cladding and attachment welds to the cladding which may propagate into the underlying ferritic steel. Volumetric examination of the circumferential shell to head weld and the weld metal between the surge nozzle and the vessel lower head will be performed each inspection interval. The Pressurizer Examinations Program will be implemented prior to the period of extended operation.

A.2.1.21 REACTOR VESSEL HEAD PENETRATION PROGRAM

The Reactor Vessel Head Penetration Program manages cracking of nickel based alloy reactor vessel head penetrations exposed to borated water to assure that the pressure boundary function is maintained. The program consists of both visual and volumetric examinations in accordance with NRC Order EA-03-009. In addition, the program includes ANO-2 commitments in response to NRC Generic Letter 97-01. The program will be modified as appropriate to implement evolving commitments in response to industry experience and regulatory requirements. The Inservice Inspection (Section A.2.1.15) and Water Chemistry Control Programs (Section A.2.1.33) are used in conjunction with this program to manage cracking of the reactor vessel head penetrations.

A.2.1.22 REACTOR VESSEL INTEGRITY PROGRAM

The Reactor Vessel Integrity Program manages reduction of fracture toughness of reactor vessel beltline materials to assure that the pressure boundary function of the reactor vessel is maintained. The program is based on ASTM E-185-82, “Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels,” and includes an evaluation of radiation damage based on pre-irradiation and post irradiation testing of Charpy V-notch and tensile specimens. Through the Reactor Vessel Integrity Program, reports are submitted as required by 10CFR Part 50 Appendix H.

A.2.1.23 REACTOR VESSEL INTERNALS CAST AUSTENITIC STAINLESS STEEL (CASS) PROGRAM

The Reactor Vessel Internals Cast Austenitic Stainless Steel (CASS) Program will manage aging effects of cast austenitic stainless steel reactor vessel internals components. This program will supplement the reactor vessel internals inspections required by the ASME Section XI Inservice Inspection Program. The program will manage cracking, reduction of fracture toughness, and dimensional changes using inspections of applicable components which will be determined based on the neutron fluence and thermal embrittlement susceptibility of the component. The Reactor Vessel Internals Cast Austenitic Stainless Steel Program will be initiated prior to the period of extended operation.

A.2.1.24 REACTOR VESSEL INTERNALS STAINLESS STEEL PLATES, FORGINGS, WELDS, AND BOLTING PROGRAM

The Reactor Vessel Internals Stainless Steel Plates, Forgings, Welds, and Bolting Program will manage aging effects of reactor vessel internals plates, forgings, welds, and bolting. This program will supplement the reactor vessel internals inspections required by the ASME Section XI Inservice Inspection Program. This program will manage the effects of crack initiation and growth due to stress corrosion cracking or irradiation assisted stress corrosion cracking, loss of fracture toughness due to neutron irradiation embrittlement, and distortion due to void swelling. This program will provide visual inspections and non-destructive examinations of reactor vessel internals. The Reactor Vessel Internals Stainless Steel Plates, Forgings, Welds, and Bolting Program will be initiated prior to the period of extended operation.

A.2.1.25 SERVICE WATER INTEGRITY PROGRAM

The Service Water Integrity Program relies on implementation of the recommendations of NRC Generic Letter (GL) 89-13 to ensure that the effects of aging on the service water (SW) system will be managed. The program includes surveillance and control techniques to manage aging effects in the SW system or structures and components serviced by the SW system.

A.2.1.26 STEAM GENERATOR INTEGRITY PROGRAM

In the industry, steam generator tubes have experienced degradation related to corrosion phenomena, such as primary water stress corrosion cracking, outside diameter stress corrosion cracking, intergranular attack, pitting, and wastage, along with other mechanically induced phenomena, such as denting, wear, impingement damage, and fatigue. The Steam Generator Integrity Program uses nondestructive examination techniques to identify tubes that are defective and need to be removed from service or repaired in accordance with the guidelines of the Technical Specifications.

A.2.1.27 STRUCTURES MONITORING - MASONRY WALL PROGRAM

The Masonry Wall Program will manage cracking of masonry walls within the scope of license renewal. Masonry walls are visually inspected as part of the Structures Monitoring Program conducted for the maintenance rule, 10CFR50.65.

A.2.1.28 STRUCTURES MONITORING - STRUCTURES MONITORING PROGRAM

Structures monitoring as required by 10CFR50.65 (the maintenance rule) is based on the guidance in NRC Regulatory Guide (RG) 1.160, Rev. 2, and NUMARC 93-01, Rev. 2. These two documents provide guidance for development of licensee-specific programs to monitor the condition of structures and structural components within the scope of the maintenance rule and the scope of license renewal, such that there is no loss of structure or structural component intended function.

A.2.1.29 SYSTEM WALKDOWN PROGRAM

The System Walkdown Program will conduct inspections to manage loss of material, loss of mechanical closure integrity and cracking, as applicable, for systems and components within the scope of license renewal. The program will use general visual inspections of readily accessible system and component surfaces during system walkdowns.

A.2.1.30 WALL THINNING MONITORING PROGRAM

The Wall Thinning Monitoring Program will manage loss of material from components, as applicable, within the scope of license renewal. Inspections will be performed to ensure wall thickness is above the minimum required in order to avoid failures. The Wall Thinning Monitoring Program will be initiated prior to the period of extended operation.

A.2.1.31 WATER CHEMISTRY CONTROL - AUXILIARY SYSTEMS WATER CHEMISTRY CONTROL PROGRAM

The Auxiliary Systems Water Chemistry Control program manages loss of material, cracking, and fouling, as applicable, of components in the scope of license renewal. The program monitors and controls the relevant chemistry conditions for components exposed to treated water environments.

A.2.1.32 WATER CHEMISTRY CONTROL - CLOSED COOLING WATER CHEMISTRY CONTROL PROGRAM

The Closed Cooling Water Chemistry Control Program includes preventive measures that manage loss of material, cracking, and fouling, as applicable, for component cooling water system components. These chemistry activities provide for monitoring and controlling component cooling water chemistry using procedures and processes based on EPRI TR-107396, "Closed Cooling Water Chemistry Guidelines."

A.2.1.33 WATER CHEMISTRY CONTROL - PRIMARY AND SECONDARY WATER CHEMISTRY CONTROL PROGRAM

The Primary and Secondary Water Chemistry Control Program manages loss of material, cracking, and fouling, as applicable, by control of contaminants. This water chemistry program relies on monitoring and control of water chemistry based on EPRI guidelines for primary water chemistry and for secondary water chemistry.

A.2.2 EVALUATION OF TIME-LIMITED AGING ANALYSES

In accordance with 10CFR54.21(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

Three analyses that address the effects of neutron irradiation embrittlement of the reactor vessel have been identified as TLAA. These analyses address:

- Charpy upper-shelf energy,
- pressurized thermal shock, and
- pressure-temperature (P-T) limits.

The analyses were updated to 48 effective full power years (EFPY) which represents the approximate end of the period of extended operation (60 years) assuming a capacity factor of 80%. The Reactor Vessel Integrity Program described in Section A.2.1.22 will ensure that the time-dependent parameters used in these TLAA remain valid through the period of extended operation. The reactor vessel neutron embrittlement TLAA are projected to the end of the period of extended operation in accordance with 10CFR54.21 (c)(1)(ii).

A.2.2.1.1 Charpy Upper Shelf Energy

Appendix G of 10CFR50 requires that reactor vessel beltline materials maintain a Charpy upper shelf energy (C_VUSE) of no less than 50 ft-lb throughout the life of the vessel. The C_VUSE values were calculated using Regulatory Guide 1.99, Revision 2, Positions 1 and 2. The C_VUSE is maintained above 50 ft-lb for all base metal (plates and forgings) and welds at 48 EFPY. Therefore, the analysis of upper shelf energy has been projected to the end of the period of extended operation in accordance with 10CFR54.21 (c)(1)(ii).

A.2.2.1.2 Pressurized Thermal Shock

10CFR50.61(b)(1) provides rules for the protection of pressurized water reactors against pressurized thermal shock. The projected values of reference temperature for pressurized thermal shock, RT_{PTS}, are required to be assessed upon request for a change in the expiration date for the facility operating license.

10CFR50.61(b)(2) establishes screening criteria for RT_{PTS}: 270°F for plates, forgings, and axial welds and 300°F for circumferential welds. Projected values for RT_{PTS} at 48 EFPY for ANO-2 were calculated using Regulatory Guide 1.99, Revision 2, Positions 1 and 2, and are all within the established screening criteria for 48 EFPY. Therefore, calculations for RT_{PTS} have been projected to the end of the period of extended operation in accordance with 10 CFR 54.21 (c)(1)(ii).

A.2.2.1.3 Pressure-Temperature Limits

Appendix G of 10CFR50 requires operation of the reactor pressure vessel within established pressure-temperature (P-T) limits. ANO-2 submitted, and the NRC approved, a license amendment request for reactor coolant system pressure-temperature curves for 32 EFPY (Reference A.2.3.6, A.2.3.7). The curves specify limits on RCS pressure and temperature for up to 32 effective full power years with a 7.5% power uprate. These P-T curves are based on a fluence analysis methodology that complies with Regulatory Guide 1.190 and utilize ASME Code Cases N-640 and N-588. The ANO-2 P-T limit analyses have been extended to 48 EFPY and the operating window at 48 EFPY is sufficient to conduct normal heatup and cooldown operations. Therefore, pressure-temperature limits for ANO-2 have been projected to the end of the period of extended operation in accordance with 10CFR54.21 (c)(1)(ii).

A.2.2.2 METAL FATIGUE

The design analysis of metal fatigue is a TLAA for Class 1 and selected non-Class 1 mechanical components within the scope of license renewal. Industry experience and new research have found additional fatigue issues, such as thermal stratification and environmentally-assisted fatigue, that were not considered in the original plant design.

A.2.2.2.1 Class 1 Metal Fatigue

Fatigue evaluations performed in the design of the Class 1 RCS components were based on a number of design cycles assumed for the life of the plant. The RCS design transients used in the fatigue evaluations for the Class 1 components were reviewed for ANO-2. The numbers of actual RCS design transients from plant operating history were extrapolated to 60 years of operation. In all instances, the number of RCS design transients assumed in the original design was greater than the extrapolated number for 60 years of operation. Therefore, the fatigue evaluations for the Class 1 components remain valid for the period of extended operation in accordance with 54.21(c)(1)(i). The RCS design transients are monitored through the Fatigue Monitoring Program, which is discussed in Section A.2.1.9.

A.2.2.2.2 Non-Class 1 Metal Fatigue

Piping components that may have normal or upset condition operating temperature in excess of 220°F for carbon steel, or 270°F for austenitic stainless steel, were evaluated for fatigue. The piping components were evaluated for their potential in 60 years of plant operation to exceed the limiting number of equivalent full temperature cycles used for the original design. Fatigue considerations for the original piping and component design are valid for the period of extended operation.

A.2.2.2.3 Environmentally-Assisted Fatigue

Recent test data indicate that certain environmental effects (such as temperature, oxygen, and stress) in the primary systems of light water reactors (LWR) could result in greater susceptibility to fatigue than would be predicted by fatigue analyses based on the ASME Section III design fatigue curves. The NRC has concluded that although not safety significant through the end of the initial license term, the environmental effects associated with fatigue life should be addressed for license renewal.

The effects of environmentally-assisted thermal fatigue for the limiting locations identified in NUREG-6260 were evaluated for ANO-2 in accordance with 54.21 c(1)(i and ii) and all locations are acceptable for the period of extended operation with the exception of the charging nozzle, shutdown cooling line, and pressurizer surge line. The approach for addressing environmental fatigue for the above locations will include one or more of the following:

(1) Further refinement of the fatigue analysis to lower the CUFs to below 1.0, or

(2) Repair of the affected locations, or

(3) Replacement of the affected locations, or

(4) Manage the effects of fatigue of the locations by an inspection program that has been reviewed and approved by the NRC (for example, periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC). The inspections are expected to be able to detect cracking due to thermal fatigue prior to loss of function. Replacement or repair will then be implemented such that the intended function will be maintained for the period of extended operation, or

(5) Monitor ASME Code activities to use the environmental fatigue methodology approved by the code committee and NRC.

Should ANO-2 select Option 4 (inspection) to manage environmentally-assisted fatigue during the period of extended operation, details such as scope, qualification, method, and frequency will be provided to the NRC prior to entering the period of extended operation.

The effects of environmental-assisted thermal fatigue for the limiting locations identified in NUREG-6260 have been evaluated for ANO-2 in accordance with

10CFR54.21(c)(1)(i and ii) and all locations are acceptable for the period of extended operation with the exception of the charging nozzle, shutdown cooling line, and pressurizer surge line. Cracking by environmentally-assisted fatigue of these locations is addressed using one of the five approaches previously discussed in accordance with 10CFR54.21(c)(1).

A.2.2.2.4 Thermal Stresses in Piping Connected to Reactor Coolant Systems

ANO-2 provided responses to NRC Bulletin 88-08, Thermal Stresses in Piping Connected to Reactor Coolant Systems, in Reference A.2.3.8 and A.2.3.9. A review of 39 reactor coolant system (RCS) connected systems determined that none of these lines were determined to be subject to temperature distributions which would result in unacceptable thermal stresses. Supplement 3 to NRC Bulletin 88-08 was addressed for ANO-2 via a Combustion Engineering Owners Group (CEOG) task which revealed that only 3 piping systems (safety injection, shutdown cooling, and hot leg injection) could possibly be subject to excessive stresses due to outleakage from the RCS. Additional evaluations of these 3 systems determined that due to ANO-2's specific configuration and existing instrumentation, these systems will either not be subject to the outleakage type stratification as described in Supplement 3, or there is sufficient instrumentation and precautions in place to detect outleakage.

Subsequently, commitments regarding inspections at ANO-2 in response to NRC Bulletin 88-08 have been superseded by the ANO-2 risk-informed inspection (RI-ISI) of ASME Class 1 piping, as approved by the NRC (Reference A.2.3.10, A.2.3.11, A.2.3.12 and A.2.3.13). Aging effects due to thermal stratification as described in Bulletin 88-08 will be managed in accordance with 10CFR54.21(c)(1)(iii) by maintaining associated thermal fatigue calculations and augmented inspections (as part of RI-ISI) through the period of extended operation.

A.2.2.2.5 Pressurizer Surge Line Thermal Stratification

The ANO-2 response to NRC Bulletin 88-11, Pressurizer Surge Line Thermal Stratification (Reference A.2.3.14 and A.2.3.15), included analyses by the CEOG that demonstrated that the bounding surge line and nozzles for Combustion Engineering plants met the ASME Code stress and fatigue requirements for a 40-year design life considering thermal stratification and thermal striping. ANO-2 verified the applicability of the CEOG report and the plant-specific stress and fatigue analysis was completed as required by NRC Bulletin 88-11. Visual inspections of the pressurizer surge line were also performed.

Subsequently, commitments regarding inspections at ANO-2 in response to NRC Bulletin 88-11 have been superseded by the ANO-2 risk-informed inspection (RI-ISI) of ASME Class 1 piping, as approved by the NRC (Reference A.2.3.10, A.2.3.11, A.2.3.12 and A.2.3.13). The RI-ISI program meets the requirements of 10CFR54.21(c)(1)(iii). Aging effects due to thermal stratification as described in Bulletin 88-11 will be managed by maintaining associated thermal fatigue calculations and augmented inspections (as part of RI-ISI) through the period of extended operation.

A.2.2.3 ENVIRONMENTAL QUALIFICATION OF ELECTRICAL COMPONENTS

The ANO-2 Environmental Qualification (EQ) of Electrical Components Program, discussed in Section A.2.1.8, manages component thermal, radiation and cyclical aging, as applicable, through the use of aging evaluations based on the qualification methods of 10CFR50.49(f). Aging evaluations for EQ components that specify a qualification of at least 40 years are considered TLAA for license renewal. The EQ Program ensures that the qualification of these EQ components will be maintained. The effects of aging will thus be managed in accordance with 54.21(c)(1)(iii).

A.2.2.4 CONCRETE CONTAINMENT TENDON PRESTRESS

The analysis of loss of prestress in the containment building post-tensioning system is a time-limited aging analysis. The final effective prestress at the end of 40 years for typical dome, vertical, and hoop tendons was calculated at the time of initial design and following steam generator replacement activities. Calculation of the acceptability of the effective prestress of the containment building post-tensioning system at 60 years has been performed to show that the containment building tendon elements will be acceptable for the period of extended operation in accordance with 54.21(c)(1)(ii). Additionally, loss of tendon prestress in the containment building post-tensioning system will be managed for license renewal in accordance with 54.21(c)(1)(iii), by the Containment Inservice Inspection Program. This program, discussed in Section A.2.1.14, includes tendon surveillance testing.

A.2.2.5 CONTAINMENT LINER PLATE AND PENETRATION FATIGUE ANALYSES

The interior surface of the containment is lined with welded carbon steel plate to provide an essentially leak tight barrier. At the penetrations, the containment liner plate is thickened to reduce stress concentrations. The criteria in SAR Sections 3.8.1.3.4 and 3.8.1.6.3 were applied to the containment design to ensure that the integrity of the liner plate is not exceeded under design basis accident conditions. The evaluation of this issue for license renewal is based on an analytical assessment of the containment liner and penetrations as described in SAR Section 3.8.1.4.2 and the results of recently completed containment liner plate evaluations for ANO-2. TLAA for the ANO-2 reactor containment structure include containment liner and containment penetration fatigue analyses.

Mechanical penetrations are leak-tight, welded assemblies. As described in SAR Section 3.8.1.4.2, containment penetrations are designed to meet the requirements of ASME Section III. The evaluation for mechanical penetrations covers the penetration assembly and the weld to the process piping, but does not include the process piping within the penetration. The closure of the pipe to the liner plate is accomplished with special heads welded to the pipe and the liner plate reinforcement. Penetration anchorage to the containment wall is designed to resist pipe rupture, seismic and thermal loads.

Liner plate stress analyses indicate a conservative maximum stress of approximately 30 ksi for worst case (DBA) conditions. Stresses from normal operating cycles such as heatup and cooldown are less than 30 ksi. Using ASME section III, Division 1 design fatigue curve, at 30 ksi the maximum cycles for the liner would be approximately 25,000. The number of normal operating cycles for the liner plate will be well below this value. On this basis, the liner plate and penetrations are suitable for the cyclic loads of normal operating conditions throughout the period of extended operation.

For license renewal, containment liner plate and penetration fatigue analyses remain valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

A.2.2.6 OTHER PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES

Other ANO-2-specific TLAA include leak-before-break (LBB) analyses, fracture mechanics evaluation of the RCP casing, steam generator flow-induced vibration analysis, alloy 600 nozzle repairs and fatigue evaluations for high energy line break analyses.

A.2.2.6.1 RCS Piping Leak-Before-Break

The leak-before-break analyses reported in CEN-367-A (Reference A.2.3.16) are TLAA since they are based on the 40-year design limits for reactor coolant system fatigue transient cycles. As described for Class 1 metal fatigue in Section A.2.2.2.1, the assumed number of RCS design transients are acceptable for 60 years so these analyses will remain valid during the period of extended operation in accordance with 54.21(c)(1)(i).

A.2.2.6.2 RCP-Code Case N-481

Compliance of the primary loop pump casings to ASME Code Case N-481 was evaluated for ANO-2. The evaluation is based on the 40-year design limits for reactor coolant system fatigue transient cycles, and is therefore a TLAA. As described for Class 1 metal fatigue in Section A.2.2.2.1, the assumed number of RCS design transients are acceptable for 60 years so the Code Case N-481 evaluation will remain valid during the period of extended operation in accordance with 54.21(c)(1)(i).

A.2.2.6.3 Steam Generator Tubes – Flow-Induced Vibration

The TLAA applicable to the steam generators is the analysis of steam generator tube flow induced vibration (FIV). The time dependent assumptions made within the FIV calculation are based on a forty-year design life of the steam generators. The ANO-2 replacement generators were installed in 2000 and their design life, which extends to 2040, surpasses the period of extended operation. Therefore, the steam generator FIV analysis remains valid for the period of extended operation.

A.2.2.6.4 Alloy 600 Nozzle Repairs

In 2000, repairs were made to a number of pressurizer heater penetrations, and resistance temperature detector (RTD) and pressure measurement nozzle penetrations on the RCS hot leg. The repair for the pressurizer heater penetration replaced the pressure boundary weld on the inside surface of the pressurizer nozzle with a weld on the outside of the pressurizer. The hot leg piping penetration modification consisted of removing a portion of the old RTD/pressure tap by cutting it near the outer wall of the RCS piping and replacing it with a new nozzle welded on the outside surface of the RCS piping.

A fracture mechanics evaluation was performed to evaluate the potential for a crack in the remaining pressurizer and RCS hot leg penetration welds to propagate in the pressurizer vessel or hot leg pipe wall. The crack growth evaluations utilized operating transient cycles which were assumed over a 40 year plant lifetime. The replacement nozzles and attachment welds were qualified for structural adequacy in accordance with ASME code criteria. This analysis included a simplified fatigue evaluation which considered cyclic loads due to pressure, thermal gradients, and mechanical loads. As described for Class 1 metal fatigue in Section A.2.2.2.1, the 40-year design limits for reactor coolant system transients are acceptable for 60 years. The fatigue crack growth analysis for the repairs will remain valid during the period of extended operation in accordance with 10CFR54.21(c)(1)(i). Similarly, the fatigue analysis for the replacement nozzles and attachment welds remains valid for the period of extended operation.

A.2.2.6.5 High Energy Line Break Analyses

For the high energy line break analyses, the selection of break locations for ASME Section III Class 1 piping was in part based on piping fatigue analyses. These fatigue evaluations, and thus the high energy line break analyses, are TLAA since they are based on the 40-year design transients limits. As described for Class 1 metal fatigue in Section A.2.2.2.1, the assumed number of RCS design transients are acceptable for 60 years so these analyses will remain valid during the period of extended operation in accordance with 54.21(c)(1)(i).

A.2.3 REFERENCES

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A.2.3.2 (NRC SER for ANO-2 License Renewal - later)

A.2.3.3 NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*, U.S. Nuclear Regulatory Commission, July 2001.

A.2.3.4 NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, U.S. Nuclear Regulatory Commission, July 2001.

A.2.3.5 NUREG-1743, *Safety Evaluation Report Related to the License Renewal of Arkansas Nuclear One, Unit 1*, U.S. Nuclear Regulatory Commission, April 2001.

A.2.3.6 Anderson, Craig, Letter to US NRC (2CAN100101), Proposed Technical Specification Change Request Regarding Revised ANO-2 Pressure/Temperature and Low Temperature Overpressure Protection Limits for 32 Effective Full Power Years, October 30, 2001.

A.2.3.7 US NRC, Letter to Craig G. Anderson (2CNA040205), Arkansas Nuclear One, Unit 2 – Issuance of Amendment Re: Reactor Vessel Pressure Temperature Limits and Exemption from the Requirements of 10CFR Part 50, Section 50.60(a), April 15, 2002.

A.2.3.8 Letter from Dan R. Howard to US NRC (0CAN108806), “Arkansas Nuclear One – Units 1 & 2, Docket Nos. 50-313 and 50-368, License Nos. DPR-51 and NPF-6, NRC Bulletin No. 88-08: Thermal Stresses in Piping Connected to Reactor Coolant Systems,” dated October 12, 1988.

A.2.3.9 Letter from James J. Fisicaro to US NRC (0CAN019102), “Arkansas Nuclear One – Units 1 & 2, Docket Nos. 50-313 and 50-368, License Nos. DPR-51 and NPF-6, NRC Bulletin No. 88-08: Thermal Stresses in Piping Connected to Reactor Coolant Systems,” dated January 31, 1991.

A.2.3.10 Letter from Dwight C. Mims to US NRC (2CAN099706), “Arkansas Nuclear One – Unit 2, Docket 50-368, License No. NPF-6, Risk-Informed Inservice Inspection Pilot Plant Submittal for ANO-2,” dated September 30, 1997.

A.2.3.11 Letter from Jimmy D. Vandergrift to US NRC (2CAN109801), “Arkansas Nuclear One – Unit 2, Docket 50-368, License No. NPF-6, Additional Information in Support of the Risk-Informed Inservice Inspection Pilot Application”, dated October 8, 1998.

A.2.3.12 Letter from Jimmy D. Vandergrift to US NRC (2CAN119804), “Arkansas Nuclear One – Unit 2, Docket 50-368, License No. NPF-6, Information to Support Risk-Informed Inservice Inspection Pilot Application”, dated November 25, 1998.

A.2.3.13 Letter from John N. Hannon (NRC) to C. Randy Hutchinson (EOI) (2CNA129805), “Request to Use Risk Informed Alternative to the Requirements of ASME Code Section XI, Table IWX-2500 at Arkansas Nuclear One, Unit No. 2 (TAC No. M99756),” dated December 29, 1998.

A.2.3.14 Combustion Engineering Report CEN-387-P, *Pressurizer Surge Line Flow Stratification Evaluation*, Revision 1.

A.2.3.15 Letter from Thomas W. Alexion (NRC) to Jerry W. Yelverton (EOI) (2CNA079307), "Safety Evaluation for Combustion Engineering Owners Group Report CEN-387-P, Revision 1, Pressurizer Surge Line Thermal Stratification Evaluation (NRC Bulletin 88-11) (TAC No. M72109)", dated July 23, 1993.

A.2.3.16 Calculation 90-R-2006-04 (CEN-367-A), Leak-Before-Break Evaluation of Primary Coolant Piping, May 1992.