APPENDIX A

UPDATED FINAL SAFETY ANALYSIS REPORT SUPPLEMENT

TABLE OF CONTENTS

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 Indian Point Energy Center (IPEC) License Renewal Application (LRA). The LRA contains the technical information required by 10 CFR 54.21(a) and (c). Appendix B of the IPEC 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 both IP2 and IP3 Updated Final Safety Analysis Reports (UFSAR) Supplement information in this appendix.

This appendix is divided into three sections. The first section identifies changes to the existing sections of each UFSAR related to license renewal. The next two sections provide new information to be incorporated into the Unit 2 UFSAR and Unit 3 UFSAR. The information presented in each section will be incorporated into the respective UFSAR following issuance of the renewed operating licenses. Upon inclusion of the UFSAR Supplement in each 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 Indian Point Energy Center Unit 2 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 IP2 UFSAR Chapter 1 Changes

Section 1.1 Introduction

(7th paragraph)

~~The reactor is presently licensed to operate until September 28, 2013.~~ The reactor power plus 12.4 MWt of energy from the reactor coolant pumps give a total rated heat input to the NSSS of 3126.6 MW, which corresponds to a design electric output from the turbine-generator of approximately 1034.1 MW.

A.1.2 IP2 UFSAR Chapter 4 Changes

Section 4.1.5 – Cyclic Loads

(1st paragraph)

During unit startup and shutdown, the rates of temperature and pressure changes are limited as indicated in Section 4.4.1. The cycles are estimated for equipment design purposes ~~(40-year life)~~ and are not intended to be an accurate representation of actual transients or actual operating experience. For example, the number of cycles for plant heatup and cooldown at 100°F per hr was selected as a conservative estimate based on an evaluation of the expected requirements. The resulting number~~, which averages five heatup and cooldown cycles per year,~~ could be increased significantly; however, it is the intent to represent a conservative realistic number rather than the maximum allowed by the design.

Section 4.1.6 – Service Life

(5th paragraph)

To establish the service life of the reactor coolant system components as required by the ASME (Section III) Boiler and Pressure Vessel Code for Class "A" vessels, the unit operating conditions have been established for the ~~40-year design~~licensed life. These operating conditions include the cyclic application of pressure loadings and thermal transients.

Table 4.1-8 – Thermal and Loading Cycles

Transient Condition

Cycles[1]	Design
1. Plant heatup at 100°F per hr	200[2] ~~(5/yr2)~~
2. Plant cooldown at 100°F per hr	200 ~~(5/yr)~~
3. Plant loading at 5-percent of full power per min	14,500 ~~(1/day)~~
4. Plant unloading at 5-percent of full power per min	14,500 ~~(1/day)~~
5. Step load increase of 10-percent of full power but not to exceed full power)	2000 ~~(1/wk)~~
6. Step load decrease of 10-percent of full power	2000 ~~(1/wk)~~
7. Step load decrease of 50-percent of full power	200 ~~(5/yr)~~
8. Reactor trip	400 ~~(10/yr)~~
9. Hydrostatic test at 3110 psig pressure	5 (preoperational)
10. Hydrostatic test at 2485 psig pressure and 400°F temperature	5 (postoperational)

11. Steady state fluctuations — the reactor coolant average temperature for purposes of design is assumed to increase and decrease a maximum of 6°F in one minute. The corresponding reactor coolant pressure variation is less than 100 psig. ~~It is assumed that an infinite number of such fluctuations will occur.~~

Notes:

1. Estimated for equipment design purposes ~~(40-yr life)~~ and not intended to be an accurate representation of actual transients, or to reflect actual operating experience.
2. This transient includes pressurizing to 2235 psig.

Section 4.2.2.5 – Reactor Coolant Pump Flywheel Integrity

(8th paragraph, bullet 3)

A fracture mechanics evaluation (WCAP-15666-A) was made on the reactor coolant pump flywheel. This evaluation justifies operation of the RCPs with flywheel inspections at least every 20 years.~~This evaluation considered the following assumptions~~:

~~1. Maximum tangential stress at an assumed overspeed of 125 percent compared with a maximum expected overspeed of 109 percent.~~

~~2. A through crack through the thickness of the flywheel at the bore.~~

~~3. 400 cycles of startup operation in 40 years.~~

~~Using critical stress intensity factors and crack growth data attained on flywheel material, the critical crack size for failure was greater than 17 in. radially and the crack growth data was 0.030 in. to 0.060 in. per 1000 cycles.~~

Section 4.2.5 – Materials of Construction

(6th paragraph)

The reactor vessel was fabricated by Combustion Engineering, Inc. A sketch of the reactor vessel showing all materials in the beltline region is shown in Figure 4.2-11. Information on each of the welds and plates in the beltline region is shown in Tables 4.2-2 through 4.2-5, and Tables 4.2-5 through 4.2-8, respectively. Information relative to weld and plate material included in the material surveillance program is shown in Tables 4.2-2 and 4.2-6 through 4.2-8. ~~Details concerning the reactor vessel radiation surveillance program are provided in WCAP-7323 (Reference 7) and in the Technical Specifications.~~

(13th paragraph)

The maximum integrated fast neutron (E > 1 MeV) exposure of the vessel for 32 EFPYs is calculated to be 1.39 x 10^{19} n/cm2 based on the measurements from the fourth surveillance Capsule V. Fast neutron fluences corresponding to 32 EFPYs at various reactor vessel thicknesses are given in Table 4.2-10.

For the extended period of operation (60 years), the maximum integrated fast neutron (E > 1 MeV) exposure of the vessel for 48 EFPYs is calculated to be 1.906 x 10^{19} n/cm^2 based on calculations using Regulatory Guide 1.99 methodology. Fast neutron fluences corresponding to 48 EFPYs at various reactor vessel thicknesses are given in Table 4.2-10.

(14th paragraph)

The calculated neutron exposure exceeds the value of 0.85 x 10^{19} n/cm^2 (E > 1MeV) reported in the First Supplement to the Preliminary Facility Description Safety Analysis Report. The reasons for the increase are:

1. Anticipated increase in reactor power from 2758 MWt to 3071.4 MWt in Cycle 10 and subsequently to 3114.4 MWt in Cycle 16.
2. Revision of analysis methodology including upgrading of neutron cross sections and codes.
3. Core design considerations involving changes in loading patterns.
4. Extended period of operation.

(16th paragraph)

The maximum reference temperature, RT_{NDT} for the Indian Point Unit 2 vessel core beltline materials at the 1/4 thickness and the 3/4 thickness after 32 effective full power years of operation are projected to be 240°F and 194°F, respectively, based on calculations performed per Regulatory Guide 1.99, Revision 2, using data obtained from evaluation of Surveillance Capsule V. (Ref.11). This data provides the basis for subsequent calculation of Adjusted Reference Temperature values for determination of allowable pressure/temperature limits for operation to 25 EFPY, as described in Reference 19.

For the extended period of operation, the maximum reference temperature RT_{NDT} for the Indian Point Unit 2 vessel core beltline materials at the 1/4 thickness and the 3/4 thickness after 48 effective full power years of operation is projected to be 238.3°F and 174.8°F, respectively (at circumferential weld 9-042), based on calculations performed per Regulatory Guide 1.99, Revision 2.

(18th paragraph)

The reference nil ductility transition temperatures for pressurized thermal shock evaluation (RT_{PTS}) have been estimated[11,12,13] in accordance with 10 CFR 50.61(b)(2). The values at 15 EFPY and also at the end of the license term are well below the screening criteria of 270°F (for plates and axial weld materials) and 300°F (for circumferential weld materials), based on a low-leakage core design. The NRC has accepted this analysis.[14] Additional information in response to Generic Letter 92-01, Revision 1, is given in reference 21.

The projected RT_{PTS} values at 48 EFPY are within the established screening criteria of 270°F (for plates and axial weld materials) and 300°F (for circumferential weld materials), based on calculations performed per Regulatory Guide 1.99, Revision 2.

Table 4.2-5 – Maximum ~~End-of-Life~~ 32 EFPY Fluence at Vessel Inner Wall Locations

Table 4.2-10 – Reactor Vessel Beltline Fluence

	Fast Neutron Fluence (>1 MeV) 32 Effective Full Power Years (n/cm^2)[1]
Reactor vessel Interior surface	1.39×10^{19}
1/4 vessel thickness (1/4 T)	9.04×10^{18}
3/4 vessel thickness (3/4 T)	3.48×10^{18}

Notes:
1. These values are calculated based upon experimental results from the measurements on the fourth surveillance capsule V. See Reference 11.

	Fast Neutron Fluence (>1 MeV) 48 Effective Full Power Years (n/cm^2)	
	Vessel Plates and Circumferential Welds (45° azimuthal position)[2]	Axial Welds (30° azimuthal position)[2]
Interior surface	1.906×10^{19}	1.295×10^{19}
1/4 vessel thickness (1/4 T)	1.136×10^{19}	7.72×10^{18}
3/4 vessel thickness (3/4 T)	4.04×10^{18}	2.74×10^{18}

Notes:
2. The 30° fluences are used to calculate embrittlement parameters for the beltline axial welds because these welds are at azimuthal locations of 0, 15, and 30 degrees. The 45° fluences are used for all other reactor vessel beltline components.

Section 4.3.1.1 – Reactor Vessel

(3rd paragraph)

The cycles specified for the fatigue analysis are the results of an evaluation of the expected plant operation coupled with experience from nuclear power plants now in service. ~~These cycles include five heatup and cooldown cycles per year, a conservative selection considering that the vessel may not complete more than one cycle per year during normal operation.~~

Section 4.5.2 – Reactor Vessel – Reactor Vessel Surveillance Program

(5th paragraph)

The following is a list of the surveillance program capsules along with the actual (past) and anticipated (future) withdrawal schedule based on the latest fluence and embrittlement calculations performed in accordance with the requirements of Regulatory Guide 1.99, Revision 2 (WCAP-15629).

Capsule	Location	Lead Factor	Withdrawal Date
T	320°	3.42	End of Cycle 1
Y	220°	3.48	End of Cycle 2
Z	40°	3.53	End of Cycle 5
V	4°	1.18	End of Cycle 8
S	140°	3.5	Retired in Place
U*	176°	1.2	Determined by Reactor Vessel Surveillance Program~~End of Cycle 19~~
W*	184°	1.2	Determined by Reactor Vessel Surveillance Program~~End of Life~~
X*	356°	1.2	Determined by Reactor Vessel Surveillance Program~~Spare~~

*The withdrawal schedule of these capsules is interchangeable due to common materials and lead factors.

A.1.3 IP2 UFSAR Chapter 7 Changes

Section 7.1.3.2 – Category 2 Valves

(3rd paragraph, Item 4)

4. Heat aging of motor. Heat aging at 180°C for 100 hr ~~(equivalent to 40-year life)~~ was performed. Comparison of insulation resistance between new and aged motor indicated no significant insulation degradation.

Section 7.1.3.3.3 – Cable and Splice Test

(4th paragraph)

Westinghouse performed additional testing on 18 cable and cable splice test specimens. The testing consisted of the following:

1. Thermal aging ~~to an equivalent of 40 years of operation~~. (Kerite cable - 150°C for 192 hr; silicone cable - 210°C for 30 days.)

A.1.4 IP2 UFSAR Chapter 9 Changes

Table 9.2-2 – Chemical and Volume Control System Performance Requirements

~~Plant design life, years~~	~~40~~
Seal water return flow rate, gpm	12

This section identifies changes to existing sections of the Indian Point Energy Center Unit 3 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.1.5 IP3 UFSAR Chapter 1 Changes

Section 1.3.2, Design Criterion 15 (Reactor Coolant System Design)

(5th paragraph)

The number of thermal and loading cycles used for design purposes ~~and the bases thereof~~ are given in Table 4.1-8. During unit startup and shutdown, the rates of temperature and pressure changes are limited as indicated in Section 4.4.1. The cycles were estimated for equipment design purposes ~~(40 year life)~~ and are not intended to be an accurate representation of actual transients of actual operating experience.

(13th paragraph)

To establish the service life of the Reactor Coolant System components as required by Section III of the ASME Boiler and Pressure Vessel Code for Class “A” vessels, the unit operating conditions were established for the ~~40 year design~~licensed life. These operating conditions include the cyclic application of pressure loadings and thermal transients.

Section 1.3.4, Design Criterion 31 (Fracture Prevention of Reactor Coolant Pressure Boundary)

(22nd paragraph)

To establish the service life of the Reactor Coolant System components as required by the ASME (Section III) Boiler and Pressure Vessel Code for Class “A” vessels, the unit operating conditions have been established for the ~~40 year design~~licensed life. These operating conditions include the cyclic application of pressure loadings and thermal transients.

A.1.6 IP3 UFSAR Chapter 4 Changes

Section 4.1.5 – Cyclic Loads

(1st paragraph)

All components in the Reactor Coolant System were designed to withstand the effects of cyclic loads due to reactor system temperature and pressure changes. These cyclic loads are introduced by normal unit load transients, reactor trip, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes ~~and the bases thereof~~ are given in Table 4.1-8. During unit startup and shutdown, the rates of temperature and pressure changes are limited as indicated in Section 4.4.1. The cycles are estimated for equipment design purposes ~~(40 year life)~~ and are not intended to be an accurate representation of actual transients or actual operating experience.

(3rd paragraph)

Clearly it is difficult to discuss in absolute terms the transients that the plant will actually experience during the ~~40 years~~ operating life. For clarity, however, each transient condition is discussed in order to make clear the nature and basis for the various transients.

2. Unit Loading and Unloading

The unit loading and unloading cases were conservatively represented by a continuous and uniform ramp power change of 5% per minute between no load and full load. The reactor coolant temperature varies with load as prescribed by the temperature control system. The number of each operation was specified at 14,500 times ~~or once per day for the 40-year plant design life~~. In practice, the plant is operated at base load.

3. Step Increase and Decrease of 10%

(6th paragraph)

The number of each operation was specified at 2000 times ~~or 50 per year for the 40-year plant design life~~.

4. Large Step Decrease in Load

(3rd paragraph)

The number of occurrences of this transient was specified at 200 times ~~or 5 per year for the 40-year plant design life~~.

5. Reactor Trip from Full Power

(2nd paragraph)

The number of occurrences of this transient was specified at 400 times ~~or 10 per year for the 40-year plant design life~~.

Section 4.1.6 – Service Life

(5th paragraph)

To establish the service life of the Reactor Coolant System components as required by the ASME (Section III) Boiler and Pressure Vessel Code for Class “A” vessels, the unit operating conditions were established~~ for the 40-year design life~~. These operating conditions included the cyclic application of pressure loadings and thermal transients.

Table 4.1-8 – Thermal and Loading Cycles

Transient Condition	Design Cycles+**	Loading Conditions
1. Plant heatup at 100°F per hour	200 ~~(5/yr*)~~	Normal
2. Plant cooldown at 100°F per hour	200 ~~(5/year)~~	Normal
3. Plant loading at 5% of full power per minute	14,500 ~~(1/day)~~	Normal
4. Plant unloading at 5% of full power per minute	14,500 ~~1/day)~~	Normal
5. Step load increase of 10% of full power (but not to exceed full power)	2,000 ~~(50/year)~~	Normal
6. Step load decrease of 10% of full power	2,000 ~~(50/year)~~	Normal
7. Step load decrease of 50% of full power	200 ~~(5/year)~~	Normal
8. Reactor trip	400 ~~(10/year)~~	Upset
9. Hydrostatic test at 3110 psig pressure, 100 F temperature	5 (pre-operational)	Test
10. Hydrostatic test at 2485 psig pressure and 400 F temperature	200 (post-operational)	Test
11. Steady state fluctuations – the reactor coolant average temperature for purposes of design is assumed to increase and decrease a maximum of 6 F in one minute. The corresponding reactor coolant pressure variation is less than 100 psig. ~~It is assumed that an infinite number of such fluctuations will occur.~~		
12. Loss of load, without immediate turbine trip or reactor trip	80 ~~(2 /year)~~	Upset
13. Partial loss of flow, one pump only	80 ~~(2/year)~~	Upset
14. Operating Basis Earthquake (OBE)	5++	Upset
15. Design Basis Earthquake (DBE)	1++	Faulted

+ Estimated for equipment design purposes ~~(40-year life)~~ and not intended to be an accurate representation of actual transients or to reflect actual operating experience. See Section 4.1.5.

Section 4.2.5 – Materials of Construction

(10th paragraph)

The maximum integrated fast neutron exposure of the vessel inner surface was computed to be 0.922 x 10^{19} n/cm^2 at ~~license expiration, or~~ 27.1 effective full power years, at a power of 3216 MWt~~)~~. Similarly, the maximum integrated fast neutron exposure at the ¼T location was computed to be 5.5 x 10^{18} n/cm^2 ~~at EOL~~ (11). With this exposure the end-of-life ΔRT_{NDT} was originally estimated to be 170.6°F, and RT_{PTS} temperature was not expected to be over 268°F, which is below the NRC screening criteria of 270^{o}F.

The maximum integrated fast neutron exposure of the vessel inner surface was computed to be 1.560 x 10^{19} n/cm^2 for the period of extended operation, or 48 effective full power years, at a power of 3216 MWt. Similarly, the maximum integrated fast neutron exposure at the 1/4T location was computed to be 9.298 x 10^{18} n/cm^2 at EOL. With this exposure, the end-of-life 1/4T RT_{NDT} was estimated to be 255.8°F (plate B2803-3), and RT_{PTS} temperature was estimated to be 279.9°F (plate B2803-3), which exceeds the NRC screening criterion of 270^{o}F. For all other locations, the RT_{PTS} temperature is projected to be less than the NRC screening criterion. As required by 10 CFR 50.61(b)(4), a plant-specific safety analysis for plate B2803-3 will be submitted to the NRC three years prior to reaching the RT_{PTS} screening criterion. Each core redesign is evaluated to assure that leakage is less than assumed in analyses to predict the effect of neutron embrittlement.

(13th paragraph)

The maximum shift in RT_{NDT} after 20 EFPYs of operation is projected to be 230.1°F at the 1/4T and 188.8°F at the 3/4T vessel wall locations for Plate B2803-3 the controlling plate. The maximum adjusted RT_{NDT} after 48 EFPYs of operation is projected to be 255.8°F at the 1/4T and 207.9°F at the 3/4T vessel wall locations for Plate B2803-3, the controlling plate. Plate B2803-3 was also the controlling plate for the operating periods of 2 EFPYs, 9 EFPYs, 11.00 EFPYs, 13.3 EFPYs and 16.2 EFPYs.

(18th paragraph)

Using the prescribed PTS Rule methodology, RT_{PTS} values were generated for all beltline region materials of the reactor vessel as a function of ~~several~~ fluence ~~values and pertinent vessel lifetimes~~. All of the projected RT_{PTS} values ~~remain below~~ are within the NRC screening values for PTS ~~using the projected fluence exposure through the expiration date of the operating license.(7)~~ at 48 EFPY with the exception of plate

B2803-3, which has an EOL RT_{PTS} of 279.9°F and is 9.9°F above the screening criterion. As required by 10 CFR 50.61(b)(4), a plant-specific safety analysis for plate B2803-3 will be submitted to the NRC three years prior to reaching the RT_{PTS} screening criterion. At present, it is estimated that plate B2803-3 will reach the screening criterion at approximately 37 EFPY.

Section 4.3.1 – Safety Factors

(Transient Analysis, 4th paragraph)

The number of occurrences of the loss of load and the loss of flow transients was generally specified at 80 ~~two (2) for each year of plant design life~~.

(Reactor Coolant Pump Flywheel, 22nd paragraph)

A fracture mechanics evaluation (WCAP-15666-A) was made on the reactor coolant pump flywheel. This evaluation justifies operation of the RCPs with flywheel inspections at least every 20 years. ~~This evaluation considered the following assumptions:~~

~~1) Maximum tangential stress at an assumed overspeed of 125% compared with maximum expected overspeed of 109%~~

~~2) A crack through the thickness of the flywheel at the bore~~

~~3) 400 cycles of startup operation in 40 years of plant design life.~~

~~Using critical stress intensity factors and crack growth data attained on flywheel material, the critical crack size for failure was greater than 17 inches radially and the crack growth data was 0.030" to 0.060" per 1000 cycles.~~

Section 4.4.1— System Heatup and Cooldown Rates

(9th paragraph)

For the limiting reactor vessel beltline materials, the original end-of-life RT_{PTS} was estimated to be within the screening criteria of 10 CFR 50.61 for plate metal and welds (Reference 11). (Reg. Guide 1.99, Rev. 2 defines the property RT_{PTS} as an indicator of vessel embrittlement.) Refer to Section 4.2.5 for a discussion of projected RT_{PTS} values for the period of extended operation.

Section 4.5.2 - Reactor Vessel Surveillance

(14th paragraph)

The tentative schedule for removal of the capsules is as follows:

CAPSULE	REMOVAL TIME
T	Removed (1978 Refueling Outage, At the Replacement of the First Region of the Core, 1.34 EFPY*)
Y	Removed (1982 Refueling Outage, 3.13 EFPY)
Z	Removed (1987 Refueling Outage, 5.55 EFPY)
S	**
X	Removed (2~~2~~003 Refueling Outage, 15.6 EFPY)
U	~~30 Years or 25.5 EFPY, assuming an 85% capacity~~) Determined by the Reactor Vessel Surveillance Program
V	Standby

*NOTE: Effective full power years from plant startup.

**Capsule S, scheduled for removal in the 2001 outage, was found to be inaccessible due to equipment interference and has therefore been removed from the program. The schedule for specimen retrieval beyond Capsule Z was revised in 2003 in order to optimize the benefits gained from specimen analysis ~~in the latter half of plant live~~.

~~This plan was developed assuming an 85% capacity factor over plant life. Accordingly, the~~ The times for removal will be adjusted based on the Reactor Vessel Surveillance Program ~~may be extended to allow for historical capacity factor below 85%~~.

A.1.7 IP3 UFSAR Chapter 9 Changes

Table 9.2-2 – Chemical and Volume Control System Performance Requirements

~~Plant design life, years~~	~~40~~
Seal water return flow rate, gpm	12

A.1.8 IP3 UFSAR Appendix 14A Changes

Section 7.5 - Program Input and Results for Non-SCC

(3rd paragraph)

This figure presents that the probability of missile generation due to Low Cycle Fatigue after N = 250 cycles is in the magnitude of $7 \bullet 10^{-17}$ ~~(here it is assumed that the unit operates approximately 40 years with six starts per year).~~

A.2 NEW UFSAR SECTION FOR UNIT 2

The following information will be integrated into the UFSAR to document aging management programs and activities credited in the 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 <u>Supplement for Renewed Operating License</u>

The Indian Point Energy Center 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 <u>Aging Management Programs and Activities</u>

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) through 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.

IPEC 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 IPEC 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. The corrective action, confirmation process, and administrative controls of the Entergy (10 CFR Part 50, Appendix B) Quality Assurance Program are applicable to all aging management programs and activities required during the period of extended operation.

A.2.1.1 Aboveground Steel Tanks Program

The Aboveground Steel Tanks Program is an existing program that manages loss of material from external surfaces of aboveground carbon steel tanks by periodic visual inspection of external surfaces and thickness measurement of locations that are inaccessible for external visual inspection.

The Aboveground Steel Tanks Program will be enhanced to include the following.

- Revise applicable procedures to perform thickness measurements of the bottom surfaces of the condensate storage tank, city water tank, and fire water tank, once during the first 10 years of the period of extended operation.

- Revise applicable procedures to require trending of thickness measurements when material loss is detected.

Enhancements will be implemented prior to the period of extended operation.

A.2.1.2 Bolting Integrity Program

The Bolting Integrity Program is an existing program that relies on recommendations for a comprehensive bolting integrity program, as delineated in NUREG-1339, industry recommendations, and 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.

The program applies to bolting and torquing practices of safety- and nonsafety-related bolting for pressure retaining components, NSSS component supports, and structural joints. The program addresses all bolting regardless of size except reactor head closure studs, which are addressed by the Reactor Head Closure Studs Program. The program includes periodic inspection of closure bolting for signs of leakage that may be due to crack initiation, loss of preload, or loss of material due to corrosion. The program also includes preventive measures to preclude or minimize loss of preload and cracking.

The Bolting Integrity Program will be enhanced to include the following.

- Revise applicable procedures to clarify that actual yield strength is used in selecting materials for low susceptibility to SCC and to clarify the prohibition on use of lubricants containing MoS_2 for bolting.

Enhancements will be implemented prior to the period of extended operation.

A.2.1.3 Boraflex Monitoring Program

The Boraflex Monitoring Program is an existing program that assures degradation of the Boraflex panels in the spent fuel racks does not compromise the criticality analysis in support of the design of the spent fuel storage racks. The program relies on (1) areal density testing, (2) use of a predictive computer code, and (3) determination of boron loss through correlation of silica levels in spent fuel water samples to assure that the required 5% subcriticality margin is maintained. Corrective actions are initiated if the test results find that the 5% subcriticality margin cannot be maintained because of current or projected Boraflex degradation.

A.2.1.4 Boric Acid Corrosion Prevention Program

The Boric Acid Corrosion Prevention Program is an existing program that relies on implementation of recommendations of NRC Generic Letter 88-05 to monitor the condition of components on which borated reactor water may leak. The program detects boric acid leakage by periodic visual inspection of systems containing borated water for deposits of boric acid crystals and the presence of moisture; and by inspection of adjacent structures, components, and supports for evidence of leakage. This program manages loss of material and loss of circuit continuity, as applicable. The program includes provisions for evaluation when leakage is discovered by other activities. Program improvements have been made as suggested in NRC Regulatory Issue Summary 2003-013.

A.2.1.5 Buried Piping and Tanks Inspection Program

The Buried Piping and Tanks Inspection Program is a new program that 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, 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.

Prior to entering the period of extended operation, plant operating experience will be reviewed to verify that an inspection occurred within the past ten years. If an inspection did not occur, a focused inspection will be performed prior to the period of extended operation. A focused inspection will be performed within the first ten years of the period of extended operation, unless an opportunistic inspection occurs within this ten-year period.

The Buried Piping and Tanks Inspection Program will be implemented prior to the period of extended operation.

A.2.1.6 Containment Leak Rate Program

The Containment Leak Rate Program is an existing program. As described in 10 CFR Part 50, Appendix J, containment leak rate tests are required to assure that (a) leakage through reactor containment and systems and components penetrating 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 containment.

A.2.1.7 Containment Inservice Inspection (CII) Program

The Containment Inservice Inspection Program is an existing program encompassing ASME Section XI Subsection IWE and IWL requirements as modified by 10 CFR 50.55a.

Visual inspections for IWE monitor loss of material of the steel containment liner and integral attachments; containment hatches and airlocks; moisture barriers; and pressure- retaining bolting by inspecting surfaces for evidence of flaking, blistering, peeling, discoloration, and other signs of distress.

Visual inspections for IWL monitor structural concrete surfaces for evidence of leaching, erosion, voids, scaling, spalls, corrosion, cracking, exposed reinforcing steel, and detached embedment.

A.2.1.8 Diesel Fuel Monitoring Program

The Diesel Fuel Monitoring Program is an existing program that entails sampling to ensure that adequate diesel fuel quality is maintained to prevent loss of material and fouling in fuel systems. Exposure to fuel oil contaminants such as water and microbiological organisms is minimized by periodically draining and cleaning tanks and by verifying the quality of new oil before its introduction into the storage tanks. Sampling and analysis activities are in accordance with technical specifications on fuel oil purity and the guidelines of ASTM Standards D4057-95 and D975-95 (or later revisions of these standards).

Thickness measurements of storage tank bottom surfaces verify that significant degradation is not occurring.

The One-Time Inspection Program describes inspections planned to verify the effectiveness of the Diesel Fuel Monitoring Program.

The Diesel Fuel Monitoring Program will be enhanced to include the following.

- Revise applicable procedures to include cleaning and inspection of the GT1 gas turbine fuel oil storage tanks, EDG fuel oil day tanks, and SBO/Appendix R diesel generator fuel oil day tank once every ten years.

- Revise applicable procedures to include quarterly sampling and analysis of the SBO/Appendix R diesel generator fuel oil day tank and security diesel fuel oil day tank. Particulates (filterable solids), water and sediment checks will be performed on the samples. Filterable solids acceptance criterion will be $\leq$ 10mg/l. Water and sediment acceptance criterion will be $\leq$ 0.05%.

- Revise applicable procedures to include thickness measurement of the bottom surface of the EDG fuel oil storage tanks, EDG fuel oil day tanks, SBO/Appendix R diesel generator fuel oil day tank, GT1 gas turbine fuel oil storage tanks, and diesel fire pump fuel oil storage tank once every ten years.

- Revise appropriate procedures to change the GT1 gas turbine fuel oil storage tanks and diesel fire pump fuel oil storage tank analysis for water and particulates to a quarterly frequency.

- Revise applicable procedures to specify acceptance criteria for thickness measurements of the fuel oil storage tanks within the scope of the program.

Enhancements will be implemented prior to the period of extended operation.

A.2.1.9 Environmental Qualification (EQ) of Electric Components Program

The EQ of Electric Components Program is an existing program that 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 are refurbished, replaced, or their qualification is extended prior to reaching the aging limits established in the evaluations. Some aging evaluations for EQ components are considered time-limited aging analyses (TLAAs) for license renewal.

A.2.1.10 External Surfaces Monitoring Program

The External Surfaces Monitoring Program is an existing program that inspects 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 that are insulated are inspected when the external surface is exposed (i.e., during maintenance). Surfaces are inspected at frequencies to assure the effects of aging are managed such that applicable components will perform their intended function during the period of extended operation.

The External Surfaces Monitoring Program will be enhanced to include the following.

- Guidance documents will be revised to require periodic inspections of systems in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4 (a)(1) and (a)(3). Inspections shall include areas surrounding the subject systems to identify hazards to those systems. Inspections of nearby systems that could impact the subject systems will include SSCs that are in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4 (a)(2).

Enhancements will be implemented prior to the period of extended operation.

A.2.1.11 Fatigue Monitoring Program

The Fatigue Monitoring Program is an existing program that tracks the number of critical thermal and pressure transients for selected reactor coolant system components. The program ensures the validity of analyses that explicitly analyzed a specified number of fatigue transients by assuring that the actual effective number of transients does not exceed the analyzed number of transients.

The Fatigue Monitoring Program will be enhanced to include the following.

- Perform an evaluation to confirm that monitoring steady state cycles is not required or revise appropriate procedures to monitor steady state cycles. Review the number of allowed events and resolve discrepancies between reference documents and monitoring procedures.

Enhancements will be implemented prior to the period of extended operation.

A.2.1.12 Fire Protection Program

The Fire Protection Program is an existing program that includes a fire barrier inspection, an RCP oil collection system inspection, and a diesel-driven fire pump inspection. The fire barrier inspection requires 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 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.

The program also includes periodic inspection and testing of the Halon fire protection system.

The Fire Protection Program will be enhanced to include the following.

- Revise appropriate procedures to explicitly state that the diesel fire pump engine sub-systems (including the fuel supply line) shall be observed while the pump is running. Acceptance criteria will be revised to verify that the diesel engine does not exhibit signs of degradation while it is running; such as fuel oil, lube oil, coolant, or exhaust gas leakage.

- Revise appropriate procedures to specify that diesel fire pump engine carbon steel exhaust components are inspected for evidence of corrosion or cracking at least once each operating cycle.

Enhancements will be implemented prior to the period of extended operation.

A.2.1.13 Fire Water System Program

The Fire Water System Program is an existing program that manages water-based fire protection systems that consist of sprinklers, nozzles, fittings, valves, hydrants, hose stations, standpipes, piping, and components that are tested in accordance with applicable National Fire Protection Association (NFPA) codes and standards. Such testing assures functionality of systems. To determine if significant corrosion has occurred in water-based fire protection systems, periodic flushing, system performance testing and inspections are conducted. 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 loss of material due to corrosion.

A sample of sprinkler heads required for 10 CFR 50.48 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.

The Fire Water System Program will be enhanced to include the following.

- Revise applicable procedures to include inspection of hose reels for corrosion. Acceptance criteria will be revised to verify no unacceptable signs of degradation.

- A sample of sprinkler heads required for 10 CFR 50.48 will be inspected using guidance of NFPA 25 (2002 edition), Section 5.3.1.1.1 before the end of the 50-year sprinkler head service life and at 10-year intervals thereafter during the extended period of operation to ensure that signs of degradation, such as corrosion, are detected in a timely manner.

- Wall thickness evaluations of fire protection piping will be performed on system components using non-intrusive techniques (e.g., volumetric testing) to identify loss of material due to corrosion. These inspections will be performed before the end of the current operating term and at intervals thereafter during the period of extended operation. Results of the initial evaluations will be used to determine the appropriate inspection interval to ensure aging effects are identified prior to loss of intended function.

Enhancements will be implemented prior to the period of extended operation.

A.2.1.14 Flow-Accelerated Corrosion Program

The Flow-Accelerated Corrosion Program is an existing program that applies to safety-related and nonsafety-related carbon and low alloy steel components in systems containing high-energy fluids carrying two-phase or single-phase high-energy fluid $\geq$ 2% of plant operating time.

The program, based on EPRI guidelines in the Nuclear Safety Analysis Center (NSAC)-202L-R2 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.2.1.15 Flux Thimble Tube Inspection Program

The Flux Thimble Tube Inspection Program is an existing program that monitors for thinning of the flux thimble tube wall, which provides a path for the incore neutron flux monitoring system detectors and forms part of the RCS pressure boundary. Flux thimble tubes are subject to loss of material at certain locations in the reactor vessel where flow-induced fretting causes wear at discontinuities in the path from the reactor vessel instrument nozzle to the fuel assembly instrument guide tube. An NDE methodology, such as eddy current testing (ECT), or other similar inspection method is used to monitor for wear of the flux thimble tubes. This program implements the recommendations of NRC Bulletin 88-09, “Thimble Tube Thinning in Westinghouse Reactors”.

The Flux Thimble Tube Inspection Program will be enhanced to include the following.

- Revise appropriate procedures to implement comparisons to wear rates identified in WCAP-12866. Include provisions to compare data to the previous performances and perform evaluations regarding change to test frequency and scope.
- Revise appropriate procedures to specify the acceptance criteria as outlined in WCAP-12866 or other plant-specific values based on evaluation of previous test results.
- Revise appropriate procedures to direct evaluation and performance of corrective actions based on tubes that exceed or are projected to exceed the acceptance criteria.
- Stipulate in procedures that flux thimble tubes that cannot be inspected over the tube length and can not be shown by analysis to be satisfactory for continued service, must be removed from service to ensure the integrity of the reactor coolant system pressure boundary.

Enhancements will be implemented prior to the period of extended operation.

A.2.1.16 Heat Exchanger Monitoring Program

The Heat Exchanger Monitoring Program is an existing plant-specific program that inspects heat exchangers for loss of material through visual or other non-destructive examination.

Heat exchanger tubes are inspected at frequencies based on plant-specific and application-specific knowledge, as well as past history, heat exchanger operating conditions, and heat exchanger availability. Inspection frequencies may be changed based on engineering evaluation of inspection results.

The Heat Exchanger Monitoring Program will be enhanced to include the following.

- Revise applicable procedures to include the following heat exchangers in the scope of the program.

    - safety injection pump lube oil heat exchangers
    - RHR heat exchangers
    - RHR pump seal coolers
    - non-regenerative heat exchangers
    - charging pump seal water heat exchangers
    - charging pump fluid drive coolers
    - spent fuel pit heat exchangers
    - secondary system steam generator sample coolers
    - waste gas compressor heat exchangers
    - SBO/Appendix R diesel jacket water heat exchanger

- Revise appropriate procedures to perform visual inspection on heat exchangers where non-destructive examination, such as eddy current inspection, is not possible due to heat exchanger design limitations

- Revise appropriate procedures to include consideration of material-environment combination when determining sample population of heat exchangers.

- Revise appropriate procedures establishing the minimum tube wall thickness for the new heat exchangers identified in the scope of the program. Revise appropriate procedures establishing acceptance criteria for heat exchangers visually inspected to include no unacceptable signs of degradation.

Enhancements will be implemented prior to the period of extended operation.

A.2.1.17 Inservice Inspection – Inservice Inspection (ISI) Program

The ISI Program is an existing program based on ASME Section XI 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.

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

On July 1, 1994, the plant entered the third ISI interval. The ASME code edition and addenda used for the third interval is the 1989 Edition with no addenda.

The ISI Program will be enhanced to include the following.

- Revise appropriate procedures to provide periodic inspections to confirm the absence of aging effects for lubrite sliding supports used in the steam generator and reactor coolant pump support systems.

The current program ensures that the structural integrity of Class 1, 2, and 3 systems and associated supports is maintained at the level required by 10 CFR 50.55a.

A.2.1.18 Masonry Wall Program

The Masonry Wall Program is an existing program that manages 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 visual inspection of 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. Structural steel components of masonry walls are managed by the Structures Monitoring Program.

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

The Masonry Wall Program will be enhanced to include the following.

- Revise applicable procedures to specify that the IP1 intake structure is included in the program.

The enhancement will be implemented prior to the period of extended operation.

A.2.1.19 Metal-Enclosed Bus Inspection Program

The Metal-Enclosed Bus Inspection Program is an existing program that performs inspections on the following non-segregated phase bus.

- 6.9kV bus between station aux transformers and switchgear buses 1/2/3/4/5/6

- 480V bus associated with substation A

- 480V bus between emergency diesel generators and switchgear buses 2A/3A/5A/6A

Inspections are performed 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.

The Metal-Enclosed Bus Inspection Program will be enhanced to include the following.

- Revise appropriate procedures to add 480V bus associated with substation A to the scope of bus inspected.

- Revise appropriate procedures to visually inspect the external surface of MEB enclosure assemblies for no unacceptable loss of material at least once every ten years. The acceptance criterion will be no significant loss of material.

- Revise appropriate procedures to inspect bolted connections visually at least once every five years or at least once every ten years using thermography.

Enhancements will be implemented prior to the period of extended operation.

A.2.1.20 Nickel Alloy Inspection Program

The Nickel Alloy Inspection Program is an existing program that manages aging effects of Alloy 600 items and 82/182 welds in the reactor coolant system that are not addressed by the Reactor Vessel Head Penetration Inspection Program, Section A.2.1.30 or the Steam Generator Integrity Program, Section A.2.1.34. The aging effect requiring management for nickel alloys exposed to borated water at an elevated temperature is primary water stress corrosion cracking (PWSCC). The Nickel Alloy Inspection Program includes elements of the Inservice Inspection (ISI) Program, Section A.2.1.17, which specifies the nondestructive examination (NDE) techniques and acceptance criteria applied to evaluation of identified cracks, and the Boric Acid Corrosion Control Program, Section A.2.1.4. Also, the Water Chemistry Control - Primary and Secondary Program, Section A.2.1.40 maintains primary water in accordance with the Electric Power Research Institute (EPRI) guidelines to minimize the potential for crack initiation and growth.

The site will continue to implement commitments associated with (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff accepted industry guidelines.

A.2.1.21 Non-EQ Bolted Cable Connections Program

The Non-EQ Bolted Cable Connections Program is a new program which monitors for loosening of bolted connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation. It provides for one-time inspections on a sample of connections that will be completed prior to the period of extended operation. The following factors are considered for sampling: application (medium and low voltage, defined as < 35 kV), circuit loading (high loading), and location (high temperature, high humidity, vibration, etc.). The technical basis for the sample selections will be documented. If an unacceptable condition or situation is identified in the selected sample, the corrective action program will be used to evaluate additional requirements.

A.2.1.22 Non-EQ Inaccessible Medium-Voltage Cable Program

The Non-EQ Inaccessible Medium-Voltage Cable Program is a new program that entails periodic inspections for water collection in cable manholes and periodic testing of cables. In scope medium-voltage cables (cables with operating voltage from 2kV to 35kV) exposed to significant moisture and voltage will be tested at least once every ten years to provide an indication of the condition of the conductor insulation. The program includes inspections for water accumulation in manholes at least once every two years.

The Non-EQ Inaccessible Medium-Voltage Cable Program will be implemented prior to the period of extended operation.

A.2.1.23 Non-EQ Instrumentation Circuits Test Review Program

The Non-EQ Instrumentation Circuits Test Review Program is a new program that assures the intended functions of sensitive, high-voltage, low-signal cables exposed to adverse localized equipment environments caused by heat, radiation and moisture; (i.e., neutron flux monitoring instrumentation); can be maintained consistent with the current licensing basis through the period of extended operation. Most neutron flux monitoring system cables and connections are included in 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 will be performed once every ten years, with the first review occurring before the period of extended operation.

For neutron monitoring system cables that are disconnected during instrument calibrations, testing using a proven method for detecting deterioration for the insulation system (such as insulation resistance tests or time domain reflectometry) will occur at least every ten years, with the first test occurring before the period of extended operation. In accordance with the corrective action program, an engineering evaluation will be performed when test acceptance criteria are not met and corrective actions, including modified inspection frequency, will be implemented to ensure that the intended functions of the cables can be maintained consistent with the current licensing basis through the period of extended operation. This program will consider the technical information and guidance provided in NUREG/CR-5643, IEEE Std. P1205, SAND96-0344, and EPRI TR-109619.

The Non-EQ Instrumentation Circuits Test Review Program will be implemented prior to the period of extended operation.

A.2.1.24 Non-EQ Insulated Cables and Connections Program

The Non-EQ Insulated Cables and Connections Program is a new program that assures the 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 within the scope of license renewal will be visually inspected for cable and connection jacket surface anomalies such as embrittlement, discoloration, cracking or surface contamination. The technical basis for sampling will be determined using EPRI document TR-109619, “Guideline for the Management of Adverse Localized Equipment Environments.”

The Non-EQ Insulated Cables and Connections Program will be implemented prior to the period of extended operation.

A.2.1.25 Oil Analysis Program

The Oil Analysis Program is an existing program that 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.

The One-Time Inspection Program includes inspections planned to verify the effectiveness of the Oil Analysis Program.

The Oil Analysis Program will be enhanced to include the following.

- Revise appropriate procedures to sample and analyze lubricating oil used in the SBO/Appendix R diesel generator consistent with oil analysis for other site diesel generators.

- Revise appropriate procedures to sample and analyze generator seal oil and turbine hydraulic control oil (electrohydraulic fluid).

- Revise appropriate procedures to formalize preliminary oil screening for water and particulates and laboratory analyses including defined acceptance criteria for all components included in the scope of the program. The controlled documents will specify corrective actions in the event acceptance criteria are not met.

- Revise appropriate procedures to formalize trending of preliminary oil screening results as well as data provided from independent laboratories.

Enhancements will be implemented prior to the period of extended operation.

A.2.1.26 One-Time Inspection Program

The One-Time Inspection Program is a new program that includes measures to verify effectiveness of an aging management program (AMP) and confirm the absence of an aging effect. For structures and components that rely on an AMP, this program will verify effectiveness of the AMP by confirming that unacceptable degradation is not occurring and the intended function of a component will be maintained during the period of extended operation. One-time inspections may be needed to address concerns for potentially long incubation periods for certain aging effects on structures and components. There are cases where either (a) an aging effect is not expected to occur but there is insufficient data to completely rule it out, or (b) an aging effect is expected to progress very slowly. For these cases, there will be confirmation that either the aging effect is indeed not occurring, or the aging effect is occurring very slowly as not to affect the component or structure intended function. A one-time inspection of the subject component or structure is appropriate for this verification.

The elements of the 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.2.1.38, A.2.1.39, and A.2.1.40].

A one-time inspection activity is used to verify the effectiveness of the Oil Analysis Program by confirming that unacceptable loss of material and fouling are not occurring on components within systems covered by the Oil Analysis Program [Section A.2.1.25].

A one-time inspection activity is used to verify the effectiveness of the Diesel Fuel Monitoring Program by confirming that unacceptable loss of material and fouling is not occurring on components within systems covered by the Diesel Fuel Monitoring Program [Section A.2.1.8].

One-time inspection activities on the following confirm that loss of material is not occurring or is so insignificant that an aging management program is not warranted.

- internal surfaces of stainless steel drain components containing raw water (drain water)

- internal surfaces of stainless steel components in the station air containment penetration exposed to condensation
- internal surfaces of stainless steel EDG starting air components exposed to condensation
- internal surfaces of carbon steel and stainless steel components in the RCP oil collection system exposed to lube oil
- internal surfaces of auxiliary feedwater system stainless steel components exposed to treated water from the city water system
- internal surfaces of stainless steel components in the containment penetration for gas analyzers exposed to condensation
- internal surfaces of circulating water stainless steel and CASS components containing raw water
- internal surfaces of intake structure system stainless steel components containing raw water
- internal surfaces of chemical feed system stainless steel components containing treated water
- internal surfaces of city water system stainless steel and CASS components containing treated water (city water)
- internal surfaces of EDG system stainless steel components containing condensation or treated water (city water)
- internal surfaces of fresh water cooling system stainless steel components containing treated water (city water)
- internal surfaces of integrated liquid waste handling system stainless steel components containing raw water
- internal surfaces of lube oil system aluminum components containing raw water
- internal surfaces of river water service system stainless steel components containing raw water
- internal surfaces of waste disposal system stainless steel and CASS components containing raw water
- internal surfaces of water treatment plant system stainless steel components containing treated water (city water)

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

The inspection will be performed prior to the period of extended operation.

A.2.1.27 One-Time Inspection – Small Bore Piping Program

The One-Time Inspection – Small Bore Piping Program is a new program applicable to small bore ASME Code Class 1 piping less than 4 inches nominal pipe size (NPS 4), which includes pipe, fittings, and branch connections. The ASME Code does not require volumetric examination of Class 1 small bore piping. The One-Time Inspection of ASME Code Class 1 Small Bore Piping Program will manage cracking through the use of volumetric examinations.

The program will include a sample selected based on susceptibility, inspectability, dose considerations, operating experience, and limiting locations of the total population of ASME Code Class 1 small bore piping locations.

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

The inspection will be performed prior to the period of extended operation.

A.2.1.28 Periodic Surveillance and Preventive Maintenance Program

The Periodic Surveillance and Preventive Maintenance Program is an existing program that includes periodic inspections and tests that manage aging effects not managed by other aging management programs. In addition to specific activities in the plant's preventive maintenance program and surveillance program, the Periodic Surveillance and Preventive Maintenance Program includes enhancements to add new activities. The preventive maintenance and surveillance testing activities are generally implemented through repetitive tasks or routine monitoring of plant operations.

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

- reactor building cranes (polar and manipulator), crane rails, and girders, and refueling platform
- recirculation pump motor cooling coils and housing
- city water system components
- charging pump casings
- plant drain components and backwater valves
- station air containment penetration piping
- HVAC duct flexible connections
- HVAC stored portable blowers and flexible trunks
- EDG exhaust components
- EDG duct flexible connections
- EDG air intake and aftercooler components
- EDG air start components
- EDG cooling water makeup supply valves
- security generator exhaust components
- security generator radiator tubes
- SBO/Appendix R diesel exhaust components
- SBO/Appendix R diesel turbocharger and aftercooler
- SBO/Appendix R jacket water heat exchanger
- SBO/Appendix R diesel fuel oil cooler
- diesel fuel oil trailer transfer tank and associated valves
- auxiliary feedwater components

- containment cooling duct flexible connections
- containment cooling fan units internals
- control room HVAC condensers and evaporators
- control room HVAC ducts and drip pans
- control room HVAC duct flexible connections
- circulating water, city water, intake structure system, emergency diesel generator, fresh water cooling, instrument air, integrated liquid waste handling, lube oil, miscellaneous, radiation monitoring, river water, station air, waste disposal, and water treatment plant system piping, piping components, and piping elements
- pressurizer relief tank

The Periodic Surveillance and Preventive Maintenance Program will be enhanced as follows.

- Program activity guidance documents will be developed or revised as necessary to assure that the effects of aging will be managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis through the period of extended operation.

Enhancements will be implemented prior to the period of extended operation.

A.2.1.29 Reactor Head Closure Studs Program

The Reactor Head Closure Studs Program is an existing program that 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.2.1.30 Reactor Vessel Head Penetration Inspection Program

The Reactor Vessel Head Penetration Inspection Program is an existing program that manages primary water stress corrosion cracking (PWSCC) of nickel-based alloy reactor vessel head penetrations exposed to borated water to ensure that the pressure boundary function is maintained. This program was developed in response to NRC Order EA-03-009. The ASME Section XI, Subsection IWB Inservice Inspection and Water Chemistry Control Programs are used in conjunction with this program to manage cracking of the reactor vessel head penetrations. Detection of cracking is accomplished through implementation of a combination of bare metal visual examination (external surface of head) and non-visual examination (underside of head) techniques. Procedures are developed to perform reactor vessel head bare metal inspections and calculations of the susceptibility ranking of the plant.

The plant will continue to implement commitments associated with (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines.

A.2.1.31 Reactor Vessel Surveillance Program

The Reactor Vessel Surveillance Program is an existing program that manages reduction in fracture toughness of reactor vessel beltline materials to assure that the pressure boundary function of the reactor pressure vessel is maintained through the period of extended operation.

The Reactor Vessel Surveillance Program will be enhanced to include the following.

- The specimen capsule withdrawal schedules will be revised to draw and test a standby capsule to cover the peak reactor vessel fluence expected through the end of the period of extended operation.

- Revise appropriate procedures to require that tested and untested specimens from all capsules pulled from the reactor vessel are maintained in storage.

Enhancements will be implemented prior to the period of extended operation.

A.2.1.32 Selective Leaching Program

The Selective Leaching Program is a new program that ensures the integrity of components made of gray cast iron, bronze, brass, and other alloys exposed to raw water, treated water, or groundwater that may lead to selective leaching. The program includes a one-time visual inspection, hardness measurement (where feasible based on form and configuration), or other industry accepted mechanical inspection techniques 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 through the period of extended operation.

The Selective Leaching Program will be implemented prior to the period of extended operation.

A.2.1.33 Service Water Integrity Program

The Service Water Integrity Program is an existing program that relies on implementation of the recommendations of GL 89-13 to ensure that the effects of aging on the service water system are managed through the period of extended operation. The program includes component inspections for erosion, corrosion, and biofouling to verify the heat transfer capability of safety-related heat exchangers cooled by service water. Chemical treatment using biocides and sodium hypochlorite and periodic cleaning and flushing of infrequently used loops are methods used to control fouling within the heat exchangers and to manage loss of material in service water components.

A.2.1.34 Steam Generator Integrity Program

The Steam Generator Integrity Program is an existing program that uses nondestructive examination (NDE) techniques to identify tubes that are defective and need to be removed from service or repaired in accordance with the guidelines of the plant technical specifications. The program also includes processes for monitoring and maintaining secondary side component integrity. The program defines when inspections and maintenance are performed, the scope of work, and the methods used.

The Steam Generator Integrity Program will be enhanced to include the following.

- Revise appropriate procedures to require that the results of the condition monitoring assessment are compared to the operational assessment performed for the prior operating cycle with differences evaluated.

Enhancements will be implemented prior to the period of extended operation.

A.2.1.35 Structures Monitoring Program

The Structures Monitoring Program is an existing program that performs inspections 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 commodities to ensure there is no loss of intended function.

The Structures Monitoring Program will be enhanced to include the following.

- Appropriate procedures will be revised to explicitly specify that the following structures are included in the program.

  -city water storage tank foundation
  -discharge canal
  -emergency lighting poles and foundations
  -fire pumphouse
  -fire water storage tank foundation
  -gas turbine 1 fuel storage tank foundation
  -maintenance and outage building–elevated passageway
  -new station security building
  -nuclear service building (IP1)
  -service water pipe chase
  -superheater stack
  -transformer/switchyard support structures
  -waste holdup tank pit

- Appropriate procedures will be revised to clarify that in addition to structural steel and concrete, the following commodities are inspected for each structure as applicable.

  -cable trays and supports
  -concrete portion of reactor vessel supports
  -conduits and supports

-cranes, rails, and girders
-equipment pads and foundations
-fire proofing (pyrocrete)
-HVAC duct supports
-jib cranes
-manholes and duct banks
-manways, hatches, and hatch covers
-monorails
-new fuel storage racks
-sumps, sump screens, strainers and flow barriers

- Guidance will be added to the Structures Monitoring Program to inspect inaccessible concrete areas that are exposed by excavation for any reason. The site will also inspect inaccessible concrete areas in environments where observed conditions in accessible areas exposed to the same environment indicate that significant concrete degradation is occurring.

- Revise applicable structures monitoring procedures for inspection of elastomers (seals, gaskets, seismic joint filler, and roof elastomers) to identify cracking and change in material properties and for inspection of aluminum vents and louvers to identify loss of material.

- Guidance to perform an engineering evaluation of groundwater samples to assess aggressiveness of groundwater to concrete on a periodic basis (at least once every five years) will be added to the Structures Monitoring Program. The site will obtain samples from a well that is representative of the groundwater surrounding below-grade site structures. Samples will be monitored for sulfates, pH and chlorides.

Enhancements will be implemented prior to the period of extended operation.

A.2.1.36 Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program

The Thermal Aging Embrittlement of CASS Program is a new program that augments the inspection of the reactor coolant system components in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI. The inspection detects the effects of loss of fracture toughness due to thermal aging embrittlement of cast austenitic stainless steel (CASS) components. This aging management program determines the susceptibility of CASS components to thermal aging embrittlement based on casting method, molybdenum content, and percent ferrite. The program provides aging management through either enhanced volumetric examination or flaw tolerance evaluation. Additional inspection or evaluations to demonstrate that the material has adequate fracture toughness are not required for components that are not susceptible to thermal aging embrittlement.

The Thermal Aging Embrittlement of CASS Program will be implemented prior to the period of extended operation.

A.2.1.37 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program

The Thermal Aging and Neutron Irradiation Embrittlement of CASS Program is a new program that augments the reactor vessel internals visual inspection in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Subsection IWB. This inspection manages the effects of loss of fracture toughness due to thermal aging and neutron embrittlement of cast austenitic stainless steel (CASS) components. This aging management program determines the susceptibility of CASS components to thermal aging or neutron irradiation (neutron fluence) embrittlement based on casting method, molybdenum content, operating temperature and percent ferrite. For each "potentially susceptible" component, aging management is accomplished through either a component-specific evaluation or a supplemental examination of the affected component as part of the inservice inspection (ISI) program during the license renewal term.

The Thermal Aging and Neutron Irradiation Embrittlement of CASS Program will be implemented prior to the period of extended operation.

A.2.1.38 Water Chemistry Control – Auxiliary Systems Program

The Water Chemistry Control – Auxiliary Systems Program is an existing program that manages loss of material and cracking for components exposed to treated water.

Program activities include sampling and analysis to minimize component exposure to aggressive environments for the house service boiler systems and stator cooling water systems.

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 aging effects.

A.2.1.39 Water Chemistry Control – Closed Cooling Water Program

The Water Chemistry Control – Closed Cooling Water Program is an existing program that includes preventive measures that manage loss of material, cracking, or fouling for components in closed cooling water systems (component cooling water (CCW), conventional closed cooling (CCC), instrument air closed cooling (IACC), emergency diesel generator cooling, security generator cooling, and SBO/Appendix R diesel generator cooling). These chemistry activities provide for monitoring and controlling closed cooling water chemistry using 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 aging effects.

The Water Chemistry Control – Closed Cooling Water Program will be enhanced to include the following.

- Revise appropriate procedures to maintain water chemistry of the SBO/Appendix R diesel generator cooling system per EPRI guidelines.

- Revise appropriate procedures to maintain the security generator cooling water system pH within limits specified by EPRI guidelines.

Enhancements will be implemented prior to the period of extended operation.

A.2.1.40 Water Chemistry Control – Primary and Secondary

The Water Chemistry Control – Primary and Secondary Program is an existing program that manages aging effects caused by corrosion and cracking mechanisms. The program relies on monitoring and control of reactor water chemistry based on the EPRI guidelines in TR-105714 for primary water chemistry and TR-102134 for secondary 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 – Primary and Secondary Program has been effective at managing aging effects.

The Water Chemistry Control – Primary and Secondary Program will be enhanced to include the following.

- Revise appropriate procedures to test sulfates monthly in the RWST with a limit of < 150 ppb.

Enhancements will be implemented prior to the period of extended operation.

A.2.1.41 Reactor Vessel Internals Aging Management Activities

To manage loss of fracture toughness, cracking, change in dimensions (void swelling), and loss of preload in vessel internals components, the site will (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

A.2.2 Evaluation of Time-Limited Aging Analyses – Unit 2

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 current licensing basis analyses evaluating reduction of fracture toughness of the reactor vessel for 40 years are TLAA. The reactor vessel neutron embrittlement TLAA is summarized below. Forty-eight effective full-power years (EFPY) are projected for the end of the period of extended operation (60 years) based on actual capacity factors from the start of commercial operation until 2005 and an average capacity factor of 95% from 2005 to the end of the period of extended operation.

A.2.2.1.1 Reactor Vessel Fluence

As part of the stretch power uprate analysis, the neutron exposure levels for the reactor pressure vessel were projected for an operating period extending to 48 EFPY. These fluence values included peak plate and weld ID fluences. The 1/4 T fluences were derived using RG 1.99 formula and conservative wall thicknesses.

A.2.2.1.2 Pressure-Temperature Limits

Appendix G of 10 CFR 50 requires operation of the reactor pressure vessel be accomplished within established pressure-temperature (P-T) limits. These limits are established by calculations that utilize the materials and fluence data obtained through the unit specific reactor surveillance capsule program.

Technical Specifications contain pressure/temperature limits valid through 25 EFPY including the effects of power uprate.

The site will submit additional P-T curves as 10 CFR 50, Appendix G requires prior to the period of extended operation as part of the Reactor Vessel Surveillance Program.

A.2.2.1.3 Charpy Upper-Shelf Energy

The predictions for percent drop in C_VUSE at 48 EFPY are based on chemistry data, unirradiated C_VUSE data, and 1/4 T fluence values. The projected 48 EFPY peak beltline fluence level was applied to all beltline materials with the exception of axial welds. Based on surveillance data, peak fluence levels at the beltline axial welds is based on the expected fluence at the 30 degree azimuth position.

One intermediate shell plate (B2002-3) and one lower shell plate (B2003-1) have projected upper shelf energy levels that fall below 50 ft-lb during the period of extended operation. All remaining plate and weld beltline materials meet the requirement of exceed 50 ft-lb at 48 EFPY.

An equivalent margins analysis performed in WCAP-13587, Rev. 1, demonstrated that the minimum acceptable USE for reactor vessel plate material in four-loop plants is 43 ft-lbs. In the safety assessment of WCAP-13587, the NRC concluded the report demonstrated margins of safety equivalent to those of the ASME code for beltline plate and forging materials. The USE values are therefore acceptable since the lowest projected USE level for the beltline plate material through the period of extended operation of 47.4 ft-lb for intermediate shell plate B2002-3 is above the 43 ft-lbs minimum acceptable USE for four-loop plants determined in WCAP-13587 Rev. 1.

A.2.2.1.4 Pressurized Thermal Shock

10 CFR 50.61(b)(1) provides rules for protection against pressurized thermal shock events for pressurized water reactors. Licensees are required to perform an assessment of the projected values of reference temperature whenever a significant change occurs in projected values of the adjusted reference temperature for pressurized thermal shock (RT_{PTS}). The screening criteria for RT_{PTS} is 270°F for plates, forgings, and axial welds and 300°F for circumferential welds.

Adjusted reference temperatures are calculated for both Positions 1 and 2 by following the guidance in Regulatory Guide 1.99, Sections 1.1 and 2.1, respectively, using copper and nickel content of beltline materials and end-of-life (EOL) best estimate fluence projections.

All projected RT_{PTS} values are within the established screening criteria at 48 EFPY.

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), reactor vessel internals, pressurizer, steam generators, reactor coolant pumps, control rod drive mechanisms, regenerative letdown heat exchanger, and Class-1 piping and in-line components.

The Fatigue Monitoring Program will assure that the analyzed number of transient cycles is not exceeded. The program requires corrective action if the analyzed number of transient cycles is 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).

A.2.2.2.2 Non-Class 1 Metal Fatigue

For non-Class 1 piping and in-line 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, the individual stress calculations require evaluation. No systems were identified with projected cycles exceeding 7000. Therefore, the TLAA for non-Class 1 piping and in-line components remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(i).

The only non-Class1, non-piping component identified with a fatigue time-limited aging analysis was the residual heat removal heat exchanger. That heat exchanger is projected to incur less than the analyzed number of cycles and therefore the analysis will remain valid for the period of extended operation.

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. At least two years prior to entering the period of extended operation, for the locations identified in NUREG/CR-6260 for Westinghouse PWRs of this vintage, the site will implement one or more of the following:

(1) Refine the fatigue analyses to determine valid CUFs less than 1 when accounting for the effects of reactor water environment. This includes applying the appropriate F_{en} factors to valid CUFs determined in accordance with one of the following.

For locations, including NUREG/CR-6260 locations, with existing fatigue analysis valid for the period of extended operation, use the existing CUF to determine the environmentally adjusted CUF.

In addition to the NUREG/CR-6260 locations, more limiting plant-specific locations with a valid CUF may be evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component.

Representative CUF values from other plants, adjusted to or enveloping the plant-specific external loads may be used if demonstrated applicable.

An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.

(2) Manage the effects of aging due to fatigue at the affected locations by 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).

(3) Repair or replace the affected locations before exceeding a CUF of 1.0.

Should the site select the option to manage the aging effects due to environmental-assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be submitted to the NRC at least two years prior to the period of extended operation.

A.2.2.3 Environmental Qualification of Electrical Components

The 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 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 Containment Liner Plate and Penetrations Fatigue Analyses

In 1973, a feedwater line cracked circumferentially, resulting in damage to the liner plate causing containment liner plate buckling at the penetration for feedwater line #22. Studies were performed to evaluate the effects of fatigue on the deformed area of the liner due to predicted high strain-limited cycle loading during its projected life. The evaluation was based on the 40-year operating life of the plant and is thus considered a TLAA. The TLAA associated with the buckled liner adjacent to the feedwater line #22 penetration remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

There are no TLAA associated with the containment penetrations.

A.2.2.5 Leak before Break

Leak before break (LBB) analyses evaluate postulated flaw growth in piping to justify changes to the structural design basis involving protection against the effect of postulated reactor coolant loop pipe ruptures. The LBB evaluations use saturated (fully aged) fracture toughness properties, and these analyses do not have a material property time-limited assumption. The fatigue crack growth for 40 years was calculated using the design transients for the reactor vessel. As these transients will not be exceeded in 60 years, these analyses will remain valid during the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.2.2.6 Steam Generator Flow-Induced Vibration and Tube Wear

The steam generators were evaluated with respect to flow-induced vibration (tube wear). The replacement steam generators went into service in January 2000 and will have less than 40 years of service at the end of the period of extended operation (September 2033). Therefore these TLAA will remain valid through the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.2.3 References

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

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

A.3 NEW UFSAR SECTION FOR UNIT 3

The following information will be integrated into the UFSAR to document aging management programs and activities credited in the 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.3.0 Supplement for Renewed Operating License

The Indian Point Energy Center license renewal application (Reference A.3-1) 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.3-2). 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.3.1) and a description of the evaluation of time-limited aging analyses for the period of extended operation (Section A.3.2). The period of extended operation is the 20 years after the expiration date of the original operating license.

A.3.1 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) through 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.

IPEC 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 IPEC 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. The corrective action, confirmation process, and administrative controls of the Entergy (10 CFR Part 50, Appendix B) Quality Assurance Program are applicable to all aging management programs and activities required during the period of extended operation.

A.3.1.1 Aboveground Steel Tanks Program

The Aboveground Steel Tanks Program is an existing program that manages loss of material from external surfaces of aboveground carbon steel tanks by periodic visual inspection of external surfaces and thickness measurement of locations that are inaccessible for external visual inspection.

The Aboveground Steel Tanks Program will be enhanced to include the following.

- Revise applicable procedures to perform thickness measurements of the bottom surfaces of the condensate storage tank and fire water tanks, once during the first 10 years of the period of extended operation.

- Revise applicable procedures to require trending of thickness measurements when material loss is detected.

Enhancements will be implemented prior to the period of extended operation.

A.3.1.2 Bolting Integrity Program

The Bolting Integrity Program is an existing program that relies on recommendations for a comprehensive bolting integrity program, as delineated in NUREG-1339, industry recommendations, and 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.

The program applies to bolting and torquing practices of safety- and nonsafety-related bolting for pressure retaining components, NSSS component supports, and structural joints. The program addresses all bolting regardless of size except reactor head closure studs, which are addressed by the Reactor Head Closure Studs Program. The program includes periodic inspection of closure bolting for signs of leakage that may be due to crack initiation, loss of preload, or loss of material due to corrosion. The program also includes preventive measures to preclude or minimize loss of preload and cracking.

The Bolting Integrity Program will be enhanced to include the following.

- Revise applicable procedures to clarify that actual yield strength is used in selecting materials for low susceptibility to SCC and to clarify the prohibition on use of lubricants containing MoS_2 for bolting.

Enhancements will be implemented prior to the period of extended operation.

A.3.1.3 Boral Surveillance Program

The Boral Surveillance Program is an existing program that provides assurance the Boral neutron absorbers in the spent fuel racks maintain the validity of the criticality analysis in support of the rack design. The program relies on representative coupon samples mounted in surveillance assemblies located in the spent fuel pool to monitor performance of the absorber material without disrupting the integrity of the storage system.

Surveillance assemblies are removed from the spent fuel pool on a prescribed schedule and physical and chemical properties are measured. From this data, the stability and integrity of the Boral in the storage cells are assessed.

A.3.1.4 Boric Acid Corrosion Prevention Program

The Boric Acid Corrosion Prevention Program is an existing program that relies on implementation of recommendations of NRC Generic Letter 88-05 to monitor the condition of components on which borated reactor water may leak. The program detects boric acid leakage by periodic visual inspection of systems containing borated water for deposits of boric acid crystals and the presence of moisture; and by inspection of adjacent structures, components, and supports for evidence of leakage. This program manages loss of material and loss of circuit continuity, as applicable. The program includes provisions for evaluation when leakage is discovered by other activities. Program improvements have been made as suggested in NRC Regulatory Issue Summary 2003-013.

A.3.1.5 Buried Piping and Tanks Inspection Program

The Buried Piping and Tanks Inspection Program is a new program that 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, 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.

Prior to entering the period of extended operation, plant operating experience will be reviewed to verify that an inspection occurred within the past ten years. If an inspection did not occur, a focused inspection will be performed prior to the period of extended operation. A focused inspection will be performed within the first ten years of the period of extended operation, unless an opportunistic inspection occurs within this ten-year period.

The Buried Piping and Tanks Inspection Program will be implemented prior to the period of extended operation.

A.3.1.6 Containment Leak Rate Program

The Containment Leak Rate Program is an existing program. As described in 10 CFR Part 50, Appendix J, containment leak rate tests are required to assure that (a) leakage through reactor containment and systems and components penetrating 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 containment.

A.3.1.7 Containment Inservice Inspection (CII) Program

The Containment Inservice Inspection (CII) Program is an existing program encompassing ASME Section XI Subsection IWE and IWL requirements as modified by 10 CFR 50.55a.

Visual inspections for IWE monitor loss of material of the steel containment liner and integral attachments; containment hatches and airlocks; moisture barriers; and pressure-retaining bolting by inspecting surfaces for evidence of flaking, blistering, peeling, discoloration, and other signs of distress.

Visual inspections for IWL monitor structural concrete surfaces for evidence of leaching, erosion, voids, scaling, spalls, corrosion, cracking, exposed reinforcing steel, and detached embedment.

A.3.1.8 Diesel Fuel Monitoring Program

The Diesel Fuel Monitoring Program is an existing program that entails sampling to ensure that adequate diesel fuel quality is maintained to prevent loss of material and fouling in fuel systems. Exposure to fuel oil contaminants such as water and microbiological organisms is minimized by periodically draining and cleaning tanks and by verifying the quality of new oil before its introduction into the storage tanks. Sampling and analysis activities are in accordance with technical specifications on fuel oil purity and the guidelines of ASTM Standards D4057-95 and D975-95 (or later revisions of these standards).

Thickness measurements of storage tank bottom surfaces verify that significant degradation is not occurring.

The One-Time Inspection Program describes inspections planned to verify the effectiveness of the Diesel Fuel Monitoring Program.

The Diesel Fuel Monitoring Program will be enhanced to include the following.

- Revise applicable procedures to include cleaning and inspection of the EDG fuel oil day tanks, Appendix R fuel oil storage tank and Appendix R fuel oil day tank once every ten years.

- Revise applicable procedures to include quarterly sampling and analysis of the Appendix R fuel oil storage tank. Particulates (filterable solids), water and sediment checks will be performed on the samples. Filterable solids acceptance criterion will be $\leq$ 10mg/l. Water and sediment acceptance criterion will be $\leq$ 0.05%.

- Revise applicable procedures to include thickness measurement of the bottom surface of the EDG fuel oil day tanks, Appendix R fuel oil storage tank, and diesel fire pump fuel oil storage tank once every ten years.

- Revise appropriate procedures to change the Appendix R fuel oil day tank and diesel fire pump fuel oil storage tank analysis for water and particulates to a quarterly frequency.

- Revise applicable procedures to specify acceptance criteria for thickness measurements of the fuel oil storage tanks within the scope of the program.

Enhancements will be implemented prior to the period of extended operation.

A.3.1.9 Environmental Qualification (EQ) of Electric Components Program

The EQ of Electric Components Program is an existing program that 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. Some aging evaluations for EQ components are considered time-limited aging analyses (TLAAs) for license renewal.

A.3.1.10 External Surfaces Monitoring Program

The External Surfaces Monitoring Program is an existing program that inspects 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 that are insulated are inspected when the external surface is exposed (i.e., during maintenance). Surfaces are inspected at frequencies to assure the effects of aging are managed such that applicable components will perform their intended function during the period of extended operation.

The External Surfaces Monitoring Program will be enhanced to include the following.

- Guidance documents will be revised to require periodic inspections of systems in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4 (a)(1) and (a)(3). Inspections shall include areas surrounding the subject systems to identify hazards to those systems. Inspections of nearby systems that could impact the subject systems will include SSCs that are in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4 (a)(2).

Enhancements will be implemented prior to the period of extended operation.

A.3.1.11 Fatigue Monitoring Program

The Fatigue Monitoring Program is an existing program that tracks the number of critical thermal and pressure transients for selected reactor coolant system components. The program ensures the validity of analyses that explicitly analyzed a specified number of fatigue transients by assuring that the actual effective number of transients does not exceed the analyzed number of transients.

The Fatigue Monitoring Program will be enhanced to include the following.

- Revise appropriate procedures to include all the transients identified. Assure all fatigue analysis transients are included with the lowest limiting numbers. Update the number of design transients accumulated to date.

Enhancements will be implemented prior to the period of extended operation.

A.3.1.12 Fire Protection Program

The Fire Protection Program is an existing program that includes a fire barrier inspection, an RCP oil collection system inspection, and a diesel-driven fire pump inspection. The fire barrier inspection requires 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 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.

The program also includes periodic inspection and testing of the CO_2 fire protection system.

The Fire Protection Program will be enhanced to include the following.

- Revise appropriate procedures to explicitly state that the diesel fire pump engine sub-systems (including the fuel supply line) shall be observed while the pump is running. Acceptance criteria will be revised to verify that the diesel engine does not exhibit signs of degradation while it is running; such as fuel oil, lube oil, coolant, or exhaust gas leakage.

- Revise appropriate procedures to specify that diesel fire pump engine carbon steel exhaust components are inspected for evidence of corrosion or cracking at least once each operating cycle.

- Revise appropriate procedures to visually inspect the cable spreading room, 480V switchgear room, and EDG room CO_2 fire suppression system for signs of degradation, such as corrosion and mechanical damage at least once every six months.

- Revise appropriate procedures to inspect the external surfaces of the RCP oil collection system for loss of material each refueling outage.

Enhancements will be implemented prior to the period of extended operation.

A.3.1.13 Fire Water System Program

The Fire Water System Program is an existing program that manages water-based fire protection systems that consist of sprinklers, nozzles, fittings, valves, hydrants, hose stations, standpipes, and components that are tested in accordance with applicable National Fire Protection Association (NFPA) codes and standards. Such testing assures functionality of systems. To determine if significant corrosion has occurred in water-based fire protection systems, periodic flushing, system performance testing and inspections are conducted. 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 loss of material due to corrosion.

A sample of sprinkler heads required for 10 CFR 50.48 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 ten years after initial field service testing.

The Fire Water System Program will be enhanced to include the following.

- Revise applicable procedures to include inspection of hose reels for corrosion. Acceptance criteria will be revised to verify no unacceptable signs of degradation.

- Revise applicable procedures to inspect the internal surface of the foam-based fire suppression tanks. Acceptance criteria will be enhanced to verify no significant corrosion.

- A sample of sprinkler heads required for 10 CFR 50.48 will be inspected using guidance of NFPA 25 (2002 edition), Section 5.3.1.1.1 before the end of the 50-year sprinkler head service life and at 10-year intervals thereafter during the extended period of operation to ensure that signs of degradation, such as corrosion, are detected in a timely manner.

- Wall thickness evaluations of fire protection piping will be performed on system components using non-intrusive techniques (e.g., volumetric testing) to identify loss of material due to corrosion. These inspections will be performed before the end of the current operating term and at intervals thereafter during the period of extended operation. Results of the initial evaluations will be used to determine the appropriate inspection interval to ensure aging effects are identified prior to loss of intended function.

Enhancements will be implemented prior to the period of extended operation.

A.3.1.14 Flow-Accelerated Corrosion Program

The Flow-Accelerated Corrosion Program is an existing program that applies to safety-related and nonsafety-related carbon and low alloy steel components in systems containing high-energy fluids carrying two-phase or single-phase high-energy fluid more than two percent of plant operating time.

The program, based on EPRI guidelines in the Nuclear Safety Analysis Center (NSAC)-202L-R2 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.3.1.15 Flux Thimble Tube Inspection Program

The Flux Thimble Tube Inspection Program is an existing program that monitors for thinning of the flux thimble tube wall, which provides a path for the incore neutron flux monitoring system detectors and forms part of the RCS pressure boundary. Flux thimble tubes are subject to loss of material at certain locations in the reactor vessel where flow-induced fretting causes wear at discontinuities in the path from the reactor vessel instrument nozzle to the fuel assembly instrument guide tube. An NDE methodology, such as eddy current testing (ECT), or other similar inspection method is used to monitor for wear of the flux thimble tubes. This program implements the recommendations of NRC Bulletin 88-09, “Thimble Tube Thinning in Westinghouse Reactors”.

The Flux Thimble Tube Inspection Program will be enhanced to include the following.

- Revise appropriate procedures to implement comparisons to wear rates identified in WCAP-12866. Include provisions to compare data to the previous performances and perform evaluations regarding change to test frequency and scope.

- Revise appropriate procedures to specify the acceptance criteria as outlined in WCAP-12866 or other plant-specific values based on evaluation of previous test results.

- Revise appropriate procedures to direct evaluation and performance of corrective actions based on tubes that exceed or are projected to exceed the acceptance criteria.

- Stipulate in procedures that flux thimble tubes that cannot be inspected over the tube length and can not be shown by analysis to be satisfactory for continued service, must be removed from service to ensure the integrity of the reactor coolant system pressure boundary.

Enhancements will be implemented prior to the period of extended operation.

A.3.1.16 Heat Exchanger Monitoring Program

The Heat Exchanger Monitoring Program is an existing plant-specific program that inspects heat exchangers for loss of material through visual or other non-destructive examination.

Heat exchanger tubes are inspected at frequencies based on plant-specific and application-specific knowledge, as well as past history, heat exchanger operating conditions, and heat exchanger availability. Inspection frequencies may be changed based on engineering evaluation of inspection results.

The Heat Exchanger Monitoring Program will be enhanced to include the following.

- Revise applicable procedures to include the following heat exchangers in the scope of the program.

  - safety injection pump lube oil heat exchangers
  - RHR heat exchangers
  - RHR pump seal coolers
  - non-regenerative heat exchangers
  - charging pump seal water heat exchangers
  - charging pump fluid drive coolers
  - instrument air heat exchangers
  - spent fuel pit heat exchangers
  - secondary system steam generator sample coolers
  - waste gas compressor heat exchangers

- Revise appropriate procedures to perform visual inspection on heat exchangers where non-destructive examination, such as eddy current inspection, is not possible due to heat exchanger design limitations

- Revise appropriate procedures to include consideration of material-environment combination when determining sample population of heat exchangers.

- Revise appropriate procedures establishing the minimum tube wall thickness for the new heat exchangers identified in the scope of the program. Enhance appropriate procedures establishing acceptance criteria for heat exchangers visually inspected to include no unacceptable signs of degradation.

Enhancements will be implemented prior to the period of extended operation.

A.3.1.17 Inservice Inspection – Inservice Inspection (ISI) Program

The ISI Program is an existing program based on ASME Section Xi Inspection Program B (Section XI, IWA-2432), which has ten-year inspection intervals. Every ten years the program is updated to the latest ASME Section XI code edition and addendum approved in 10 CFR 50.55a.

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

On July 21, 2000, the site entered the third ISI interval. The ASME code edition and addenda used for the third interval is the 1989 Edition with no addenda.

The ISI Program will be enhanced to include the following.

- Revise appropriate procedures to periodic inspections to confirm the absence of aging effects for lubrite sliding supports used in the steam generator and reactor coolant pump support systems.

The current program ensures that the structural integrity of Class 1, 2, and 3 systems and associated supports is maintained at the level required by 10 CFR 50.55a.

A.3.1.18 Masonry Wall Program

The Masonry Wall Program is an existing program that manages 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 visual inspection of 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. Structural steel components of masonry walls are managed by the Structures Monitoring Program.

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

The Masonry Wall Program will be enhanced to include the following.

- Revise applicable procedures to specify that the IP1 intake structure is included in the program.

Enhancements will be implemented prior to the period of extended operation.

A.3.1.19 Metal-Enclosed Bus Inspection Program

The Metal-Enclosed Bus Inspection Program is an existing program that performs inspections on the following non-segregated phase bus.

- 6.9kV bus between station aux transformers and switchgear buses 1/2/3/4/5/6

- 6.9kV bus associated with the gas turbine substation

- 480V bus between emergency diesel generators and switchgear buses 2A/3A/5A/6A

Inspections are performed 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.

The Metal-Enclosed Bus Inspection Program will be enhanced to include the following.

- Revise appropriate procedures to visually inspect the external surface of MEB enclosure assemblies for no unacceptable loss of material at least once every ten years. The acceptance criterion will be no significant loss of material.

- Revise appropriate procedures to inspect bolted connections visually at least once every five years or at least once every ten years using thermography.

Enhancements will be implemented prior to the period of extended operation.

A.3.1.20 Nickel Alloy Inspection Program

The Nickel Alloy Inspection Program is an existing program that manages aging effects of Alloy 600 items and 82/182 welds in the reactor coolant system that are not addressed by the Reactor Vessel Head Penetration Inspection Program, Section A.3.1.30 or the Steam Generator Integrity Program, Section A.3.1.34. The aging effect requiring management for nickel alloys exposed to borated water at an elevated temperature is primary water stress corrosion cracking (PWSCC). The Nickel Alloy Inspection Program includes elements of the Inservice Inspection (ISI) Program, Section A.3.1.17, which specifies the nondestructive examination (NDE) techniques and acceptance criteria applied to evaluation of identified cracks, and the Boric Acid Corrosion Control Program, Section A.3.1.4. Also, the Water Chemistry Control - Primary and Secondary Program, Section A.3.1.40, maintains primary water in accordance with the Electric Power Research Institute (EPRI) guidelines to minimize the potential for crack initiation and growth.

The site will continue to implement commitments associated with (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines.

A.3.1.21 Non-EQ Bolted Cable Connections Program

The Non-EQ Bolted Cable Connections Program is a new program which monitors for loosening of bolted connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation. It provides for one-time inspections that will be completed on a sample of connections that will be completed prior to the period of extended operation. The following factors are considered for sampling: application (medium and low voltage, defined as < 35 kV), circuit loading (high loading), and location (high temperature, high humidity, vibration, etc.). The technical basis for the sample selections will be documented. If an unacceptable condition or situation is identified in the selected sample, the corrective action program will be used to evaluate additional requirements.

A.3.1.22 Non-EQ Inaccessible Medium-Voltage Cable Program

The Non-EQ Inaccessible Medium-Voltage Cable Program is a new program that entails periodic inspections for water collection in cable manholes and periodic testing of cables. In scope medium-voltage cables (cables with operating voltage from 2kV to 35kV) exposed to significant moisture and voltage will be tested at least once every ten years to provide an indication of the condition of the conductor insulation. The program includes inspections for water accumulation in manholes at least once every two years.

The Non-EQ Inaccessible Medium-Voltage Cable Program will be implemented prior to the period of extended operation.

A.3.1.23 Non-EQ Instrumentation Circuits Test Review Program

The Non-EQ Instrumentation Circuits Test Review Program is a new program that assures the intended functions of sensitive, high-voltage, low-signal cables exposed to adverse localized equipment environments caused by heat, radiation and moisture; (i.e., neutron flux monitoring instrumentation); can be maintained consistent with the current licensing basis through the period of extended operation. Most neutron flux monitoring system cables and connections are included in 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 will be performed once every ten years, with the first review occurring before the period of extended operation.

For neutron monitoring system cables that are disconnected during instrument calibrations, testing using a proven method for detecting deterioration for the insulation system (such as insulation resistance tests or time domain reflectometry) will occur at least every ten years, with the first test occurring before the period of extended operation. In accordance with the corrective action program, an engineering evaluation will be performed when test acceptance criteria are not met and corrective actions, including modified inspection frequency, will be implemented to ensure that the intended functions of the cables can be maintained consistent with the current licensing basis for the period of extended operation. This program will consider the technical information and guidance provided in NUREG/CR-5643, IEEE Std. P1205, SAND96-0344, and EPRI TR 109619.

The Non-EQ Instrumentation Circuits Test Review Program will be implemented prior to the period of extended operation.

A.3.1.24 Non-EQ Insulated Cables and Connections Program

The Non-EQ Insulated Cables and Connections Program is a new program that assures the 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 within the scope of license renewal will be visually inspected for cable and connection jacket surface anomalies such as embrittlement, discoloration, cracking or surface contamination. The technical basis for sampling will be determined using EPRI document TR-109619, “Guideline for the Management of Adverse Localized Equipment Environments.”

The Non-EQ Insulated Cables and Connections Program will be implemented prior to the period of extended operation.

A.3.1.25 Oil Analysis Program

The Oil Analysis Program is an existing program that 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.

The One-Time Inspection Program includes inspections planned to verify the effectiveness of the Oil Analysis Program.

The Oil Analysis Program will be enhanced to include the following.

- Revise appropriate procedures to sample and analyze generator seal oil and turbine hydraulic control oil (electrohydraulic fluid).

- Revise appropriate procedures to formalize preliminary oil screening for water and particulates and laboratory analyses including defined acceptance criteria for all components included in the scope of the program. The controlled documents will specify corrective actions in the event acceptance criteria are not met.

- Revise appropriate procedures to formalize trending of preliminary oil screening results as well as data provided from independent laboratories.

Enhancements will be implemented prior to the period of extended operation.

A.3.1.26 One-Time Inspection Program

The One-Time Inspection Program is a new program that includes measures to verify effectiveness of an aging management program (AMP) and confirm the absence of an aging effect. For structures and components that rely on an AMP, this program will verify effectiveness of the AMP by confirming that unacceptable degradation is not occurring and the intended function of a component will be maintained during the period of extended operation. One-time inspections may be needed to address concerns for potentially long incubation periods for certain aging effects on structures and components. There are cases where either (a) an aging effect is not expected to occur but there is insufficient data to completely rule it out, or (b) an aging effect is expected to progress very slowly. For these cases, there will be confirmation that either the aging effect is indeed not occurring, or the aging effect is occurring very slowly as not to affect the component or structure intended function. A one-time inspection of the subject component or structure is appropriate for this verification.

The elements of the 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.3.1.38, A.3.1.39, and A.3.1.40).

A one-time inspection activity is used to verify the effectiveness of the Oil Analysis Program by confirming that unacceptable loss of material and fouling are not occurring on components within systems covered by the Oil Analysis Program (Section A.3.1.25).

A one-time inspection activity is used to verify the effectiveness of the Diesel Fuel Monitoring Program by confirming that unacceptable loss of material and fouling is not occurring on components within systems covered by the Diesel Fuel Monitoring Program (Section A.3.1.8).

One-time inspection activities on the following confirm that loss of material is not occurring or is so insignificant that an aging management program is not warranted.

- internal surfaces of stainless steel drain components containing raw water (drain water)
- internal surfaces of stainless steel components in the station air containment penetration exposed to condensation
- internal surfaces of stainless steel EDG starting air components exposed to condensation
- internal surfaces of carbon steel and stainless steel components in the RCP oil collection system exposed to lube oil
- internal surfaces of auxiliary feedwater system stainless steel components exposed to treated water from the city water system
- internal surfaces of stainless steel components in the containment penetration for gas analyzers exposed to condensation
- internal surfaces of circulating water stainless steel and CASS components containing raw water
- internal surfaces of ammonia/morpholine addition system stainless steel components containing treated water
- internal surfaces of boron and layup chemical addition system stainless steel components containing treated water
- internal surfaces of city water makeup system stainless steel and CASS components containing treated water (city water)
- internal surfaces of gaseous waste disposal system CASS components containing condensation
- internal surfaces of hydrazine addition system stainless steel components containing treated water
- Internal surfaces of liquid waste disposal system stainless steel and CASS components containing raw water or treated water (city water)
- internal surfaces of nuclear equipment drain system stainless steel components containing raw water

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

The inspection will be performed prior to the period of extended operation.

A.3.1.27 One-Time Inspection – Small Bore Piping Program

The One-Time Inspection - Small Bore Piping Program is a new program applicable to small bore ASME Code Class 1 piping less than 4 inches nominal pipe size (NPS 4), which includes pipe, fittings, and branch connections. The ASME Code does not require volumetric examination of

Class 1 small bore piping. The One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program will manage cracking through the use of volumetric examinations.

The program will include a sample selected based on susceptibility, inspectability, dose considerations, operating experience, and limiting locations of the total population of ASME Code Class 1 small-bore piping locations.

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

The inspection will be performed prior to the period of extended operation.

A.3.1.28 Periodic Surveillance and Preventive Maintenance Program

The Periodic Surveillance and Preventive Maintenance Program is an existing program that includes periodic inspections and tests that manage aging effects not managed by other aging management programs. In addition to specific activities in the plant's preventive maintenance program and surveillance program, the Periodic Surveillance and Preventive Maintenance Program includes enhancements to add new activities. The preventive maintenance and surveillance testing activities are generally implemented through repetitive tasks or routine monitoring of plant operations.

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

- reactor building cranes (polar and manipulator), crane rails, and girders, and refueling platform
- containment spray system sodium hydroxide tank
- recirculation pump motor cooling coils and housing
- city water system components
- charging pump casings
- plant drain components
- station air containment penetration piping
- HVAC duct flexible connections
- HVAC stored portable blowers and flexible trunks
- EDG exhaust components
- EDG duct flexible connections
- EDG air intake and aftercooler components
- EDG air start components
- EDG cooling water makeup supply valves
- security generator exhaust components
- security generator radiator tubes
- Appendix R diesel generator exhaust components
- Appendix R diesel generator radiator

- Appendix R diesel generator aftercooler
- Appendix R diesel generator starting air components
- Appendix R diesel generator crankcase exhaust components
- diesel fuel oil trailer transfer tank and associated valves
- auxiliary feedwater components
- containment cooling duct flexible connections
- containment cooling fan units internals
- control room HVAC condensers and evaporators
- control room HVAC ducts and drip pans
- control room HVAC duct flexible connections
- chlorination, circulating water, city water makeup, emergency diesel generator, floor drain, gaseous waste disposal, instrument air, liquid waste disposal, nuclear equipment drain, river water, station air piping, steam generator sampling, and secondary plant sampling piping components, and piping elements
- pressurizer relief tank

The Periodic Surveillance and Preventive Maintenance Program will be enhanced as follows.

- Program activity guidance documents will be developed or revised as necessary to assure that the effects of aging will be managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis through the period of extended operation.

Enhancements will be implemented prior to the period of extended operation.

A.3.1.29 Reactor Head Closure Studs Program

The Reactor Head Closure Studs Program is an existing program that 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.3.1.30 Reactor Vessel Head Penetration Inspection Program

The Reactor Vessel Head Penetration Inspection Program is an existing program that manages primary water stress corrosion cracking (PWSCC) of nickel-based alloy reactor vessel head penetrations exposed to borated water to ensure that the pressure boundary function is maintained. This program was developed in response to NRC Order EA-03-009. The ASME Section XI, Subsection IWB Inservice Inspection and Water Chemistry Control Programs are used in conjunction with this program to manage cracking of the reactor vessel head penetrations. Detection of cracking is accomplished through implementation of a combination of bare metal visual examination (external surface of head) and non-visual examination (underside of head) techniques. Procedures are developed to perform reactor vessel head bare metal inspections and calculations of the susceptibility ranking of the plant.

The plant will continue to implement commitments associated with (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines.

A.3.1.31 Reactor Vessel Surveillance Program

The Reactor Vessel Surveillance Program is an existing program that manages reduction in fracture toughness of reactor vessel beltline materials to assure that the pressure boundary function of the reactor pressure vessel is maintained through the period of extended operation.

The Reactor Vessel Surveillance Program will be enhanced to include the following.

- The specimen capsule withdrawal schedules will be revised to draw and test a standby capsule to cover the peak reactor vessel fluence expected through the end of the period of extended operation.

- Appropriate procedures will be revised to require that tested and untested specimens from all capsules pulled from the reactor vessel are maintained in storage.

Enhancements will be implemented prior to the period of extended operation.

A.3.1.32 Selective Leaching Program

The Selective Leaching Program is a new program that ensures the integrity of components made of gray cast iron, bronze, brass, and other alloys exposed to raw water, treated water, or groundwater that may lead to selective leaching. The program includes a one-time visual inspection, hardness measurement (where feasible based on form and configuration), or other industry accepted mechanical inspection techniques 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 through the period of extended operation.

The Selective Leaching Program will be implemented prior to the period of extended operation.

A.3.1.33 Service Water Integrity Program

The Service Water Integrity Program is an existing program that relies on implementation of the recommendations of GL 89-13 to ensure that the effects of aging on the service water system are managed through the period of extended operation. The program includes component inspections for erosion, corrosion, and biofouling to verify the heat transfer capability of safety-related heat exchangers cooled by service water. Chemical treatment using biocides and chlorine and periodic cleaning and flushing of infrequently used loops are methods used to control fouling within the heat exchangers and to manage loss of material in service water components.

A.3.1.34 Steam Generator Integrity Program

The Steam Generator Integrity Program is an existing program that performs nondestructive examination (NDE) techniques to identify tubes that are defective and need to be removed from service or repaired in accordance with the guidelines of the plant technical specifications. The program also includes processes for monitoring and maintaining secondary side component integrity. The program defines when inspections and maintenance are performed, the scope of work, and the methods used.

The Steam Generator Integrity Program will be enhanced to include the following.

- Revise appropriate procedures to require that the results of the condition monitoring assessment are compared to the operational assessment performed for the prior operating cycle with differences evaluated.

Enhancements will be implemented prior to the period of extended operation.

A.3.1.35 Structures Monitoring Program

The Structures Monitoring Program is an existing program that performs inspections 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 commodities to ensure there is no loss of intended function.

The Structures Monitoring Program will be enhanced to include the following.

- Appropriate procedures will be revised to explicitly specify that the following structures are included in the program.

  -Appendix R emergency diesel generator foundation
  -Appendix R emergency diesel generator fuel oil tank vault
  -Appendix R emergency diesel generator switchgear and enclosure
  -city water storage tanks foundation
  -condensate storage tank foundation
  -containment access facility and annex
  -discharge canal
  -emergency lighting poles and foundations
  -fire protection pumphouse
  -fire water storage tank foundation
  -gas turbine 1 fuel storage tank foundation
  -primary water storage tank foundation
  -refueling water storage tank foundation
  -security access and office building
  -service water pipe chase

-service water valve pit
-superheater stack
-waste holdup tank pit

- Appropriate procedures will be revised to clarify that in addition to structural steel and concrete, the following commodities are inspected for each structure as applicable.

  -cable trays and supports
  -concrete portion of reactor vessel supports
  -conduits and supports
  -cranes, rails, and girders
  -equipment pads and foundations
  -fire proofing (pyrocrete)
  -HVAC duct supports
  -jib cranes
  -manholes and duct banks
  -manways, hatches, and hatch covers
  -monorails
  -new fuel storage racks
  -sumps, sump screens, strainers and flow barriers

- Guidance will be added to the Structures Monitoring Program to inspect inaccessible concrete areas that are exposed by excavation for any reason. The site will also inspect inaccessible concrete areas in environments where observed conditions in accessible areas exposed to the same environment indicate that significant concrete degradation is occurring.

- Revise applicable structures monitoring procedures for inspection of elastomers (seals, gaskets, seismic joint filler, and roof elastomers) to identify cracking and change in material properties and for inspection of aluminum vents and louvers to identify loss of material.

- Guidance to perform an engineering evaluation of groundwater samples to assess aggressiveness of groundwater to concrete on a periodic basis (at least once every five years) will be added to the Structures Monitoring Program. The site will obtain samples from a well that is representative of the ground water surrounding below-grade site structures. Samples will be monitored for sulfates, pH and chlorides.

Enhancements will be implemented prior to the period of extended operation.

A.3.1.36 Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program

The Thermal Aging Embrittlement of CASS Program is a new program that augments the inspection of the reactor coolant system components in accordance with the American Society of

Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI. The inspection detects the effects of loss of fracture toughness due to thermal aging embrittlement of cast austenitic stainless steel (CASS) components. This aging management program determines the susceptibility of CASS components to thermal aging embrittlement based on casting method, molybdenum content, and percent ferrite. The program provides aging management through either enhanced volumetric examination or flaw tolerance evaluation. Additional inspection or evaluations to demonstrate that the material has adequate fracture toughness are not required for components that are not susceptible to thermal aging embrittlement.

The Thermal Aging Embrittlement of CASS Program will be implemented prior to the period of extended operation.

A.3.1.37 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program

The Thermal Aging and Neutron Irradiation Embrittlement of CASS Program is a new program that augments the reactor vessel internals visual inspection in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Subsection IWB. This inspection manages the effects of loss of fracture toughness due to thermal aging and neutron embrittlement of cast austenitic stainless steel (CASS) components. This aging management program determines the susceptibility of CASS components to thermal aging or neutron irradiation (neutron fluence) embrittlement based on casting method, molybdenum content, operating temperature and percent ferrite. For each "potentially susceptible" component, aging management is accomplished through either a component-specific evaluation or a supplemental examination of the affected component as part of the inservice inspection (ISI) program during the license renewal term.

The Thermal Aging and Neutron Irradiation Embrittlement of CASS Program will be implemented prior to the period of extended operation.

A.3.1.38 Water Chemistry Control – Auxiliary Systems Program

The Water Chemistry Control – Auxiliary Systems Program is an existing program that manages loss of material and cracking for components exposed to treated water.

Program activities include sampling and analysis to minimize component exposure to aggressive environments for NaOH components in the containment spray system, house service boiler systems, and stator cooling water systems.

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 aging effects.

A.3.1.39 Water Chemistry Control – Closed Cooling Water Program

The Water Chemistry Control – Closed Cooling Water Program is an existing program that includes preventive measures that manage loss of material, cracking, and fouling for components in closed cooling water systems (component cooling water (CCW), instrument air closed cooling (IACC), emergency diesel generator cooling, security generator cooling, Appendix R diesel generator cooling, and turbine hall closed cooling (THCC)). These chemistry activities provide for monitoring and controlling closed cooling water chemistry using 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 aging effects.

The Water Chemistry Control – Closed Cooling Water Program will be enhanced to include the following.

- Revise appropriate procedures to maintain security generator cooling water pH within limits specified by EPRI guidelines.

Enhancements will be implemented prior to the period of extended operation

A.3.1.40 Water Chemistry Control – Primary and Secondary

The Water Chemistry Control – Primary and Secondary Program is an existing program that manages aging effects caused by corrosion and cracking mechanisms. The program relies on monitoring and control of reactor water chemistry based on the EPRI guidelines in TR-105714 for primary water chemistry and TR-102134 for secondary 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 – Primary and Secondary Program has been effective at managing aging effects.

A.3.1.41 Reactor Vessel Internals Aging Management Activities

To manage loss of fracture toughness, cracking, change in dimensions (void swelling), and loss of preload in vessel internals components, the site will (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

A.3.2 Evaluation of Time-Limited Aging Analyses – Unit 3

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.3.2.1 Reactor Vessel Neutron Embrittlement

The current licensing basis analyses evaluating reduction of fracture toughness of the reactor vessel for 40 years are TLAA. The reactor vessel neutron embrittlement TLAA is summarized below. Forty-eight effective full-power years (EFPY) are projected for the end of the period of extended operation (60 years) based on actual capacity factors from the start of commercial operation until 2005 and an average capacity factor of 95% from 2005 to the end of the period of extended operation.

A.3.2.1.1 Reactor Vessel Fluence

As part of the stretch power uprate analysis, the neutron exposure levels for the reactor pressure vessel were projected for an operating period extending to 48 EFPY. These fluence values included peak vessel ID fluences. The 1/4 T fluence was derived using RG 1.99 formula and conservative wall thicknesses.

A.3.2.1.2 Pressure-Temperature Limits

Appendix G of 10 CFR 50 requires operation of the reactor pressure vessel be accomplished within established pressure-temperature (P-T) limits. These limits are established by calculations that utilize the materials and fluence data obtained through the unit specific reactor surveillance capsule program.

Technical Specifications contain pressure/temperature limits valid through 34 EFPY including the effects of power uprate.

The site will submit additional P-T curves as 10 CFR 50, Appendix G requires prior to the period of extended operation as part of the Reactor Vessel Surveillance Program.

A.3.2.1.3 Charpy Upper-Shelf Energy

The predictions for percent drop in C_VUSE at 48 EFPY are based on chemistry data, unirradiated C_VUSE data, and 1/4 T fluence values. The projected 48 EFPY peak beltline fluence level was conservatively applied to all beltline materials. All plate and weld beltline materials meet the requirement of exceeding a C_VUSE value of 50 ft-lb at 48 EFPY.

A.3.2.1.4 Pressurized Thermal Shock

10 CFR 50.61(b)(1) provides rules for protection against pressurized thermal shock events for pressurized water reactors. Licensees are required to perform an assessment of the projected values of reference temperature whenever a significant change occurs in projected values of the adjusted reference temperature for pressurized thermal shock (RT_{PTS}). The screening criteria for RT_{PTS} is 270°F for plates, forgings, and axial welds and 300°F for circumferential welds.

Adjusted reference temperatures are calculated for both Positions 1 and 2 by following the guidance in Regulatory Guide 1.99, Sections 1.1 and 2.1, respectively, using copper and nickel content of beltline materials and end-of-life (EOL) best estimate fluence projections.

All projected RT_{PTS} values are within the established screening criteria for 48 EFPY with the exception of plate B2803-3, which exceeds the screening criterion by 9.9°F.

As required by 10 CFR 50.61(b)(4), a plant-specific safety analysis for plate B2803-3 will be submitted to the NRC three years prior to reaching the RT_{PTS} screening criterion. Alternatively, the site may choose to implement the revised PTS (10 CFR 50.61) rule when approved, which would permit use of Regulatory Guide 1.99, Revision 3. Application of Regulatory Guide 1.99, Revision 3, to plate B2803-3 is expected to result in an acceptably low probability of a through-wall crack at 48 EFPY.

A.3.2.2 Metal Fatigue

A.3.2.2.1 Class 1 Metal Fatigue

Class 1 components evaluated for fatigue and flaw growth include the reactor pressure vessel (RPV), reactor vessel internals, pressurizer, steam generators, reactor coolant pumps, control rod drive mechanisms, regenerative letdown heat exchanger, and Class-1 piping and in-line components.

The Fatigue Monitoring Program will assure that the analyzed number of transient cycles is not exceeded. The program requires corrective action if the analyzed number of transient cycles is 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).

A.3.2.2.2 Non-Class 1 Metal Fatigue

For non-Class 1 piping and in-line 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, the individual stress calculations require evaluation. No systems were identified with projected cycles exceeding 7000. Therefore, the TLAA for non-Class 1 piping and in-line components remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(i).

The only non-Class1, non-piping component, identified with a fatigue time-limited aging analysis was the residual heat removal heat exchanger. That heat exchanger is projected to incur less than the analyzed number of cycles and therefore the analysis will remain valid for the period of extended operation.

A.3.2.2.3 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. At least two years prior to entering the period of extended operation, for the locations identified in NUREG/CR-6260 for Westinghouse PWRs of this vintage, the site will implement one or more of the following:

(1) Refine the fatigue analyses to determine valid CUFs less than 1 when accounting for the effects of reactor water environment. This includes applying the appropriate F_{en} factors to valid CUFs determined in accordance with one of the following.

   For locations, including NUREG/CR-6260 locations, with existing fatigue analysis valid for the period of extended operation, use the existing CUF to determine the environmentally adjusted CUF.

   In addition to the NUREG/CR-6260 locations, more limiting plant-specific locations with a valid CUF may evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component.

   Representative CUF values from other plants, adjusted to or enveloping the plant-specific external loads may be used if demonstrated applicable.

   An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.

(2) Manage the effects of aging due to fatigue at the affected locations by 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).

(3) Repair or replace the affected locations before exceeding a CUF of 1.0.

Should the site select the option to manage the aging effects due to environmental-assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be submitted to the NRC at least two years prior to the period of extended operation.

A.3.2.3 Environmental Qualification of Electrical Components

The 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.3.2.4 Containment Liner Plate and Penetrations Fatigue Analyses

There are no TLAA associated with the containment liner plate or the containment penetrations.

A.3.2.5 Leak before Break

Leak before break (LBB) analyses evaluate postulated flaw growth in piping to justify changes to the structural design basis involving protection against the effect of postulated reactor coolant loop pipe ruptures. The LBB evaluations use saturated (fully aged) fracture toughness properties, these analyses do not have a material property time-limited assumption. The fatigue crack growth for 40 years was calculated using the design transients for the reactor vessel. As these transients will not be exceeded in 60 years, these analyses will remain valid during the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.3.2.6 Steam Generator Flow-Induced Vibration and Tube Wear

The steam generators were evaluated with respect to flow-induced vibration. The projected tube wear is 2.8 mils (~5.7% through-wall wear) by the end of the period of extended operation. Therefore, the TLAA associated with tube wear has been projected to the end of the period of extended operation in accordance with 10 CFR 54(21)(c)(1)(ii).

A.3.3 References

A.3-1 [IPEC License Renewal Application—later]

A.3-2 [NRC SER for IPEC License Renewal—later]