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NUREG-2223
Volume 2

Safety Evaluation Report

Related to the License Renewal of South Texas Project Units 1 and 2

Docket Nos. 50-498, and 50-499

South Texas Project Nuclear Operating Company

Sections 4 to 6

Office of Nuclear Reactor Regulation

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ABSTRACT

This safety evaluation report (SER) documents the technical review of the South Texas Project (STP), Units 1 and 2, license renewal application (LRA) by the U.S. Nuclear Regulatory Commission (NRC) staff (the staff). By letter dated October 25, 2010, South Texas Nuclear Operating Company (STPNOC or the applicant) submitted the LRA in accordance with Title 10, Part 54, of the *Code of Federal Regulations*, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants" (10 CFR Part 54). The applicant requests renewal of the STP operating licenses (Facility Operating License Numbers DPR-76 and DPR-80, respectively) for a period of 20 years beyond the current license periods ending August 20, 2027 (Unit 1), and December 15, 2028 (Unit 2).

STP is located near the town of Matagorda, Texas, in Matagorda County, Texas. The staff issued the original construction permits for STP on December 22, 1975 (both units), and the operating licenses on August 20, 1987 (Unit 1), and December 15, 1988 (Unit 2). Each unit's nuclear steam supply system consists of a 4-loop pressurized-water reactor (PWR) designed by Westinghouse Electric Corporation. The primary containment for each unit is a dry ambient design. The balance of plant was designed and constructed by Bechtel Corporation. Both units operate at a licensed power output of 3,853 MWt, with a net electrical power output of 1,250 MWe each. The updated final safety analysis report contains details of the plant and the site.

Unless otherwise indicated, this SER presents the status of the staff's review of information submitted through May 2, 2017, the cutoff date for consideration in this SER. The open item previously identified in the SER with Open Items, issued October 2016, has been closed (see Section 1.5); therefore, no open items remain to be resolved before the final determination is reached by the staff on the LRA.

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ABBREVIATIONS

AAC	all aluminum conductors
ACAR	aluminum conductor alloy reinforced
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
ACSR	aluminum core, steel reinforced
ACT	analysis confirmatory test
ADAMS	Agencywide Documents Access and Management System
AEA	Atomic Energy Act
AERM	aging effect requiring management
AFST	auxiliary feedwater storage tank
AFW	auxiliary feedwater
AHU	air handling unit
AISC	American Institute of Steel Construction
A/LAI	applicant/licensee action item
AMP	aging management program
AMR	aging management review
ANS	American Nuclear Society
ANSI	American National Standards Institute
APCSB	Auxiliary and Power Conversion Systems Branch
ART	adjusted reference temperature
ASM	American Society for Metals
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
AVB	anti-vibration bar
AWAA	American Water Works Association
BMI	bottom-mounted instrumentation
BOP	balance of plant
BTP	Branch Technical Position
BWR	boiling-water reactor
C	Celsius
CAP	Corrective Action Program
CASS	cast austenitic stainless steel
CBF	cycle-based fatigue
CCCW	closed-cycle cooling water
CCW	component cooling water
CE	Combustion Engineering
CEA	control element assembly
CEO	Chief Executive Officer
CEOG	Combustion Engineering Owners Group
CETNA	core exit thermocouple nozzle assembly
CFO	Chief Financial Officer
CFR	*Code of Federal Regulations*
CLB	current licensing basis
cm^2	square centimeter
CMAA	Crane Manufacturers Association of America

CMTR	certified material test report
CMU	concrete masonry unit
COMS	cold overpressure mitigation system
CR	condition report
CRD	control rod drive
CRDM	control rod drive mechanism
CRGT	control rod guide tube
CUF	cumulative usage factor
CUF_{en}	environmentally correct cumulative usage factor
CVCS	chemical and volume control system
DBA	design basis accident
DBE	design basis event
DGB	diesel generator building
DPI	digital pressure indicator
E	energy
EAB	electrical auxiliary building
EAF	environmentally-assisted fatigue
ECCS	emergency core cooling system
ECP	essential cooling pond
	essential cooling water pond
ECW	essential cooling water
ECWIS	essential cooling water intake structure
ECWS	essential cooling water system
EFPY	effective full power year
EOL	end of life
EPDM	ethylene-propylene-diene
EPRI	Electric Power Research Institute
EQ	environmental qualification
ERFDADS	Emergency Response Facilities Data Acquisition and Display System
ESF	engineered safety feature
EW	essential cooling water
F	Fahrenheit
F_{en}	environmental adjustment factor
FERC	Federal Energy Regulatory Commission
FHAR	fire hazards analysis report
FHB	fuel handling building
FOCD	foreign ownership, control, or domination
FOCI	foreign ownership, control, or influence
FRN	*Federal Register Notice*
FSAR	final safety analysis report
ft	foot
ft-lb	foot-pound (energy)
FTIR	Fourier transform infrared
FWIV	feedwater isolation valve
FWST	firewater storage tank
GALL	Generic Aging Lessons Learned (NUREG-1801)
GDC	general design criterion

GEIS	generic environmental impact statement
GL	generic letter
gpm	gallons per minute
HAZ	heat-affected zone
HELB	high-energy line break
HPSI	high-pressure safety injection
HVAC	heating, ventilation, and air conditioning
I&C	instrumentation and control
I&E	inspection and flaw evaluation
IASCC	irradiation-assisted stress corrosion cracking
IEEE	Institute of Electrical and Electronics Engineers
IGSCC	intergranular stress corrosion cracking
ILRT	integrated leak rate testing
IN	Information Notice
in^2	square inch
INPO	Institute for Nuclear Power Operations
IPA	integrated plant assessment
IR	insulation resistance
ISA	International Society of Automation
ISG	interim staff guidance
ISI	inservice inspection
ISIT	initial structural integrity test
ksi	kilopounds per square inch
kV	kilovolt
lb	pound
LBB	leak-before-break
LCO	limiting condition for operation
LER	licensee event report
LOCA	loss-of-coolant accident
LRA	license renewal application
LR-ISG	License Renewal Interim Staff Guidance
LSS	low safety significance
LTOP	low temperature overpressure protection
LTW	long-term weighting
LWR	light-water reactor
M	margin term
MAB	mechanical auxiliary building
MDPE	medium-density polyethylene
MEAB	mechanical-electrical auxiliary building
MEB	metal-enclosed bus
MED	master equipment database
MeV	million electron volts
MIC	microbiologically-influenced corrosion
MRP	Materials Reliability Program
MRV	minimum required value
MSIV	main steam isolation valve

MWe	megawatt(s) electric
MWt	megawatt(s) thermal
n	neutron
NACE	National Association of Corrosion Engineers
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Agency
Ni	nickel
NOC	Nuclear Operations Committee
NPS	nominal pipe size
NRC	U.S. Nuclear Regulatory Commission
NRG	NRG Energy, Inc.
NRS	non-risk significant
NSSS	nuclear steam supply system
OBE	operating-basis earthquake
OD	outside diameter
ODSCC	outside diameter stress-corrosion cracking
OEP	Operating Experience Program
OI	open item
OOS	out of specification
OTSG	once-through steam generator
PDI	performance demonstration initiative
PDMS	plant data management system
PE	profile examination
PMWO	preventive maintenance work order
PORV	power operated relief valve
ppm	parts per million
PRT	pressurizer relief tank
psig	pounds per square inch gauge
P-T	pressure-temperature
PTS	pressurized thermal shock
PUCT	Public Utilities Commission of Texas
PVC	polyvinyl-chloride
PWR	pressurized-water reactor
PWSCC	primary water stress corrosion cracking
QA	quality assurance
RAI	request for additional information
RCB	reactor containment building
RCCA	rod control cluster assembly
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RCSC	Research Council on Structural Connections
RFO	refueling outage
RG	regulatory guide
RHR	residual heat removal
RIS	Regulatory Issue Summary

RP	Regulatory Position
RPV	reactor pressure vessel
RRVCH	replacement reactor vessel closure head
RSG	replacement steam generator
RT_{NDT}	reference temperature (nil ductility)
$RT_{NDT(U)}$	reference temperature (nil ductility) - unirradiated
RT_{PTS}	reference temperature (pressurized thermal shock)
RV	reactor vessel
RVI	reactor vessel internal
RVIIP	reactor vessel internals inspection plan
RVWLIS	RV water level indicator system
RWST	refueling water storage tank
S_a	allowable stress value
SBO	station blackout
SC	structure and component
SCC	stress corrosion cracking
SDG	standby diesel generator
SE	safety evaluation
SEC	Securities and Exchange Commission
SECY	Secretary of the Commission, Office of the Nuclear Regulatory Commission
SER	safety evaluation report
SG	steam generator
SI	spatial interaction
SIT	structural integrity test
S_m	design stress intensity
SRP-LR	standard review plan-license renewal (NUREG-1800)
SSC	structures, systems, and components
SSER	Supplemental Safety Evaluation Report
SSPC	Steel Structures Painting Council
STP	South Texas Project
STPNOC	South Texas Project Nuclear Operating Company
STW	short-term weighting
SWOL	structural weld overlay
TE	thermal embrittlement
TGB	turbine generator building
TLAA	time-limited aging analysis
TOFD	time-of-flight-diffraction
TS	technical specifications
TSC	technical service center
TSP	trisodium phosphate
UFSAR	updated final safety analysis report
USAR	updated safety analysis report
U.S.C.	U.S. Code
USE	upper-shelf energy
UT	ultrasonic testing

V	volt
WCAP	Westinghouse Commercial Atomic Power
wt-%	weight percent
XL	extra-long
Zn	zinc

SECTION 4

TIME-LIMITED AGING ANALYSES

4.1 Identification of Time-Limited Aging Analyses

This section of the safety evaluation report (SER) provides the U.S. Nuclear Regulatory Commission (NRC) staff (the staff) evaluation of the applicant's basis for identifying those plant-specific or generic analyses that need to be identified as time-limited aging analyses (TLAAs) for the applicant's license renewal application (LRA) and the list of TLAAs for the LRA. TLAAs are certain plant-specific safety analyses that involve time-limited assumptions defined by the current operating term. This section of the SER also provides the staff's evaluation of the applicant's basis for identifying regulatory exemptions that need to be identified in the LRA.

In accordance with the requirements in Section 54.21(c)(1) of Title 10 of the *Code of Federal Regulations* (10 CFR 54.21(c)(1)), an applicant for license renewal must list all evaluations, analyses, and calculations in the current licensing basis (CLB) that conform to the definition of a TLAA as specified in 10 CFR 54.3. A plant-specific or generic evaluation, analysis, or calculation is a TLAA as defined in 10 CFR 54.3 if it meets all six of the following TLAA identification criteria:

(1) involves a system, structure, or component (SSC) within the scope of license renewal, as delineated in 10 CFR 54.4(a)

(2) considers the effects of aging

(3) involves time-limited assumptions that are defined by the current operating term (e.g., 40 years)

(4) was determined to be relevant by the applicant in making a safety determination

(5) involves conclusions or provides the basis for conclusions related to the capability of the SSC to perform its intended functions, as described in 10 CFR 54.4(b)

(6) is contained or incorporated by reference in the CLB

In addition, pursuant to 10 CFR 54.21(c)(2), applicants must list all plant-specific exemptions in the CLB that were granted in accordance with the exemption approval criteria in 10 CFR 50.12 and that are based on a TLAA. For such exemptions, the applicant must evaluate and justify the continuation of the exemptions during the period of extended operation.

The NRC's guidance recommendations for reviewing LRA Chapter 4.1 sections are provided in NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR), Section 4.1, "Identification of Time Limiting Aging Analyses." SRP-LR Section 4.1.1 summarizes the areas of review. SRP-LR Section 4.1.2 provides the staff's acceptance criteria for performing TLAA and LRA exemption identification reviews. SRP-LR Section 4.1.3 provides the staff's review procedures for performing the TLAA and LRA exemption identification reviews. SRP-LR Table 4.1-1 provides case-by-case examples on whether a given analysis category would be required to be identified as a TLAA for an LRA. SRP-LR Table 4.1-2 provides a generic list of those analyses or calculations that are commonly identified as TLAAs for an LRA. SRP-LR Table 4.1-3 provides a generic list of those analyses or calculations that may be identified as plant-specific TLAAs for an LRA.

4.1.1 Summary of Technical Information in the Application

4.1.1.1 Identification of Time-Limited Aging Analyses

LRA Section 4.1 states that the applicant reviewed and evaluated the evaluations, analyses, and calculations in the CLB for South Texas Project (STP), Units 1 and 2, against the six criteria for TLAAs in 10 CFR 54.3. The LRA also states that the applicant reviewed the list of TLAAs in the SRP-LR to determine if each TLAA is applicable to and included as part of the applicant's CLB. The applicant stated that it used the following guidance documents as part of the basis for its TLAA identification methodology:

- NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR), Chapter 4
- Nuclear Energy Institute (NEI) 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 – the License Renewal Rule"
- 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants"
- prior LRAs
- plant-specific document reviews and interviews with plant personnel

The applicant stated that its review of the CLB included a review of the following plant-specific or generic sources (documents or records):

- STP's updated final safety analysis report (UFSAR)
- STP's technical specifications (TS)
- NRC SERs for the original operating licenses
- subsequent NRC safety evaluations (SEs)
- STP and NRC-docketed licensing correspondence
- vendor, NRC-sponsored, and licensee topical reports
- STP design calculations
- Code stress reports or Code design reports
- STP plant drawings and specifications

The staff noted that LRA Table 4.1-1 identifies the following evaluations, analyses, or calculations in the CLB that meet the six criteria for TLAAs in 10 CFR 54.3:

- Reactor Pressure Vessel (RPV) Neutron Embrittlement Analyses in LRA Section 4.2:
  - LRA Section 4.2.1, "Neutron Fluence Values"
  - LRA Section 4.2.2, "Pressurized Thermal Shock [PTS]"
  - LRA Section 4.2.3, "Upper-Shelf Energy (USE)"
  - LRA Section 4.2.4, "Pressure-Temperature (P-T) Limits"
  - LRA Section 4.2.5, "Low Temperature Overpressure Protection [LTOP]"
- Metal Fatigue Analyses in LRA Section 4.3:
  - LRA Section 4.3.2, American Society of Mechanical Engineers (ASME) Code Section III Class 1 Fatigue Analyses of Vessels, Piping, and Components:

        - RPV, nozzles, head, and studs (LRA Section 4.3.2.1)
        - control rod drive mechanism (CRDM) pressure housings and core exit thermocouple nozzle assemblies (CETNAs) (LRA Section 4.3.2.2)
        - reactor coolant pump (RCP) pressure boundary components (LRA Section 4.3.2.3)
        - pressurizer and pressurizer nozzles (LRA Section 4.3.2.4)
        - steam generator ASME Code Section III Class 1, Class 2 secondary side, and feedwater nozzle fatigue analyses (LRA Section 4.3.2.5)
        - ASME Code Section III Class 1 valves (LRA Section 4.3.2.6)
        - ASME Code Section III Class 1 piping and piping nozzles (LRA Section 4.3.2.7)
        - intermittent thermal cycling analysis performed in response to NRC Bulletin No. 88-08 (LRA Section 4.3.2.8)
        - revised fatigue analysis for the pressurizer surge line performed in response to NRC Bulletin No. 88-11 (LRA Section 4.3.2.9)
        - high-energy line break (HELB) postulation based on fatigue cumulative usage factor (CUF) (LRA Section 4.3.2.10)
        - fatigue crack growth assessments and fracture mechanics stability analyses for leak-before-break (LBB) elimination of dynamic effects of primary loop piping failures (LRA Section 4.3.2.11)
        - ASME Code Section III Class 1 design of ASME Code Class 3 feedwater control valves (LRA Section 4.3.2.12)
    - LRA Section 4.3.3, ASME Code Section III Subsection NG Fatigue Analysis for Reactor Pressure Vessel Internals
    - LRA Section 4.3.4, Effects of Reactor Coolant System Environment on Fatigue Life of Piping and Components (Generic Safety Issue (GSI)-190)
    - LRA Section 4.3.5, Assumed Thermal Cycle Count for Allowable Secondary Stress Range Reduction Factor for American National Standards Institute (ANSI) B31.1 and ASME Code Section III Class 2 and 3 Piping
    - LRA Section 4.3.6, Fatigue Analyses of Metal Bellows and Expansion Joints
- Environmental Qualification (EQ) of Electric Equipment in LRA Section 4.4
- Concrete Containment Tendon Prestress Analysis in LRA Section 4.5
- Containment Liner Plate, Metal Containments, and Penetrations Fatigue Analyses in LRA Section 4.6:
    - LRA Section 4.6.1, “Fatigue Waivers for the Personnel Airlocks and Emergency Airlocks”
    - LRA Section 4.6.2, “Fatigue Design of Containment Penetrations”
- Plant-Specific TLAAs in LRA Section 4.7:

- LRA Section 4.7.1, “Load Cycle Limits for Cranes, Lifts, and Fuel Handling Equipment Designed to CMAA-70 [Crane Manufacturers Association of America Specification 70]”
- LRA Section 4.7.3, “TLAA for the Corrosion Effects in the Essential Cooling Water (ECW) System”
- LRA Section 4.7.5, Reactor Coolant Pump (RCP) Flywheel Fatigue Flaw Growth Analysis

The applicant provided its bases for dispositioning these TLAAs in accordance with the requirements in either 10 CFR 54.21(c)(1)(i), (ii), or (iii) in the applicable subsections of LRA Sections 4.2-4.7.

In addition, LRA Table 4.1-1 identifies the “Disposition Category” as “Not Applicable” for TLAAs related to “In-service Flaw Growth Analyses that Demonstrate Structural Stability for 40 years” (LRA Section 4.7.2) and “Absence of a TLAA for Reactor Vessel Underclad Cracking Analyses” (LRA Section 4.7.4).

The staff noted that LRA Table 4.1-2 states that the following analyses, which are listed in SRP-LR Tables 4.1-2 and 4.1-3 as potential or plant-specific TLAAs, do not meet the definition of a TLAA for STP:

- inservice local metal containment corrosion analyses
- intergranular separation in the heat-affected zone (HAZ) of reactor vessel (RV) low-alloy steel under austenitic stainless steel cladding
- fatigue analysis for the main steam supply lines to turbine-driven auxiliary feedwater pumps
- flow-induced vibration endurance limit
- ductility reduction of fracture toughness for reactor vessel internals (RVIs)
- fatigue analysis for the containment liner plate
- RPV circumferential weld inspection relief (boiling water reactor (BWR))

4.1.1.2 Identification of Regulatory Exemptions

LRA Section 4.1.4 states that the applicant’s review of the CLB identified seven exemptions, granted pursuant to the criteria in 10 CFR 50.12, that are currently in effect for the STP CLB. The LRA states that pursuant to 10 CFR 54.21(c)(2), of these exemptions, the exemption for implementation of the LBB analysis was the only exemption that was based in part on a TLAA. The applicant stated that its basis for extending the acceptance of the LBB analysis for the period of extended operation is given in LRA Section 4.3.2.11.

4.1.2 Staff Evaluation

4.1.2.1 Identification of TLAAs

The staff reviewed the applicant’s methodology for identifying the TLAAs and the TLAA results for the LRA against the six criteria for TLAA identification in 10 CFR 54.3 and the generic list of TLAAs in SRP-LR Section 4.1, including those in SRP-LR Tables 4.1-2 and 4.1-3, as applicable to the STP CLB. The staff used the acceptance criteria in SRP-LR Section 4.1.2 and the review procedures in SRP-LR Section 4.1.3 as the basis for its review.

4.1.2.1.1 Evaluations, Analyses, and Calculations in the CLB Conforming to 10 CFR 54.3 TLAA Criteria

The staff confirmed that the applicant included its TLAAs for the RPV neutron irradiation embrittlement analyses in the applicable referenced subsections of LRA Section 4.2, which includes the TLAAs for the neutron fluence, PTS, USE, P-T limits, and LTOP. The staff noted that these analyses should be included as TLAAs for the LRA because the analyses are mandated by applicable NRC requirements (e.g., 10 CFR 50.61 for PTS; 10 CFR Part 50, Appendix G, for USE, P-T limit, and LTOP analyses; and 10 CFR Part 50, Appendix H, for RPV surveillance capsule neutron dosimetry and fracture toughness analyses). Additionally, the analyses conform to all six of the criteria for identifying TLAAs in 10 CFR 54.3. Thus, the staff noted that the applicant's identification of these TLAAs is consistent with the recommendations in SRP-LR Sections 4.1 and 4.2, which provide the bases for identifying these types of neutron irradiation embrittlement analyses as TLAAs in accordance with the requirements in 10 CFR 54.21(c)(1). Based on this review, the staff finds that the identification of these analyses as TLAAs is acceptable because it complies with 10 CFR 54.21(c)(1). The staff evaluated the applicant's basis for the disposition of each of these TLAAs in accordance with either 10 CFR 54.21(c)(1)(i), (ii), or (iii) in the applicable subsections of SER Section 4.2.

The staff confirmed that the applicant included its TLAAs on metal fatigue analyses in the applicable subsections of LRA Section 4.3, as referenced in the "Summary of Technical Information" above. The staff noted that these analyses should be included as TLAAs for the LRA because the analyses are mandated by applicable design rules (e.g., those in Section III of the ASME Code or in the ANSI B31.1 design code) or applicable NRC requirements (e.g., 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 4, for the LBB analyses), or were implemented as part of the applicant's commitments to applicable NRC generic communications (e.g., the supplemental fatigue analyses that were performed in response to the recommendations in NRC Bulletins 88-08 and 88-11). Additionally, the analyses conform to all six of the criteria for identifying TLAAs in 10 CFR 54.3. The staff noted that the applicant's identification of these TLAAs is consistent with SRP-LR Sections 4.1 and 4.3, which provide the bases for identifying these types of fatigue analyses as TLAAs in accordance with the requirements in 10 CFR 54.21(c)(1). Based on this review, the staff finds that the identification of these analyses as TLAAs is acceptable because it complies with 10 CFR 54.21(c)(1). The staff evaluated the applicant's basis for dispositioning each of these TLAAs in accordance with either 10 CFR 54.21(c)(1)(i), (ii), or (iii) in the applicable subsections of SER Section 4.3.

The staff confirmed that the applicant included its TLAA on environmental qualification (EQ) of electric equipment in LRA Section 4.4. The staff noted that the EQ analysis should be included as a TLAA for the LRA because the analysis is mandated by the requirements in 10 CFR 50.49 and conforms to all six of the criteria for identifying TLAAs in 10 CFR 54.3. The staff confirmed that the applicant's identification of the EQ TLAA is consistent with the staff recommendations in SRP-LR Sections 4.1 and 4.4, which provide the bases for identifying EQ analyses as TLAAs in accordance with 10 CFR 54.21(c)(1). Based on this review, the staff finds that the identification of the EQ TLAA is acceptable because it complies with 10 CFR 54.21(c)(1). The staff evaluated the applicant's basis for dispositioning the EQ TLAA in accordance with 10 CFR 54.21(c)(1)(iii) in SER Section 4.4.

The staff confirmed that the applicant included its TLAA on concrete containment tendon prestress analysis in LRA Section 4.5. The staff noted that the concrete containment prestress analysis should be included as a TLAA for the LRA because the analysis is mandated by applicable ASME Code Section III CC-3000 design rules, and the analysis conforms to all six of the criteria for identifying TLAAs in 10 CFR 54.3. The staff confirmed that the applicant's identification of the concrete containment tendon prestress TLAA is consistent with the staff recommendations in SRP-LR Sections 4.1 and 4.5, which provide the staff's bases for identifying concrete containment tendon prestress analyses as TLAAs in accordance with 10 CFR 54.21(c)(1). Based on this review, the staff finds that the identification of the concrete containment tendon prestress TLAA is acceptable because it complies with 10 CFR 54.21(c)(1). The staff evaluated the applicant's basis for dispositioning concrete containment tendon prestress analysis in accordance with 10 CFR 54.21(c)(1)(iii) in SER Section 4.5.

The staff confirmed that the applicant included its TLAAs on fatigue analyses for the containment structure and other structural components in LRA Section 4.6. The waiver analysis exempting the containment personnel and emergency airlocks from the performance of a CUF based fatigue analysis is provided in LRA Section 4.6.1. The CUF fatigue analyses for the containment penetrations are evaluated in LRA Section 4.6.2 and identified in LRA Table 4.6.2-1. The fatigue analysis of fuel transfer tube bellows are also evaluated in LRA Section 4.6.2. The staff noted that these analyses should be included as TLAAs for the LRA because the analyses are mandated by the applicable fatigue calculation or fatigue waiver rules in Section III of the ASME Code and conform to the six criteria for TLAAs in 10 CFR 54.3. The staff noted that the applicant's identification of these TLAAs is consistent with staff recommendations in SRP-LR Sections 4.1 and 4.6, which provide the staff's bases for identifying containment structural analyses as TLAAs in accordance with 10 CFR 54.21(c)(1). Based on this review, the staff finds that the identification of these containment component TLAAs is acceptable because it complies with 10 CFR 54.21(c)(1). The staff evaluated the applicant's basis for dispositioning these TLAAs in accordance with either 10 CFR 54.21(c)(1)(i), (ii), or (iii) in SER Sections 4.6.1 and 4.6.2.

The staff confirmed that the applicant included the following plant-specific TLAAs for the LRA in LRA Section 4.7:

- TLAA in LRA Section 4.7.1 on load cycle limits for the applicant's cranes, lifts, and fuel handling equipment
- TLAA in LRA Section 4.7.3 for the corrosion rate analysis for the ECW system, as performed in support of discontinuing the use of biocide inhibitors in this system
- TLAA in LRA Section 4.7.5 for the RCP flywheel flaw growth analysis

The staff noted that the applicant's identification of these TLAAs is consistent with the staff recommendations for identifying plant-specific TLAAs in SRP-LR Sections 4.1 and 4.7. Based on this review, the staff finds that the identification of these plant-specific TLAAs is acceptable because it complies with 10 CFR 54.21(c)(1). The staff evaluated the applicant's basis for dispositioning these plant-specific TLAAs in accordance with either 10 CFR 54.21(c)(1)(i), (ii), or (iii) in the applicable subsections of SER Section 4.7.

For the items identified as "Not Applicable" in LRA Table 4.1-1, specifically "In-service Flaw Growth Analyses that Demonstrate Structural Stability for 40 years" (LRA Section 4.7.2) and "Absence of a TLAA for Reactor Vessel Underclad Cracking Analyses" (LRA Section 4.7.4), the staff's evaluations of the information in the LRA are provided in SER Sections 4.7.2 and 4.7.4, respectively.

4.1.2.1.2 Evaluations, Analyses, and Calculations in the CLB That Do Not Conform to TLAA Criteria, or Absence of a TLAA Due to Absence in the CLB

Absence of a TLAA for Inservice Local Metal Containment Corrosion Analyses. LRA Table 4.1-2 shows that the applicant's review of the CLB did not identify any time-dependent local metal corrosion analyses for the containment structures. Therefore, the applicant stated that the LRA does not need to include a localized metal corrosion TLAA for the containment structures because the generic "inservice local metal corrosion analysis" TLAA in SRP-LR Table 4.1-2 is not applicable to its CLB.

The staff reviewed the UFSAR for relevant information. The staff noted that the applicant addresses design features for managing corrosion of steel containment tendons in UFSAR Section 3.8.1 and the steel containment liners in UFSAR Section 3.8.5. The staff noted that UFSAR Section 3.8.1.7.3.1.2 indicates that the applicant does not use a time-dependent analysis to serve as the design basis for managing the impact of postulated corrosion effects on the steel containment tendons. The staff confirmed that the applicant uses its Concrete Containment Tendon Prestress Program (LRA Section B3.3) to manage the impact of postulated corrosion effects on the steel containment tendons. The staff also noted that this is the same aging management program (AMP) that is used to disposition the applicant's time-dependent prestress analysis for the tendons in accordance with the TLAA acceptance criterion in 10 CFR 54.21(c)(1)(iii).

In addition, the staff noted that UFSAR Section 3.8.5.1 indicates that the applicant does not use a time-dependent analysis as the design basis for managing the impact of postulated corrosion effects on the steel containment liners. The staff confirmed that UFSAR Section 3.8.5.1 indicates that the applicant uses a combination of cathodic protection and a waterproofing membrane as the basis for protecting the below-grade portions of the steel containment liner against the effects of corrosion. The staff confirmed that the applicant uses AMP B2.1.27, ASME Code Section XI, Subsection IWE, as its basis for managing the effects of aging (including potential loss of material due to corrosion) that are applicable to the metal containment liners. The staff's evaluation of the applicant's AMP B2.1.27 is provided in SER Section 3.0.3.2.22.

Based on this review, the staff confirmed that the applicant does not use any time-dependent corrosion analyses as the basis for protecting containment structure metal components against the effects of corrosion. The staff finds that the LRA does not need to identify any localized metal containment corrosion as TLAAs because the staff has confirmed that the applicant's CLB does not include these types of analyses. Additionally, the applicant uses either applicable design features or surveillance programs (i.e., the ASME Code Section XI, Subsection IWE, condition monitoring program) to manage the impacts of corrosion on the integrity of containment structure metal components.

Absence of a TLAA for Intergranular Separation in the HAZ of Reactor Vessel Low-Alloy Steel Under Austenitic Stainless Steel Cladding (RPV Underclad Cracks). In LRA Table 4.1-2 and LRA Section 4.7.4, the applicant stated that its review of the CLB did not identify any time-dependent flaw growth, flaw tolerance, or fracture mechanics evaluations to assess RPV underclad cracks. The applicant stated that, although there is an applicable Westinghouse topical report that assesses fatigue flaw growth analysis of postulated RPV underclad cracks, the report is not credited as part of its CLB for managing the potential for underclad cracks to develop in welds used to join the stainless steel cladding to RPV SA-508, Class 2, forging components (henceforth cladding-to-forging welds.) The applicant stated that its design basis uses the application of Regulatory Guide (RG) 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," as the basis for precluding or mitigating the occurrence of underclad cracks in the RPV cladding-to-forging welds.

The staff reviewed the UFSAR for relevant information. The staff noted that UFSAR Section 5.2.3.3.2 states that all welding is conducted using procedures that are qualified in accordance with the applicable weld qualification rules of the ASME Code Sections III and IX. Additionally, the UFSAR states that control of welding variables, as well as examination and testing methods, during procedure qualification and production welding is performed in accordance with the applicable ASME Code requirements. The staff also noted that UFSAR Section 5.2.3.3.2 states that Westinghouse (the nuclear steam supply system (NSSS) vendor for the RPV) met the intent of RG 1.43 by requiring qualification of any high-heat-input welding process (including the submerged-arc wide-strip and submerged-arc-6-wire welding processes) through implementation of a performance test, as recommended in Regulatory Position 2 of RG 1.43. The staff also noted, however, that UFSAR Section 5.3.1.2 states that the applicant would perform an additional "special evaluation" to verify and validate the special procedure qualification in its ability to assure freedom from RPV underclad cracking.

The staff also noted that LRA Section 4.7.4 did not make any reference to the "special evaluation" referenced in UFSAR Section 5.3.1.2 for underclad cracks. Specifically, the staff noted that the basis in LRA Section 4.7.4 did not identify how the applicant fulfilled the UFSAR Section 5.3.1.2 protocol for performing the special evaluation or describe what the "special evaluation" involved. The staff noted that the basis did not assess how the special evaluation, as implemented, compared to the six criteria for TLAAs in 10 CFR 54.3, and it did not justify whether the evaluation would need to be identified as a TLAA pursuant to the requirements in 10 CFR 54.21(c)(1).

By letter dated September 22, 2011, the staff issued a request for additional information (RAI) 4.1-3 to address this issue. In this RAI, the staff asked the applicant to clarify how it fulfilled the UFSAR Section 5.3.1.2 protocol for performing a "special evaluation" to confirm and validate the special procedure qualification in its ability to assure freedom from RPV underclad cracking and to summarize what the special evaluation involved, with an appropriate CLB reference. The staff also asked the applicant to summarize how the "special evaluation," if implemented as part of the CLB, compares to the six criteria for TLAAs in 10 CFR 54.3 and to justify whether the evaluation would need to be identified as a TLAA pursuant to the requirements in 10 CFR 54.21(c)(1). The staff also asked the applicant to justify its basis for not performing the "special evaluation," if—contrary to the statement in UFSAR Section 5.3.1.2—the applicant had not performed the "special evaluation" as part of its CLB.

The applicant responded to RAI 4.1-3 by letter dated November 21, 2011, that the weld qualification process discussed in UFSAR Section 5.2.3.3.2 provides the "special evaluation" referred to in UFSAR Section 5.3.1.2. The applicant stated that the "special evaluation" is a performance test that was implemented consistent with the recommended Position 2 of RG 1.43. The applicant also stated that its review of the welding qualification test recommendations in Position 2 of RG 1.43 did not indicate that the tests would need to account for an aging mechanism or a time-dependent parameter that was defined in terms of the life of the plant. The applicant further stated that, based on this review, it concluded that the special evaluation referred to in UFSAR Section 5.3.1.2 did not meet the definition of a TLAA in 10 CFR 54.3.

The staff noted that, in its response to RAI 4.1-3, the applicant based its "absence of a TLAA" conclusion for the RV SA-508 Class 2 forging components on the criteria that were established in RG 1.43 and not on the applicant's own plant-specific basis that was implemented pursuant to the CLB to conform to the NRC's regulatory position in RG 1.43. Specifically, the staff noted that the applicant did not indicate which plant-specific document in its CLB was implemented to conform to the RG 1.43 basis. The staff also noted that the applicant did not summarize which type of tests or evaluations were performed as part of its CLB to meet the recommended weld qualification criteria in RG 1.43 and, if an evaluation was performed as part of this qualification process, whether the evaluation would meet the six criteria for TLAAs in 10 CFR 54.3. Therefore, the staff did not have sufficient information to determine whether the applicant's CLB basis for conforming to RG 1.43 included an analysis that, when assessed against the six criteria for TLAAs in 10 CFR 54.3, would need to be identified as a TLAA in accordance with 10 CFR 54.21(c)(1).

As a result of an audit of the applicant's RV underclad cracking references, the staff also noted that the applicant referenced Westinghouse Non-Proprietary Class 3 Report WCAP-15338-A, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants," as an applicable RV underclad cracking reference. This report includes a generic fatigue flaw growth analysis for underclad cracks that would constitute a TLAA for a pressurized water reactor (PWR) LRA if the report was being relied upon as part of the license renewal applicant's CLB. Thus, the staff also needed additional information on whether WCAP-15338-A was being relied upon as part of the CLB.

By letter dated February 15, 2012, the staff issued RAI 4.1-3a, requesting that the applicant reference the specific report, calculation, or analysis document that was used in the CLB to conform to the NRC's regulatory position in RG 1.43. The staff also asked the applicant to summarize the types of tests or evaluations that were performed as part of this CLB to be consistent with the NRC's regulatory position in RG 1.43. Additionally, if the CLB included any evaluations, analyses, or calculations in support of the RG 1.43 conformance basis, the staff asked the applicant to justify why the evaluations, analyses, or calculations would not need to be identified as TLAAs for the LRA.

The applicant responded to RAI 4.1-3a by letter dated March 29, 2012. In its response, the applicant stated that it was amending its CLB and design basis to adopt Westinghouse Report WCAP-15338-A as the basis for managing potential underclad cracking in the RV nozzles that are made from SA-508 Class 2 alloy steel forging materials. The applicant also stated that it was amending the LRA to identify the fatigue flaw growth analysis in WCAP-15338-A as a TLAA for the LRA. The applicant also stated that it was amending the following sections of the LRA in accordance with the updated basis for managing RV undercladding cracking:

- LRA Section 3.1.2.2.5, which provides the applicant's aging management further evaluation response to SRP-LR Section 3.1.2.2.5
- LRA Tables 4.1-1 and 4.1-2, which amend the LRA to identify that the fatigue flaw growth analysis in WCAP-15338-A is a TLAA for evaluating the stability of potential RV underclad cracks in those RV nozzles that are fabricated from SA 508, Class 2 alloy steel forging materials

- LRA Section 4.7.4, which amends the LRA to provide the applicant's summary and discussion on why the analysis in WCAP-15338-A is acceptable for evaluating and managing potential RV underclad cracks in the associated RV nozzles and the applicant's basis for dispositioning the fatigue flaw growth analysis in WCAP-15338-A in accordance with 10 CFR 54.21(c)(1)(i)
- Inclusion of LRA Section A.3.6.5, which provides the applicant's UFSAR supplement summary description for the fatigue flaw growth TLAA in WCAP-15338-A

The staff reviewed the applicant's response to RAI 4.1-3a and determined that the applicant's amended basis is consistent with SRP-LR Section 3.1.2.2.5, "Crack Growth Due to Cyclical Loading," which states the following:

> Crack growth due to cyclic loading could occur in reactor vessel shell forgings clad with stainless steel using a high-heat-input welding process. Growth of intergranular separations (underclad cracks) in the heat-affected zone under austenitic stainless steel cladding is a TLAA to be evaluated for the period of extended operation for all the SA-508-Cl-2 forgings where the cladding was deposited with a high heat input welding process. The methodology for evaluating the underclad flaw should be consistent with the flaw evaluation procedure and criterion in the ASME Code Section XI Code, 2004 Edition 1. See the SRP-LR, Section 4.7, "Other Plant-Specific Time-Limited Aging Analysis," for generic guidance for meeting the requirements of 10 CFR 54.21(c).

The staff noted that the applicant amended LRA Section 3.1.2.2.5, LRA Tables 4.1-1 and 4.1-2, LRA Section 4.7.4, and the UFSAR supplement summary description for the RV underclad cracking TLAA in LRA Section A.3.6.5. However, the staff also noted that the LRA amendments associated with this revision should have amended LRA aging management review (AMR) item 3.1.1.21 to identify that the basis for managing cracking in the RV nozzles made from SA 508, Class 2 steel forging materials is consistent with AMR, item 21, in SRP-LR Table 3.1-1. The staff noted that the applicant should have amended LRA Table 3.1.2-1 to include a new Table 2 AMR item for these RV nozzles that use the associated TLAA as the basis for managing fatigue-induced cracking in the nozzles. Resolution of this issue is documented in the staff's evaluation provided in SER Section 3.1.2.2.5. All other aspects of RAIs 4.1-3 and 4.1-3a are resolved.

The staff's evaluation of the amended LRA Section 4.7.4 is documented in SER Section 4.7.4.

<u>Absence of Fatigue Analyses for Main Steam Supply Lines to the Turbine-Driven Auxiliary Feedwater Pumps</u>. In LRA Table 4.1-2, the applicant stated that its review of the CLB did not identify any time-dependent fatigue analyses for the main steam supply lines to the turbine-driven auxiliary feedwater (AFW) pumps. Therefore, the applicant stated that the LRA does not need to include a fatigue TLAA for these components because the generic "fatigue analysis for the main steam supply lines to the turbine-driven auxiliary feedwater pumps" in SRP-LR Table 4.1-3 is not applicable to its CLB.

The staff reviewed the UFSAR for relevant information. The staff confirmed that UFSAR Table 10.1-1 indicates that the applicant's units are each designed with three motor-driven AFW pumps and one turbine-driven AFW pump. The staff also confirmed that UFSAR Table 3.2.A-1 indicates that the main steam supply line to the turbine-driven AFW pump was designed to either ASME Code Section III, subarticle NC or ND, design requirements for ASME Code Class 2 or 3 components.

The staff noted that the ASME Code Section III design code of record (1974 edition inclusive of the winter 1975 addenda) did not require explicit CUF or I_t fatigue analyses of these main steam supply lines. The staff noted, however, that the ASME Code Section III, subarticle NC or ND, requirements may have required the applicant to perform a maximum allowable stress range reduction analysis for the main steam supply line to the turbine-driven AFW pump. The staff also noted that LRA Section 4.3.5 identifies the maximum allowable stress range reduction analyses for the ASME Code Class 2 and 3 piping as TLAAs for the LRA. The staff further noted that GALL Report AMR VIII.B1-10 identifies that fatigue is to be managed using a TLAA for steel main steam piping that is exposed to steam or secondary water environments and that the applicant included the applicable AMR items for its steel main steam piping components in LRA Table 3.4.2-1. Thus, the staff noted that the applicant would need to provide further clarification and justification on why the maximum allowable stress range reduction TLAA discussed in LRA Section 4.3.5 would not be applicable to the main steam supply line that supplies steam to the turbine-driven AFW pump during a turbine-driven AFW system actuation.

By letter dated September 22, 2011, the staff issued RAI 4.1-4 to address this issue. In this RAI, the staff asked the applicant to provide its basis on why the cumulative fatigue damage in the main steam supply lines to the turbine-driven AFW pumps would not need to be managed using the maximum allowable stress range reduction TLAA in LRA Section 4.3.5.

The applicant responded to RAI 4.1-4 by letter dated November 21, 2011. In its response, the applicant clarified that ASME Code Section III requirements would have required it to include the main steam supply lines to the turbine-driven AFW pumps in accordance with the maximum allowable stress range reduction analysis (implicit fatigue analysis) methodology that is defined as a TLAA and evaluated in LRA Section 4.3.5. Therefore, the applicant stated that the main steam supply lines to the turbine-driven AFW pumps are within the scope of the components that are included in the implicit fatigue TLAA in LRA Section 4.3.5. The applicant stated that, in order to create the link between the AMR for these main steam supply lines in LRA Section 3.4 and the TLAA in LRA Section 4.3.5, it amended LRA Table 3.4.2-1 to include an AMR item for the main steam supply lines to the turbine-driven AFW pumps, which indicates that the applicant credits the implicit fatigue TLAA in LRA Section 4.3.5 for management of cumulative fatigue damage of these components.

The staff reviewed the applicant's response to RAI 4.1-4 and confirmed that the applicant amended the LRA to include the following additional AMR item for the main steam piping components, including the supply lines to the turbine-driven AFW pumps. Based on this response, the staff finds that the applicant's amended basis is acceptable because:

- The applicant identified that the main steam supply lines to the turbine-driven AFW pumps are within the scope of the components that are included in the implicit fatigue TLAA in LRA Section 4.3.5.
- The applicant amended the LRA to include the appropriate TLAA-based AMR item for the main steam system piping components, including the steam line piping to the turbine-driven AFW pumps.
- The amended basis creates the link in the LRA between the components and the basis for managing cumulative fatigue damage in the components using the stated TLAA.
- This complies with the aging management requirement in 10 CFR 54.21(a)(3) and with the requirement for identifying the applicable metal fatigue TLAA in 10 CFR 54.21(c)(1).

The staff's concerns expressed in RAI 4.1-4 are resolved.

Absence of Flow-Induced Vibration Endurance Limit TLAAs for Reactor Vessel. In LRA Table 4.1-2 and LRA Section 4.3.3, the applicant stated that its review of the CLB did not identify any time-dependent flow-induced vibration endurance limit analyses for the RVI components. The applicant stated that the CLB does not describe any time-limited effects for a licensed operating period associated with flow-induced vibration; therefore, there are no analyses in the CLB associated with flow-induced vibrations of the RVI components that would meet the definition of a TLAA in accordance with 10 CFR 54.3. The applicant concluded that the LRA does not need to include these types of TLAAs because the generic "flow-induced vibration endurance limit for the reactor vessel internals" TLAA in SRP-LR Table 4.1-3 is not applicable to or part of the CLB.

The staff reviewed the UFSAR for relevant information. The staff confirmed that the applicant's flow-induced vibration analysis basis for RVI components is accounted for in the following sections and tables of the UFSAR:

- Section 3.9.2.3, Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady-State Conditions
- Section 3.9.2.4, Preoperational Flow-Induced Vibration Testing of Reactor Internals
- Section 3.9.2.6, Correlations of Reactor Internals Vibration Tests with the Analytical Results
- Section 1.6, Material Incorporated By Reference, and Table 1.6-2, Westinghouse Topical Reports Incorporated By Reference—with the following WCAP Reports invoked by reference as part of the flow-induced vibrational analysis basis:
  - Proprietary NRC-Approved WCAP-8303-P-A, Revision 0, "Prediction of the Flow-Induced Vibration of Reactor Internals by Scale Model Tests"
  - Proprietary NRC-Approved WCAP-8516-P-A, Revision 0, "UHI Plant Internals Vibration Measurement Program and Pre and Post Hot Functional Examinations"
  - Proprietary NRC-Approved WCAP-8766-P-A, Revision 0, "Verification of Neutron Pad and 17x17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant"
  - Proprietary WCAP-9395-P, "4XL Scale Model Internal Flow Test Structural Response Test" (UFSAR Section 1.5 indicates that this WCAP includes an assessment of the vibrational levels in the internals)
  - WCAP-9646, "Verification of Upper Head Injection Reactor Vessel Internals by Preoperational Test of the Sequoyah Power Plant"
  - Proprietary WCAP-10865, "South Texas Plant (TGX) Reactor Internals Flow-Induced Vibration Assessment"

The staff confirmed that, collectively, these UFSAR sections indicate that the applicant uses consistency with the NRC's position in RG 1.20, Revision 3 "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," as the basis for protecting the integrity of the RVI components against those aging effects that may be induced by flow-induced vibrations (e.g., cracking induced by flow-induced vibrations or loss of material/wear induced by the vibrations.) The staff noted that RG 1.20 provides an acceptable position that, if followed, can be used to demonstrate how an applicant for an operating license would comply with the technical information requirements for flow-induced vibrations in 10 CFR 50.34. It also permits applicants applying the RG basis to assess flow-induced vibrations of their RVI components using prototypical data and tests results from other U.S. PWRs whose RVI components were well analyzed for their responses to flow-induced vibrations.

The staff also noted that UFSAR Section 3.9.2.3 provides the applicant's basis for conforming to the prototypical analysis basis in RG 1.20. This UFSAR section states that the applicant applies the flow-induced vibration analysis for the Indian Point Unit 2 internals, with some additional prototypical data and test results from the Trojan and Sequoyah Unit 1 reactors, as the prototypical basis for analyzing the response of the STP RVI components to flow-induced vibrations. UFSAR Section 3.9.2.4 provides the list of confirmatory preoperational testing examinations that the applicant will perform of its RVI components (in lieu of performing instrument-implemented vibrational testing of the RVI components) to validate the prototypical flow-induced vibration analysis basis for STP and to demonstrate conformance of the STP RVI components with the NRC's position in RG 1.20. UFSAR Section 3.9.2.6 provides the applicant's basis for correlating the data from flow-vibration behavioral test studies to the data obtained from the Sequoyah and Trojan instrument tests to demonstrate the conservatism in the behavioral test studies estimates.

The staff noted that LRA Section 4.3.3 states that the CLB did not include any flow-induced vibration analyses that would need to be identified as a TLAA for the LRA. It also states that any flow-induced vibration analyses in the CLB either did not involve an assessment of an applicable aging effect (i.e., did not conform to 10 CFR 54.3 Criterion 2) or were not based on time-dependent assumptions defined by the life of the plant (i.e., did not conform to 10 CFR 54.3 Criterion 3). The staff also noted that, although LRA Section 4.3.3 referenced the applicability of UFSAR Section 3.9.2.3, it did not mention that the applicant's flow-induced vibrational basis for the RVI components was based on consistency with the NRC position in RG 1.20 or that the flow-induced vibrational bases in UFSAR Sections 3.9.2.4 and 3.9.2.6 were also part of the applicant's RG 1.20 basis. The staff also noted that the applicant did not identify in LRA Section 4.3.3 that WCAP-8303-P-A, WCAP-8516-P-A, WCAP-8766-P-A, WCAP-9395-P, WCAP-9646, and WCAP-10865-P were being relied upon as part of the applicant's RG 1.20 conformance basis, and it did not provide an assessment on whether the analyses in these WCAP reports would need to be identified as TLAAs when compared to the six criteria for TLAAs in 10 CFR 54.3. The staff further noted that LRA Section 4.3.3 also did not mention that the applicant credits its plant-specific PWR Reactor Internals Program (i.e., LRA AMP B2.1.35) with the management of the aging effects that are applicable to the RVI components, including those from a flow-induced vibration mechanism (e.g., cracking or loss of material).

By letter dated September 22, 2011, the staff issued RAI 4.1-5, requesting details on how the applicant's consistency with RG 1.20 for flow-induced vibrations was accounted for in the current design basis. The staff also asked the applicant to explain whether any analyses that are part of this RG basis (when assessed against the six criteria for TLAAs in 10 CFR 54.3) would need to be identified as TLAAs for the LRA pursuant to the criterion in 10 CFR 54.21(c)(1). In RAI 4.1-5, Part 1, the staff asked the applicant to clarify which edition of RG 1.20 was being used as the current basis for assessing flow-induced vibrations of the RVI components and to provide a summary of how the information in UFSAR Sections 3.9.2.3, 3.9.2.4, and 3.9.2.5 is related to the information in other referenced UFSAR sections. In RAI 4.1-5, Part 2, the staff asked the applicant to identify which of the WCAPs in UFSAR Table 1.6-2 were currently being relied upon as part of the applicant's RG 1.20 basis. The staff also asked the applicant to provide a summary of all analyses, evaluations, or calculations that were included in WCAP reports as part of the RG 1.20 basis and to perform a comparison of these analyses, evaluations, or calculations (if any) to the six criteria for defining TLAAs in 10 CFR 54.3. In RAI 4.1-5, Part 3, the staff asked the applicant to justify whether or not the analyses, evaluations, or calculations provided in response to RAI 4.1-5, Part 2, would need to be identified as TLAAs for the LRA in accordance with the TLAA identification requirements in 10 CFR 54.21(c)(1).

The applicant responded to RAI 4.1-5, Parts 1, 2, and 3, by letter dated November 21, 2011. In its response to RAI 4.1-5, Part 1, the applicant stated that it is committed to the NRC regulatory position in RG 1.20, Revision 2 (May 1976). The applicant further stated that, under this basis, its units are "Non-Prototype, Category 1" plants that rely on the tests and analyses for evaluating the impacts of flow-induced vibrations on the structural integrity of RVI components at the three "prototype" Westinghouse units in the United States (i.e., as performed for the Indian Point Unit 2, Trojan, and Sequoyah Unit 1 reactors). The applicant clarified that UFSAR Section 3.9.2.3 specifically describes the portions of the analyses and tests at the "prototype" reactors that are applicable to the CLB and RG 1.20 conformance basis. The applicant clarified that UFSAR Section 3.9.2.4 specifically describes its basis for conforming to the regulatory position in RG 1.20, Revision 2, by demonstrating that the design differences between the applicant's reactor and the "prototype" reactors would not have any significant effect on the vibratory responses of the RVI components and by describing the pre-service inspections that would be performed during the initial startups of the applicant's units. The applicant clarified that UFSAR Section 3.9.2.5 is not related to the RG 1.20 consistency.

The staff found that the applicant resolved the administrative requests, which were addressed by RAI 4.1-5, Part 1, because the applicant clarified which version of RG 1.20 is being relied upon as part of the flow-vibrational analysis in the CLB. In addition, the applicant clarified how the design basis in the UFSAR addresses the applicant's flow-vibrational analysis basis for its RVI components. Therefore, the staff's concerns in RAI 4.1-5, Part 1—with respect to how the UFSAR establishes the design basis for conforming to the regulatory position in RG 1.20—are resolved.

In its response to RAI 4.1-5, Parts 2 and 3, the applicant stated that the following Westinghouse WCAP report bases[1] are included in the CLB consistent with the RG 1.20 recommendations: (1) WCAP-7879, (2) WCAP-8303-P-A, (3) WCAP-8516-P-A, (4) WCAP-8766-P-A, (5) WCAP-9395-P-A, (6) WCAP-9946, and (7) WCAP-10865. The applicant stated that the bases in these WCAP reports do not include any TLAAs because the reports do not include any analyses that are based on time-dependent assumptions defined by the life of the plant (i.e., the bases in the reports do not conform to Criterion 3 for identifying TLAAs in 10 CFR 54.3(a)).

The staff reviewed the applicant's response to RAI 4.1-5, Parts 2 and 3, to determine whether it provided a valid basis for concluding that the referenced WCAP reports do not include any analysis that would need to be identified as TLAAs for the LRA. The staff noted that the applicant identified WCAP-10865 as the report that established how the applicant is consistent with RG 1.20 and why the applicant does not need to perform vibratory functional testing of the RVI components. The staff noted that WCAP-10865 references many of the flow-induced vibration WCAPs that were issued in regard to the flow-vibrational studies performed at the

[1] These reports contain proprietary information. Therefore, the staff will not discuss the details of these reports in this SER to protect Westinghouse's privileged information in the WCAP reports in accordance with the privileged information withholding requirements in 10 CFR 2.390.

prototypical Westinghouse units (i.e., at the Trojan, Indian Point, Unit 2, and Sequoyah, Unit 1, nuclear plants). The staff confirmed that WCAP-10865 does not include any analyses that would need to be identified as TLAAs for the LRA because it only serves as a basis for why the WCAPs for the prototypical Westinghouse units could be used as the RG 1.20 consistency basis for the applicant's units.

The staff also determined that the applicant's response to RAI 4.1-5 provided an acceptable basis for concluding that the assessments in WCAP-8303-NP-A, WCAP-8516-P-A, and WCAP-9395-P did not include any TLAAs because the staff confirmed that the reports only summarized flow-vibration measurement test results and the acceptability of these results, and they did not involve any high-cycle modeling analyses that would need to be compared to the six criteria for TLAAs in 10 CFR 54.3.

However, the staff noted that the applicant stated that the methodologies in the WCAP-7879, WCAP-8766-P-A, and WCAP-9946 reports did include high-cycle modeling analyses, which evaluated the impact of flow-induced vibrations on the measured strains for the components. The staff further noted that, contrary to the applicant's determination, the high-cycle modeling analyses in these reports included a time dependency because the analyses assessed the strains in the components based on an assumed number of flow-induced vibration cycles. The staff noted that the analyses in the reports address applicable aging effects because the reports assess whether the flow-induced vibrations could induce high-cycle fatigue-induced cracking or changes in dimensions (i.e., strain-induced plastic deformation) in the components and whether the intended functions of the sister-plant RVI components that are within the scope of the WCAP reports would be impacted. The staff also noted that the applicant's response to RAI 4.1-5 indicated that the analytical bases in these WCAP reports were relied upon in the CLB as part of the applicant's basis for conforming to the recommended NRC position in RG 1.20. The staff confirmed that the sister-plant components in the analyses directly correlate to those RVI components that the applicant had identified as being within the scope of the AMR items in LRA Table 3.1.2-1.

Thus, the staff concluded that the analyses in these reports would meet Criteria 1, 2, 4, 5, and 6 for defining TLAAs in 10 CFR 54.3 for the following reasons:

- The analyses involve sister-plant RVI components that are being used as the RG 1.20 basis for analogous RVI components within the scope of the applicant's LRA.
- The analyses involve the effects of aging.
- The analyses are being relied upon as part of a safety-basis decision in the CLB for conforming to the NRC's regulatory position in RG 1.20.
- The analyses involve conclusions relative to the ability of the analogous RVI components to perform their intended safety functions.
- The WCAP reports are incorporated by reference in the UFSAR.

Hence, the staff concluded that the applicant's "absence of a TLAA" basis that cited Criterion 3—the conclusion that the assessments in the reports did not include time-dependencies—would only be acceptable if the applicant could establish that the time-dependent variable (i.e., high-cycle vibrations) in the report was not defined in terms of the life of the plant (e.g., a 40-year operating basis).

By letter dated February 15, 2012, the staff issued RAI 4.1-5a, requesting clarification on whether the analysis of vibratory cycles (the time-dependent parameter in the analyses) in the WCAP-7879, WCAP-8766-P-A, and WCAP-9946 reports was defined in terms of the life of the applicant's units (e.g., for a 40-year design life). The staff also requested further justification on why the analyses would not need to be identified as TLAAs for the LRA.

The applicant responded to RAI 4.1-5a by letter dated March 12, 2012. In its response, the applicant clarified that the high-cycle vibratory analyses in WCAP-7879, WCAP-8766-P-A, and WCAP-9946 are not considered dependent upon a time-dependent parameter defined by the life of the plant because the stress ranges associated with vibratory fatigue cycles are well below the lower bound stress endurance limit in which a high-cycle fatigue-induced failure could be postulated. The applicant stated that the RVI components within the scope of these reports could tolerate an infinite number of low-stress vibratory cycles. The applicant stated that the high-cycle vibratory fatigue analyses in WCAP-7879, WCAP-8766-P-A, and WCAP-9946 do not include any time dependency; therefore, high-cycle fatigue analyses in these reports do not need to be identified as TLAAs for the LRA because they do not conform to TLAA identification Criterion 3 in 10 CFR 54.3a.

The staff noted that the applicant's basis for concluding that high-cycle fatigue analyses in these WCAP reports are not TLAAs is based on the concept that the RVI components would not initiate high-cycle fatigue cracks if the stresses in the components were lower than that associated with the endurance limits for the materials of fabrication for the components. The staff finds that the basis provided in the response to RAI 4.1-5a is a valid basis for drawing this conclusion because the stresses associated with the high-cycle vibratory fatigue analyses for the RVI components within the scope of these reports would permit the components to withstand an extremely high number of low-stress, vibratory cycles beyond the number of vibratory cycles associated with the end of the period of extended operation. Additionally, the analyses would not conform to the TLAA Criterion 3 in 10 CFR 54.3(a), in that the analyses are not time-dependent analyses that are defined by the life of the plant (e.g., 40 years). Based on this evaluation, the staff concludes that the applicant provided an acceptable basis for concluding that there are not any time-dependent, high-cycle vibratory analyses for the RVI components that need to be identified as TLAAs for the LRA.

In addition, the staff noted that the applicant is crediting its PWR Reactor Internals Program (LRA AMP B2.1.35) as its condition monitoring program for managing cracking in the RVI components. Therefore, the staff has additional assurance that the applicant will have an acceptable AMP in place to manage cracking of RVI components during the period of extended operation. The staff's evaluation of the PWR Reactor Internals Program is provided in SER Section 3.0.3.3.2. Therefore, the staff's concerns described in RAIs 4.1-5, Parts 1, 2, and 3, and 4.1-5a are resolved.

<u>Absence of Ductility Reduction or Fracture Toughness Reduction TLAAs for Reactor Vessel Internal (RVI) Components</u>. In LRA Table 4.1-2, the applicant identified that its review of the CLB did not identify any time-dependent ductility reduction analyses or reduction of fracture toughness analyses for RVI components. Therefore, the applicant stated that the LRA does not need to include these types of TLAAs because the generic "ductility reduction of fracture toughness" TLAA in SRP-LR Table 4.1-3 is not applicable to or part of the CLB.

The staff reviewed the UFSAR for relevant information and confirmed that the UFSAR does not include or make any references to reduction of ductility analyses or reduction of fracture toughness analyses for the RVI components. The staff also noted that the applicant credits its

PWR Reactor Internals Program as the basis for managing the effects of aging during the period of extended operation and that the program manages loss of fracture toughness in the RVI components as a result of neutron irradiation embrittlement, void swelling, and thermal aging for RVI components made from cast austenitic stainless steel (CASS), precipitation hardened stainless steels, and X-750 material.

Based on its review, the staff finds that the applicant provided an acceptable basis for concluding that the LRA does not need to include a TLAA related to ductility reduction or reduction of fracture toughness because the staff has confirmed that the CLB does not currently include these types of analyses for the RVI components.

The staff's evaluation of the applicant's PWR Reactor Internals Program to manage reduction of fracture toughness in the RVI components is provided in SER Section 3.0.3.3.2.

<u>Absence of a Fatigue Analysis TLAA for the Containment Liner Plate</u>. LRA Section 4.6 states that the applicant's review of the CLB did not identify any fatigue analyses for the containment liner plate. The staff's evaluation of the applicant's conclusions is provided in SER Section 4.6.

<u>Absence of a Fatigue Analysis TLAA for the Containment Equipment Hatches</u>. LRA Section 4.6 states that the applicant's review of the CLB did not identify any fatigue analyses for the containment equipment hatches. The staff's evaluation of the applicant's conclusions is provided in SER Section 4.6.

<u>Absence of a Fatigue Analysis TLAA for the Containment Polar Crane Brackets</u>. LRA Section 4.6 states that the applicant's review of the CLB did not identify any fatigue analyses for the containment polar crane brackets. The staff's evaluation of the applicant's conclusions is provided in SER Section 4.6.

<u>Absence of TLAA on Reactor Vessel Circumferential Weld Inspection Relief (BWR)</u>. In LRA Table 4.1-2, the applicant stated that the TLAA associated with inspection relief of RPV circumferential welds does not apply to the applicant because STP is a PWR and the analysis only applies to BWRs; thus, this item is not applicable to its CLB.

The staff noted that SRP-LR Section 4.2 identifies that circumferential weld and axial weld probability of failure analyses that are used in support of 10 CFR 50.55a relief requests from applicable inservice inspection (ISI) requirements (i.e., those that are mandated by 10 CFR 50.55a(g)(6) and applicable ASME Code Section XI Category B-A inspection requirements) are only applicable to BWRs. The staff also noted that the applicant's UFSAR identify the applicant's units as four-loop Westinghouse design PWRs. Based on this review, the staff finds that the applicant provided an acceptable basis for concluding that the LRA does not need to include any RPV circumferential weld or axial weld probability of failure TLAAs because these types of assessments are only applicable to BWRs, and the staff confirmed that the applicant's units are PWRs.

<u>Relevance of UFSAR Appendix 9A to the LRA</u>. As part of its review, the staff noted that UFSAR Appendix 9A provides the applicant's "Assessment of the Potential Effects of Through-Wall Cracks in the ECWS Piping." The staff noted that UFSAR Appendix 9A states that through-wall cracks were identified in the applicant's ECW system piping (aluminum bronze components), which were initiated by pre-existing weld defects and propagated by a dealloying phenomenon. The staff noted that UFSAR Appendix 9A states, "STPEGS has analyzed the effects of the cracking and found that the degradation is slow so that rapid or catastrophic failure is not a

consideration, and determined that the leakage can be detected before the flaw reaches a limiting size that would affect the operability of the [ECW system].”

The staff also noted that UFSAR Appendix 9A states that potential effects of leakage in the ECW system piping were assessed for the following impacts at the plant:

- internal flooding in rooms containing these pipes and other rooms that receive drains from these sources
- electrical shorts or grounds caused by water spray from the crack
- reduction in ECW system flow through the heat exchangers served by the affected ECW system piping train
- water losses from the essential cooling pump not accounted for in the existing analysis
- possible effects on the transient pressures when the pump is started or stopped

The staff also noted that UFSAR Appendix 9A then referenced the following flaw-related evaluations and analyses that were performed to support the applicant’s basis that any potential leakage from the ECW system piping would be detected before a fast fracture of the piping would occur:

- HL&P Laboratory Report MT-3512A, “Evaluation of Cracked Elbow-to-Nozzle Weld from South Texas Project Unit 1 Essential Cooling Water System”
- HL&P Laboratory Report MT-3512B, “Evaluation of Cracked Aluminum Bronze Pipe-to-Pipe Weld from South Texas Project Unit 2 Essential Cooling Water System”
- Aptech Calculation No. AES-C-1630-2, “Calculation of Critical Bending Stress for Flawed Pipe Welds in the ECW System”

The staff noted that the MT-3512A, MT-35612B, and AES-C-1630-2 evaluations referenced in UFSAR Appendix 9A appeared to be using an leakage detection basis (LDB)-type of logic (leakage detection basis) to assess the potential flaws in the aluminum bronze ECW system components, and the apparent cause basis for UFSAR Appendix 9A was predicated on the conclusion that the existing flaws in the aluminum bronze components would be propagated by an aluminum bronze dealloying flaw growth mechanism. The staff also noted that the applicant did not mention UFSAR Appendix 9A and the three associated evaluations, and it did not provide an assessment in the LRA on whether these evaluations would need to be identified as TLAAs for the LRA, in accordance with 10 CFR 54.21(c)(1), when assessed against the six criteria for defining TLAAs in 10 CFR 54.3.

During the staff’s onsite audit of the applicant’s LRA AMPs the week of June 20-24, 2011, the staff noted that the applicant’s LDB-type approach to the assessment of potential flaws in aluminum bronze ECW system components appeared to be based on three additional assessments that were not referenced as being relevant in UFSAR Appendix 9A:

(1) a vendor-specific leakage/seepage and soil diffusion calculation
(2) an applicant-specific leakage/seepage and soil diffusion calculation that was used to confirm the conclusions in the vendor-specific calculation

(3) an applicant-specific engineering report that summarized the applicant's results in the above vendor-specific and applicant-specific calculations, which appears to have been the basis for the conclusions in UFSAR Appendix 9A

The staff also noted that these evaluations did not include any flaw tolerance evaluations, which support the applicant's claim that a leak in an ECW system aluminum bronze component would be detected before a catastrophic fast fracture in the system's aluminum bronze piping.

The staff finds that if the leakage detection basis in UFSAR Appendix 9A was to be relied upon for aging management, it would need to be supported by an appropriate time-dependent flaw tolerance evaluation to demonstrate that the critical flaw size for the applicable piping would not be less than the flaw size that would lead to a detectable leak at the soil or soil/gravel surface. Furthermore, if the critical crack size was greater than the flaw size that would lead to a detectable leak (i.e., the leak-detection size), the analysis would need to demonstrate that a flaw the size of the leak-detection size would not grow and reach the critical flaw size limit for the piping before the time it would take the applicant to detect such a leak at the soil surface or soil/gravel surface. In addition, any evaluations used to support this type of analysis would be relevant, even if the applicant had repaired the relevant indications pursuant to applicable ASME Code Section XI repair criteria, because the evaluations would still be needed to support the applicant's basis that visual examinations of the piping would be capable of detecting leakage from aluminum bronze ECW system components before a postulated fast fracture (i.e., catastrophic failure) of the piping.

The staff noted that the basis in UFSAR Appendix 9A was predicated on the assumption that flaw growth was occurring from an aluminum bronze dealloying mechanism. However, upon its audit of the HL&P MT-3512A and MT-35612B lab reports, the staff noted that the lab reports also indicated the occurrence of some failure striations in the weld failure photographs that could indicate that the flaws in the aluminum bronze materials had also been, at times, propagating by a low-cycle to high-cycle fatigue growth mechanism.

By letter dated September 22, 2011, the staff issued RAI 4.1-6, requesting that the applicant provide additional clarifications on the UFSAR Appendix 9A basis and whether the LRA should have included any relevant UFSAR Appendix 9A-based flaw tolerance TLAAs for the ECW system in accordance with the identification requirement in 10 CFR 54.21(c)(1). Specifically, in RAI 4.1-6, Part 1, the staff asked the applicant to explain why the applicable vendor-specific and applicant-specific leakage seepage and soil diffusion analyses, and the applicable engineer report, for the ECW system aluminum bronze components had not been referenced as applicable reports to the UFSAR Appendix 9A basis in the reference section of that UFSAR appendix. In RAI 4.1-6, Part 2, the staff asked the applicant to clarify whether the vendor-specific and applicant-specific leakage seepage and soil diffusion analyses, used for the UFSAR Appendix 9A safety basis, were supported by any flaw tolerance analyses to demonstrate that the critical flaw size for the applicable piping would not be less than the flaw size that would lead to a detectable leak at the soil or soil/gravel surface. The staff asked the applicant to clarify, if the limiting critical flaw size was greater than the flaw size that would lead to a detectable leak (i.e., the leak-detection size), whether a flaw the size of the leak-detection size would not grow and reach the critical flaw size for the piping before it would be detected at the soil surface or soil/gravel surface. The staff also asked the applicant to clarify whether such a flaw tolerance analysis, if performed as part of the CLB, would need to be identified as a TLAA for the LRA in accordance with the criteria in 10 CFR 54.21(c)(1). In RAI 4.1-6, Part 3, the staff asked the applicant to perform a comparison of the evaluations in HL&P Report Nos. MT-3512A and MT-3512B and in Aptech Calculation No. AES-C-1630-2 to the six criteria for defining analyses as TLAAs in 10 CFR 54.3. The staff also asked the applicant to provide its basis on why any evaluations, analyses, or calculations in these reports would not need to be identified as TLAAs pursuant to the requirements in 10 CFR 54.21(c)(1). In RAI 4.1-6, Part 4, the staff asked the applicant to justify why the basis in UFSAR Appendix 9A did not need to consider and evaluate the possibility of fatigue flaw growth in these aluminum bronze components.

The applicant responded to RAI 4.1-6, Parts 1-4, in a letter dated December 8, 2011. In its response to RAI 4.1-6, Part 1, the applicant stated that the applicable leakage analysis is Calculation No. CC-5089, which is referenced on page 9A-2 of UFSAR Appendix 9A, and that the vendor-specific analysis is included as an attachment to Calculation No. CC-5089. The staff noted that the UFSAR Appendix 9A basis relied on more than one vendor-specific or applicant-specific analysis. The staff noted that Tables 1 and 2 of the applicant's letter, dated December 8, 2011, listed all of the ECW aluminum bronze cast components and piping components that had degraded by either a selective leaching (dealloying) or crack propagation mechanism as part of its response to another RAI issued on this UFSAR basis (RAI B2.1.37-1). The staff also noted that these tables referenced the applicable engineering analyses, material test reports, and condition reports that were issued relevant to the applicant's leakage detection basis for these components. Therefore, based on the collective responses to RAIs B2.1.37-1 and 4.1-6, Part 1, the staff found that the applicant provided a definitive basis on the types of reports, calculations, and analyses that were being relied upon as part of the applicant's UFSAR Appendix 9A leakage management basis for the ECW system. The staff's evaluation of the applicant's UFSAR Appendix 9A basis to manage loss of material and cracking in the ECW system is provided in SER Section 3.0.3.3.3. Therefore, the staff's concerns expressed in RAI 4.1-6, Part 1, are resolved.

In its response to RAI 4.1-6, Parts 2 and 3, the applicant stated that the leakage detection basis for UFSAR Appendix 9A was based on the leakage detection threshold that was established in the applicant's Calculation No. CC-5089 and that the critical crack was established in Aptech Calculation No. AES-C-1630-2. The applicant also stated that the crack length needed to produce a leak rate of 10 gallons per minute (gpm) was less than the critical crack length established in Aptech Calculation No. AES-C-1964-7. The applicant further stated that the referenced calculations do not involve any predictions of wastage (loss of material) progression by a selective leaching mechanism or flaw growth by a crack propagation mechanism such as fatigue or stress corrosion cracking (SCC). The applicant clarified that laboratory examinations indicate that a pre-existing crack at the root of a weld will support dealloying at the crack tip and that the crack would propagate through the dealloyed material until non-dealloyed material was reached. The applicant stated that the process could repeat itself until the crack extends fully through the wall of the component. However, the applicant also stated that the rate at which a crack would propagate could not be determined for this type of process. The applicant stated that since the calculations do not involve time-dependent assumptions, the analyses in the calculations do not conform to the criterion in 10 CFR 54.3(a), Criterion 3, and do not need to be identified as TLAAs for the LRA.

The staff reviewed the calculations in these documents and determined that any flaw tolerance evaluations in the Aptech calculations used limit-load or linear-elastic fracture mechanics analyses for the crack stability analyses. The staff also noted that these analyses only assessed a conservatively assumed through-wall flaw size against the critical crack size for the analyzed component. The staff further noted that the flaw tolerance analyses did not include any time-dependent flaw growth calculations (e.g., growth by fatigue or by SCC) for the assumed flaws. The staff noted that the leak detection analysis basis in Calculation No. CC-5089 would not meet the definition of a TLAA because the period analyzed did not fully cover a 40-year life basis. The staff concluded that the applicant provided an acceptable basis for stating that the flaw tolerance analyses in these reports are not TLAAs, because the analyses do not involve time-dependent assumptions defined by the life of plant, and thus do not conform to Criterion 3 in 10 CFR 54.3(a). Therefore, the staff's concerns expressed in RAI 4.1-6, Parts 2 and 3—with respect to identifying whether Calculation No. CC-5089 and the flaw tolerance evaluations in the Aptech calculations need to be identified as TLAAs for the LRA—are resolved.

The staff also noted that some of the applicant's material test reports indicated that some aluminum bronze components in the ECW system had failed and leaked as a result of an SCC propagation mechanism, sometimes with and sometimes without dealloying as a contributing cause for the failure of the components. Thus, the staff questioned whether the applicant's leakage detection basis for aluminum bronze components in Calculation No. CC-5089 is adequate because the supporting flaw tolerance bases did not account for potential SCC-initiated growth of the analyzed flaws in the calculations. The staff did not have sufficient assurance that the leaks from the analyzed components would be detected at the soil surface prior to a complete guillotine-type failure of the components because the flaw tolerance basis did not account for SCC-initiated growth of the analyzed flaws in the calculations. Additionally, the applicant did not sufficiently demonstrate that leakage from a pre-existing through-wall flaw would be detected before a full failure of an aluminum bronze component in the ECW system. The staff's concerns and evaluations related to the potential of SCC-initiated crack growth are provided in SER Section 3.0.3.3.3. The staff's evaluation in SER Section 3.0.3.3.3 includes an assessment of whether additional inspections and time-dependent flaw tolerance evaluations will be needed for the cast aluminum bronze components and aluminum bronze piping components in the ECW system during the period of extended operation.

In its response to RAI 4.1-6, Part 4, the applicant stated that, although fatigue is a phenomenon that could occur in any piping system, selective leaching (dealloying) was the main contributing factor for the aluminum bronze components in the ECW system. The applicant also stated that the laboratory material test report photographs of the failed aluminum bronze components did not exhibit any evidence that fatigue was a contributing cause for the components that had failed by a crack growth mechanism. The staff reviewed the photographs in the material test reports and determined that the components had failed either by a selective leaching (dealloying) pitting mechanism or by crack initiation and growth where SCC was the main contributing mechanism for crack growth (i.e., with or without synergistic contributions of dealloying on the crack growth mechanism or on the fracture toughness property of the aluminum bronze material in the component). The staff concludes that the applicant provided an acceptable basis for concluding that fatigue was not a contributing mechanism for the failures in the aluminum bronze ECW components. Therefore, the staff's concerns expressed in RAI 4.1-6, Part 4, are resolved.

Based on its review, the staff concludes that the applicant does not need to identify a TLAA relative to the UFSAR Appendix 9A basis in the CLB because the analyses and calculations that are relied upon in the CLB are not based on any time-dependencies defined by the life of the plant, and therefore do not satisfy TLAA identification Criterion 3 in 10 CFR 54.3(a)

SER Section 3.0.3.3.3 provides the staff's evaluation of the applicant's plans for managing loss of material by dealloying or cracking in the aluminum bronze ECW components.

4.1.2.2 Identification of Exemptions in the LRA

As required by 10 CFR 54.21(c)(2), the applicant must identify and evaluate all exemptions granted in accordance with 10 CFR 50.12 that are based on a TLAA and justify their use during the period of extended operation. The LRA states that each active exemption was reviewed to determine whether it was based on a TLAA.

The staff also reviewed the applicant's CLB to see if it included any exemptions that were granted in accordance with 10 CFR 50.12 and that were based on a TLAA. The staff's review included a review of the current operating license for the facility and the applicant's UFSAR. The staff's review also included an "exemption" keyword search of the NRC's Agencywide Documents Access and Management System (ADAMS).

LRA Section 4.1.4 states that the CLB includes seven exemptions that were granted pursuant to the provisions in 10 CFR 50.12. Of these exemptions, the LRA states that only one exemption is based in part on a time-limited aging analysis—the LBB analysis (which forms the applicant's basis for complying with "dynamic effect" analysis relaxation provisions in 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 4) was the only exemption that was based in part on a TLAA. The applicant stated that the LBB analysis would be needed for the period of extended operation to justify continued removal of the dynamic effect analyses from the scope of the UFSAR and to justify removal of the pipe whip restraints for the scope of the reactor coolant loop design during the period of extended operation.

The applicant indicated that the LBB analysis is identified as a TLAA in LRA Section 4.3.2.11. The staff confirmed that LRA Section 4.3.2.11 identifies the LBB analysis as a TLAA and that the LRA section gives the applicant's basis for accepting the LBB TLAA in accordance with the acceptance criterion in 10 CFR 54.21(c)(1)(iii). The staff also confirmed that the effect of fatigue flaw growth on the intended pressure boundary function of the main coolant loop, and its impact of compliance with GDC 4, will be adequately managed for the period of extended operation. The staff evaluated the LBB TLAA and the basis for accepting this TLAA in accordance with 10 CFR 54.21(c)(1)(iii) in SER Section 4.3.2.11.

The staff noted that the applicant did not identify any additional exemptions in the CLB that were granted pursuant to the provisions in 10 CFR 50.12 and were based on a TLAA. The staff could not determine whether the remaining six exemptions mentioned in LRA Section 4.1.4 would need to be identified as exemptions in the LRA pursuant to 10 CFR 54.21(c)(2) because the applicant did not identify upon which regulations the exemptions were based. The staff also noted that, in LRA AMP B2.1.15, "Reactor Vessel Surveillance," the applicant stated that an exemption was granted in the original license from meeting the requirements in 10 CFR Part 50, Appendix H. However, the applicant did not provide any discussion in the LRA on why this exemption would not need to be identified in the LRA pursuant to the criteria in 10 CFR 54.21(c)(2).

Based on the results of its ADAMS Legacy Library search, the staff noted that on May 4, 1999 (NRC Microfiche Accession No. 9905110094, Microfiche Address A7956, pages 355-359), the staff granted an exemption that permitted the applicant to apply the alternative methods in ASME Code Case N-514 as the basis for establishing the LTOP system pressure lift and arming temperature set points for the power-operated relief valves (PORVs) that are credited for relieving pressure when the LTOP system is actuated. Specifically, the staff noted that, based on the Code Case methodology, this exemption permits the applicant to set the LTOP system pressure lift set points for the PORVs to a pressure value that is equivalent to 110 percent of the limiting pressure established in the approved P-T limits curve for the system's arming temperature set point. The staff also noted that the exemption granting the use of Code Case N-514 also permits the applicant to set the arming temperature based on the Code Case's arming temperature set point methodology.

In addition, the staff noted that in LRA Section 4.2 the applicant identified P-T limit analyses for Units 1 and 2 as TLAAs in the LRA. The staff also noted that the LTOP system set points are currently within the scope of TS limiting condition of operation (LCO) 3.4.9.3 and surveillance requirement (SR) 4.4.9.3, and the P-T limits are currently within the scope of LCO 3.4.9.2 and SR 4.4.9.2.

By email dated December 3, 2010, the staff issued RAI 4.1-1 to the applicant, requesting further clarification on why the exemption allowing the use of Code Case N-514 (i.e., the exemption on the LTOP methodology) had not been identified as an exemption that was based on a TLAA. The applicant responded to RAI 4.1-1 by letter dated December 9, 2010. In its response, the applicant stated that the exemption regarding use of Code Case N-514 should have been identified as an exemption for the LRA that conforms to the exemption identification requirement in 10 CFR 54.21(c)(2). The applicant also amended the LRA to add the exemption on Code Case N-514 as an exemption that was based on a TLAA. The applicant clarified that the exemption would be applied during the period of extended operation and that the basis for accepting both P-T limit and LTOP TLAAs during the period of extended operation is provided in LRA Sections 4.2.4 and 4.2.5, respectively, and includes the application of the exemption on the use of the Code Case to the LTOP methodology. The staff finds that the applicant resolved the concerns raised in RAI 4.1-1 because the applicant amended the LRA to include the exemption for Code Case N-514 as an exemption that is based on a TLAA and because this conforms to the exemption criterion requirement in 10 CFR 54.21(c)(2). The staff's evaluation of the LTOP TLAA is provided in SER Section 4.2.5. The staff's evaluation includes the basis for applying the exemption on the use of Code Case N-514 to the LTOP methodology.

In addition, the staff noted that, by Letter No. NOC-AE-000518, dated July 13, 1999, and as supplemented by letters dated October 14 and 22, 1999; January 26 and August 31, 2000; January 15, 18, and 23, 2001; March 19, 2001; and May 8 and 21, 2001, the applicant requested several other exemptions pursuant to the criteria in 10 CFR 50.12. Some of these exemptions were based on risk-informed approaches, but the staff was not able to confirm which of these were in the LRA. Therefore, the staff could not: (a) identify how many exemptions had been granted to the applicant in the CLB pursuant to the criteria in 10 CFR 50.12; (b) determine the appropriate regulations that the specific exemptions were based on, and what the exemptions involved; nor (c) identify how many of the exemptions would need to be identified as TLAAs in the LRA.

By letter dated September 22, 2011, the staff issued RAI 4.1-7, requesting further clarifications on the exemptions that the applicant referenced in LRA Section 4.1.4. In RAI 4.1-7, Part 1, the staff asked the applicant to identify all exemptions that were granted in accordance with the criteria in 10 CFR 50.12, and, of these exemptions, to identify the regulation for which each exemption was requested, summarize what the exemption involved, and state whether it remained in effect for the CLB. In RAI 4.1-7, Part 2, the staff asked the applicant to justify why each of the exemptions discussed in the response to Part 1 of RAI 4.1-7 would not need to be identified as an exemption in the LRA, in accordance with the exemption identification criterion in 10 CFR 54.21(c)(2). The staff also asked the applicant to account for the exemption to the requirements in 10 CFR Part 50, Appendix H, which was referred to in LRA Section B2.1.15, and the risked-informed exemptions that were requested in the applicant's letter of July 13, 1999.

The applicant responded to RAI 4.1-7, Parts 1 and 2, in a letter dated November 21, 2011. In its response, the applicant included a table that identified all of the regulatory exemptions that were granted to the applicant in accordance with the requirements in 10 CFR 50.12 and summarized the bases for these exemptions in the CLB. The table also included the applicant's bases for comparing the exemptions to the NRC's exemption identification criteria in 10 CFR 54.21(c)(2) and for concluding whether the exemptions were based on a TLAA. The applicant also amended LRA Section 4.1.4 for consistency with its RAI response. The following paragraphs discuss the exemptions in more detail.

The applicant identified that the CLB includes an exemption from the requirements in 10 CFR 70.24 for criticality monitoring during spent fuel handling operations. The applicant stated that the NRC's exemption granted permits the applicant to perform spent fuel handling operations without the use of any criticality monitoring equipment. The exemption was granted because the applicant had adequately demonstrated that the probability of a criticality accident would be sufficiently low during spent fuel handling operations by meeting seven operational criteria. The applicant also stated that these criteria did not involve any time-dependent parameters. The staff confirmed that the NRC's granting of the fuel handling operation exemption was based only on the applicant's conformance with seven fuel handling operational criteria and that the exemption was not based on any analysis that conformed to a TLAA. Based on this review, the staff concludes that the exemption from 10 CFR 70.24 does not need to be identified as an exemption in the LRA because the granting of the exemption is not based on an analysis that is a TLAA. Therefore, concerns raised in RAI 4.1-7 with respect to this exemption are resolved.

The applicant stated that the exemption from 10 CFR Part 50, Appendix J, containment leak rate testing requirements was in relation to compliance with the leak rate testing requirements in paragraph III.D.2(b)(ii) of the appendix. This paragraph requires full pressure testing of the air locks following opening during periods when containment integrity is not required (i.e., during Operating Modes 5 or 6). The applicant stated that the exemption permits the applicant to use the 10 CFR Part 50, Appendix J, paragraph III.D.2(b)(iii) seal leakage test as an alternative to the full pressure test required by paragraph III.D.2(b)(ii) of the appendix. The applicant stated that the exemption is based on the NRC's acceptance of the position that, if the tests required by paragraphs III.D.2.(b)(i) and III.D.2(b)(iii) are current and if maintenance is performed on the air lock such that it is properly sealed, then there is no reason to expect the air lock to leak excessively. The applicant stated that, as such, this exemption is not based on any analysis that would need to be identified as a TLAA for the LRA. The staff noted that the applicant's description of the Appendix J exemption confirms that the exemption was based solely on substituting one 10 CFR Part 50, Appendix J requirement for another, which may be done as long as the applicant continues to perform appropriate maintenance on the containment air locks. The staff noted that the applicant's discussion of the exemption demonstrates that the exemption is not based on any analysis that would need to be identified as a TLAA. Therefore, the staff concludes that the exemption from the Appendix J testing requirements does not need to be identified as an exemption for the LRA because the exemption is not based on any analysis that is a TLAA. Therefore, the staff concerns raised in RAI 4.1-7 with respect to this exemption are resolved.

The applicant stated that the CLB includes an exemption from the requirements in 10 CFR 50.71(e) with regard to the schedule for reporting UFSAR revisions to the NRC.

The staff noted that this exemption involves relaxations in schedule only and is not based on an analysis that is a TLAA. Based on its review, the staff concludes that the exemption from 10 CFR 50.71(e) does not need to be identified as an exemption for the LRA because the granting of the exemption is not based on an analysis that is a TLAA for the LRA. Therefore, concerns raised in RAI 4.1-7 with respect to this exemption are resolved.

The applicant stated that the CLB includes an exemption from the requirements in 10 CFR Part 50, Appendix A, GDC 4, for analyzing dynamic effects associated with a postulated rupture of reactor coolant pressure boundary (RCPB) piping. The applicant stated that the exemption granted from GDC 4 is based on a TLAA because it is based on the applicant's LBB analysis, which is identified as a TLAA in LRA Section 4.3.2.11. The staff's basis for granting the exemption from the requirements of GDC 4 has been previously discussed and evaluated above in this section. Therefore, the concerns raised in RAI 4.1-7 with respect to this exemption are resolved.

The applicant stated that the CLB includes an exemption from the requirements in 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," allowing use of ASME Code Case N-514 for the pressure lift and temperature actuation setpoints on the applicant's LTOP system. The applicant identified that this exemption is based on the applicant's P-T limits TLAA, as described in LRA Section 4.2.5. Therefore, concerns raised in RAI 4.1-7 with respect to this exemption are resolved.

The applicant also identified that the CLB includes an exemption from certain requirements in 10 CFR 50.46 and 10 CFR Part 50, Appendix K. The staff confirmed that the exemption permitted the applicant to use Optimized ZIRLO™ as the fabrication materials for fuel cladding on up to eight lead test assemblies containing fuel rods, guide thimble tubes, and instrumentation tubes instead of the already-approved ZIRLO™ material approved for the facility. The staff also confirmed that the granting of the exemption was not based on an analysis that conforms to the definition of a TLAA in 10 CFR 54.3. Based on this review, the staff concludes that the exemption from 10 CFR 50.46 and 10 CFR Part 50, Appendix K, does not need to be identified in the LRA because the granting of the exemption is not based on an analysis that is a TLAA for the LRA. Therefore, concerns raised in RAI 4.1-7 with respect to this exemption are resolved.

The applicant also identified that the CLB includes an exemption that was requested in accordance with the risk-informed regulation in 10 CFR 50.69 from meeting specific requirements in 10 CFR Parts 21, 50, and 100 and was granted in accordance with the exemption provisions in 10 CFR 50.12. The applicant stated that the "non risk significant" (NRS) and "low safety significance" (LSS) components within the scope of the special exemption no longer fall within the scope of the EQ of electrical component requirements in 10 CFR 50.49. However, the applicant also stated that the qualification of the safety-related components at the facility is still part of the CLB and remains within the scope of the applicant's EQ requirements and that the exemption is based in part on the EQ TLAA that is given in LRA Section 4.4.

The staff noted that the special exemption requests from meeting the specific requirements in 10 CFR Parts 21, 50, and 100 were approved in the NRC's SE dated August 3, 2001

(ADAMS Accession Nos. ML011990368 and ML012040470) and granted 10 CFR 50.12-based exemptions from the following requirements:

- 10 CFR Part 50, Appendix B, quality assurance requirements
- 10 CFR 50.55a requirements for inservice testing and inservice inspection
- 10 CFR 50.49 requirements for EQ of safety-related electrical equipment

The staff noted that, of these exemptions, the exemption from the EQ requirements in 10 CFR 50.49 was the only exemption that was based on a TLAA. The staff confirmed that the applicant included its EQ TLAA in LRA Section 4.4. The staff evaluated the applicant's basis for accepting the EQ TLAA in accordance with 10 CFR 54.21(c)(1)(iii) and the impact that this TLAA will have on the 10 CFR 50.49-based "special treatment requirements" exemption in SER Section 4.4. The 10 CFR 50.49-based "special treatment requirements" exemptions apply to the applicant's request for an exemption from 10 CFR Part 49(b), to exclude certain low-safety significance (LSS) and non-risk significant (NRS) components from the scope of electrical equipment important to safety under 10 CFR 50.49(b) (see Section 4.4 of this SER for a more detailed discussion on this exemption). Based on its review, the staff concluded that applicant met the requirements in 10 CFR 54.21(c)(2) because the applicant's letter of November 21, 2011, appropriately amended the LRA to identify the 10 CFR 50.49-based "special treatment requirements" exemption as an exemption that was granted in accordance with 10 CFR 50.12 and that was based on a TLAA. Therefore, concerns raised in RAI 4.1-7, with respect to 10 CFR 50.49-based "special treatment requirements" exemption, are resolved.

The staff confirmed that the remaining "special requirements" exemptions had risk-informed bases that were reviewed and approved by the staff in accordance with the exemption approval criteria in 10 CFR 50.12. The staff also confirmed that the non-10 CFR 50.49 "special treatment requirements" exemptions were not based on any time-dependent analyses that would need to be identified as TLAAs in the LRA. Based on its review, the staff concludes that the applicant does not need to identify these remaining "special treatment requirements" exemptions as exemptions for the LRA because they are not based on any analyses that would need to be identified as TLAAs in the LRA. Therefore, they do not fall within the scope of exemptions that would need to be identified in accordance with the requirements in 10 CFR 54.21(c)(2), and the staff's concerns expressed in RAI 4.1-7, with respect to the non-10 CFR 50.49-based "special treatment requirements" exemptions, are resolved. RAI 4.1-7 are resolved with respect to compliance with 10 CFR 54.21(c)(2) exemption identification requirements.

In its letter dated November 21, 2011, the applicant clarified that the CLB does not include any exemptions from 10 CFR Part 50, Appendix H, RV Surveillance Program requirements. The applicant stated that the statement in LRA Section B2.1.15 regarding an exemption in the program was in reference to a footnote in the UFSAR on page 5.3-4. The applicant stated that the footnote clarifies that weld coupons for the program are not samples from specimens taken from the actual manufacturing of the vessel but, instead, represent weld metal that is identical to the wire heat and flux lot used to fabricate the RV intermediate-to-lower-shell girth weld. The staff noted that the footnote on UFSAR page 5.3-4 demonstrates compliance with the RV surveillance program requirements in 10 CFR Part 50, Appendix H, because it documents that the program includes RV weld test coupons that are representative of the RV beltline welds. Therefore, the staff has confirmed that the clarification on the UFSAR section does not constitute an exemption from 10 CFR Part 50, Appendix H, requirements. The staff's concerns in RAIs 4.1-1 and 4.1-7 with respect to compliance with 10 CFR Part 50, Appendix H, requirements are resolved, because the CLB does not include any exemption from those requirements.

Based on the information provided by the applicant, the amendment to LRA Section 4.1.4, and the scope of staff's review, the staff concludes that, in accordance with 10 CFR 54.21(c)(2), the LRA includes the appropriate exemptions that were granted pursuant to 10 CFR 50.12 and that were based on a TLAA.

4.1.3 Conclusion

Based on its review, the staff concludes the applicant provided an acceptable list of TLAAs, as required by 10 CFR 54.21(c)(1). The staff confirmed that, as required by 10 CFR 54.21(c)(2), the applicant identified the appropriate exemptions that were granted pursuant to 10 CFR 50.12 and that are based on a TLAA.

4.2 Reactor Vessel Neutron Embrittlement Analysis

During plant service, neutron irradiation reduces the fracture toughness of ferritic steel in the beltline region of the RV. As fracture toughness decreases with cumulative fast neutron exposure, the material's resistance to crack propagation decreases. The projected reduction in fracture toughness is a function of fluence, temperature and certain material parameters (e.g., weld or base metal, copper and nickel content). Areas of review to ensure that the RV materials have adequate fracture toughness to prevent brittle failure during normal and off-normal operating conditions are as follows:

- Neutron Fluence Values (Section 4.2.1)
- Pressurized Thermal Shock (Section 4.2.2)
- Upper-Shelf Energy (Section 4.2.3)
- Pressure-Temperature (P-T) Limits (Section 4.2.4)
- Low Temperature Overpressure Protection (Section 4.2.5)

4.2.1 Neutron Fluence Values

4.2.1.1 Summary of Technical Information in the Application

LRA Section 4.2.1 describes the applicant's TLAA for neutron fluence. LRA Section 4.2.1 states that the fluence values which cover the period of extended operation were projected based on the results of the Capsule V and U analyses for STP, Units 1 and 2, respectively. The revised fluences were determined with transport calculations using the DORT discrete ordinates code and the BUGLE-96 cross section library, which is derived from ENDF/B-VI. The neutron transport and dosimetry evaluation methodologies follow the guidance and meet the requirements of the most recent issue of RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (March 2001). The fluence projections were developed with dosimeter data for which all measurement-to-calculation comparisons fall well within the 20 percent limit, which is specified as the acceptance criteria in RG 1.190.

LRA Table 4.2-1 provides 60-year peak projections for neutron fluence values for each unit.

The applicant dispositioned the neutron fluence TLAA in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

4.2.1.2 Staff Evaluation

The staff reviewed LRA Section 4.2.1 and the neutron fluence TLAA, to confirm pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions will be adequately managed for the period of extended operation. The staff reviewed the applicant's TLAA and the corresponding disposition consistent with the review procedures in SRP-LR Section 4.2.3. The applicant stated that the neutron fluence calculations adhere to the NRC position detailed in RG 1.190, and it described the technique used to determine the STP, Units 1 and 2, neutron fluence values. To confirm this information, the staff reviewed the following reports, which provide additional details about the neutron fluence calculations and RV dosimetry analyses:

- WCAP-16093, "Analysis of Capsule V from the South Texas Project Nuclear Operating Company South Texas Unit 1 Reactor Vessel Radiation Surveillance Program," (ADAMS Accession No. ML072500123)
- WCAP-16149, "Analysis of Capsule U from the South Texas Project Nuclear Operating Company South Texas Unit 2 Reactor Vessel Radiation Surveillance Program," (ADAMS Accession No. ML072490211)

Chapter 6 of each report describes the neutron fluence calculations and states that they were performed using the nuclear data described above and that the uncertainties were within the RG 1.190 acceptance criterion of 20 percent. The reports provide additional information concerning the neutron transport calculations. The DORT calculations were used to perform a 3D flux synthesis, and the calculations employed a P_5 Legendre polynomial expansion and S_{16} angular quadrature.

The staff finds the applicant's neutron fluence calculations acceptable because the applicant performed the neutron fluence calculations per RG 1.190, and the fluence projections fall within the 20 percent limits of the RG. Fluence is managed for the period of extended operation by the Reactor Vessel Surveillance Program, which is described in LRA Section B2.1.15. The validity of these parameters, and the analyses that depend upon them, will be managed to the end of the period of extended operation. The staff finds that the applicant demonstrated pursuant to 10 CFR 54.21(c)(1)(iii), that the neutron fluence will be adequately managed for the period of extended operation.

4.2.1.3 UFSAR Supplement

LRA Section A3.1.3 provides the UFSAR supplement summarizing the TLAA evaluation of LTOP. The staff reviewed LRA Section A3.1.3 consistent with the review procedures in SRP-LR Section 4.2.3.2, which state that the applicant should provide a summary description of the evaluation of the RV neutron embrittlement TLAA, equivalent to that in SRP-LR Table 4.2-1. Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.2.2.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the TLAA for neutron fluence, as required by 54.21(d).

4.2.1.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging caused by neutron fluence will be adequately managed for the period of extended operation.

4.2.2 Pressurized Thermal Shock

4.2.2.1 Summary of Technical Information in the Application

LRA Section 4.2.2 describes the PTS evaluation of the STP, Units 1 and 2, RV beltline and extended beltline materials for the period of extended operation, against the screening criteria established in accordance with the PTS Rule, 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

For STP, Unit 1, the applicant stated that the limiting PTS reference temperature (RT_{PTS}) material is intermediate shell R1606-3 with an RT_{PTS} value of 83.6 °F at 54 effective full power years (EFPY), based on the information provided in LRA Table 4.2-2. For STP, Unit 2, the applicant stated that the limiting RT_{PTS} material is intermediate shell R2507-2 with an RT_{PTS} value of 63.7 °F at 54 EFPY, based on the information provided in LRA Table 4.2-3. The applicant concluded that each material in the STP, Units 1 and 2, RVs that has a surface neutron fluence value exceeding 1.0×10^{17} n/cm^2 (E > 1.0 MeV) at 54 EFPY has been demonstrated to have an RT_{PTS} value less than the applicable screening criterion; therefore, the RT_{PTS} value analyses have been satisfactorily projected for 60 years of operation.

The applicant dispositioned the PTS evaluation TLAA in accordance with 10 CFR 54.21(c)(1)(ii) to demonstrate that the analyses have been projected to the end of the period of extended operation.

4.2.2.2 Staff Evaluation

The staff reviewed LRA Section 4.2.2 to confirm that the PTS analyses have been projected to the end of the period of extended operation, pursuant to 10 CFR 54.21(c)(1)(ii). The staff reviewed the applicant's TLAA and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.2.3.1.2.2, which state that the documented results of the revised PTS analysis based on the projected neutron fluence at the end of the period of extended operation are reviewed for compliance with 10 CFR 50.61 (the PTS Rule). The SRP-LR also states that the staff should confirm that the applicant provided sufficient information for PTS for the period of extended operation. Based on the requirements of the PTS Rule, license holders shall have projected RT_{PTS} values for each RPV beltline material through the end of its operating license. The RT_{PTS} value for each beltline material is evaluated from:

$$RT_{PTS} = RT_{NDT(u)} + \Delta RT_{PTS} + M$$

$RT_{NDT(u)}$ is the unirradiated reference temperature (RT_{NDT}) (as defined in the ASME Code Section III, paragraph NB-2331); ΔRT_{PTS} is the shift in RT_{NDT} caused by neutron irradiation; and M is the margin term to account for uncertainties in the calculation. The methodology used for determining ΔRT_{PTS} and the margin term M are described in the PTS Rule, including provisions for the use of surveillance data. The PTS Rule also provides the NRC-approved screening criteria for plates, forgings, axial weld materials (270 °F), and circumferential weld materials (300 °F).

In LRA Tables 4.2-2 and 4.2-3, the applicant presented the projected RT_{PTS} values at 54 EFPY for STP, Units 1 and 2, respectively. These tables also present the input parameters necessary for calculating the applicant's RT_{PTS} values. The staff identified discrepancies and insufficient information for the input parameters. Therefore, by letter dated January 13, 2012, the staff issued RAI 4.2.2-1, requesting that the applicant provide complete material descriptions and describe the procedures used to determine the chemistry data, initial RT_{NDT}, and margins for the extended beltline materials to demonstrate that it has applied consistent approaches for both the beltline and the extended beltline materials. (Note that RAI 4.2.2-1 also requested information related to Charpy USE,[2] as discussed in SER Section 4.2.3)

By letter dated April 17, 2012, the applicant submitted the requested information for the beltline and extended beltline materials that are expected to receive neutron fluence values greater than 1.0×10^{17} n/cm^2 (E > 1.0 MeV). The applicant revised Tables 4.2-2 and 4.2-3 to include projected RT_{PTS} values at 54 EFPY for beltline and extended beltline materials for Units 1 and 2, respectively. A revision to LRA Section 4.2.2 states that the fluence projections for the nozzle (upper) shell to intermediate shell circular weld and lower shell to lower head torus circular weld bound the extended beltline materials both above and below the beltline.

The staff notes that neutron fluence decreases as distance from the core increases. The applicant, in its analyses of neutron fluence for beltline components, assigned the neutron fluence value for the circumferential weld between the nozzle (upper) shell and intermediate shell to the RV beltline components above this location, and the neutron fluence for the circumferential weld between the lower shell and lower head torus to RV beltline components below this location. Since actual neutron fluence values would decrease above or below those points, respectively, because of the increasing distance from the core, the staff finds this approach for these extended beltline materials to be conservative and to provide acceptable projections of neutron fluence values for the period of extended operation.

The staff compared the unirradiated materials' properties in Tables 4.2-2 and 4.2-3 to the information in the current UFSAR. The staff noted that the initial RT_{NDT} values for the Unit 1 bottom head torus (R1617-1) and bottom head dome (R-1618-1) were both -50 °F in the UFSAR; however, each has a value of -30 °F in the LRA. Since the LRA value is more conservative, the staff finds these changes to be acceptable. The staff also noted that Tables 4.2-2 and 4.2-3 contain several extended beltline materials not listed in UFSAR Tables 5.3-3 and 5.3-4. For Units 1 and 2, these are: inlet/outlet nozzle to shell welds; nozzle (upper) shell longitudinal welds; nozzle (upper) shell to intermediate shell circumferential weld; lower shell to lower head torus circumferential weld; lower head torus longitudinal weld; and lower head torus to dome circumferential weld. The RAI response states that values of copper and nickel contents for these extended beltline materials were obtained from weld certification records and the STP RV specification. Where nickel values were not listed in the UFSAR or weld certification records, the RAI response states that a nickel value of 1.0 percent was assumed based on 10 CFR 50.61(c)(1)(iv)(A), which states the following:

> CF (°F) is the chemistry factor, which is a function of copper and nickel content. CF is given in Table 1 for welds and in Table 2 for base metal (plates and forgings). Linear interpolation is permitted. In Tables 1 and 2, "Wt-% copper" and "Wt-% nickel" are the best-estimate values for the material, which will normally be the mean of the measured values for a plate or forging. For a weld, the best estimate values will normally be the mean of the measured values for a weld deposit made using the same weld wire heat number as the critical vessel weld. If these values are not available, the upper limiting values given in the material specifications to which the vessel material was fabricated may be used.

[2] USE values are derived from Charpy-impact testing.

> If not available, conservative estimates (mean plus one standard deviation) based on generic data may be used if justification is provided. If none of these alternatives are available, 0.35% copper and 1.0% nickel must be assumed.

Therefore, the staff finds that the assumption of 1.0 percent for these nickel values is acceptable.

The applicant stated that, according to the Weld Inspection Forms, the Unit 1 inlet nozzle to shell circumferential weld was fabricated using manual E-8018 type welds, and the initial RT_{NDT} values for the E-8018 type welds are bounded by the generic Linde 0091 flux type weld properties. Based on a review of measured initial RT_{NDT} values for E-8018 welds at other plants, the staff determined that the generic bounding initial RT_{NDT} value of -56 °F for Linde 0091 from 10 CFR 50.61(c)(1)(ii) provides an appropriate estimate of the initial RT_{NDT} of the E-8018 welds in the Unit 1 RV. The staff's concerns in RAI 4.2.2-1 related to PTS are resolved.

As part of its review to confirm acceptability of the applicant's analysis, the staff performed confirmatory calculations of RT_{PTS} values for each of the extended beltline materials in LRA Tables 4.2-2 and 4.2-3 and concluded that the applicant's projected RT_{PTS} values are consistent with those calculated by the staff. With the addition of the extended beltline materials, the limiting material for Unit 1 was determined to be inlet nozzle R1613-4 with an RT_{PTS} values of 127.3 °F, and the limiting material for Unit 2 was determined to be outlet nozzle R2012-1 with an RT_{PTS} values of 111.1 °F. These values are below the screening criterion of 270 °F for plates, forgings, and axial weld materials.

Although the staff's confirmatory calculations yielded RT_{PTS} values consistent with those provided in LRA Tables 4.2-2 and 4.2-3, LRA Section 4.2.2 identifies the limiting material for each unit as an intermediate shell material—which has an RT_{PTS} value less than (i.e., less limiting than) that for the nozzle materials identified in the SER paragraph above—as the limiting material for the respective unit. To address the inconsistency between the text and the tables in LRA Section 4.2.2, the applicant, in a letter dated December 11, 2012, revised LRA Section 4.2.2. The revised section states that, while the limiting RT_{PTS} value for the beltline material for each unit is an intermediate shell material (as discussed above), the component with the most limiting RT_{PTS} value for the unit is the nozzle shell material as listed in the respective LRA Tables 4.2-2 and 4.2-3. For Unit 1, the limiting material is Inlet Nozzle R1613-4. For Unit 2, the most limiting material is Outlet Nozzle R2012-1. The staff finds the LRA section revision acceptable and consistent with its own confirmatory calculations.

Based on the above discussion, the staff concludes that the Units 1 and 2 RV beltline and extended beltline materials will satisfy the PTS requirements in 10 CFR 50.61 through the period of extended operation. The applicant's TLAA is acceptable because it meets the requirements in 10 CFR 54.21(c)(1)(ii) and will ensure that the Units 1 and 2 RV materials will have adequate RT_{PTS} values and fracture toughness through the period of extended operation.

4.2.2.3 UFSAR Supplement

LRA Section A3.1.2 provides the UFSAR supplement summarizing the PTS TLAA. The staff reviewed LRA Section A3.1.2, consistent with the review procedures in SRP-LR Section 4.2.3.2, which state that the applicant should provide a summary description of the evaluation of the RV neutron embrittlement TLAA equivalent to that in SRP-LR Table 4.2-1. Based on its review of the UFSAR supplement, the staff finds that the applicant provided an adequate summary description of its actions to address PTS, as required by 10 CFR 54.21(d).

4.2.2.4 Conclusion

Based on its review of the LRA and the applicant's response to RAI 4.2.2-1 (related to PTS), the staff concludes that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the PTS analyses have been projected to the end of the period of extended operation and will continue to meet the requirements of the PTS Rule (10 CFR 50.61). The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d), and therefore, is acceptable.

4.2.3 Upper-Shelf Energy

4.2.3.1 Summary of Technical Information in the Application

LRA Section 4.2.3 describes the applicant's TLAA for the evaluation of Charpy USE values for the 60-year period of extended operation. The applicant projected the Charpy USE using the 54 EFPY fluences described in LRA Section 4.2.1, as attenuated to the 1/4T location in the RV wall thickness.

Charpy USE values for all of the beltline materials of the STP, Units 1 and 2, RVs were determined in accordance with RG 1.99, Revision 2, without the use of surveillance data (Position 1.2 of the RG), although the surveillance data were available and found to be credible. This approach results in lower (more conservative) projections for the USE at the end of the 60-year period of extended operation than the alternative (Position 2.2 of the RG). The projected USE values for the beltline and extended beltline materials remain above the 50 ft-lb requirement through the period of extended operation, as indicated in LRA Tables 4.2-4 and 4.2-5 for STP, Units 1 and 2, respectively.

The applicant dispositioned the USE TLAA in accordance with 10 CFR 54.21(c)(1)(ii) to demonstrate that the analyses have been projected to the end of the period of extended operation.

4.2.3.2 Staff Evaluation

The staff reviewed LRA Section 4.2.2 and the USE TLAA to confirm, pursuant to 10 CFR 54.21(c)(1)(ii), that the Charpy USE analyses have been projected to the end of the period of extended operation.

The staff reviewed the applicant's TLAA and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.2.3.1.1.2, which state that the documented results of the revised USE analysis based on the projected neutron fluence at the end of the period of extended operation are reviewed for compliance with 10 CFR Part 50, Appendix G. Appendix G to 10 CFR Part 50 contains the screening criteria that establish limits on the USE values for RV materials after neutron irradiation exposure. The regulation requires the USE value to be greater than 50 ft-lb in the irradiated condition throughout the licensed life of the plant. USE values of less than 50 ft-lb may be acceptable to the staff if it can be demonstrated that these lower values will provide margins of safety against brittle fracture equivalent to those required by ASME Code Section XI, Appendix G.

RG 1.99, Revision 2, states that the predicted decrease in USE values due to neutron embrittlement during plant operation is dependent upon the amount of copper in the material and the predicted neutron fluence for the material. RG 1.99 outlines two ways to project the USE values for ferritic steels: Position 1.2 uses Figure 2 of RG 1.99, and Position 2.2 uses reactor surveillance data. As indicated above in SER Section 4.2.3.1, the applicant stated that it used Position 1.2 to determine the Charpy USE values at the end of the period of extended operation for the RPV beltline materials, because Position 1.2 projected lower (more conservative) USE values for each of these materials.

The staff identified discrepancies and insufficient information for the USE input parameters. Therefore, the staff issued RAI 4.2.2-1, requesting that the applicant provide complete material descriptions and describe the procedures used to determine the chemistry data and initial USE values for the extended beltline materials to demonstrate that it has applied consistent approaches for both the beltline and the extended beltline materials.

By letter dated April 17, 2012, the applicant provided the requested information in revised Tables 4.2-4 and 4.2-5 for Units 1 and 2, respectively. As discussed in Section 4.2.2.2, the staff reviewed the copper values in Tables 4.2-2 and 4.2-3 and determined that the values were acceptable. The copper values in Tables 4.2-4 and 4.2-5 are identical to the values in Tables 4.2-2 and 4.2-3 for Units 1 and 2, respectively. The staff compared the unirradiated USE values to the UFSAR. Initial USE values for the Units 1 and 2 bottom head torus longitudinal welds were obtained from measured values recorded in weld certification records. For welds lacking measured values, generic USE values from NRC-approved report CEN-622-A, "Generic Upper-Shelf Values for Linde 1092, 124 and 0091 Reactor Vessel Welds, CEOG Task 839," were used. The generic "mean minus 2 sigma" values for Linde 0091 and Linde 124 flux types are 101 ft-lb and 84 ft-lb, respectively. The staff compared these generic values to measured unirradiated USE values for E-8018 welds at other plants and concluded that the generic values are appropriate for estimating the initial USE values of E-8018 welds in the STP RV in lieu of measured unirradiated USE values. Therefore, the staff concluded that the unirradiated USE values in the revised Tables 4.2-4 and 4.2-5 for Units 1 and 2, respectively, are acceptable. The concerns in RAI 4.2.2-1 related to unirradiated USE values are resolved.

The staff used Position 1.2 of RG 1.99, Revision 2, and determined that, based upon the analysis for all beltline and extended beltline materials, the applicant's projected USE values were determined conservatively and resulted in 71 ft-lb for the limiting material (intermediate shell R1606-2) for Unit 1, and 72 ft-lb for the limiting materials (lower shell to lower head torus circumferential weld and nozzle (upper) shell to intermediate shell circumferential weld) for Unit 2.

By letter dated December 11, 2012, the applicant revised LRA Section 4.2.3. The revised section specifies the limiting material for each RV (for Unit 1, intermediate shell R1606-2, and for Unit 2, the lower shell to lower head torus circumferential weld and the nozzle (upper) shell to intermediate shell circumferential weld) and states that the embrittlement projections for these limiting materials also bound the other materials above and below the beltline. The staff finds the LRA section revision acceptable and consistent with its own confirmatory calculations.

In summary, the staff finds that the Units 1 and 2 beltline and extended beltline materials have projected USE values at 1/4 T greater than 50 ft-lb and, pursuant to 10 CFR 54.21(c)(1)(ii), meet the 10 CFR Part 50, Appendix G, USE requirement to the end of the period of extended operation; therefore, the applicant's USE analyses are acceptable.

4.2.3.3 UFSAR Supplement

LRA Section A3.1.3 provides the UFSAR supplement summarizing the USE TLAA. The staff reviewed LRA Section A3.1.3, consistent with the review procedures in SRP-LR Section 4.2.3.2, which state that the applicant should provide a summary description of the evaluation of the RV neutron embrittlement TLAA equivalent to that in SRP-LR Table 4.2-1. Based on its review of the UFSAR supplement, the staff finds that the applicant provided an adequate summary description of its actions to address USE, as required by 10 CFR 54.21(d).

4.2.3.4 Conclusion

Based on its review of the LRA and the applicant's response to RAI 4.2.2-1 (related to USE), the staff concludes that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the USE analyses have been projected to the end of the period of extended operation and will meet the criteria defined in Appendix G to 10 CFR Part 50. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.2.4 Pressure-Temperature Limits

4.2.4.1 Summary of Technical Information in the Application

LRA Section 4.2.4 describes the applicant's TLAA for the evaluation of the RV P-T limits for the period of extended operation. The applicant developed the adjusted RT values (ART values) at the 1/4T and 3/4T RV wall thickness locations using neutron fluences for those locations. The current P-T limit curves are valid through 32 EFPY.

The LRA states that the Reactor Vessel Surveillance Program (LRA Section B2.1.15) monitors RV embrittlement. This program provides data to update the P-T limits; therefore, it permits the applicant to manage the P-T limits going forward in accordance with 10 CFR 54(c)(1)(iii). The applicant will submit updates to the P-T limits for STP, Units 1 and 2, to the NRC at the appropriate time to comply with 10 CFR Part 50, Appendix G.

The applicant dispositioned the RV P-T limits TLAA in accordance with 10 CFR 54.21(c)(iii) to demonstrate that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

4.2.4.2 Staff Evaluation

The staff reviewed LRA Section 4.2.4 and the P-T limits TLAA to confirm, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the P-T limits will be adequately managed by the applicant for the period of extended operation. The staff reviewed the applicant's TLAA and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.2.3.1.3.3, which state that the updated P-T limits for the period of extended operation must be available prior to entering the period of extended operation. The staff noted that the P-T limits are contained in the applicant's TS, Section 3.4.9.1, "Pressure/Temperature Limits, Reactor Coolant System."

Prior to the expiration of the current P-T limit curves for STP, Units 1 and 2 (32 EFPY), the applicant is required to submit revised P-T limit curves in accordance with 10 CFR Part 50, Appendix G, considering the impact of all reactor coolant system (RCS) components, the increase of the limiting ART, and plant-specific embrittlement information from additional surveillance data provided by the RV Surveillance Program.

Ferritic RCPB components that are <u>not</u> RV beltline shell materials (i.e., consistent with GALL Report definitions, those RV components that will receive neutron fluence less than 1.0×10^{17} n/cm^{2,}) may have calculated P-T curve limits, irrespective of the components' neutron fluence values, that are more restrictive than those calculated for RV beltline shell materials. For example, this could be because of such factors as a component that exhibits significantly higher stresses, due to having a complex geometry, than components in the beltline, or an RCPB component having a higher initial nil-ductility reference transition temperature, which leads to a more restrictive P-T limitation than those for RV shell components. The staff noted that the information in LRA Section 4.2.2.4 describing the applicant's approach for revising its P-T limit curves beyond their currently approved 32 EFPY did not address how the approach considers all ferritic RCPB materials and the most restrictive service temperatures among all ferritic RCPB materials, consistent with the requirements in 10 CFR Part 50, Appendix G.

By letter dated June 25, 2012, the staff issued RAI 4.2.2.4-1, requesting that the applicant address this issue as it relates to its P-T curve methodology and explain how it will manage its P-T limit curves during the period of extended operation.

By letter dated July 17, 2012, the applicant stated the following:

> The development of the revised P-T limit curves to extend the curves beyond 32 EFPY and into the PEO [period of extended operation] will be in accordance with 10 CFR [Part] 50 Appendix G. The revised P-T limit curves will consider the effects of neutron embrittlement on the adjusted reference temperature for RV beltline and extended-beltline locations and the higher stresses in the inlet/outlet nozzle corner region. The revised P-T limit curves also will consider the ferritic RCPB components outside the beltline and extended-beltline locations when determining the lowest service temperature.

In addition, the applicant revised LRA Section 4.2.4 and Appendix A3.1.4, "Pressure-Temperature (P-T) Limits" to describe how the P-T limit curves will be revised to be consistent with the requirements in 10 CFR Part 50, Appendix G, during the period of extended operation. Enclosure 2 to the July 17, 2012, letter provides the line-in/line-out changes to LRA Section 4.2.4 and Appendix A3.1.4. These changes demonstrate that the approach for revising the P-T limit curves beyond 32 EFPY will be consistent with the requirements in 10 CFR Part 50, Appendix G. The staff's concerns in RAI 4.2.2.4-1 are resolved.

Based on this review, the staff finds that the applicant's plan to manage the P-T limits in accordance with 10 CFR 54.21(c)(1)(iii) is acceptable because revised P-T limit curves (as contained in TS 3.4.9.1) meeting the requirements in 10 CFR Part 50, Appendix G, will be implemented by the license amendment process (i.e., through revision of the plant's TS).

4.2.4.3 UFSAR Supplement

LRA Section A3.1.4, as revised by the applicant in its letter dated July 17, 2012, provides the UFSAR supplement summarizing the P-T limits TLAA. The staff reviewed LRA Section A3.1.4, consistent with the review procedures in SRP-LR Section 4.2.3.2, which state that the applicant should provide a summary description of the evaluation of the RV neutron embrittlement TLAA and provide information equivalent to SRP-LR Table 4.2-1. Based on its review of the UFSAR supplement, as revised, the staff finds that the applicant provided an adequate summary description of its actions to address P-T limits, as required by 10 CFR 54.21(d).

4.2.4.4 Conclusion

Based on its review, the staff concludes that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the P-T limits will be adequately managed by the applicant for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.2.5 Low Temperature Overpressure Protection

4.2.5.1 Summary of Technical Information in the Application

LRA Section 4.2.5 describes the applicant's TLAA for the evaluation of LTOP. The LRA states that LTOP is required by TS LCO 3.4.9.3 and is provided by the cold overpressure mitigation system (COMS), which opens the pressurizer PORVs at a setpoint calculated to prevent violation of the P-T limits. The LRA states that changes to the P-T limit curves require an evaluation of the LTOP temperature and PORV pressure setpoints, and that the LTOP analyses depend only on ART values at critical locations and the P-T limits, and not on any other time-dependent values.

The applicant dispositioned the LTOP TLAA in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

4.2.5.2 Staff Evaluation

The staff reviewed LRA Section 4.2.5 and the LTOP TLAA, and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.2.3.1.3 to confirm, pursuant to 10 CFR 54.21(c)(1)(iii), that LTOP will be adequately managed by the applicant for the period of extended operation. The staff noted that the LTOP requirements are contained in the applicant's TS, Section 3.4.9.3, "Overpressure Protection Systems."

LRA Section 4.2.4 states that the current P-T limits are projected and approved through 32 EFPY. Prior to the expiration of the current P-T limit curves, the applicant is required to submit revised P-T limit curves in accordance with 10 CFR Part 50, Appendix G, considering all applicable RCS materials, the increase of the limiting ART, and plant-specific embrittlement information from additional surveillance data provided by the Reactor Vessel Surveillance Program. Revised P-T limit curves will require evaluation of the LTOP temperature and PORV pressure setpoints; the revised P-T limit curves and the revised ART values are the only time-dependent inputs to the LTOP analyses.

Based on this review, the staff finds that the applicant's plan to manage LTOP in accordance with 10 CFR 54.21(c)(1)(iii) is acceptable because it meets the requirements in 10 CFR Part 50, Appendix G, and will be implemented by the license amendment process (i.e., through revision of TS 3.9.4.3 and the associated TS bases).

4.2.5.3 UFSAR Supplement

LRA Section A3.1.5 provides the UFSAR supplement summarizing the LTOP TLAA. The staff reviewed LRA Section A3.1.5, consistent with the review procedures in SRP-LR Section 4.2.3.2, which state that the applicant should provide a summary description of the evaluation of the RV neutron embrittlement TLAA and provide information equivalent to SRP-LR Table 4.2-1. Based on its review of the UFSAR supplement, the staff finds that the applicant provided an adequate summary description of its actions to address LTOP, as required by 10 CFR 54.21(d).

4.2.5.4 Conclusion

Based on its review, the staff concludes that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that LTOP will be adequately managed by the applicant for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3 Metal Fatigue Analysis

LRA Section 4.3 provides the applicant's assessment of metal fatigue as a TLAA for license renewal. The applicant's assessment is divided into the following major subsections of LRA Section 4.3:

- fatigue cycles and the monitoring activities performed by the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program (Section 4.3.1)
- ASME Code Section III Class I fatigue analysis of vessels, piping, and components (Section 4.3.2)
- ASME Code Section III Subsection NG fatigue analysis of reactor pressure vessel internals (Section 4.3.3)
- effects of the RCS environment on fatigue life of piping and components (Section 4.3.4)
- assumed thermal cycle count for allowable secondary stress range reduction factor in ANSI B31.1 and ASME Code Section III Class 2 and 3 piping (Section 4.3.5)
- ASME Code Section III fatigue analysis of metal bellows and expansion joints (Section 4.3.6)

The staff's evaluation of LRA Section 4.3.1 is documented in SER Section 4.3.1.2. The description and staff's evaluation of above-listed Sections 4.3.2 to 4.3.6 are documented in SER Sections 4.3.2 to 4.3.6, respectively.

4.3.1 Metal Fatigue of Reactor Coolant Pressure Boundary Program

4.3.1.1 Summary of Technical Information in the Application

LRA Section 4.3.1 describes the design transients and associated number of design cycles that are significant fatigue contributors in the applicant's assessment of metal fatigue TLAAs. LRA Section 4.3.1 also indicates that the Metal Fatigue of Reactor Coolant Pressure Boundary Program (B3.1) is required by STP TS 5.7.1 and 6.8.3.f. UFSAR Section 3.9.1 discusses the design cycles as historical numbers used in the original design basis fatigue evaluations for equipment design purposes. The ASME Code does not require inclusion of emergency or faulted conditions in fatigue evaluations. Therefore, the Metal Fatigue of Reactor Coolant Pressure Boundary Program does not monitor emergency and faulted conditions.

The Metal Fatigue of Reactor Coolant Pressure Boundary Program tracks the occurrences of the transients listed in LRA Table 4.3-2 and manages the CUFs by using either the cycle-counting monitoring method or cycle-based fatigue (CBF) monitoring method. The Metal Fatigue of Reactor Coolant Pressure Boundary Program ensures that the number of transients actually experienced during the period of extended operation remains below the assumed number or that appropriate corrective actions maintain the design and licensing basis.

The applicant reviewed the operating history of STP, Units 1 and 2, from initial startup to year-end 2008 to baseline the transient event count for the Metal Fatigue of Reactor Coolant Pressure Boundary Program. These baselined results were then extrapolated to 60 years. LRA Table 4.3-2 includes the accumulated cycle counts through 2008 and the projections to 60 years. The LRA states that the cycle projections are based on a long-term weighting (LTW) and short-term weighting (STW) to obtain the most accurate projections of the future behavior of that event. These projections are intended to be a best estimate of the actual cycles expected.

4.3.1.2 Staff Evaluation

The staff reviewed LRA Section 4.3.1 to confirm that the transients that are significant fatigue contributors are monitored to ensure that the applicant's fatigue evaluations remain valid. The staff also reviewed the methodology used by the applicant to obtain the 60-year projections. The staff's evaluation of the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program is documented in the SER Section 3.0.3.2.28.

LRA Table 4.3-2 indicates that the current cycle count for Transient 41 (Charging Trip with Prompt Return to Service) for Unit 1 is 10 as of the end of 2008. During its audit, the staff reviewed the applicant's design basis documents and noted that the cycle count for Transient 41 for Unit 1 was 11 as of April 2005. By letter dated September 22, 2011, the staff issued RAI 4.3-1 requesting that the applicant justify the discrepancy and provide the correct current cycle count and, as applicable, the 60-year projected cycles for Transient 41 for Unit 1.

In its response dated November 21, 2011, the applicant stated that its corrective action document noted 11 occurrences of the loss of charging events, including the April 12, 2005, event in which letdown was temporarily reduced. Upon further review of plant data recordings when developing the baseline cycles for license renewal (LRA Table 4.3-2), the applicant determined that the April 12, 2005, event was not a loss of charging event because charging flow remained above 35 gpm, while the flow rate varied during the entire day. The applicant confirmed that the correct current cycle count for Transient 41 as of the end of 2008 is 10 occurrences. The staff noted that the applicant is continuing to manage this transient, which is used in metal fatigue evaluations, during the period of extended operation with its Metal Fatigue of Reactor Coolant Pressure Boundary Program that ensures the validity of its fatigue analyses or calculates accrued usage to ensure that the Code design limit of 1.0 is not exceeded.

The applicant also stated that the April 12, 2005, event should be classified as "Charging Flow Step Decrease and Return to Normal," which assumes 24,000 occurrences for the design number of cycles, and its Metal Fatigue of Reactor Coolant Pressure Boundary Program does not specifically count this event because the number of assumed cycles is far greater than the number expected over 60 years. However, it was not clear to the staff why this transient does not need to be monitored by the applicant's program to ensure any fatigue analysis that assumed the occurrence of this transient remains valid. By letter dated January 31, 2012, the staff issued RAI 4.3-1a (followup) requesting that the applicant clarify the baseline number of events up to the end of 2008 and the 60-year projected cycles for the charging flow step decrease and return to normal transient.

In its response dated February 16, 2012, the applicant stated that “charging flow step decrease and return to normal” transient is not included in the baseline because the transient is not monitored. Furthermore, this transient occurs when there is a power change, typically during plant heatup and cooldown, and the estimated number of events based on the plant heatup and cooldown events that have occurred up to the year ending of 2008 are 87 (Unit 1) and 55 (Unit 2). The applicant estimated that the 60-year projected events would be 172 (Unit 1) and 154 (Unit 2). The staff noted that there is significant margin between the expected number of cycles through the period of extended operation and 14,400 cycles and finds it reasonable that sufficient margin exists to account for unanticipated shutdowns or power reductions. In addition, the staff finds it reasonable that the “charging flow step decrease and return to normal” transient is not monitored because the applicant’s units do not practice load-following operation but operate as base-loaded plants. The staff also noted that the units’ projected occurrences are far less than the limiting value of 14,400 cycles identified in the applicant’s response to RAI 4.3-2, dated November 21, 2011.

The staff finds it reasonable that the “charging flow step decrease and return to normal” transient does not require monitoring by the Metal Fatigue of Reactor Coolant Pressure Boundary Program because the transient is correlated with the occurrences of the heatup and cooldown transients that are monitored, and that there is a substantial margin between the 60-year projected occurrence (less than 200) and the limiting value of 14,400. Therefore, the staff finds the applicant’s response to RAI 4.3-1a acceptable. The staff’s concern described in RAI 4.3-1a (followup) is resolved.

The staff finds the applicant’s response to RAI 4.3.1 acceptable because the applicant clarified and justified the discrepancy for the cycle count of Transient 41 based on actual plant data and the applicant’s Metal Fatigue of Reactor Coolant Pressure Boundary Program is monitoring this transient. The staff’s concern described in RAI 4.3-1 is resolved.

LRA Section 4.3.1.2 states that the occurrences of the transients listed in LRA Table 4.3-2 are tracked, and the CUFs at the locations listed in LRA Table 4.3-1 are managed using either the cycle-counting monitoring method or the CBF monitoring method. In addition, the LRA states that the most limiting number of cycles for each transient is listed as the “Program Limiting Value” and will be used for the Metal Fatigue of Reactor Coolant Pressure Boundary Program. It was not clear to the staff whether the components identified in LRA Table 4.3-1 are the only components monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program to manage cumulative fatigue damage and whether there are any TLAAs or evaluations other than the environmentally-assisted fatigue (EAF) evaluations that use the 60-year projected cycles.

By letter dated September 22, 2011, the staff issued RAI 4.3-11 requesting that the applicant clarify the monitoring method used by the Metal Fatigue of Reactor Coolant Pressure Boundary Program for the components and locations in which the applicant’s metal fatigue TLAAs were dispositioned in accordance with 10 CFR 54.21(c)(1)(iii). The staff also asked the applicant to clarify whether the cycle-counting monitoring method accounts for the use of the 60-year projected cycles for those TLAAs or evaluations other than the EAF evaluations.

In its response dated November 21, 2011, the applicant stated that the components identified in LRA Table 4.3-1 are monitored by CBF monitoring, and all other components that are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii) are monitored by cycle counting. In addition, the applicant clarified that there are no other fatigue analyses, other than the EAF evaluations, which use the 60-year projected cycles. The staff noted that this method provides a “real-time” usage factor and allows the applicant to ensure that the ASME Code design limit of 1.0 is not exceeded, consistent with the recommendations of GALL Report AMP X.M1, “Fatigue Monitoring.” This method allows the determination of cumulative fatigue usage for a specific location based on the actual number of transient occurrences and the assumption that the fatigue usage contributed by each transient is equal to the design transient severity. The staff finds the applicant’s use of CBF monitoring to be capable of managing metal fatigue because it periodically calculates cumulative fatigue usage based on the cycle counts and design transient severity to ensure that the design limit is not exceeded during the period of extended operation. The staff’s review of the Metal Fatigue of Reactor Coolant Pressure Boundary Program, specifically the management of cumulative fatigue usage, is documented in SER Section 3.0.3.2.28. The staff noted that the EAF evaluations that use the 60-year projected cycles will be monitored by CBF. Since there are no other fatigue analyses that rely on the 60-year projected cycles, the staff finds it appropriate that the applicant’s program limiting values on number of cycles does not need to be based on these projected cycles.

Based on its review, the staff finds the applicant’s response to RAI 4.3-11 acceptable because the applicant clarified that the CBF method, which calculates real-time usage to ensure that the Code design limit is not exceeded, is used for those components identified in LRA Table 4.3-1. Additionally, the cycle-counting method is used for all other components to ensure on an ongoing basis that the analysis that calculated the CUF to be less than 1.0 remains valid. Both methods are consistent with the recommendations of GALL Report AMP X.M1 to manage cumulative fatigue damage. The staff’s concerns in RAI 4.3-11 are resolved.

LRA Section 4.3.1.2 states that the Metal Fatigue of Reactor Coolant Pressure Boundary Program tracks the occurrences of the transients listed in LRA Table 4.3-2, which includes the following transients:

- Transient 5, “Unit Loading at 5% of Full Power/min”
- Transient 6, “Unit Unloading at 5% of Full Power/min”
- Transient 10, “Steady State Fluctuations, Initial”
- Transient 11, “Steady State Fluctuations, Random”
- Transient 15, “Unit Loading Between 0-15% of Full Power”
- Transient 16, “Unit Unloading Between 0-15% of Full Power”
- Transient 17, “Boron Concentration Equalization”

The staff noted that LRA Table 4.3-2 does not provide baseline numbers of cycles for Units 1 and 2 for the transients listed above; therefore, it is not clear how the Metal Fatigue of Reactor Coolant Pressure Boundary Program tracks the occurrences of these transients. Since 60-year projections were not provided for the transients listed above, it was not clear to the staff whether they were used as part of the applicant’s EAF CUF calculations.

By letter dated September 22, 2011, the staff issued RAI 4.3-13 requesting that the applicant justify how the Metal Fatigue of Reactor Coolant Pressure Boundary Program tracks the occurrences of Transients 5, 6, 10, 11, 15, 16, and 17 without having a baseline number of cycles for each of them. The staff also asked the applicant to clarify whether these transients were included in the EAF CUF calculation.

In its response dated November 21, 2011, the applicant stated that Transients 5, 6, 10, 11, 15, 16, and 17 are not projected; therefore, they are not tracked by the Metal Fatigue of Reactor Coolant Pressure Boundary Program. The applicant explained that Transient 17, "Boron Concentration Equalization," occurs following any large change in boron concentration in the RCS by initiating spray in order to equalize boron concentration between the RCS loops and the pressurizer. For design purposes, it is assumed that this operation is performed after each load change in the load-following design cycle, and Transient 17 is assumed to coincide with Transients 5 and 6, which are listed in Footnote 3 of LRA Table 4.3-2 as transients for a load-following plant. Similarly, Footnote 4 of LRA Table 4.3-2 indicates that Transients 15 and 16 are transients for a load-following plant. The applicant further clarified that it does not operate as a load-following plant, which sets the power level of a unit in accordance with the electrical grid. The applicant stated that LRA Table 4.3-2, Footnote 5, will be revised to note that Transient 17 is a load-following transient.

The staff noted that the design number of cycles for Transients 5, 6, 15, 16, and 17 were based on the assumption that the plant operated in a load-following mode. The applicant explained in the footnotes of LRA Table 4.3-2, and further clarified in its response, that the units do not load-follow. The staff finds it acceptable that the applicant does not monitor these transients because the design number of cycles was based on load-following operations. Also, since the units do not load-follow, it would not be expected that the design number of cycles would be approached.

The applicant also stated in its response that Transients 10 and 11 are both subcategories of steady-state fluctuations. Transient 10 identifies fluctuations that are assumed to occur only during the first 20 full-power months of operation; therefore, Transient 10 is not applicable for future operation and does not need to be managed for fatigue. The applicant stated that the number of cycles for Transient 11 is below the endurance limit of the ASME Code fatigue curves; therefore, Transient 11 does not need to be managed for fatigue. When compared to the ASME fatigue curve, the staff was unable to determine the meaning of the applicant's statement regarding Transient 11. The staff held a conference call on August 9, 2012 with the applicant in order for the staff to obtain clarification on the applicant's intended response. The applicant stated during the call, as documented in the conference call summary (ADAMS Accession No. ML12227A560), that the intent of the statement was to read as follows: "The stress range of Transient 11 is below the endurance limit of the ASME fatigue curves, therefore this transient is not significant to fatigue." Based on this clarification, the staff finds it reasonable that if the stress caused by Transient 11 is less than the S_a associated with the endurance limit on the ASME Code fatigue curves then fatigue life can be considered infinite because the alternating stress from this transient is less than the stress that would result in metal fatigue. Therefore, the staff finds that this transient does not need to be monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program. The staff also finds it acceptable that Transient 10 is not monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program because it was only applicable during the first 20 full-power months of operation and is not applicable for future operation.

The applicant clarified that if a transient is not projected for the period of extended operation, then the design number of events is used in the EAF CUF calculations. The applicant stated that Transients 5, 6, 10, 11, and 17 are used in the hot leg surge nozzle EAF CUF calculation. In addition, Transients 5, 6, 10, 11, 15, 16, and 17 have a negligible effect on EAF CUF calculations for the charging nozzles and are not included in those calculations. The staff finds it conservative that the above-mentioned transients were included in the hot leg surge nozzle EAF CUF calculation because the design number of events for a load-following plant was assumed to occur even though the applicant's site does not practice load-following operation. Because these transients were meant for a plant designed for load-following operation, the staff also finds it reasonable that the above-mentioned transients were not included in the charging nozzle EAF CUF calculations because the applicant's site does not practice load-following operation and operates as a base-loaded plant.

Based on its review, as described above, the staff finds that the applicant's response clarified why Transients 5, 6, 10, 11, 15, 16, and 17 are not monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program and that it is acceptable. The staff's concern described in RAI 4.3-13 is resolved.

LRA Table 4.3-2 provides the baseline and 60-year projected numbers of cycle for STP, Units 1 and 2, for the following transients:

- Transient 19, "Primary Side Leak Test"
- Transient 22, "Turbine Roll Test"
- Transient 43, "Primary Side Hydrostatic Test"
- Transient 44, "Secondary Side Hydrostatic Test"

The staff noted that LRA Section 4.3.4 states that a method used to reduce the EAF CUF values includes using 60-year projected occurrences of transient events in LRA Table 4.3-2, instead of using the 40-year design number of events. For the transients listed above, LRA Table 4.3-2 indicates that these transients are not expected to occur again through 60 years of operation, except Transient 19 for Unit 2. Since these projections may have been used in reducing the EAF CUF, it is not clear why these transients are not expected to occur again and whether this is conservative.

By letter dated September 22, 2011, the staff issued RAI 4.3-14 requesting that the applicant justify why Transients 19 (except for Unit 2), 22, 43, and 44 are not expected to occur again through 60 years of operation. The staff also asked the applicant to justify that the use of these projections is conservative for the EAF CUF calculations.

In its response dated November 21, 2011, the applicant stated that Transients 19, 22, 43 and 44 are tests performed during initial startup, and no more tests are expected. The applicant also explained that for Unit 2 Transient 19, it chose to project one assumed event since no cycles have accumulated to date. In addition, the applicant stated that these projections were used in the EAF CUF calculations, but these startup tests are not expected to be performed again. Since these are test transients that are performed during initial startup, the staff finds it reasonable that the applicant assumed these transients would not occur again during the period of extended operation. However, the staff also noted that the applicant is not relying on the 60-year projections to justify that its fatigue analyses are valid for the period of extended operation. The applicant is continuing to manage the cumulative fatigue damage during the period of extended operation with its Metal Fatigue of Reactor Coolant Pressure Boundary Program that ensures the validity of its fatigue analyses or calculates accrued usage to ensure the Code design limit of 1.0 is not exceeded. The applicant stated that if these transients were to occur again, they would be tracked and incorporated in CBF-generated EAF CUFs, which will ensure that corrective actions are taken as the EAF CUFs approach the action limit and the Code design limit.

The staff finds the applicant's response acceptable because the applicant clarified that these transients were performed as part of the initial start-up process for both units and are not expected to occur again, and the applicant has not relied on these 60-year projections to justify that its fatigue analyses are valid for the period of extended operation. In the event these test transients were to occur again, the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program is monitoring these transients. The staff's concern described in RAI 4.3-14 is resolved.

LRA Section 4.3.1.3 states that the applicant captured all the necessary transient events, and the event history was taken primarily from existing manual or computer-assisted cycle-counting records. LRA Section 4.3.1.3 also states that the baseline cycle-counting results were projected to 60 years, and the projected cycle counts were computed based on the actual accumulation history since the start of plant life. In addition, the cycle projections are based on LTW and STW to obtain the "most accurate projections of the future behavior of that event."

It was not clear to the staff if, during the applicant's review of the transient event history, the applicant had confirmed that the severity of the transients that occurred was bounded by the severity of the design transient. In addition, since the applicant used the 60-year transient projections in its EAF fatigue analyses, additional information was needed about the LTW and STW used by the applicant in its projection methodology for the staff to determine if the methodology used was appropriate and reasonable.

By letter dated September 22, 2011, the staff issued RAI 4.3-12, requesting that the applicant describe actions taken to confirm that the severity of all transients that have occurred is bounded by the design severity of the transient and to describe the LTW and STW used for the 60-year projection methodology of design transients. The staff also asked the applicant to justify that this 60-year projection methodology is reasonable.

In its response dated November 21, 2011, the applicant stated that it did not confirm that the severity of all transients that have occurred is bounded by the design severity of the transient during the preparation of the LRA. However, the plant operating procedures and TS are designed to ensure that the severity of plant events is bounded by those described in the design analyses. The applicant explained that its current procedure requires a daily screening of transients that have occurred. The applicant further explained that a transient-specific datasheet is completed to record the plant's conditions during the event, and such information is forwarded to system engineering for validation and review. The staff finds it acceptable that the applicant did not confirm the severity of all past transients during the development of the LRA because the applicant's procedures and TS ensure that transients are recorded on a daily basis and will receive validation and review by the applicant's engineering staff at the time the transients occur.

The applicant also stated that the LTW and STW values used for each transient are estimated by taking into account the history of each transient, number of cycles, distribution, and transient qualities. In general, the applicant assumed that the short-term history was three times more likely to predict future performance than the long-term history (i.e., STW = 3, LTW = 1), and the short-term is 10 years, which is approximately one-third of the plant operating period. The applicant identified exceptions, which are those transients that occur randomly with a low number of occurrences and those that only occurred during initial plant testing. The applicant identified the transients that did not rely on the 3-to-1 short-term-to-long-term ratio described above and provided the corresponding STW, LTW, and short-term period in a table in its response to RAI 4.3-12.

The applicant also stated that the short-term-to-long-term ratio projection method is not used for transients that had never occurred, in which case at least one event was assumed for future operation. The staff noted that the applicant has not relied upon this methodology to determine the 60-year projections to justify that any fatigue analysis is valid for the period of extended operation. The staff further noted that the applicant is managing the validity of its design basis fatigue analyses (which did not use the 60-year projected cycles) and ensuring that the CUF for those components selected for EAF does not exceed the Code design limit of 1.0 with its Metal Fatigue of Reactor Coolant Pressure Boundary Program on an ongoing basis. The staff finds that the applicant's methodology for determining 60-year projections provides an estimate of the margin between the number of cycles that have been used in the fatigue analyses and the expected number of cycles for 60 years of operation.

The staff finds the applicant's response acceptable because the applicant's procedures and TS have ensured, and will ensure, that the severity of a transient does not exceed the assumptions in the fatigue analysis. In addition, the staff finds the response acceptable because the applicant is not relying on the 60-year projections to justify that its fatigue analysis is valid for the period of extended operation. Instead, the applicant is continuing to manage the cumulative fatigue damage during the period of extended operation with its Metal Fatigue of Reactor Coolant Pressure Boundary Program, which ensures the validity of its fatigue analyses or calculates accrued usage to ensure the Code design limit of 1.0 is not exceeded. The staff's concern described in RAI 4.3-12 is resolved.

Based on its review, the staff finds that the applicant demonstrated that it monitors all transients that cause cyclic strains, which are significant contributors to the fatigue usage factor with its Metal Fatigue of Reactor Coolant Pressure Boundary Program, such that corrective actions are taken prior to the design limit exceeding 1.0, including environmental effects when applicable.

4.3.1.3 UFSAR Supplement

LRA Sections A2.1 and A3.2 provide the UFSAR supplement summarizing the applicant's basis of its fatigue analyses and describing its Metal Fatigue Reactor Coolant Pressure Boundary Program to ensure that the number of cycles for each transient actually experienced remains below the assumed number. The staff reviewed LRA Sections A2.1 and A3.2, consistent with the review procedures in SRP-LR Section 4.3.3.3, which state that the reviewer should confirm that the applicant has provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA.

Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.3.2.3. Additionally, the staff finds that the applicant provided an adequate summary description of its Metal Fatigue Reactor Coolant Pressure Boundary Program to monitor the number of transients actually experienced, as required by 10 CFR 54.21(d).

4.3.1.4 Conclusion

Based on its review, the staff concludes that the applicant provided an adequate description and acceptable basis for monitoring design transients and cycles with its Metal Fatigue of Reactor Coolant Pressure Boundary Program. The program ensures that corrective actions are taken prior to exceeding the design limit during the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the monitoring bases of transients and design cycles, as required by 10 CFR 54.21(d).

4.3.2 Fatigue of ASME Code Class 1 Components

4.3.2.1 Reactor Pressure Vessel, Nozzles, Head, and Studs

4.3.2.1.1 Summary of Technical Information in the Application

LRA Section 4.3.2.1 describes the applicant's TLAA for fatigue of the RPV, nozzles, head, and studs. LRA Section 4.3.2.1 states that the Units 1 and 2 RPVs are designed to ASME Code Section III, 1971 edition with addenda through summer 1973. The STP vessels were built and analyzed for the assumed 40-year number of transient cycles. The applicant subdivided the TLAA discussion into three cases: (1) the replacement reactor vessel closure heads (RRVCHs); (2) the repaired bottom-mounted instrument (BMI) nozzles; and (3) all remaining components of the RPV, nozzles, and studs.

The applicant replaced the Units 1 and 2 RPV heads in the fall of 2009 and spring of 2010, respectively. The RRVCHs were designed to ASME Code Section III, 1989 edition (no addendum). The applicant stated that the fatigue CUF analyses for the RPV heads and any similarly replaced and analyzed appurtenances are analyzed for the design number of transient cycles starting from the time of installation. The applicant dispositioned the fatigue CUF analyses in accordance with 10 CFR 54.21(i) to demonstrate that the analyses remain valid for the period of extended operation.

Pressure-retaining and support components of the RPV are listed in LRA Table 4.3-3 and are subject to a fatigue CUF analysis in accordance with ASME Code Section III. The applicant updated the fatigue CUF analysis to incorporate redefinitions of loads and design basis events (DBEs), operating changes, power uprate, replacement steam generators (RSGs), and minor modifications. The applicant concluded that the currently applicable fatigue CUF analyses of the reactor pressure boundary and its supports are TLAAs, and dispositioned the analyses in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

The Unit 1 BMI nozzles are made of Alloy 600 and are attached to the clad inner surface of the RV bottom head by Alloy 182 J-groove welds. During refueling outage (RFO) 11 (1RE11, spring 2003), the applicant discovered leaks at Unit 1 BMI nozzles 1 and 46, which were repaired by the "half-nozzle" method. A half-nozzle repair leaves the existing flaw(s) in the original, inner-wall J-groove weld in place. In addition, the repair exposes a small portion of the low-alloy steel base metal of the lower RV head to reactor coolant and, therefore, to possible corrosion. These repairs were evaluated for growth of postulated residual flaws due to fatigue and corrosion. The flaw growth analysis, corrosion analysis, and fatigue CUF analysis qualify the repaired BMI nozzles for operation from the time of the repair through the period of extended operation. These are the only Alloy 600 half-nozzle repairs performed at STP. The applicant concluded that the analyses for the two BMI nozzle repairs are TLAAs and dispositioned them in accordance with 10 CFR 54.21(c)(1)(i) to demonstrate that the analyses remain valid for the period of extended operation.

4.3.2.1.2 Staff Evaluation

The staff reviewed LRA Section 4.3.2.1 and the fatigue CUF analyses or crack growth analyses, or both, for the RPV, head, nozzles, or studs, to confirm pursuant to 10 CFR 54.21(c)(1)(i) that the analyses remain valid for the period of extended operation. Otherwise, the staff confirmed that the effect of fatigue will be adequately managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii). The staff reviewed the applicant's TLAA and the corresponding disposition, consistent with the review procedures in SRP-LR Sections 4.3.2.1 and 4.7.3, which state that the review of the TLAA provides assurance that the aging effect is properly addressed through the period of extended operation. The staff evaluated three major component categories: (1) the RRVCH with associated CRDM penetration nozzles, (2) the repaired BMI nozzles, and (3) the remaining RPV components as listed in LRA Table 4.3-3.

Reactor Vessel Closure Head. By letter dated April 14, 2011, the staff issued RAI 4.3.2.1-1, requesting that the applicant discuss the condition of the RRVCHs in both units and measures that have been taken to minimize the degradation in CRDM penetration nozzles. By letter dated May 12, 2011, the applicant responded that no relevant indications were identified during the pre-service inspection of the Unit 1 and 2 RRVCHs. Currently, the applicant uses ASME Code Section XI, 2004 edition (no addenda) for the Inservice Inspection (ISI) Program.

Based on the third interval 10-year ISI Program, the applicant performs visual examinations of the RRVCHs every third RFO. The RRVCH and CRDM nozzles and partial penetration welds are monitored by performing volumetric or surface examinations (or both) once per 10-year ISI interval. To minimize degradation in CRDM penetration nozzles, the applicant used thermally treated Alloy 690 material for CRDM penetrations, Alloy 52 weld filler metal for J-groove welds, and automatic J-groove welding technology (including water cooling to improve stress distribution through the CRDM adapter wall).

The staff noted that the RRVCHs use material that is less susceptible to pressurized water stress corrosion cracking (PWSCC) and welding technology that would produce sound welds. The applicant followed the ASME Code ISI Program to monitor potential degradation in the RRVCH, and the staff finds that aging effects of the RRVCH will be managed by inspection satisfactorily. Based on the above evaluation, the staff's concern as described in RAI 4.3.2.1-1 is resolved.

By letter dated April 14, 2011, the staff issued RAI 4.3.2.1-5 requesting that the applicant provide the basis for its conclusion that the fatigue CUF analyses for the RRVCH are valid for the period of extended operation. By letter dated May 12, 2011, the applicant responded that the Unit 1 and 2 CRDM pressure housings, the CETNAs, and the internal disconnect devices were replaced with the RRVCHs in 2009 and 2010, respectively. The new CRDMs and CETNAs were qualified for 40 years. This means that the RRVCHs are qualified and applicable for use up to 2049 and 2050 for Units 1 and 2, respectively. The renewed operating licenses for STP, Units 1 and 2, would expire in 2047 and 2048, respectively, and thus the fatigue CUF analysis is valid for the period of extended operation. The staff finds that the fatigue CUF analyses of the RRVCHs remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i). Therefore, the staff's concern described in RAI 4.3.2.1-5 is resolved.

Unit 1 Reactor Vessel Bottom Mounted Instrument Nozzle Repairs. In RAI 4.3.2.1-8 (April 14, 2011), the staff asked the applicant to identify any flaws or indications that remain in service in the RPV components and discuss how these flaws or indications will be managed throughout the period of extended operation. By letter dated May 12, 2011, the applicant responded that after searching the UFSAR, TS, the NRC SERs for the original operating licenses, subsequent NRC SEs, and South Texas Project Nuclear Operating Company (STPNOC) and NRC docketed licensing correspondence, the only flaws remaining in service in the RPV are the flaws in Unit 1 BMI nozzles 1 and 46. The staff's concern described in RAI 4.3.2.1-8 is resolved because the applicant confirmed that the only flaws are the specific

Unit 1 BMI nozzles. The issue of their acceptability for the period of extended operation is evaluated below.

LRA Section 4.3.2.1 states that the 48-year fatigue crack growth analysis, CUF analysis, and the corrosion analysis for the Unit 1 BMI nozzles and lower head repairs are valid for the period of extended operation. In RAI 4.3.2.1-4 (April 14, 2011), the staff asked the applicant to demonstrate how these three analyses remain valid for the period of extended operation.

By letter dated May 12, 2011, the applicant responded that the fatigue crack growth analysis for the repaired Unit 1 BMI nozzles assumes the number of transient cycles equivalent to 48 years of operation by using 120 percent (48 years/40 years) of the design number of transients in UFSAR Table 3.9-8. Because this fatigue crack growth analysis covers an additional 48 years of operation from the repair date of 2003 (i.e., effective to 2051), the fatigue crack growth analysis is valid for the period of extended operation, which ends in 2047. The staff finds that the fatigue crack growth analysis for the repaired BMI nozzles remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

For the CUF analysis of the repaired Unit 1 BMI nozzles, the applicant stated that the analysis assumed transient cycles equivalent to 50 years of operation. The applicant stated that the validity of the CUF analysis of the repaired BMI penetrations extends from the repair date of the condition in 2003 to 2053, which is beyond the end of the period of extended operation in 2047. The staff notes that for the repaired Unit 1 BMI nozzles, the applicant calculated a CUF less than the allowable of 1.0. The staff finds that the CUF analysis for the Unit 1 repaired BMI nozzles remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

For the corrosion analysis of the repaired Unit 1 BMI nozzles, the applicant used a corrosion rate of 0.00153 inch per year to project the total metal corrosion in 50 years. The applicant doubled the rate to give the diametral corrosion rate of 0.00306 inch per year, or 0.153 inch in the 50 years from the repair in 2003, which extends the analysis to 2053 and through the end of the period of extended operation (2047). The applicant calculated that the base metal corrosion in the repaired BMI can increase the bore diameter from 1.562 inches to 1.95 inches (a diametral increase of 0.388 inch) and still meet the stress requirements of ASME Code Section III.

The applicant stated that the application of the corrosion rate through the period of extended operation is conservative because general corrosion will decrease after a period of time because of the lack of oxygen, tight geometry, and the lack of RCS flow at the location. The applicant derived the corrosion rate using the methodology documented in Combustion Engineering report, CE NPSD-1198-P, Revision 0, which the staff has approved. In support of relief request RR-ENG-2-33, the applicant provided information concerning the effects of corrosion on the BMI half-nozzle repairs in letters dated July 3, 2003 (ADAMS Accession No. ML031920109), and July 17, 2003 (ADAMS Accession No. ML032020109). The NRC approved relief request RR-ENG-2-33 in a letter dated August 1, 2003 (ADAMS Accession No. ML032130454).

To confirm the applicant's corrosion rate, the staff used information from Westinghouse topical report WCAP-15973-P, Revision 1, "Low-Alloy Steel Component Corrosion Analysis Supporting Small Diameter Alloy 600/690 Nozzle Repair/Replacement Program," which the NRC approved on January 12, 2005 (ADAMS Accession No. ML050180528). The corrosion rate of 1.53 mils per year used by the applicant is the same as the corrosion rate specified in WCAP-15973-P.

Therefore, the staff finds that the applicant's corrosion rate to project the metal loss in the affected BMI nozzles is acceptable. The staff finds that the applicant's corrosion analysis remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

The staff notes that the fatigue crack growth analysis showed that the repaired BMI nozzles in Unit 1 are acceptable for operation up to 48 years, and the CUF and corrosion analyses showed that the repaired BMI nozzles are acceptable for 50 years. The 48-year duration is more limiting than the 50-year duration. Therefore, the staff concludes that the repaired BMI nozzles are acceptable for 48 years. As stated above, the applicant repaired the two BMI nozzles in Unit 1 in 2003. Extending 48 years from 2003, the repaired Unit 1 BMI nozzles are acceptable for operation up to the end of the period of extended operation in 2047. Based on the above evaluation, the staff's concern as described in RAI 4.3.2.1-4 is resolved.

<u>Reactor Pressure Vessel Nozzles, Flange, and Studs</u>. LRA Table 4.3-2 shows the 40-year design transient cycle counts along with the 60-year projected number of (actual) cycles for Units 1 and 2. Table 4.3-3 shows both the 40-year (design) and the 60-year (projected) fatigue CUF for both units. The staff noted that the 40-year CUFs for the RV components such as the vessel flange, studs, and RPV nozzles, and the 60-year CUFs for these components, are all within the ASME Code allowable of 1.0, except for the 60-year value for the stud hole inserts. Footnote 2 to LRA Table 4.3-3 states that the 40-year design basis number of events "should be sufficient for 60 years of operation." The staff also noted that the applicant stated that it will manage these components to limit the number of transients to below the 40-year design limits through its Metal Fatigue of Reactor Coolant Pressure Boundary Components AMP. However, the staff found that the meaning of "should be sufficient" in Footnote 2 to LRA Table 4.3-3 was ambiguous. In RAI 4.3.2.1-6 (April 14, 2011), the staff asked the applicant to demonstrate that the 40-year design basis transient cycles are, in fact, sufficient for 60 years.

By letter dated May 12, 2011, the applicant responded that the term "should be [sufficient]" refers to a possibility that a unit would exceed a 40-year design basis number of cycles. The applicant stated that, when the 60-year projections of Table 4.3-3 are compared to the 40-year design basis quantities, the 40-year design basis number of events are bounding for 60 years. The applicant also stated that by using the Metal Fatigue of Reactor Coolant Pressure Boundary Program, the applicant ensures that the actual transients remain below the projected number of events for 60 years; thus, the CUFs for these RPV components during the period of extended operation would be maintained less than the allowable of 1.0. In the case of the stud hole inserts, LRA Table 4.3-3 shows that the projected 60-year CUF for the stud hole inserts is 1.3278, which exceeds the allowable of 1.0. The applicant stated that the Metal Fatigue of Reactor Coolant Pressure Boundary Components AMP will monitor the actual transient cycles to ensure that the CUF will not exceed the allowable of 1.0 for the stud hole inserts. When the CUF approaches 1.0, the applicant will take appropriate actions in accordance with the Metal Fatigue of Reactor Coolant Pressure Boundary Program. Based on the above evaluation, the staff's concern as described in RAI 4.3.2.1-6 is resolved.

In RAI 4.3.2.1-7 (April 14, 2011), the staff asked the applicant to demonstrate that multiplying a factor of 1.5 to the 40-year CUF is appropriate, or at a minimum, conservative. By letter dated May 12, 2011, the applicant responded that this approach has been shown to be conservative through operating history, as shown in LRA Table 4.3-2. The applicant calculated the CUF in accordance with ASME Code Section III, paragraph NB-3222.4(e)(5).

The applicant stated that when the calculated 60-year CUF approaches 1.0, the CUF analysis will be managed through the Metal Fatigue of Reactor Coolant Pressure Boundary Program, and the TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

The staff finds that the Metal Fatigue of Reactor Coolant Pressure Boundary Program ensures that the number of transients actually experienced during the period of extended operation remains below the assumed number, or that appropriate corrective actions maintain the design and licensing basis by other means. The effects of fatigue will therefore be managed for the period of extended operation. Those TLAAs will be dispositioned in accordance with 10 CFR 54.21(c)(1)(iii). The staff also finds that the applicant showed that the CUF is directly proportional to the transient cycle count, in accordance with ASME Code Section III, paragraph NB-3222.4(e)(5). Therefore, the staff's concern as described in RAI 4.3.2.1-7 is resolved.

The staff reviewed LRA Table 4.3-3 and noted that, the 40-year CUF for the reactor studs of Units 1 and 2 is 0.3372, and the 40-year CUF value for the stud hole inserts is 0.885. During its audit, the staff noted that Stud No. 30 of Unit 2 had rotated inadvertently during a de-tensioning process, causing it to partially engage inside the stud hole insert and causing damage to both Stud No. 30 and its stud hole insert. The applicant's design change package to address the issue conservatively estimated the damaged areas of the stud hole insert bearing surfaces to be 17 percent of the original area of contact. The applicant replaced Stud No. 30 and performed an evaluation of the stud hole insert, determining that the nonconforming condition of the stud insert should be dispositioned as "Use-As-Is."

The staff noted that the reduced load-bearing surfaces of the partially rolled stud hole insert would increase the stress level applied to the stud and to the stud hole insert, which could affect assumptions used in the fatigue analyses. The staff also noted that the stud nut, washer, and associated collar were not damaged during this event, and that the stud was replaced. By letter dated September 22, 2011, the staff issued RAI 4.3-8, requesting that the applicant justify that the assumptions and results of the fatigue analyses of these components remain valid, when considering the operating experience related to the stud hole insert, and that cumulative fatigue damage will be managed for the period of extended operation.

In its response dated November 21, 2011, the applicant stated that the damage to the stud hole insert was along only about 17 percent of the length of the lug. The applicant clarified that the damage was radially inward from the location of the maximum usage factor (which would occur at the intersection of the lug and the vertical cylinder surface of the insert). In addition, the applicant explained that the current CUF calculation of 0.8852 is very conservative; the stress pairing that contributes the most to fatigue was analyzed for 13,177 events (when only 10 events were required), which adds about 0.4 to the CUF. Therefore, the applicant concluded that the reported CUF of 0.8852 is bounding, and the damage will not affect the number of analyzed design transients. The applicant also stated that the Metal Fatigue of Reactor Coolant Pressure Boundary Program will maintain this margin for the original fatigue CUF analysis during the period of extended operation by ensuring that the specified quantity of 10 events is not exceeded.

Based on the applicant's response, the staff was not clear as to what "event" was analyzed for 13,177 cycles and what document (e.g., design specification, Code, or Standard) required only 10 of these events to be analyzed. The staff was also not clear as to which transient is being monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program for the "specified 10 events." The staff reviewed LRA Table 4.3-2, and it was not clear which transient is being monitored. By letter dated January 31, 2012, the staff issued RAI 4.3-8a (follow-up) to request these clarifications.

In its response to RAI 4.3-8a (follow-up) dated February 16, 2012, the applicant stated that the primary side hydrostatic test transient (10 cycles) was paired with 13,177 of the 13,200 unit unloading at 5 percent of full power per minute transient in the design fatigue CUF analysis for the stud hole insert. In addition, the applicant clarified that the transients used in the design fatigue CUF analysis for the stud hole insert are specified in the RPV design specification, which was provided as part of the response. The staff's evaluation associated with the overall aging management of the damaged stud hole insert is documented in SER Section 3.0.3.2.2 for the Reactor Head Closure Studs Program.

With respect to metal fatigue, the staff noted that that applicant's response to RAI 4.3-8, dated November 21, 2011, stated that damage to the stud hole insert—along only about 17 percent of the length of the lug and radially inward from the location of the maximum usage factor (at the intersection of the lug and the vertical cylinder surface of the insert)—is such that the bending moment loading on the lugs at the maximum usage factor location is not as great as at the damaged location. Therefore, the increase in stress at the maximum usage factor location would be less than 17 percent. It was not clear to the staff how the applicant made these determinations. Therefore, by letter dated March 21, 2012, the staff issued RAI B2.1.3-2b requesting in Part 6, that the applicant justify how it determined that the increase in the stress at the maximum usage factor location would be less than 17 percent and that the increase in stress at this location would not result in exceeding the Code design limit CUF of 1.0.

In its response to RAI B2.1.3-2b, Part 6, dated April 17, 2012, the applicant provided an explanation related to the effects of the damaged stud hole insert on metal fatigue. The applicant described the design and configuration of the stud hole insert, including an explanation of the damaged area and location of the calculated maximum CUF. The staff noted that the deformation of the stud hole insert occurred away from the location of the calculated maximum CUF.

The applicant stated that the bearing damage does not create higher peak stress intensities that would cause the CUF to increase as a result of additional stress concentration, and the bending stress is less at the edge of the bearing deformation; because it is radially inward from the location of the maximum usage factor at the lug-ID wall juncture, the moment arm is reduced. In addition, the applicant stated that, according to the calculation for the stud hole insert, the fillet radii could not be modeled in the 3-D finite element analysis, and the results already include high stress concentration. The staff noted that the calculated peak stress values would be higher than the actual values because of the difference between the modeling and the actual layout of the stud hole insert.

The applicant clarified that the largest contribution to the design CUF value of 0.8852 is due to the pairing of cold hydrostatic test and unit loading at 5 percent of full power per minute. The staff noted that the applicant used 13,177 cycles for this pairing when only 10 occurrences of cold hydrostatic test needed to be considered, as defined by the design specification. The applicant stated that using 10 cycles of the cold hydrostatic test would reduce the calculated CUF by 0.470. The applicant also provided an explanation of how the CUF value could be further refined consistent with the provisions defined by ASME Code Section III, paragraph NB-3222.4

The staff noted that ASME Code Section III, paragraph NB-3222.4(e), requires that the CUF not exceed 1.0. Based on the available refinement in the maximum CUF value, the staff finds it reasonable that there is margin in the calculated maximum CUF for the stud hole insert because of (a) the conservative methods used in the design transients to calculate the CUF, as described above, and (b) the applicant's use of a combination of primary, secondary, and peak stresses based on a high stress concentration when compared to the actual layout of the stud hole insert.

The staff also noted that the applicant stated in LRA Section 4.3.2.1 that the effect of fatigue for the closure studs and stud hole inserts will be managed by the Metal Fatigue of Reactor Coolant Pressure Boundary Program. The program ensures that the number of transients actually experienced by the component during the period of extended operation remains below the assumed number of cycles in the analysis; otherwise, corrective actions will be taken.

The staff finds the applicant's response to RAI B2.1.3-2b, Part 6, acceptable because the ASME Code design limit CUF of 1.0 is not exceeded, as discussed above, and the applicant is managing the effects of fatigue with its Metal Fatigue of Reactor Coolant Pressure Boundary Program to ensure the validity of CUF analyses through the period of extended operation. Therefore, the staff's concern identified in RAI B2.1.3-2b, Part 6, is resolved. The staff's evaluation of the applicant's responses to RAI B2.1.3-2b, Parts 1-5, is documented in SER Section 3.0.3.2.2.

The staff finds that, for the reactor pressure nozzles, flange, and studs, the Metal Fatigue of Reactor Coolant Pressure Boundary Program will adequately manage the effects of CUF by ensuring that the number of cycles for each transient actually experienced during the period of extended operation remains below the respective 40-year design basis value. Otherwise, appropriate corrective actions to maintain the design and licensing basis by other acceptable means will be taken. The effects of fatigue will, therefore, be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

4.3.2.1.3 UFSAR Supplement

LRA Section A3.2.1.1 provides the UFSAR supplement summarizing its TLAA for the RV, nozzles, head, and studs. The staff reviewed LRA Section A3.2.1.1, consistent with the review procedures in SRP-LR Section 4.7.3.2, which state that the staff confirms that the UFSAR supplement includes a summary description of the evaluation of each TLAA. Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.7.2.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the TLAA for the RV, nozzles, head, and studs, as required by 10 CFR 54.21(d).

4.3.2.1.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the CUF analyses for the RRVCHs and the analyses for the repaired Unit 1 BMI nozzles (fatigue crack growth, CUF, and corrosion) remain valid for the period of extended operation. The staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions of RPV nozzles, flange, and studs (including the stud hole inserts) will be adequately managed for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation of the RV, nozzles, head, and studs, as required by 10 CFR 54.21(d).

4.3.2.2 Control Rod Drive Mechanism Pressure Housings and Core Exit Thermocouple Nozzle Assemblies

4.3.2.2.1 Summary of Technical Information in the Application

LRA Section 4.3.2.2 describes the applicant's metal fatigue TLAA for Unit 1 and Unit 2 CRDM pressure housings and CETNAs. The applicant stated that these components were replaced with the RRVCHs. In addition, the CRDM pressure housings and CETNAs were designed to the Class 1 requirements of the ASME Code Section III, 1989 edition (no addenda).

The applicant stated the Unit 1 and 2 replacement RV heads, including CRDMs and CETNAs, were analyzed for a 40-year design life at the time of replacement; therefore, they are valid for the period of extended operation. The applicant dispositioned the TLAA for the CRDMs and CETNAs in accordance with 10 CFR 54.21(c)(1)(i) to demonstrate that the analysis remains valid for the period of extended operation.

4.3.2.2.2 Staff Evaluation

The staff reviewed LRA Section 4.3.2.2 and the metal fatigue TLAA for the CRDMs and CETNAs to confirm, pursuant to 10 CFR 54.21(c)(1)(i), that the analysis remains valid for the period of extended operation.

The staff reviewed the applicant's TLAA and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.3.3.1.1.1, which state that the operating transient experience and a list of the assumed transients used in the existing CUF calculations for the current operating term are reviewed to ensure that the number of assumed transients would not be exceeded during the period of extended operation.

LRA Section 4.3.2.1 states that the Unit 1 RPV head was replaced during the fall of 2009, and the Unit 2 RPV head was replaced during the spring of 2010. The staff noted that the CRDM pressure housings and CETNAs were designed to the Class 1 requirements of the ASME Code Section III, 1989 edition (no addenda.) In addition, these components were designed and qualified for 40 years, which extends the design lives (2049 for Unit 1 and 2050 for Unit 2) beyond the period of extended operation. Since these components were designed to ASME Code Section III, they were required to have a CUF value of less than 1.0 for the design life (i.e., 40 years) in order to be qualified for service. The staff reviewed LRA Table 4.3-3, which provides the CUF values for RV head components, and noted that the 40-year CUF values were less than the Code design limit of 1.0. Because the fatigue analyses for these components determined a CUF less than the Code limit beyond the period of extended operation, the staff finds that these analyses will remain valid for the period of extended operation.

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses for the CRDM pressure housings and CETNAs remain valid for the period of extended operation. Additionally, the analyses meet the acceptance criteria in SRP-LR Section 4.3.2.1.1.1 because the design life of the RV head for Units 1 and 2 include the CRDM pressure housings and CETNAs, and the associated fatigue analyses extend beyond the period of extended operation.

4.3.2.2.3 UFSAR Supplement

LRA Section A3.2.1.2 provides the UFSAR supplement summarizing the metal fatigue TLAA for the CRDM pressure housings and CETNAs. The staff reviewed LRA Section A3.2.1.2, consistent with the review procedures in SRP-LR Section 4.3.3.2, which state that the reviewer should confirm that the applicant provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA.

Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.3.2.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the metal fatigue TLAA for the CRDM pressure housings and CETNAs, as required by 10 CFR 54.21(d).

4.3.2.2.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue analyses for the CRDM pressure housings and CETNAs remain valid for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.2.3 Reactor Coolant Pump Pressure Boundary Components

4.3.2.3.1 Summary of Technical Information in the Application

LRA Section 4.3.2.3 describes the applicant's metal fatigue TLAA for the Unit 1 and Unit 2 RCP pressure boundary components. The applicant stated that there are four Model 100 RCPs for each reactor that were designed to the Class 1 requirements of ASME Code Section III, 1971 edition, with addenda through summer 1973. Furthermore, this design code requires a fatigue analysis per NB-3222.4(e) or a fatigue waiver per NB-3222.4(d).

The fatigue analyses for the RCP pressure boundary components were performed with transients consistent with those assumed in UFSAR Table 3.9-8, with additional cooling water and seal injection transients. The LRA states that the analyses demonstrated code compliance for most RCP components by satisfying the six criteria for a fatigue waiver. The exceptions are those components for which the range of primary plus secondary stress intensity exceeds 3 S_M (design stress intensity) for normal and upset conditions, which include the casing (CUF of 0.4), thermal barrier flange (CUF of 0.8287), cooling coils (CUF of 0.25), seal injection nozzle (CUF of 0.85), and thermal barrier cooling water nozzle (CUF of 0.4525). The applicant stated that Westinghouse equipment specifications include safety injection and thermal barrier cooling water transients that are specific to the RCP auxiliary nozzles, cooling coils, and the thermal barrier flange at the holes.

The applicant dispositioned the TLAA for the thermal barrier flange at the holes and the seal injection nozzles in accordance with 10 CFR 54.21(c)(1)(i) to demonstrate that the analyses remain valid for the period of extended operation. The applicant also dispositioned the TLAA for the RCP casing, thermal barrier cooling coils, and the thermal barrier water nozzles in accordance with 10 CFR 54.21(c)(1)(ii) to demonstrate that the analyses have been projected to the end of the period of extended operation. In addition, the applicant dispositioned the TLAA for the fatigue waivers of RCP pressure boundary components in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of fatigue on the RCP pressure-retaining components will be adequately managed by the Metal Fatigue of Reactor Coolant Pressure Boundary Program for the period of extended operation.

4.3.2.3.2 Staff Evaluation

The staff reviewed LRA Section 4.3.2.3 and the metal fatigue TLAAs for the RCP pressure boundary components to confirm, pursuant to 10 CFR 54.21(c)(1)(i), that the analysis remains valid for the period of extended operation. The staff also confirmed, pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses have been projected to the end of the period of extended operation and, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue will be adequately managed by the Metal Fatigue of Reactor Coolant Pressure Boundary Program for the period of extended operation.

The staff reviewed the applicant's TLAAs for the thermal barrier flange at the holes and the seal injection nozzles and the corresponding disposition of 10 CFR 54.21(c)(1)(i), consistent with the review procedures in SRP-LR Section 4.3.3.1.1.1. These procedures state that the operating transient experience and a list of the assumed transients used in the existing CUF calculations for the current operating term are reviewed to ensure that the number of assumed transients would not be exceeded during the period of extended operation.

The staff also reviewed the applicant's TLAAs for the RCP casing, thermal barrier cooling coils, thermal barrier water nozzles, and the corresponding disposition of 10 CFR 54.21(c)(1)(ii), consistent with the review procedures in SRP-LR Section 4.3.3.1.1.2. These procedures state that the revised CUF calculations are reviewed to ensure that the CUF remains less than or equal to 1.0 at the end of the period of extended operation.

In addition, the staff reviewed the applicant's TLAA for the fatigue waivers of RCP pressure boundary components and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), consistent with the review procedures in SRP-LR Section 4.3.3.1.1.3. These procedures state that the reviewer should confirm the appropriateness of the applicant's program for monitoring and tracking the number of critical thermal and pressure transients for the selected RCS components.

The applicant stated that the fatigue and fatigue waiver analyses have been updated to incorporate redefinitions of loads and DBEs, operating changes, power uprate, and other modifications. The staff finds it appropriate that the applicant updated the fatigue and fatigue waiver analyses for the RCP pressure boundary components because these analyses currently account for the actual equipment configuration and actual stresses caused by the operating conditions for these components at the applicant's site.

The staff noted that the fatigue analyses for the thermal barrier flange at the holes and seal injection nozzles indicate that the only transient that is significant to fatigue is the step change in seal injection flow temperature (180 cycles). The applicant described, in LRA Section 4.3.2.3, that this transient will occur when the charging pump suction is switched from the volume control tank to the refueling water storage tank (RWST) and back. In addition, the site does not operate in this manner, and the equipment failure that would cause the auto-swap inadvertently has never happened. The staff noted that the fatigue analyses for the thermal barrier flange at the holes and seal injection nozzles were performed with transients consistent with those assumed in UFSAR Table 3.9-8 (reproduced in LRA Table 4.3-2) and the additional transient described above. The staff reviewed LRA Table 4.3-2 and noted that there is margin between the UFSAR design cycles and the 60-year projected cycles. The staff finds it reasonable that the 40-year design CUF for the thermal barrier flange at the holes (0.8287) and seal injection nozzles (0.85) will not be exceeded during the period of extended operation because there is margin between the 60-year projected cycles and the UFSAR design cycles, and the most significant transient that contributes to these CUFs has not occurred and is not expected to occur during normal operation (consistent with the way the applicant operates its units).

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses for the thermal barrier flange at the holes and seal injection nozzles remain valid for the period of extended operation. Additionally, it meets the acceptance criteria in SRP-LR Section 4.3.3.1.1.1 because the number of transients used in the fatigue analyses to calculate the CUF will not be exceeded during the period of extended operation, and the most significant contributor to CUF values is not expected to occur at the site.

The staff noted that for the CUF values of the RCP casing (0.4), thermal barrier cooling coils (0.25), and thermal barrier water nozzles (0.4525), the applicant extrapolated to 60 years by multiplying the CUFs by a factor of 1.5, which still demonstrated that the design Code limit of 1.0 was not exceeded, as described in LRA Section 4.3.2.3. The staff finds the use of this 1.5 factor reasonable to be applied to the 40-year design CUF values because the resulting estimated 60-year CUF values provide a gauge of how much margin is available before the design limit of 1.0 is reached. For the RCP casing, thermal barrier cooling coils, and thermal barrier water nozzles, the staff noted that there is 32 percent margin or more between the 60-year projected CUF values and the Code design limit of 1.0.

Based on the above, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses for the RCP casing, thermal barrier cooling coils, and thermal barrier water nozzles have been projected to the end of the period of extended operation. Additionally, it meets the acceptance criteria of SRP-LR Section 4.3.2.1.1.2 because the applicant demonstrated that the 60-year projected CUF values will be less than the ASME Code Section III, design limit of 1.0 through the period of extended operation with significant margin.

The staff noted that the components of the RCP that form part of the RCPB are subject to an ASME Code fatigue analysis per NB-3222.4(e) or a fatigue waiver per NB-3222.4(d). In addition, these analyses demonstrated ASME Code compliance for most RCP components by satisfying the six criteria for a fatigue waiver, with the exception of those components described above. The applicant dispositioned these fatigue waiver TLAAs in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the Metal Fatigue of Reactor Coolant Pressure Boundary Program will manage effects of fatigue for the period of extended operation. However, it is not clear how the applicant's program will ensure that the fatigue waiver for RCP pressure-retaining components will remain valid for the period of extended operation. By letter dated September 22, 2011, the staff issued RAI 4.3-15, requesting that the applicant describe how the Metal Fatigue of Reactor Coolant Pressure Boundary Program will manage the effects of cumulative fatigue damage through the period of extended operation for those RCP components and associated TLAAs that satisfied the six criteria for a fatigue waiver per NB-3222.4(d).

In its response dated November 21, 2011, the applicant stated that fatigue waiver requirements are dependent on the numbers of anticipated transients over the life of the plant. In addition, the fatigue waiver for the RCPs was performed with transients consistent with those identified in UFSAR Table 3.9-8. The staff noted that as long as the number of transients that occur for a unit remain bounded by the 40-year numbers of cycles assumed in the fatigue waiver, the waiver will remain valid. The staff noted that the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program ensures that the number of transients actually experienced during the period of extended operation remains below the assumed number in the fatigue waiver or that corrective actions will be taken. The staff's evaluation of the Metal Fatigue of Reactor Coolant Pressure Boundary Program is documented in SER Section 3.0.3.2.28.

The staff finds the applicant's response acceptable because the applicant confirmed that the transients assumed in the fatigue waiver are consistent with those in the UFSAR and the Metal Fatigue of Reactor Coolant Pressure Boundary Program. This will ensure that corrective actions will be taken if the assumptions made in the fatigue waiver are approached. The staff's concern described in RAI 4.3-15 is resolved.

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the intended functions of the RCP pressure boundary components with fatigue waivers will be adequately managed for the period of extended operation. Additionally, it meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.3 because the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program monitors and tracks the number of design basis transients that will occur through the period of extended operation. Additionally, this program includes action limits and corrective actions that will ensure that the assumptions made in the fatigue waiver will not be exceeded during the period of extended operation.

4.3.2.3.3 UFSAR Supplement

LRA Section A3.2.1.3 provides the UFSAR supplement summarizing the metal fatigue TLAA for the RCP pressure boundary components. The staff reviewed LRA Section A3.2.1.3 consistent with the review procedures in SRP-LR Section 4.3.3.2, which state that the reviewer should confirm that the applicant provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA.

Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.3.2.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the metal fatigue TLAA for the RCP pressure boundary components, as required by 10 CFR 54.21(d).

4.3.2.3.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue analyses for the thermal barrier flange at the holes and the seal injection nozzles remain valid for the period of extended operation. The staff also concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that the fatigue analyses for the RCP casing, thermal barrier cooling coils, and thermal barrier water nozzles have been projected to the end of the period of extended operation. In addition, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the RCP pressure boundary components with fatigue waivers will be adequately managed for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.2.4 Pressurizer and Pressurizer Nozzles

4.3.2.4.1 Summary of Technical Information in the Application

LRA Section 4.3.2.4 describes the applicant's metal fatigue TLAA for the pressurizer and pressurizer nozzles. LRA Section 4.3.2.4 states that the Westinghouse Series 100 pressurizers are vertical cylindrical vessels with hemispherical top and bottom heads, constructed of carbon steel, with austenitic stainless steel cladding on all surfaces exposed to the reactor coolant. The pressurizers and their integral support skirts are ASME Code Class 1, designed to ASME Code Section III, 1974 edition. As such, pressure-retaining and support components of the pressurizer are subject to an ASME Code Section III fatigue CUF analysis.

The LRA states that the applicant identified new DBEs that were not included in the original fatigue CUF analyses. The applicant re-evaluated the fatigue CUF analyses considering the new parameters and transient cycles. The applicant found that the impact of 10 cold over-pressurization mitigation system activation events on the fatigue CUF of the pressurizer is negligible.

Two components, the safety and relief nozzles and the manway, were previously exempt from a fatigue usage factor analysis by a waiver under ASME Code Section III, NB-3222.4(d). However, with an additional 6,000 pressure fluctuations (10 events with 600 pressure cycles per event), these two components are no longer exempt from the fatigue analysis requirement. The applicant included the fatigue CUF in the stress analysis results for these components, as shown in LRA Table 4.3-4. The applicant stated that the TLAA for the fatigue CUF analyses of the safety and relief nozzles and the seismic support lugs are dispositioned in accordance with 10 CFR 54.21(c)(1)(ii); the TLAAs for the fatigue CUF analyses of the manway and the remaining items identified in LRA Table 4.3-4 are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

In addition, the applicant evaluated the effects of pressurizer insurge-outsurge transient cycles based on the Westinghouse report, WCAP-14950, and plant operations from the last seven heatups and seven cooldowns for Units 1 and 2, combined. These heatups and cooldowns are assumed to represent all past and future operations in terms of pressurizer insurge-outsurge and surge line stratification activity. All components were qualified using the 40-year CLB cycles and incorporated into the Metal Fatigue of Reactor Coolant Pressure Boundary Program.

The applicant installed preemptive structural weld overlays (SWOLs) on pressurizer spray, relief, safety, and surge nozzles in both units in accordance with NRC-approved relief request RR-ENG-2-43. This modification is to mitigate the Alloy 82/182 dissimilar metal welds, which are susceptible to PWSCC, at the subject pressurizer nozzles.

As part of the weld overlay design, the applicant performed crack growth analyses using the transients listed in UFSAR Table 3.9-8, spread evenly over a 40-year and a 60-year plant life. These analyses determine the amount of time necessary for a crack to propagate from 3/4 wall thickness to the interface between the SWOL and the pipe. This acceptance criterion is to confirm that an unidentified crack will not propagate to the SWOL interface during a 10-year ISI interval. If the crack is projected to propagate into the SWOL, then an inspection interval is to be established to ensure that the crack will not propagate into the SWOL before the next inspection. Since the crack is not qualified for the life of the plant, but only the inspection interval, the fatigue crack growth analysis is not a TLAA in accordance with 10 CFR 54.3(a), Criterion 3 (i.e., subparagraph to 10 CFR 54.3(a)(3)).

The applicant performed fatigue CUF evaluations for the limiting locations outside the SWOL. In the region within the SWOL, the stresses due to pressure and piping reaction loads are lower because of the increase in the pipe wall thickness as a result of the SWOL. Therefore, these stress evaluations are not TLAAs in accordance with 10 CFR 54.3(a), Criterion 2.

4.3.2.4.2 Staff Evaluation

The staff reviewed LRA Section 4.3.2.4 and the metal fatigue TLAA for the pressurizer and pressurizer nozzles to confirm that the fatigue CUF analyses and crack growth analyses for the pressurizer and associated nozzles remain valid for the period of extended of operation, that they are projected to the end of the period of extended operation, or they will be adequately managed for the period of extended operation, in accordance with 10 CFR 54.21(c)(i), (ii), or (iii), respectively. The staff reviewed the applicant's TLAA and the corresponding disposition, consistent with the review procedures in SRP-LR Sections 4.3.2.1 and 4.7.3, which state that the review of the TLAA provides assurance that the aging effect is properly addressed through the period of extended operation.

<u>Impact of Plant Modifications, Redefined Loads and New Design Basis Events</u>. LRA Section 4.3.2.4 discusses plant modifications, redefined loads, and newly identified DBEs. In RAI 4.3.2.4-1 (April 14, 2011), the staff asked the applicant to describe in detail how the plant modifications affect the loads on the pressurizer and associated components (e.g., closures, nozzles, heaters, and support skirts) and to discuss the redefined loads and newly identified DBEs.

By letter dated May 12, 2011, the applicant responded that the pressurizer weld overlay plant modifications, and the associated effects (i.e., redefined loads), have been considered in the design analyses. Other plant modifications, such as T_{hot} reduction, RSGs, and reactor thermal power uprate, did not affect the loads on the pressurizer and associated components.

The applicant stated further that the newly identified DBEs are the COMS actuation and the pressurizer insurge-outsurge events. Actuation of the COMS, which was implemented to satisfy the TS LCO 3.4.9.3 requirement for LTOP, was not initially incorporated into the fatigue CUF analyses. Later, the applicant incorporated the 6,000 pressure cycles, as defined by the NSSS vendor, into the ASME Code design specification. Based on the above evaluation, the staff's concern described in RAI 4.3.2.4-1 is resolved.

LRA Section 4.3.2.4, page 4.3-18, states that "[t]he stress reports evaluated the effect on the pressurizer of 10 cold over-pressurization mitigation system activation events. The contribution of these thermal effects to the fatigue usage can be neglected." In RAI 4.3.2.4-3 (April 14, 2011), the staff asked the applicant to explain why the contribution of these thermal effects to the fatigue CUF can be neglected.

By letter dated May 12, 2011, the applicant responded that it had evaluated the effects of 10 COMS activation events on the pressurizers and found that many existing transient loadings that have already been included in the design basis of the pressurizer are much more severe than the RCS cold over-pressurization event. The addition of the less-severe COMS transients in the design basis of the pressurizer had a minimal effect on the component ASME Code analysis.

The applicant stated further that COMS actuation is a pressure transient. The thermally induced stresses associated with the COMS transient are small and the number of cycles of thermal events is low (10). Therefore, the contribution of these thermal effects to the fatigue CUF can be neglected.

The staff finds the RAI response acceptable in that the applicant considered the redefined loads and newly identified DBEs in the fatigue CUF and fatigue crack growth calculations of the pressurizer. The staff also finds the RAI response acceptable in that the applicant analyzed the impact of the COMS pressure and that the impact is minimal to the pressurizer. Based on the above evaluation, the staff's concern described in RAI 4.3.2.4-3 is resolved.

Crack Growth Analysis of Overlaid Pressurizer Nozzles. LRA Section 4.3.2.4 states, "[t]he fatigue crack growth analysis for pressurizer spray, relief, safety, and surge nozzle preemptive overlays is not a TLAA because the crack is not qualified for the life of the plant, but only the inspection interval." In RAI 4.3.2.4-4 (April 14, 2011), the staff asked the applicant to confirm that every overlaid Alloy 82/182 weld in pressurizer spray, relief, safety, and surge nozzles will be inspected every 10 years and that the transient cycles will be monitored to confirm that the fatigue crack growth analysis bounds the actual transient cycles.

By letter dated May 12, 2011, the applicant responded that the overlaid Alloy 82/182 welds in pressurizer spray, relief, safety, and surge nozzles will be inspected every 10 years using a qualified performance demonstration initiative (PDI) ultrasonic technique in accordance with the ISI Program and Materials Reliability Program (MRP)-139/ASME Code Case N-770-1 requirements. The applicant inspected the subject welds in spring 2010 for Unit 2 (2RE14) and fall 2009 for Unit 1 (1RE15), and no flaws were identified. The third ISI interval, which is scheduled to end in 2020, will adopt ASME Code Case N-770-1, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities Section XI, Division 1."

The applicant used 40 years of transient cycles in the fatigue crack growth calculation as part of the weld overlay design. The applicant stated that the fatigue crack growth analyses were not identified as a TLAA; thus, they did not require a disposition. However, the fatigue crack growth analyses, which support the weld overlay work, were performed with the same number of transients as the design fatigue analyses for the pipe. These transients will be monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program.

The staff notes that ASME Code Case N-770-1 has been incorporated by reference with conditions in the final rule for 10 CFR 50.55a, which was published in the *Federal Register* (FR) on June 21, 2011 (76 FR 36232). All licensees and applicants must follow ASME Code Case N-770-1, as conditioned in 10 CFR 50.55a, in the inspection of Alloy 82/182 welds. The staff no longer accepts guidance in MRP-139 for the inspection of Alloy 82/182 welds. The staff notes that the applicant needs to follow the inspection requirements in the NRC-approved relief request for overlaid pressurizer nozzles first. In general, the staff authorizes the weld overlay relief request for only one 10-year ISI interval. After the relief request expires at the end of the 10-year ISI interval, the applicant may follow the inspection requirements of Code Case N-770-1 with conditions in 10 CFR 50.55a, or it may resubmit the weld overlay relief request for the subsequent 10-year ISI interval and follow the requirements in the re-approved relief request. The staff finds that the fatigue crack growth analyses for weld overlays are not a TLAA because the overlaid Alloy 82/182 dissimilar metal welds will be inspected once every 10 years. Further, the staff finds that the applicant will monitor the transient cycles used in the fatigue crack growth analysis for the overlaid pressurizer nozzles through the Metal Fatigue of Reactor Coolant

Pressure Boundary Program. Based on the above evaluation, the staff's concern as described in RAI 4.3.2.4-4 is resolved.

Fatigue Usage Factor Analyses for Pressurizer Components with CUF Less Than 0.4. LRA Section 4.3.2.4, page 4.3-19, states that, as shown in LRA Table 4.3-4, the fatigue usage factor analyses of the pressurizer safety and relief nozzles and the seismic support lugs demonstrate the 40-year CUFs to be less than 0.4. When multiplied by 1.5 (60/40) to account for the 60-year period of extended operation, the projected CUFs do not exceed 0.6, providing a large margin to the ASME Code acceptance criterion of 1.0. The staff finds that the CUF values for safety and relief nozzles and the seismic support lugs have been satisfactorily projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

Fatigue Usage Factor Analyses for Pressurizer Components with CUF Greater Than 0.4. LRA Table 4.3-4 shows that several pressurizer components (other than the safety and relief nozzles and seismic support lugs) will have their CUF greater than the allowable of 1.0 at the end of 60 years, using a simple multiplier of 1.5 on the 40-year CUF values. The Metal Fatigue of Reactor Coolant Pressure Boundary Program will ensure that the number of transients actually experienced during the period of extended operation remains below the assumed number in the CUF calculations. When the CUF approaches the allowable limit of 1.0, the applicant will take appropriate corrective actions to maintain the CUF to less than 1.0 by acceptable means.

In RAI 4.3.2.4-6 (April 14, 2011), the staff asked the applicant to describe exactly how the Metal Fatigue Reactor Coolant Pressure Boundary AMP will monitor the transient cycles and track the CUFs of components. The staff also asked the applicant to describe in detail the corrective actions and acceptable means if the CUF exceeds the allowable limit.

By letter dated May 12, 2011, in response to RAI 4.3.2.4-2 (which also addresses RAI 4.3.2.4-6), the applicant stated that the Metal Fatigue of Reactor Coolant Pressure Boundary Program monitors the number of actual plant transients to ensure that they do not exceed the number of transients used in the design fatigue analyses for the pressurizer components. The applicant reviewed the pressurizer design basis to ensure that it was within the scope of the AMP or that the AMP was enhanced to consider the additional transients included in the pressurizer design basis. From this review, the staff confirmed that the monitoring program also ensures that the number of transient cycles experienced by the plant will be within the cycles used in the pressurizer design basis during the period of extended operation.

The applicant stated further that the current procedure for the Metal Fatigue of Reactor Coolant Pressure Boundary Program requires the control room to complete daily screening data sheets. If a transient occurs, a transient-specific datasheet is completed to record the plant's conditions during the event. This process will be changed for the period of extended operation to run computer software to assess plant instrumentation data recorded by the plant process computer and to identify the transients that have occurred. At least once per fuel cycle, the information will be validated to ensure that an accurate transient count and the actual transient severity remains within the design basis. The cycle counts are then compared to the action limits, and corrective action is initiated when actual transient cycles exceed 80 percent of their design limit. Corrective actions are discussed in more detail in LRA Section B3.1 and in the applicant's response to RAI 4.3.2.11-3, and are evaluated in SER Section 4.3.2.11. The term "other acceptable means" refers to actions other than counting cycles, which are meant to address fatigue at the plant. When other acceptable corrective action is required, a 10 CFR 50.59

review is performed to determine if the methods and results are in line with the plant's CLB or if regulatory review is needed.

The applicant explained that the appropriate corrective actions are described in LRA Section B3.1 and LRA Table A4-1, Commitment No. 30. As part of its response to RAI 4.3.2.11-3, the applicant revised the corrective actions (Element 7) in Metal Fatigue of Reactor Coolant Pressure Boundary Program in LRA Section B3.1, as documented in the applicant's letter dated May 12, 2011. The enhanced corrective action description includes fatigue reanalysis, repair, replacement, or augmented inspections. The staff finds that the revised LRA Section B3.1 provides additional detailed monitoring on the fatigue CUF calculation and associated corrective actions. The staff finds it is acceptable that the applicant clarified and enhanced the Metal Fatigue Reactor Coolant Pressure Boundary AMP to monitor the transients in the CUF calculations for pressurizer components with 40-year CUF values greater than 0.4 in LRA Table 4.3-4. Based on the above evaluation, the staff's concern described in RAI 4.3.2.4-6 is resolved.

The staff finds that, pursuant to 10 CFR 54.21(c)(1)(iii), the effects of fatigue in these pressurizer subcomponents, as shown in LRA Table 4.3-4, will be adequately managed for the period of extended operation.

4.3.2.4.3 UFSAR Supplement

LRA Section A3.2.1.4 provides the UFSAR supplement summarizing description of its TLAA for the pressurizer and pressurizer nozzles. The staff reviewed LRA Section A3.2.1.4, consistent with the review procedures in SRP-LR Section 4.7.3.2, which state that the staff confirms that the UFSAR supplement includes a summary description of the evaluation of each TLAA. Based on its review, the staff finds that the UFSAR supplement meets the acceptance criteria in SRP-LR Section 4.7.2.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the TLAA for the pressurizer and pressurizer nozzles, as required by 10 CFR 54.21(d).

4.3.2.4.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that the fatigue CUF analyses of the pressurizer safety and relief nozzles and seismic support lugs have been projected to the end of the period of extended operation. The staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the pressurizer components that have 40-year CUF values greater than 0.4, as shown in LRA Table 4.3-4, will be adequately managed in terms of fatigue usage factor for the period of extended operation. The staff concludes that the fatigue crack growth analysis for the overlaid alloy dissimilar metal welds at pressurizer nozzles is not a TLAA because the overlaid welds will be inspected periodically depending on the allowable time calculated by the flaw growth analysis, the required inspection interval in accordance with the NRC-approved weld overlay relief request, or ASME Code Case N-770-1 with conditions in 10 CFR 50.55a. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation of the pressurizer and pressurizer nozzles, as required by 10 CFR 54.21(d).

4.3.2.5 Steam Generator ASME Code Class 1, Class 2 Secondary Side, and Feedwater Nozzle Fatigue Analyses

4.3.2.5.1 Summary of Technical Information in the Application

LRA Section 4.3.2.5 describes the applicant's metal fatigue TLAA for the Units 1 and 2, RSGs. The applicant stated that the Units 1 and 2 steam generators (SGs) were replaced (in 2000 and 2002, respectively) with Westinghouse Model Delta 94 SGs and are designed for 40 years of operation (for operation until 2040 and 2042, respectively), based on design transients. In addition, the RSGs are designed and fabricated to the requirements of ASME Code Section III, 1998 edition with no addenda. The primary side of each RSG is ASME Code Class 1, and the secondary side of each RSG is ASME Code Class 2; however, the entire pressure boundary of the component is constructed in accordance with ASME Code Section III, Class 1 requirements. The applicant stated that fatigue usage factors in the SG components do not depend on flow-induced vibration or other effects that are time-dependent at steady-state conditions but depend only on effects of operational and upset transient events specified in the design specification.

The applicant dispositioned the TLAAs for the Unit 1 and 2 RSGs in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of fatigue on the RSG components will be adequately managed by the Metal Fatigue of Reactor Coolant Pressure Boundary Program for the period of extended operation.

4.3.2.5.2 Staff Evaluation

The staff reviewed LRA Section 4.3.2.5 and the TLAAs for the RSGs to confirm, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue will be adequately managed by the Metal Fatigue of Reactor Coolant Pressure Boundary Program for the period of extended operation.

The staff reviewed the applicant's disposition of the TLAA for the RSGs consistent with the review procedures in SRP-LR Section 4.3.3.1.1.3. These procedures state that the reviewer should confirm the appropriateness of the applicant's program for monitoring and tracking the number of critical thermal and pressure transients for the selected RCS components.

LRA Table 4.3-5 provides the 40-year CUF values for the RSG components, which are all less than the ASME Code design limit of 1.0, except for the primary manway studs, which have a 40-year CUF value of 7.13. In LRA Section 4.3.2.5, the applicant stated that the primary manway studs have a fatigue usage factor that exceeds the allowable of 1.0, but that they are qualified for 40 years by fatigue testing. The staff noted that the TLAA for RSGs, which includes the primary manway studs, was dispositioned in accordance with 10 CFR 54.21(c)(1)(iii). The applicant did not describe the details of the fatigue testing that was performed to qualify the primary manway studs for the Unit 1 and 2 RSGs; therefore, it was not clear how the applicant's program will manage fatigue of the primary manway studs. By letter dated September 22, 2011, the staff issued RAI 4.3-16, requesting that the applicant describe how the primary manway studs for the Unit 1 and 2 RSGs were qualified for 40 years by fatigue testing and to identify the sections of the applicable design codes that were used for the fatigue testing. In addition, the staff requested that the applicant justify how the Metal Fatigue of Reactor Coolant Pressure Boundary Program will manage cumulative fatigue damage of the primary manway studs for the Unit 1 and 2 RSGs.

In its response dated November 21, 2011, the applicant stated that the bolt fatigue testing was performed on bolts that represent the same thread size and material as the primary manway studs, and the number of fatigue test cycles was calculated to envelop the SG design transients based on a 40-year life. In addition, the fatigue tests were performed in accordance with ASME Code Section III, Appendix II, Article II-1500, as allowed per ASME Code Section III, paragraph NB-3222.4(a); therefore, the primary manway studs were qualified for fatigue by testing. In addition, the applicant stated that the fatigue test data envelop the number of cycles and the severity of the transients required by the design specification. The staff noted that the applicant's program ensures that the number and severity of transients actually experienced by the plant during the period of extended operation remain below the assumptions in the design specification or that corrective actions will be taken. The staff's evaluation of the Metal Fatigue of Reactor Coolant Pressure Boundary Program is documented in SER Section 3.0.3.2.28.

The staff finds the applicant's response acceptable because the primary manway studs were fatigue tested, in accordance with ASME Code Section III, Subsection NB and Appendix II, and the tests envelop the number of cycles and severity of the SG design transients specified in the design specification. Additionally, the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program ensures that the number and severity of transients actually experienced does not exceed the assumptions made to qualify these primary manway studs for fatigue. The staff's concerns described in RAI 4.3-16 are resolved.

The applicant stated, in LRA Section 4.3.2.5, that Westinghouse evaluated the thermal-hydraulic performance and structural integrity of the replacement Model Delta 94 SGs. This Westinghouse evaluation concludes that there is no need to revise any data point on the design transient curves as a result of the 1.4 percent uprating; therefore, the original design transient curves remain applicable. The structural evaluation performed by Westinghouse focused on critical SG components that were determined by the stress ratios and fatigue usage as reported in the analyses of record. The applicant stated that, by demonstrating that these most highly stressed components remain qualified for operation at the uprated power conditions, it may be concluded that these SG components remain structurally qualified. In addition, since the SG primary stresses remain the same while the secondary pressures are reduced as a result of uprating, all other SG components also remain structurally qualified. In the staff's SE documenting the approval of a 1.4 percent increase in reactor core thermal power levels for Units 1 and 2 (from 3,800 megawatts thermal (MWt) to 3,853 MWt), dated April 12, 2002 (ADAMS Accession No. ML021130083), the staff determined that the applicant's structural evaluation of the SG components is acceptable and that the original design parameters bound the power uprated conditions. The staff finds it appropriate that the applicant considered the 1.4 percent uprated conditions and its effect on the fatigue analyses for the RSG components.

The applicant stated in LRA Section 4.3.2.5 that, as part of its RSG program, a new upset transient, COMS, was added to the original design basis for the RCS. This transient, which potentially occurs during startup or shutdown conditions at low temperatures, has been added to the UFSAR and was assumed to occur 10 times during the 40-year design life. The staff reviewed LRA Table 4.3-2 and confirmed that this transient is listed and tracked by the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program. The staff finds it appropriate that the applicant included this new upset transient as part of its UFSAR and Metal Fatigue of Reactor Coolant Pressure Boundary Program because it is consistent with the recommendations of GALL Report AMP X.M1 to monitor all plant design transients that cause cyclic strains that are significant contributors to the fatigue usage factor. The staff noted that as long as the number of cycles of transients that occur per unit remains bounded by the 40-year numbers of cycles assumed by the analysis, the design basis fatigue evaluation remains valid. The staff noted that the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program ensures that the number of transients actually experienced during the period of extended operation remains below the assumed number or that corrective actions will be taken. The staff's evaluation of the Metal Fatigue of Reactor Coolant Pressure Boundary Program is documented in SER Section 3.0.3.2.28.

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the intended functions of the RSG components will be adequately managed for the period of extended operation. Additionally, it meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.3 because the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program monitors and tracks the number of design basis transients that will occur through the period of extended operation, and it includes action limits and corrective actions that will ensure the Code design limit of 1.0, or assumptions made to fatigue qualify the primary manway studs, will not be exceeded during the period of extended operation. Additionally, the use of the applicant's program is consistent with the recommendations of GALL Report AMP X.M1.

4.3.2.5.3 UFSAR Supplement

LRA Section A3.2.1.5 provides the UFSAR supplement summarizing the metal fatigue TLAA for the RSG components. The staff reviewed LRA Section A3.2.1.5 consistent with the review procedures in SRP-LR Section 4.3.3.2, which state that the reviewer should confirm that the applicant provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA.

Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.3.2.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the metal fatigue TLAA for the RSG components, as required by 10 CFR 54.21(d).

4.3.2.5.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the RSG components will be adequately managed for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.2.6 ASME Code Class 1 Valves

4.3.2.6.1 Summary of Technical Information in the Application

LRA Section 4.3.2.6 describes the applicant's metal fatigue TLAA for the ASME Code Section III, Class 1, valves. The applicant stated that its ASME Code Class 1 valves are designed to ASME Code Section III, Subsection NB, 1974 edition with summer 1975 addenda (pressurizer safety and control valves) or the 1974 edition with winter 1975 addendum (motor-operated, manual valves 3 inches and larger and all valves 2 inches and smaller). In addition, ASME Code Section III requires a fatigue analysis only for Class 1 valves with an inlet piping connection greater than 4-inch nominal pipe size.

The applicant dispositioned the TLAAs for the following ASME Code Class 1 valves in accordance with 10 CFR 54.21(c)(1)(ii) to demonstrate that the analyses have been projected to the end of the period of extended operation:

- 6-inch pressurizer safety relief valves
- 6-inch hi-head safety injection pump discharge check valves
- 8-inch hi-head safety injection pump discharge check valves
- 8-inch lo-head safety injection to hot leg check valves
- 12-inch safety injection to cold leg injection check valves
- 12-inch safety injection accumulator outlet valves
- 2-inch chemical and volume control system (CVCS) auxiliary spray check valves
- 2-inch RCP seal injection first and second check valves

The applicant also dispositioned the TLAA for the 12-inch residual heat remover (RHR) pump suction isolation valves in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of fatigue will be adequately managed by the Metal Fatigue of Reactor Coolant Pressure Boundary Program for the period of extended operation.

4.3.2.6.2 Staff Evaluation

The staff reviewed LRA Section 4.3.2.5 and the metal fatigue TLAAs for ASME Code Class 1 valves to confirm, pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses have been projected to the end of the period of extended operation. The staff also confirmed, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue will be adequately managed by the Metal Fatigue of Reactor Coolant Pressure Boundary Program for the period of extended operation.

The staff reviewed the applicant’s TLAAs for the ASME Code Class 1 valves, described above, and the corresponding disposition of 10 CFR 54.21(c)(1)(ii), consistent with the review procedures in SRP-LR Section 4.3.3.1.1.2. These procedures state that the revised CUF calculations are reviewed to ensure that the CUF remains less than or equal to 1.0 at the end of the period of extended operation.

The staff also reviewed the applicant’s TLAA for the 12-inch RHR pump suction isolation valves and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), consistent with the review procedures in SRP-LR Section 4.3.3.1.1.3. These procedures state that the reviewer should confirm the appropriateness of the applicant’s program for monitoring and tracking the number of critical thermal and pressure transients for the selected RCS components.

For those TLAAs for Class 1 valves dispositioned by the applicant in accordance with 10 CFR 54.21(c)(1)(ii), the staff reviewed LRA Table 4.3-6 and noted that the 60-year CUF values are less than 0.33 for each valve (SER Table 4.3-1).

Table 4.3-1 Class 1 Valves Dispositioned in Accordance with 10 CFR 54.21(c)(1)(ii)

Valve Description	40-year CUF	60-year CUF
6" pressurizer safety relief valves	0.0276	0.0414
6" hi-head safety injection pump discharge check valves	0.15	0.225
8" hi-head safety injection pump discharge check valves	0.14	0.21
8" lo-head safety injection to hot leg check valves	0.14	0.21
12" safety injection to cold leg injection check valves and safety injection accumulator outlet valves	0.05	0.075
2" CVCS auxiliary spray check valves	0.2063	0.3095
2" RCP seal injection first check valves and RCP seal injection second check valves	0.2186	0.3279

During its review of LRA Section 4.3.2.6 and LRA Table 4.3-6, the staff noted that the applicant did not provide a disposition, in accordance with 10 CFR 54.21(c)(1), of the fatigue TLAAs for the “8-inch lo-head safety injection train A/B/C to loop 1(2)A/B/C cold leg check valve” or the “3-inch [by] 6-inch pressurizer power operated relief valve (PORV).” By letter dated September 22, 2011, the staff issued RAI 4.3-22, requesting that the applicant provide and justify the dispositions for the fatigue TLAA for these two Class 1 valves in accordance with 10 CFR 54.21(c)(1).

In its response dated November 21, 2011, the applicant stated that LRA Section 4.3.2.6 and Appendix A3.2.1.6 will be revised to note that the lo-head safety injection cold leg check valve and PORV are dispositioned in accordance with 10 CFR 54.21 (c)(1)(ii). The staff confirmed that the applicant dispositioned these two valves in accordance with 10 CFR 54.21(c)(1)(ii) in LRA Section 4.3.2.6 and Appendix A3.2.1.6.

The staff finds the applicant’s response acceptable because the applicant revised its LRA to provide a disposition of the fatigue TLAAs for the “8-inch Lo-Head Safety Injection Train A/B/C To Loop 1(2)A/B/C Cold Leg Check Valve” and the “3-inch [by] 6-inch Pressurizer Power Operated Relief Valve” in accordance with 10 CFR 54.21(c)(1). The staff’s review of the applicant’s disposition is documented below, and the concern described in RAI 4.3-22 is resolved. Based on the applicant’s response, the staff noted from amended LRA Section 4.3.2.6 that the 60-year CUF values are less than 0.25 for these valves (SER Table 4.3-2).

Table 4.3-2 Additional Class 1 Valves Dispositioned in Accordance with 10 CFR 54.21(c)(1)(ii)

Valve Description	40-year CUF	60-year CUF
8" lo-head safety injection train A/B/C to loop 1(2)A/B/C cold leg check valve	0.14	0.21
3"x6" pressurizer power operated relief valve	0.16	0.24

The applicant stated that the 60-year CUF values were calculated by multiplying the 40-year CUF value by a factor of 1.5 (60/40). The staff noted that the 60-year CUF values for these ASME Code Class 1 valves remain below the ASME Code design limit of 1.0. The staff finds the use of this 1.5 factor reasonable for the 40-year design CUF values because the resulting estimated 60-year CUF values provide a gauge of how much margin is available before the design limit of 1.0 is reached. For the ASME Code Class 1 valves listed in the two tables above, the staff noted that there is 65 percent margin or more between the 60-year projected CUF values and the ASME Code design limit of 1.0.

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses for the ASME Code Class 1 valves, described above, have been projected to the end of the period of extended operation. Additionally, it meets the acceptance criteria of SRP-LR Section 4.3.2.1.1.2 because the applicant demonstrated that the 60-year projected CUF values will be less than the ASME Code Section III design limit of 1.0 through the period of extended operation with significant margin.

The staff reviewed LRA Table 4.3-6 and noted that the 40-year CUF value and 60-year CUF value for 12-inch RHR pump suction isolation valves are 0.64 and 0.96, respectively. The applicant stated that the fatigue CUF for these valves do not depend on effects that are time-dependent at steady-state conditions but depend only on effects of operational, abnormal, and upset transient events. The staff noted that as long as the number of transients that occur at the site remain bounded by the 40-year numbers of cycles assumed by the analysis, the design basis fatigue evaluation remains valid. The staff noted that the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program ensures that the number of transients actually experienced during the period of extended operation remains below the assumed number or that corrective actions will be taken. The staff's evaluation of the Metal Fatigue of Reactor Coolant Pressure Boundary Program is documented in SER Section 3.0.3.2.28.

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging related to fatigue analysis of the 12-inch RHR pump suction isolation valves will be adequately managed for the period of extended operation. Additionally, the applicant's disposition meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.3 because the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program monitors and tracks the number of design basis transients that will occur through the period of extended operation and includes action limits and corrective actions that will ensure that the Code design limit of 1.0 will not be exceeded during the period of extended operation. Additionally, the use of the applicant's program is consistent with the recommendations of GALL Report AMP X.M1.

4.3.2.6.3 UFSAR Supplement

LRA Section A3.2.1.6 provides the UFSAR supplement summarizing the metal fatigue TLAAs for the Class 1 valves. The staff reviewed LRA Section A3.2.1.6, consistent with the review procedures in SRP-LR Section 4.3.3.2, which state that the reviewer should confirm that the applicant provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA. As discussed above in RAI 4.3-22, the staff requested that the applicant provide a disposition in accordance with 10 CFR 54.21(c)(1) for the fatigue TLAAs related to the "8-inch Lo-Head Safety Injection Train A/B/C To Loop 1(2)A/B/C Cold Leg Check Valve" and the "3-inch [by] 6-inch Pressurizer Power Operated Relief Valve" and any appropriate revisions to the LRA. In its response dated November 21, 2011, the applicant amended LRA Section A3.2.1.6 to disposition the fatigue TLAAs for these two valves, in accordance with 10 CFR 54.21(c)(1)(ii).

Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.3.2.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the metal fatigue TLAAs for ASME Code Section III Class 1 valves, as required by 10 CFR 54.21(d).

4.3.2.6.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that the cumulative fatigue analyses for the ASME Code Section III Class 1 valves, except the RHR pump suction isolation valves, have been projected to the end of the period of extended operation. The staff also concludes that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the cumulative fatigue analyses for the RHR pump suction isolation valves will be adequately managed for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.2.7 ASME Code Class 1 Piping and Nozzles

4.3.2.7.1 Summary of Technical Information in the Application

LRA Section 4.3.2.7 describes the applicant's metal fatigue TLAA for the ASME Code Section III Class 1 piping and piping nozzles. The applicant stated its Class 1 reactor coolant main loop piping, surge line piping, and other ASME Code Section III Class 1 piping is designed to ASME Code Section III, Subsection NB, 1974 edition with addenda through winter 1975. In addition, the Class 1 piping fatigue analyses were performed to the ASME Code Section III, Subsections NB-3200 and 3600, 1974 edition with addenda through winter 1975. The applicant stated that all Class 1 piping, Class 1 nozzles, and Class 1 thermowells were analyzed using the 40-year design transients, and the most limiting calculated design basis CUF occur in the 6-inch pressurizer safety lines and approach the limit of 1.0.

The applicant dispositioned the TLAAs for ASME Code Section III Class 1 piping and piping nozzles in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of fatigue on the ASME Code Section III Class 1 piping, piping nozzles, and thermowells will be adequately managed by the Metal Fatigue of Reactor Coolant Pressure Boundary Program for the period of extended operation.

4.3.2.7.2 Staff Evaluation

The staff reviewed LRA Section 4.3.2.7 and the TLAAs for the ASME Code Section III Class 1 piping and piping nozzles to confirm, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue will be adequately managed by the Metal Fatigue of Reactor Coolant Pressure Boundary Program for the period of extended operation.

The staff reviewed the applicant's TLAA for ASME Code Section III Class 1 piping and piping nozzles and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), consistent with the review procedures in SRP-LR Section 4.3.3.1.1.3. These procedures state that the reviewer should confirm the appropriateness of the applicant's program for monitoring and tracking the number of critical thermal and pressure transients for the selected RCS components.

LRA Sections 4.3.2.7 and A3.2.1.7 state that fatigue usage factors in ASME Code Section III Class 1 piping and piping nozzles do not depend on effects that are time-dependent at steady-state conditions but depend only on effects of normal, upset, and emergency transient events. Furthermore, the Metal Fatigue of Reactor Coolant Pressure Boundary Program ensures that the number of transients actually experienced during the period of extended operation remains below the assumed number. However, LRA Section 4.3.1.1 states that the ASME Code does not require inclusion of emergency or faulted conditions in fatigue evaluations; therefore, the Metal Fatigue of Reactor Coolant Pressure Boundary Program does not monitor emergency and faulted conditions. The staff reviewed UFSAR Section 3.9.1.1.8 and noted that the small loss-of-coolant accident (LOCA), small steam line break, and complete loss of flow transients are considered emergency conditions, but they are not listed in LRA Table 4.3-2. By letter September 22, 2011, the staff issued RAI 4.3-9 requesting that the applicant clarify whether emergency conditions are included in the fatigue analyses of ASME Code Section III Class 1 piping and piping nozzles. If so, the staff requested that the applicant justify why the Metal Fatigue of Reactor Coolant Pressure Boundary Program does not monitor emergency transients. If not, the staff requested that the applicant clarify why the dispositions for the fatigue analyses of ASME Code Section III Class 1 piping and piping nozzles in LRA Sections 4.3.2.7 and A3.2.1.7 discuss emergency transients. RAI 4.3-9 also requested the same information for RVIs, as documented in SER Section 4.3.3.2.

In its response dated November 21, 2011, the applicant stated that ASME Code Section III, paragraph NB-3222.4, requires the inclusion of those transients expected during normal service conditions. Therefore, the emergency conditions noted in the staff's question (small LOCA, small steam line break, and complete loss of flow) are not required to be included in the ASME Code Section III Class 1 fatigue analyses. In addition, the applicant stated that emergency transients would constitute a significant event and would require initiation of a corrective action document and thorough analysis of the event; therefore, emergency transients do not need to be monitored.

The staff finds the applicant's response acceptable because, consistent with ASME Code Section III, emergency and faulted conditions are not required to be considered in fatigue analyses for ASME Code Section III Class 1 piping and would not be a contributor to the calculated CUF value. Therefore, consistent with the recommendations of GALL Report AMP X.M1, the applicant is monitoring those plant design transients that cause cyclic strains, which are significant contributors to the fatigue usage factor. The staff's concern described in RAI 4.3-9 related to ASME Code Section III Class 1 piping is resolved.

The staff noted that the design basis fatigue evaluation remains valid as long as the number of transients that occur at the site remain bounded by the 40-year numbers of cycles assumed by the analysis. The staff noted that the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program ensures that the number of transients actually experienced during the period of extended operation remains below the assumed number or corrective actions will be taken. The staff's evaluation of the Metal Fatigue of Reactor Coolant Pressure Boundary Program is documented in SER Section 3.0.3.2.28.

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging related to fatigue analyses for ASME Code Section III Class 1 piping and piping nozzles will be adequately managed for the period of extended operation. Additionally, the applicant's disposition meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.3 because the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program monitors and tracks the number of design basis transients that will occur through the period of extended operation and includes action limits and corrective actions that will ensure that the ASME Code design limit of 1.0 will not be exceeded during the period of extended operation. Additionally, the use of the applicant's program is consistent with the recommendations of GALL Report AMP X.M1.

4.3.2.7.3 UFSAR Supplement

LRA Section A3.2.1.7 provides the UFSAR supplement summarizing the metal fatigue TLAA for the ASME Code Section III Class 1 piping and piping nozzles. The staff reviewed LRA Section A3.2.1.7 consistent with the review procedures in SRP-LR Section 4.3.3.2, which state that the reviewer should confirm that the applicant provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA.

Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.3.2.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the metal fatigue TLAA for ASME Code Section III Class 1 piping and piping nozzles, as required by 10 CFR 54.21(d).

4.3.2.7.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the cumulative fatigue analyses for the ASME Code Section III Class 1 piping and piping nozzles will be adequately managed for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.2.8 Response to NRC Bulletin 88-08: Intermittent Thermal Cycles Due to Thermal-Cycle-Driven Interface Valve Leaks and Similar Cyclic Phenomena

4.3.2.8.1 Summary of Technical Information in the Application

LRA Section 4.3.2.8 describes the applicant’s TLAA associated with the response to NRC Bulletin 88-08. The applicant stated that NRC Bulletin 88-08 describes the mechanism of thermal cycles in normally isolated, dead-end branches, due to leaking interface valves. Because valves often leak, an unrecognized phenomenon and possibly unanalyzed cyclic thermal stresses on valves, piping, and nozzles may exist for those reactors with these conditions. Under these conditions, thermal fatigue of the unisolable piping can result in crack initiation.

The applicant stated, for the RHR Lines, that Westinghouse compared the STP and Genkai RHR lines and determined that it is very unlikely for thermal cycling phenomenon, as described in NRC Bulletin 88-08, supplement 3, to occur. Therefore, the safety determination does not consider the effects of aging, and the evaluation of the RHR line is not a TLAA in accordance with 10 CFR 54.3(a), Criterion 2. The applicant dispositioned the TLAAs for charging, alternate charging, and auxiliary spray lines due to thermal stratification in accordance with 10 CFR 54.21(c)(1)(ii) to demonstrate that the analyses have been projected to the end of the period of extended operation.

4.3.2.8.2 Staff Evaluation

The staff reviewed LRA Section 4.3.2.8 and the metal fatigue TLAA associated with the applicant’s response to NRC Bulletin 88-08 to confirm, pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses have been projected to the end of the period of extended operation. In addition, the staff reviewed the applicant’s determination that the evaluation for the RHR lines is not a TLAA in accordance with 10 CFR 54.3(a), Criterion 2.

The staff also reviewed the applicant's TLAAs for the charging, alternate charging, and auxiliary spray lines due to thermal stratification and the corresponding disposition of 10 CFR 54.21(c)(1)(ii), consistent with the review procedures in SRP-LR Section 4.3.3.1.1.2. These procedures state that the revised CUF calculations are reviewed to ensure that the CUF remains less than or equal to 1.0 at the end of the period of extended operation.

LRA Section 4.3.2.8 states that the NRC's SE of the STP lines concluded that the normal charging, alternate charging, and the auxiliary spray lines at STP are not susceptible to thermal cycling. The LRA further states the analyses that support inspection interval determinations for these lines are independent of the life of the plant, and thus they are not TLAAs in accordance with 10 CFR 54.3(a), Criterion 3, in that the fatigue analyses do not involve a time-limited assumption.

The staff reviewed the SE related to the resolution of Bulletin 88-08, dated May 6, 1998 (ADAMS Accession No. 9805110004). Based on its review, the staff noted that the applicant estimated that the CUF limit of 1.0, when considering design transients and inadvertent thermal stratification cycling, would be achieved in a time span of 11.4 years based on a fatigue evaluation performed by Westinghouse of the weld between the check valve and the unisolable piping. In this SE, the staff noted that the time span was calculated using the assumption that thermal cycling occurred at the check valve weld and that the ASME Code CUF limit would not be achieved at the weld during the life of the 40-year plant without the assumption of thermal cycling. It is not clear to the staff why the fatigue analyses performed by the applicant, which included time-limited assumptions, would not be defined as a TLAA, in accordance with 10 CFR 54.3(a).

By letter dated September 22, 2011, the staff issued RAI 4.3-21 requesting that the applicant justify why the fatigue analyses related to thermal cycling, as discussed in the staff's SE dated May 6, 1998, were not identified as TLAAs, as defined in 10 CFR 54.3(a). Otherwise, the staff requested that the applicant provide and justify the TLAA disposition for the fatigue analyses of the weld between the check valve and the unisolable piping related to thermal cycling for the normal charging, alternate charging, and the auxiliary spray lines.

In its response dated November 21, 2011, the applicant stated that the analysis noted in the staff SE related to the resolution of Bulletin 88-08, dated May 6, 1998, was generated to form the interim basis for continuing normal operation at STP assuming that thermal cycling at the check valve weld was occurring. The staff noted that the following was concluded in the SE dated May 6, 1998: "The [applicant] has reasonably demonstrated that the normal and alternate charging lines and the auxiliary spray line at STP, Units 1 and 2, are not susceptible to the thermal cycling phenomena described in Bulletin 88-08 for the life of the plant, and is therefore not required to monitor these lines for leakage."

Based on its review and the staff's conclusions in the SE dated May 6, 1998, the staff finds the applicant's response acceptable and finds that the fatigue analysis described above is not part of the applicant's CLB and is not a TLAA in accordance with 10 CFR 54.3(a), Criterion 6. The staff's concern described in RAI 4.3-21 is resolved.

The staff noted that, for the RHR lines, the applicant stated Westinghouse compared the STP and Genkai RHR lines and determined that it is very unlikely for the thermal cycling phenomenon described in NRC Bulletin 88-08, supplement 3, to occur at STP. The staff reviewed SE Section 2.2.2 related to the resolution of Bulletin 88-08, dated May 6, 1998, which states that the RHR lines were shown by Westinghouse not to be susceptible to the phenomenon in supplement 3 of NRC Bulletin 88-08 because of the sufficient distance of the isolation valves from the turbulent penetration source. The staff, in its SE, found this to be reasonable and acceptable and considered the issue resolved for the STP RHR lines. Since it was determined that these RHR lines are not susceptible to the phenomenon in supplement 3 of NRC Bulletin 88-08, the staff finds acceptable the applicant's determination that this evaluation is not a TLAA in accordance with 10 CFR 54.3(a), Criterion 2.

LRA Section 4.3.2.8 states that the applicant evaluated the observed stratification of the charging, alternate charging, and auxiliary spray lines and determined that the incremental fatigue usage increase was less than 0.001 for the charging and alternate charging lines and less than 0.03 for the auxiliary spray lines. In addition, these evaluations demonstrated that the ASME Code limit would not be reached during the life of the plant since they are based on 40-year design transient cycles.

After reviewing the CUF values for the lines in the LRA and UFSAR Section 3, the staff noted that, when projected out to 60 years considering the increased incremental fatigue usage, the 60-year CUF values are still less than the ASME Code design limit of 1.0. During the AMP audit, the staff confirmed that the low CUF of these lines, when considering thermal stratification, would not exceed the ASME Code design limit of 1.0 when projected to 60 years. The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses for thermal stratification of the charging, alternate charging, and auxiliary spray lines have been projected to the end of the period of extended operation. Additionally, the applicant's disposition meets the acceptance criteria of SRP-LR Section 4.3.2.1.1.2 because the applicant demonstrated that the 60-year projected CUF values will be less than the ASME Code Section III, design limit of 1.0 through the period of extended operation.

4.3.2.8.3 UFSAR Supplement

LRA Section A3.2.1.8 provides the UFSAR supplement summarizing the metal fatigue TLAA in response to Bulletin 88-08. The staff reviewed LRA Section A3.2.1.8, consistent with the review procedures in SRP-LR Section 4.3.3.2, which state that the reviewer should confirm that the applicant provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA.

Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.3.2.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the TLAA for the pressurizer surge line, including thermal stratification, as required by 10 CFR 54.21(d).

4.3.2.8.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses associated with the applicant's response to NRC Bulletin 88-08 to address thermal cycles of the charging, alternate charging, and auxiliary spray lines have been projected to the end of the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.2.9 Response to NRC Bulletin 88-11: Revised Fatigue Analysis of the Pressurizer Surge Line for Thermal Cycling and Stratification

4.3.2.9.1 Summary of Technical Information in the Application

LRA Section 4.3.2.9 describes the applicant's metal fatigue TLAA for the pressurizer surge line to account for thermal cycling and stratification in response to NRC Bulletin 88-11. The applicant stated that NRC Bulletin 88-11 requested that applicants establish and implement a program to confirm pressurizer surge line integrity in view of the occurrence of thermal stratification and require addressees to inform the staff of the actions taken to resolve this issue. The applicant stated that the surge line was originally designed to ASME Code Section III, 1974 edition with addenda through winter 1975 and was re-evaluated to the 1986 Code in response to the NRC Bulletin 88-11 thermal stratification concerns.

The applicant dispositioned the metal fatigue TLAA for the pressurizer surge line to account for thermal cycling and stratification, in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of fatigue on the pressurizer surge line, including thermal cycling and stratification, will be adequately managed by the Metal Fatigue of Reactor Coolant Pressure Boundary Program for the period of extended operation.

4.3.2.9.2 Staff Evaluation

The staff reviewed LRA Section 4.3.2.3 and the metal fatigue TLAA for the pressurizer surge line, including thermal cycling and stratification, to confirm, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue will be adequately managed by the Metal Fatigue of Reactor Coolant Pressure Boundary Program for the period of extended operation.

The staff reviewed the applicant's TLAA for the pressurizer surge line, including thermal cycling and stratification, and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), consistent with the review procedures in SRP-LR Section 4.3.3.1.1.3. These procedures state that the reviewer should confirm the appropriateness of the applicant's program for monitoring and tracking the number of critical thermal and pressure transients for the selected RCS components.

The applicant stated that, in response to NRC Bulletin 88-11, Westinghouse performed a generic analysis of all domestic Westinghouse PWRs and a plant-specific evaluation of the STP pressurizer surge lines. In addition, the Surge Line Stratification Program for Units 1 and 2 performed ASME Code Section III stress, fatigue CUF, fatigue crack growth, and LBB analyses. The staff noted that the applicant's fatigue crack growth and LBB analyses for the pressurizer surge line are evaluated in SER Section 4.3.2.11.2. The applicant also stated that the new fatigue usage factors were calculated with thermal transients redefined to account for thermal stratification, and the design basis number of cyclic events was unchanged. However, a simplified elastic-plastic analysis was performed in accordance with ASME Code Section III, paragraph NB-3653.6, which resulted in a lower CUF than previous evaluations. The staff noted that a simplified elastic-plastic analysis performed per NB-3653.6 is an alternative analysis, permitted by ASME Code Section III, which may still allow the component to be qualified under NB-3650, "Analysis of Piping Products."

The applicant stated that the revised fatigue analyses, which incorporate thermal stratification, do not depend on effects that are time-dependent at steady-state conditions but depend only on effects of operational, abnormal, and upset transient conditions. The staff noted that, as long as the number of transients that occur per unit remains bounded by the 40-year numbers of cycles assumed by the analysis, the design basis fatigue evaluation remains valid. The staff noted that the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program ensures that the number of transients actually experienced during the period of extended operation remains below the assumed number or that corrective actions will be taken. The staff's evaluation of the Metal Fatigue of Reactor Coolant Pressure Boundary Program is documented in SER Section 3.0.3.2.28.

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging related to fatigue analysis of the pressurizer surge line, including thermal stratification, will be adequately managed for the period of extended operation, because the applicant is managing the number of transient cycles consistent with the 40-year design numbers. Additionally, it meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.3 because the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program monitors and tracks the number of design basis transients that will occur through the period of extended operation and includes action limits and corrective actions that will ensure that the ASME Code design limit of 1.0 will not be exceeded during the period of extended operation. Additionally, the use of the applicant's program is consistent with the recommendations of GALL Report AMP X.M1.

4.3.2.9.3 UFSAR Supplement

LRA Section A3.2.1.9 provides the UFSAR supplement summarizing the metal fatigue TLAA for the pressurizer surge line, including thermal stratification. The staff reviewed LRA Section A3.2.1.9, consistent with the review procedures in SRP-LR Section 4.3.3.2, which state that the reviewer should confirm that the applicant provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA.

Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.3.2.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the TLAA for the pressurizer surge line, including thermal stratification, as required by 10 CFR 54.21(d).

4.3.2.9.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the pressurizer surge line, including thermal stratification, will be adequately managed by the Metal Fatigue of Reactor Coolant Pressure Boundary Program for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.2.10 High-Energy Line Break Postulation Based on Fatigue Cumulative Usage Factor

4.3.2.10.1 Summary of Technical Information in the Application

LRA Section 4.3.2.10 describes the applicant's TLAA for HELB postulation based on fatigue CUF. The applicant stated that the staff's Branch Technical Position (BTP) MEB 3-1 from the SRP-LR provides guidance for determining the types and locations of postulated HELBs outside containment and has historically been used for the same purpose inside containment. BTP MEB 3-1 guidance for ASME Code Section III Class 1 piping requires postulating breaks at intermediate locations where the design basis CUF equals or exceeds 0.1. In addition, UFSAR Section 3.6.1 states that selection of pipe failure locations and evaluation of the consequences on nearby essential SSCs are presented and are in accordance with the requirements in 10 CFR Part 50, Appendix A, GDC 4. Selections and evaluations comply with the guidance of NRC BTP MEB 3-1.

The applicant dispositioned the TLAA for the welded attachments to Class 2 and 3 piping, which support the elimination of arbitrary intermediate break locations, other than those for the charging system and the main feedwater system, in accordance with 10 CFR 54.21(c)(1)(ii) to demonstrate that the analyses have been projected to the end of the period of extended operation. The applicant also dispositioned the TLAA for the Class 1 break locations and welded attachments to charging and main feedwater lines in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of fatigue on the RVIs will be adequately managed by the Metal Fatigue of Reactor Coolant Pressure Boundary Program for the period of extended operation.

4.3.2.10.2 Staff Evaluation

The staff reviewed LRA Section 4.3.2.10 and the TLAAs for HELB postulation based on fatigue CUF to confirm, pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses have been projected to the end of the period of extended operation for those locations identified as such in the LRA. The staff also confirmed, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue will be adequately managed by the Metal Fatigue of Reactor Coolant Pressure Boundary Program for the period of extended operation for those locations identified as such in the LRA.

The staff reviewed the applicant's metal fatigue TLAA for the welded attachments to Class 2 and 3 piping, other than those for the charging system and the main feedwater system, and the corresponding disposition of 10 CFR 54.21(c)(1)(ii), consistent with the review procedures in SRP-LR Section 4.3.3.1.1.2. These procedures state that the revised CUF calculations are reviewed to ensure that the CUF remains less than or equal to 1.0 at the end of the period of extended operation.

The staff also reviewed the applicant's TLAA for the Class 1 break locations and welded attachments to charging and main feedwater lines and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), consistent with the review procedures in SRP-LR Section 4.3.3.1.1.3. These procedures state that the reviewer should confirm the appropriateness of the applicant's program for monitoring and tracking the number of critical thermal and pressure transients for the selected RCS components.

In the fatigue analyses performed to postulate pipe break location for Class 2 and 3 systems, the applicant identified five CUF values that were calculated for integral welded attachments of Class 2 and 3 piping supports. The applicant stated that two of the five welded attachments—in the main feedwater system and in the charging system—will possibly experience CUFs greater than 1.0 during the period of extended operation. The remaining three Class 2 and 3 weld attachments of piping supports are validated for license renewal because their 60-year CUF values show a large margin from 1.0.

The staff noted that LRA Section 4.3.2.10 did not provide the 40-year CUF and corresponding 60-year projected CUF values for the integral pipe supports, other than those for the charging system and the main feedwater system, to support the applicant's disposition in accordance with 10 CFR 54.21(c)(1)(ii). Therefore, the staff could not confirm the adequacy of the applicant's TLAA disposition. By letter dated September 22, 2011, the staff issued RAI 4.3-3, requesting that the applicant provide the 40-year CUF and corresponding 60-year projected CUF values in the fatigue analysis for those welded attachments to Class 2 and Class 3 piping. The staff also asked the applicant to justify that the disposition for this TLAA is in accordance with 10 CFR 54.21 (c)(1)(ii) in that the analyses have been projected to the end of the period of extended operation.

In its response dated November 21, 2011, the applicant provided the 40-year and 60-year CUF values for the welded attachment to Class 2 and 3 piping. Specifically, for CVCS letdown, the 40-year CUF and 60-year CUF were 0.3704 and 0.5556, respectively. For auxiliary feedwater, the 40-year CUF and 60-year CUF were 0.4385 and 0.65775, respectively. For main steam, the 40-year CUF and 60-year CUF were 0.0985 and 0.14775, respectively. The staff noted that that the 40-year CUF values were projected to 60 years by multiplying by 1.5, which demonstrated that the ASME Code design limit of 1.0 was not exceeded. The staff finds the use of this 1.5 factor reasonable for the 40-year design CUF values because the resulting estimated 60-year CUF values provide a gauge of how much margin is available before the design limit of 1.0 is reached. For these welded attachments, the staff noted that there is 34 percent margin or more between the 60-year projected CUF values and the ASME Code design limit of 1.0.

The staff finds the applicant's response acceptable because the applicant provided the CUF values for the welded attachments that were dispositioned in accordance with 10 CFR 54.21(c)(1)(ii) to demonstrate that, even when projected to 60 years, there is margin before the ASME Code design limit of 1.0 is exceeded for these welded attachments.

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses for the welded attachments to Class 2 and 3 piping, other than those for the charging system and the main feedwater system, have been projected to the end of the period of extended operation. Additionally, it meets the acceptance criteria of SRP-LR Section 4.3.2.1.1.2 because the applicant demonstrated that the 60-year projected CUF values will be less than the ASME Code Section III design limit of 1.0 through the period of extended operation with significant margin.

For the main feedwater piping support and the charging system piping support, the applicant was not able to demonstrate that the analyses would be valid for the period of extended operation; therefore, the applicant will manage the effects of aging with its Metal Fatigue of Reactor Coolant Pressure Boundary Program for the period of extended operation. The staff's evaluation of the use of this program to manage metal fatigue for these two supports is discussed below.

The staff noted that a CUF value less than 0.1 is one criterion for HELB location selection that is discussed in UFSAR Section 3.6.2.1.1. It also noted that, for the pressurizer surge line and accumulator safety injection lines, the applicant uses a criterion of 0.4 instead of 0.1 for the CUF value. In addition, it was noted that it may be possible that the design cycle limit applicable to HELB piping locations can be less than the "UFSAR Design Cycles" and "Program Limiting Value" identified in LRA Table 4.3-2. The "acceptance criteria" program element in the Metal Fatigue of Reactor Coolant Pressure Boundary Program did not address how the acceptance criteria will be different for HELBs and cumulative fatigue damage. The applicant's program indicates that, when the accumulated cycles approach the design cycles, corrective actions will be taken to ensure that the analyzed number of cycles is not exceeded; however, it is not clear to the staff if the applicant's program addresses the situation when the accumulated cycles approach the limit in the HELB analyses.

By letter dated September 22, 2011, the staff issued RAI 4.3-2, requesting that the applicant identify the ASME Code Class 1 piping locations discussed in UFSAR Section 3.6.2.1.1 that are within the scope of LRA Section 4.3.2.10. For each location identified, the staff asked the applicant to provide the applicable design basis transients and associated cycle limits. In addition, the staff requested that the applicant justify that the Metal Fatigue of Reactor Coolant Pressure Boundary Program can adequately ensure the CUF for HELB locations remains below 0.1 (or 0.4 for the pressurizer surge line and the accumulator safety injection line) by using systematic counting of plant transient cycles associated with the HELB analysis.

In its response dated November 21, 2011, the applicant stated that all ASME Code Class 1 piping locations are within the scope of LRA Section 4.3.2.10 except the reactor coolant loops, which were excluded based on the LBB analysis discussed in LRA Section 4.3.2.11. The applicant clarified that the fatigue analyses that support the determination of the HELB location are discussed in LRA Section 4.3.2.7, and the specific HELB locations are identified in UFSAR Table 3.6.2-1 and Figure 3.6.1-1.

The applicant stated that most of these transients are already considered in the Metal Fatigue of Reactor Coolant Pressure Boundary Program. However, some transient counts assumed in the analyses are less than the program limiting values presented in LRA Table 4.3-2, and the program limiting values will be revised to include these lower values to ensure that corrective actions will be taken for the respective components prior to reaching their lower values. The applicant provided the revision to LRA Table 4.3-2, and the staff confirmed that the revision is consistent with the limiting values used in the fatigue analyses that support the determination of HELB locations. Certain transients are included in these fatigue analyses but are not included in the Metal Fatigue of Reactor Coolant Pressure Boundary Program. The staff's assessment as to whether it is acceptable that each of these transients is not included in the Metal Fatigue of Reactor Coolant Pressure Boundary Program is documented below.

The "reduce temperature return to power" transient was included in pressurizer surge line and spray line fatigue analyses, and this transient is designed to improve capabilities of the plant during load-follow operations. In addition, the "charging flow 50% decrease and return" and "letdown flow 50% increase and return" transients were included in the normal and alternate charging line fatigue analyses. These transients are designed to compensate for RCS volume changes resulting from changes in reactor power, and the number of transients is based on load-follow operations. The applicant stated that it does not practice load-follow operations, and this is not applicable to its units' operation. The staff finds it acceptable that the applicant does not monitor those transients that occur during load-following operation because the applicant does not operate as load-following units (i.e., setting the power level of a unit in accordance with the electrical grid); therefore, it is not credible for the occurrences of these transients to approach the design limit. The staff noted that the number of cycles for transients used in Normal/Charging fatigue analyses is based on alternating between the normal and alternate charging paths and the number of cycles used for this transient, "Charging flow 50% step decrease and return," is 14,400.

The "injection flow temperature change" was included in RCP seal injection line fatigue analyses, which will occur when the charging pump suction is switched back and forth from the volume control tank to the RWST. The applicant stated that, as discussed in LRA Section 4.3.2.3, it does not normally operate in this manner, and an inadvertent switching of charging pump suction sources due to equipment failure has not occurred to date. In addition, LRA Section 4.3.2.3 states there have been no events of this transient in the history of its plant operation. The staff finds it reasonable that the applicant does not monitor this transient because the circumstances in which this transient occurs are not consistent with normal plant operation. Additionally, after approximately 24 years of operation, this transient, with a design limit of 180 cycles, has not occurred at the applicant's site.

The "loss of seal injection flow" transient was included in RCP seal injection line fatigue analyses and is assumed to occur 40 times over plant life. The applicant clarified that this transient occurs whenever charging is lost, that there are two types of loss of charging transients, and that each is monitored to a 20-event limit. The staff confirmed that the applicant monitors each of the loss of charging transients (charging trip with prompt return to service and charging trip with delayed return to service) against its respective design limit of 20 cycles. The staff finds it acceptable that the applicant does not specifically monitor the loss of seal injection flow transient because the applicant is managing the two loss of charging transients that result in a loss of seal injection flow, with a combined design limit of 40 cycles (consistent with the design limit for the loss of seal injection flow transient).

The "accumulator check valve testing" transient is assumed to occur every refueling. The staff noted that the "refueling" transient is monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program. Based on the table provided in the applicant's response and LRA Table 4.3-2, the staff confirmed that the design limit of 80 cycles is applicable for both the "refueling" transient and "accumulator check valve testing" transient. The staff finds it reasonable that the "accumulator check valve testing" transient is managed for the period of extended operation through monitoring the "refueling" transient that has the same design limit of 80 cycles.

The applicant stated that the "letdown flow 50% decrease and return" transient was included in normal and alternate charging line fatigue analyses and is not a normal operating event with the plant at power. The applicant clarified that this transient was included for conservatism and assumed to occur approximately once a week for 40 years. The number experienced will not approach the limit given the conservatism of this assumption; therefore, this transient is not counted in the Metal Fatigue of Reactor Coolant Pressure Boundary Program. The staff noted a design limit of 1,200 cycles was included in the normal and alternate charging line fatigue analyses. It is not clear to the staff what the expected number of cycles is over 60 years for the "letdown flow 50% decrease and return" transient. In addition, if this transient was used as an input into a fatigue TLAA, it is not clear to the staff why this transient does not need to be monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program to ensure the analysis remains valid.

By letter dated January 30, 2012, the staff issued RAI 4.3-2a (followup), requesting that the applicant clarify the baseline number of events up to year-end 2008 and the 60-year projected cycles for the "letdown flow 50% decrease and return" transient. In addition, based on the 40-year and 60-year cycles, the staff asked the applicant to justify how it supports the statement in its response that, "the number experienced will not approach the limit given the conservatism of this assumption; therefore, this transient is not counted in the Metal Fatigue of Reactor Coolant Pressure Boundary Program."

In its response to RAI 4.3-2a (followup) dated February 16, 2012, the applicant clarified that the transient description of "letdown flow 50% decrease and return" should read "letdown flow 70% decrease and return" in the response to RAI 4.3-2 dated November 21, 2011. The applicant explained that the "letdown flow 70% decrease and return transient" was analyzed for 2,000 cycles for 40 years (50 cycles per year times 40 years) and was not the number of projected events. The applicant stated that the "letdown flow 70% decrease and return to normal" transient is not expected to occur because STP operates with continuous letdown at nominal flow, and letdown flow reduction is not part of normal operating practices. Therefore, the 60-year projected events are estimated to be zero for both units.

The staff finds it reasonable that the "letdown flow 70% decrease and return" transient is not monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program because the applicant's normal operating practices, which involve continuous letdown at nominal flow and do not consist of letdown flow reduction, preclude the occurrence of this transient. Hence, there is margin between the analyzed number of 1,200 cycles and the expected number of cycles, zero, to account for unanticipated occurrences of this transient through the period of extended operation. Therefore, the staff finds the applicant's response to RAI 4.3-2a acceptable. The staff's concern described in followup RAI 4.3-2a is resolved.

Based on its review, the staff finds the applicant's response to RAI 4.3-2 acceptable because the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program monitors and tracks the number of transients that occur through the period of extended operation, except as justified above, and includes corrective actions to ensure that the design limit will not be exceeded during the period of extended operation. Additionally, the applicant revised the "program limiting value" for each monitored transient to correspond to the lowest number of cycles assumed in the fatigue analyses to ensure corrective actions are taken before exceeding these assumptions. The staff's concern described in followup RAI 4.3-2 is resolved.

LRA Section 4.3.2.10 states that the fatigue crack growth analyses for the pressurizer surge line and accumulator safety injection lines established that flaws would not reach the flaw depths allowed in paragraph IWB-3640 of the ASME Code during the plant life. The applicant also stated that the analyses that evaluated fatigue crack growth and CUF in the pressurizer surge line and the accumulator safety injection line depend on the standard number of cycles for a 40-year reactor lifetime. LRA Section 4.3.2.10 provides two TLAA dispositions: (1) "Projection, 10 CFR 54.21(c)(1)(ii)," and (2) "Aging Management, 10 CFR 54.21 (c)(1)(iii)." However, it is not clear to the staff how these analyses for fatigue crack growth were dispositioned.

By letter dated September 22, 2011, the staff issued RAI 4.3-20, requesting that the applicant provide the TLAA disposition for the analyses that evaluated fatigue crack growth of the pressurizer surge line and the accumulator safety injection lines. If the TLAA is dispositioned in accordance with 10 CFR 54.21 (c)(1)(i) or 10 CFR 54.21 (c)(1)(ii), the staff asked the applicant to provide sufficient information related to the fatigue crack growth analyses to justify the selected disposition. In addition, if the TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(iii) and the Metal Fatigue of Reactor Coolant Pressure Boundary Program will be used, the staff asked the applicant to justify the use of cycle counting for these fatigue crack growth analyses without an update to the cycle-counting procedure and the inclusion of enhancements to the applicable program elements.

In its response dated November 21, 2011, the applicant clarified that the fatigue crack growth analyses for the pressurizer surge line and the accumulator safety injection lines are dispositioned in accordance with 10 CFR 54.21 (c)(1)(iii). The applicant also revised LRA Sections 4.3.2.10 and A3.2.1.10 to clarify that these fatigue crack growth analyses for the pressurizer surge line and the accumulator safety injection lines are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii). The staff noted that, in response to RAIs 4.3.2.11-1 and B3.1-3, the Metal Fatigue of Reactor Coolant Pressure Boundary Program was revised to include additional enhancements to manage fatigue flaw growth analyses. The staff's evaluation of the

Metal Fatigue of Reactor Coolant Pressure Boundary Program is documented in SER Section 3.0.3.2.28.

The staff finds the applicant's response acceptable because the applicant dispositioned the fatigue crack growth analyses for the pressurizer surge line and the accumulator safety injection lines, as required by 10 CFR 54.21(c)(1). Additionally, the applicant is ensuring these analyses remain valid for the period of extended operation on an ongoing basis by confirming the assumptions (number of transient cycles) are not exceeded with its Metal Fatigue of Reactor Coolant Pressure Boundary Program. The staff's concern described in RAI 4.3-20 is resolved.

The staff noted that analyses associated with the welded attachments to charging and main feedwater lines, HELB postulation based on CUF, and the fatigue crack growth for the pressurizer surge line and the accumulator safety injection lines have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii). The analyses credit the Metal Fatigue of Reactor Coolant Pressure Boundary Program to manage aging through the period of extended operation. The staff noted that as long as the number of transients that occur at the site remain bounded by the 40-year numbers of cycles assumed in these analyses, the evaluation remains valid. The staff noted that the applicant's AMP ensures that the number of transients actually experienced during the period of extended operation remains below the assumed number or that corrective actions are taken. The staff's evaluation of the Metal Fatigue of Reactor Coolant Pressure Boundary Program is documented in SER Section 3.0.3.2.28.

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the intended functions of the welded attachments to charging and main feedwater lines, HELB postulated locations based on CUF, and pressurizer surge line and the accumulator safety injection lines analyzed for fatigue crack growth will be adequately managed for the period of extended operation. Additionally, it meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.3 because the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program monitors and tracks the number of design basis transients that will occur through the period of extended operation and includes action limits and corrective actions that will ensure that theses analyses remain valid during the period of extended operation.

4.3.2.10.3 UFSAR Supplement

LRA Section A3.2.1.10 provides the UFSAR supplement summarizing the TLAA for welded attachments to Class 2 and 3 lines, HELB postulation based on CUF, and fatigue crack growth of the pressurizer surge line and the accumulator safety injection lines. The staff reviewed LRA Section A3.2.1.10, consistent with the review procedures in SRP-LR Section 4.3.3.2, which state that the reviewer should confirm that the applicant provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA.

Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.3.2.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the TLAA for welded attachments to Class 2 and 3 lines, HELB postulation based on CUF, and fatigue crack growth of the pressurizer surge line and the accumulator safety injection lines, as required by 10 CFR 54.21(d).

4.3.2.10.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that the fatigue analyses for welded attachments to Class 2 and 3 piping, which support the elimination of arbitrary intermediate break locations other than those for the charging system and the main feedwater system, have been projected to the end of the period of extended operation. In addition, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the welded attachments to charging and main feedwater lines, fatigue on HELB postulated locations based on CUF, and fatigue crack growth for the pressurizer surge line and the accumulator safety injection lines will be adequately managed for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.2.11 Fatigue Crack Growth Assessments and Fracture Mechanics Stability Analyses for Leak-Before-Break Elimination of Dynamic Effects of Primary Loop Piping Failures

4.3.2.11.1 Summary of Technical Information in the Application

LRA Section 4.3.2.11 describes the applicant's TLAA for LBB analysis. LRA Section 4.3.2.11 states that an LBB analysis eliminated the need to postulate longitudinal and circumferential breaks in the primary coolant loop piping. Elimination of these breaks omitted the need to install pipe whip restraints in the primary loop and eliminated the requirement to design for dynamic (jet and whip) effects of primary loop breaks. The LBB application will not affect the containment pressurization, emergency core cooling system, and EQ large-break design bases. The NRC approved the use of LBB in the RCS primary loop piping in NUREG-0781, Supplement 2, "Safety Evaluation Report Related to the Operation of South Texas Project, Units 1 and 2, Docket Nos 50-498 and 50-499." By letter dated October 22, 2014, the applicant amended LRA Section 4.3.2.11. The applicant deleted the heading "Primary Coolant System" in Section 4.3.2.11. This is an editorial change and it does not affect the contents of Section 4.3.2.11.

The LBB evaluation included a fatigue crack growth assessment for a range of materials at a high-stress location bounding the primary coolant system. The LBB evaluation concluded that the effects of low- and high-cycle fatigue on the integrity of primary piping are negligible. The Metal Fatigue of Reactor Coolant Pressure Boundary Program, evaluated in Section 4.3.1, ensures that the actual transient cycles remain below the assumed transient cycles in the analyses; otherwise, appropriate corrective actions will be taken. The effects of fatigue will, therefore, be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

The LBB analysis also includes a fracture mechanics evaluation, which depends on the crack initiation energy integral, J_{IN}. The primary coolant loops at STP are SA 351 Grade CF8A CASS, which at PWR operating temperatures is subject to time-dependent thermal embrittlement that would reduce the J_{IN}-integral value. Thermal embrittlement effects depend logarithmically on time (more rapid initially and approaching a saturation value over time.) The LBB analysis determined the effects of thermal aging on piping integrity for a material at thermal embrittlement saturation. Therefore, the applicant stated that the fracture mechanics evaluation for the CASS piping components in the LBB application is dependent on material properties not plant life; therefore, it is not a TLAA in accordance with 10 CFR 54.3(a), Criterion 3.

4.3.2.11.2 Staff Evaluation

The staff reviewed the fatigue crack growth calculation in LRA Section 4.3.2.11 to confirm, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the intended function of LBB piping will be adequately managed for the period of extended operation. Although the applicant stated that thermal embrittlement of the CASS piping is not a TLAA, the staff reviewed the issue to confirm, pursuant to 10 CFR 54.21(c)(1)(i), that the fracture mechanics analysis of the CASS LBB piping remains valid for the period of extended operation. The staff reviewed the applicant's TLAA and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.7.3, which state that the review of the TLAA provides assurance that the aging effect is properly addressed through the period of extended operation.

In RAI 4.3.2.11-1 (April 14, 2011), the staff asked the applicant to list the piping systems that have been approved for LBB and that are within the scope of license renewal. By letter dated May 12, 2011, the applicant responded that LBB analyses were performed for the reactor coolant piping, pressurizer surge line piping, safety injection accumulator piping, and the RHR suction piping. The applicant confirmed the specific piping systems that have been approved for LBB; therefore, the staff's concern in RAI 4.3.2.11-1 is resolved.

<u>Thermal Embrittlement of CASS Material in LBB Piping</u>. In RAI 4.3.2.11-2 (April 14, 2011), the staff asked the applicant to clarify whether the saturated (i.e., worst-case) fracture toughness value due to thermal embrittlement was used in the LBB analyses. By letter dated May 12, 2011, the applicant responded that the saturated fracture toughness value was used in all LBB analyses. The applicant stated further that although the fracture mechanics calculation considers aging of the material property, aging is not based on the plant life. Aging is based on the minimum material properties possible, and the value used by the calculation will be the same whether the plant life is 40 years, 60 years, or 100 years. Therefore, the applicant concluded that the fracture mechanics calculation is not a TLAA in accordance with 10 CFR 54.3(a), Criterion 3. Westinghouse Report WCAP-10456, "The Effects of Thermal Aging on the Structural Integrity of Cast Stainless Steel Piping for Westinghouse Nuclear Steam Supply Systems," November 1983, provides equations to predict end-of-life fracture toughness for thermal aging of CASS materials based on silicon, chromium, molybdenum, and ferrite contents. Testing found that the material properties reached saturated conditions after 30,000 hours during a 60,000-hour test. The selection of fracture toughness properties is discussed in enclosure C, item 2, of the applicant's letter dated March 12, 1986, "Alternative Pipe Break Criteria for Pressurizer Surge Line."

The staff noted that the applicant's response to Part 3 of RAI 4.3.2.11-2 cites the 1983 Westinghouse technical report WCAP-10456, "The Effects of Thermal Aging on the Structural Integrity of Cast Stainless Steel Piping for Westinghouse Nuclear Steam Supply Systems," as the basis for the saturated fracture toughness assumed in its analyses. The staff noted further that considerable information has been developed since 1983, by O. Chopra of Argonne National Laboratory, C. Faidy of Electricité de France, and others, to provide improved understanding of the thermal embrittlement of CASS materials. The following documents are examples of such reports that provide data since the 1980s:

- NUREG/CR-4513, Revision 1, “Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems” (1994)
- Appendix A of draft Electric Power Research Institute (EPRI) report 1024966, “Probabilistic Reliability Model for Thermally Aged Cast Austenitic Stainless Steel Piping”
- ASME Code paper PVP2010-25085, “Flaw Evaluation in Elbows Through French RSEM Code [a French Nuclear Code for PWR mechanical equipment],” by C. Faidy

Although the applicant’s RAI response states that the material property aging is based on the “minimum material properties possible,” the RAI response did not provide justification to support that statement in light of additional information on thermal aging of CASS over the last 29 years. In particular, it did not demonstrate that the aging after 60 years of operation is bounded by the thermal embrittlement saturation values assumed in the existing analysis. To address these issues, the staff issued RAI 4.3.2.11-6 by letter dated November 19, 2012, requesting that the applicant:

- provide justification that the assumed saturated fracture toughness in the CASS LBB evaluations bounds the expected toughness at 60 years of operation, considering the information sources cited above and others as necessary
- specify the information sources used in the response to Part 1
- identify, based on its response to Part 1, whether it will retain the current disposition of the LBB evaluation of CASS piping in LRA Section 4.3.2.11 or will instead treat it as: (a) a TLAA evaluated in accordance with 10 CFR 54.21(c)(1)(i) (i.e., the analysis “remains valid for the period of extended operation”); (b) a TLAA evaluated in accordance with 10 CFR 54.21(c)(1)(ii) (i.e., the analysis “has been projected to the end of the period of extended operation”); or (c) some other determination (please describe in full)

The issue of thermal aging embrittlement of CASS was identified in the SER with Open Items as Open Item (OI) 4.3.2.11-1.

By letter dated February 27, 2014, in response to RAI 4.3.2.11-6, the applicant stated that the reactor coolant loop LBB fracture mechanics analysis for STP is documented in Westinghouse report WCAP 10559, “Technical Bases for Eliminating the Large Primary Loop Pipe Rupture as the Structural Design Bases for South Texas Project Units 1 and 2.” The applicant stated that the reference material fracture toughness properties are shown to bound the fully aged fracture toughness properties of the STP RCPB cast stainless steel by comparing the STP fracture toughness properties and chemistry data from the certified material test reports.

The applicant compared the fracture toughness values used in the LBB analysis to the most recent data for the state of the industry and found it to be conservative. The applicant compared the gas tungsten arc welds (GTAW) prepared with Type 308 stainless steel filler materials against the results of NUREG/CR-6428, “Effects of Thermal Aging on Fracture Toughness and Charpy-Impact Strength of Stainless Steel Pipe Welds,” Revision 0, May 1996. Also, the applicant compared the SA-351 Grade CF8A base metal against the results of NUREG/CR-4513, “Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems,” Revision 1, May 1994. The straight pipe segments are centrifugal castings and the elbows are static castings. The applicant conservatively assumed the material to be static-cast CF-8 steel with ferrite content greater than 15 percent. The applicant noted that Jmax (maximum J-integral value) defines the range of applicability of the data used and is not affected by the updated data.

The applicant revised LRA Section 4.3.2.11 to identify the fracture mechanics evaluation as a TLAA and disposition it in accordance with 10 CFR 54.21(c)(1)(i). The applicant changed its position because the fracture mechanics evaluation does consider the thermal embrittlement aging mechanism and is defined by the current operating term. The applicant stated that the material fracture toughness properties selected for use in the LBB analysis are sufficiently embrittled that they bound the amount of thermal embrittlement that will occur in 60 years. Therefore, the applicant concluded that this TLAA is valid for the period of extended operation and is dispositioned in accordance with 10 CFR 54.21(c)(1)(i). The staff confirmed that the applicant used a bounding fracture toughness value in its LBB analysis and that the fracture toughness used is applicable to 60 years.

The staff concluded that the applicant's revisions to LRA Section 4.3.2.11 and Appendix A3.2.1.11 are acceptable because the revisions adequately resolve the staff's concern regarding the material fracture toughness properties used in the LBB analysis with respect to the period of extended operation.

The staff finds that the applicant satisfactorily responded to RAI 4.3.2.11-6 and, therefore, the issues associated with OI 4.3.2.11-1 are resolved.

Fatigue Flaw Growth Calculations of LBB Piping. LRA Section 4.3.2.11 states that the Metal Fatigue of Reactor Coolant Pressure Boundary Program ensures that the number of transients experienced during the period of extended operation remains below the assumed number in the fatigue CUF analysis. Appropriate corrective actions will maintain the design and licensing basis by other acceptable means. In RAI 4.3.2.11-3 (April 14, 2011), the staff asked the applicant to discuss whether the Metal Fatigue of Reactor Coolant Pressure Boundary Program specifically identifies all transients in the LBB analyses that will be monitored and to describe the appropriate corrective actions and other acceptable means that may be taken.

By letter dated May 12, 2011, the applicant responded that the transients used in the LBB analyses are consistent with those transients presented in LRA Table 4.3-2, with the exception of the following two transients not listed in LRA Table 4.3-2. The first transient, “accumulator actuation, accident operation,” is a combination of the “inadvertent RCS depressurization” transient and “LOCA” transient. The “LOCA” transient is a faulted event and, therefore, is not counted. The “inadvertent RCS depressurization” transient listed in Table 4.3-2 is monitored and counted. The second transient, “reduce temperature return to power” is identified in the pressurizer surge line fatigue crack growth analysis but not included in the STP design bases. This transient was designed to improve capabilities of the plant during load-follow operations. STP does not practice load-follow operations; therefore, this transient is not applicable to STP.

When “other acceptable corrective action” is needed, a 10 CFR 50.59 review is performed to determine if the methods and results are in line with the CLB or if regulatory review is needed. The term “appropriate corrective actions” is in reference to the corrective action described in LRA Section B3.1 and LRA Table A4-1, Commitment No. 30. As part of its response to RAI 4.3.2.11-3, the applicant revised the corrective actions (Element 7) in LRA Section B3.1, as documented in the applicant’s letter dated May 12, 2011. The staff finds that the revised LRA Section B3.1 provides additional detailed monitoring on the fatigue usage factor calculation and associated corrective actions. The staff finds that the enhanced Metal Fatigue of Reactor Coolant Pressure Boundary Program is acceptable to monitor the transient cycles used in the fatigue crack growth calculation for the LBB piping. Therefore, the fatigue aging effect for the LBB piping will be adequately managed for the period of extended operation, pursuant to 10 CFR 54.21(c)(1)(iii). The staff's evaluation of the Metal Fatigue of Reactor Coolant Pressure Boundary Program is documented in SER Section 3.0.3.2.28. Based on the above evaluation, the staff's concern described in RAI 4.3.2.11-3 is resolved.

In RAI 4.3.2.11-5 (April 14, 2011), the staff asked the applicant to discuss whether the fracture mechanics calculations and fatigue crack growth calculations for all LBB piping have been updated to include the new loads and DBEs discussed in LRA Section 4.3.2.4. By letter dated May 12, 2011, the applicant responded that LRA Section 4.3.2.11 addressed the effects of power uprate and SG replacement on the LBB analysis. The applicant reconciled the LBB analyses with the current plant-design basis, including new loads. The applicant determined that the conclusions of the previous LBB analyses for the reactor coolant piping, pressurizer surge line, and accumulator lines remain valid. The staff confirmed that the applicant evaluated the impact of power uprate and SG replacement on the original LBB evaluation and the staff found that the LBB evaluation remains valid. Based on the above evaluation, the staff's concern described in RAI 4.3.2.11-5 is resolved.

Primary Water Stress Corrosion Cracking. The staff notes that NUREG-0800, "Standard Review Plan" (SRP), Section 3.6.3, prohibits the LBB application to piping that experiences active degradation. PWR operating experience has shown that nickel-based Alloy 82/182 weld material is susceptible to PWSCC. In RAI 4.3.2.11-4 (April 14, 2011), the staff asked the applicant to identify the LBB pipes that are constructed using Alloy 82/182 weld metal, to identify the LBB pipes with and without mitigated Alloy 82/182 welds, and to discuss whether the LBB evaluation has been updated for the mitigated Alloy 82/182 welds.

By letter dated May 12, 2011, the applicant responded that STP, Units 1 and 2, reactor coolant piping (RV inlet and outlet nozzles) and the pressurizer surge line are the LBB lines that contain Alloy 82/182 filler weld metal. The applicant installed SWOLs on the Alloy 82/182 filler weld metal in the STP, Units 1 and 2, pressurizer surge lines in fall 2006 and spring 2007, respectively. Subsequently, the applicant inspected the overlaid surge line welds in fall 2009 for Unit 1 (1RE15) and spring 2010 for Unit 2 (2RE14) and found no flaws. These locations will continue to be inspected every 10 years using a qualified PDI ultrasonic technique.

The applicant stated that the Units 1 and 2 RV inlet and outlet nozzles contain unmitigated Alloy 82/182 filler weld metal, and they will be inspected with a qualified PDI ultrasonic technique in accordance with the ASME Code ISI Program and MRP-139/ASME Code Case N-770-1. The applicant performed ultrasonic testing (UT) on the reactor coolant piping inlet and outlet nozzles during 1RE15 and 2RE14 and found no flaws. The hot leg dissimilar metal welds are also visually inspected from the outside diameter every outage per ASME Code Case N-722-1.

The applicant stated that the LBB evaluations for the Units 1 and 2 pressurizer surge lines were updated to account for the effects of PWSCC in the leak rate calculations. The results of the LBB evaluation for the surge lines show that the LBB margin recommendations of SRP Section 3.6.3 are satisfied. The applicant stated further that the original LBB analysis conclusions remain valid.

The applicant's response to RAI 4.3.2.11-4 stated that the applicant was performing a 10 CFR 50.59 evaluation for the overlaid Alloy 82/182 weld in the pressurizer surge line in response to NRC Regulatory Issue Summary (RIS) 2010-007, "Regulatory Requirements for

Application of Weld Overlays and Other Mitigation Techniques in Piping Systems Approved for Leak-Before-Break." The applicant stated that the 10 CFR 50.59 review will conclude that the methodology used for the updated LBB analysis is the same method that the staff approved for use at Waterford Unit 3, as documented in its SE dated February 28, 2011 (ADAMS Accession No. ML110410119).

The staff finds that because the Alloy 82/182 dissimilar metal welds at the main primary loop nozzles have not yet been mitigated, the applicant is required to inspect these unmitigated Alloy 82/182 welds more frequently than if the welds were mitigated as discussed in ASME Code Case N-770-1 with conditions in 10 CFR 50.55a.

As stated above, the applicant's 10 CFR 50.59 evaluation of the updated LBB analysis for the surge line was ongoing at the time the staff issued RAI 4.3.2.11-4. The staff notes that the 10 CFR 50.59 evaluation of the updated LBB analysis was not directly related to the TLAA requirements in 10 CFR 54.21(c)(1) and, therefore, the 10 CFR 50.59 evaluation was not needed to completed the staff's TLAA review. The staff finds it is acceptable that the applicant installed weld overlays on and updated the LBB analysis for the pressurizer surge line, and it will perform necessary inspections to manage PWSCC in LBB piping with unmitigated Alloy 82/182 welds. Based on the above evaluation, the staff's concern described in RAI 4.3.2.11-4 is resolved.

4.3.2.11.3 UFSAR Supplement

LRA Section A3.2.1.11 provides the UFSAR supplement summarizing a description of the TLAA for the fatigue crack growth assessments and fracture mechanics stability analyses for the LBB piping. By letter dated October 22, 2014, the applicant amended LRA Section A3.2.1.11 (ADAMS Accession No. ML14308A073). In Enclosure 2 of the October 22, 2014, letter, the applicant deleted the pressurizer surge line and the accumulator line in Section A3.2.1.11 because the pressurizer surge line and accumulator line are discussed in Section A3.2.1.10 and do not belong in Section A3.2.1.11.

Section A3.2.1.10 addresses the TLAA for fatigue CUFs of high-energy lines that include the pressurizer surge lines and accumulator lines. Section A3.2.1.11 addresses the TLAA for the fatigue crack growth assessments and fracture mechanics stability analyses of primary coolant loop piping.

The staff reviewed amended LRA Section A3.2.1.11 consistent with SRP-LR Section 4.7.3.2, which state that the staff is to confirm that the UFSAR supplement includes a summary description of the evaluation of each TLAA. Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.7.2.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the TLAA for the fatigue crack growth assessments and fracture mechanics stability analyses for the LBB of primary coolant loop piping, as required by 10 CFR 54.21(d).

4.3.2.11.4 Conclusion

Based on its review, the staff concludes that, pursuant to 10 CFR 54.21(c)(1)(iii), the applicant demonstrated that the effects of fatigue crack growth on the intended function of LBB piping will be adequately managed for the period of extended operation. In addition, the staff concludes that, pursuant to 10 CFR 54.21(c)(1)(i), the applicant demonstrated that the fracture mechanics analysis of CASS primary coolant loop piping remains valid for the period of extended operation.

The staff concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation of the subject LBB piping, as required by 10 CFR 54.21(d)

4.3.2.12 Class 1 Design of Class 3 Feedwater Control Valves

4.3.2.12.1 Summary of Technical Information in the Application

LRA Section 4.3.2.12 describes the applicant's metal fatigue TLAA for the Class 3 feedwater control valves with a Class 1 design. The applicant stated that its feedwater control valves were purchased as ASME Code Section III, Class 3 valves, and UFSAR Table 3.9-8 associates a limiting number of occurrences of unit loading and unloading at 5 percent of full power for these valves. In addition, the methods and acceptance criteria for the evaluation of the valves for these occurrences were based on Class 1 methods of paragraph NB-3545 of ASME Code Section III, 1977 edition through the winter 1978 addenda.

The applicant dispositioned the metal fatigue TLAA for the Class 3 feedwater control valves with a Class 1 design in accordance with 10 CFR 54.21(c)(1)(i) to demonstrate that the analyses remain valid for the period of extended operation.

4.3.2.12.2 Staff Evaluation

The staff reviewed LRA Section 4.3.2.12 and the TLAA for the Class 3 feedwater control valves with a Class 1 design to confirm, pursuant to 10 CFR 54.21(c)(1)(i), that the analysis remains valid for the period of extended operation.

The staff reviewed the applicant's TLAAs for the thermal Class 3 feedwater control valves with a Class 1 design and the corresponding disposition of 10 CFR 54.21(c)(1)(i), consistent with the review procedures in SRP-LR Section 4.3.3.1.1.1. These procedures state that the operating transient experience and a list of the assumed transients used in the existing CUF calculations for the current operating term are reviewed to ensure that the number of assumed transients would not be exceeded during the period of extended operation.

LRA Section 4.3.2.12 states that the main feedwater control valves were analyzed for a new set of operating design transient conditions during the RSG project, and it was found that they could not be qualified for the full number of loading and unloading transients defined for the life of the plant. In order to obtain acceptable fatigue limits, the number of loadings and unloadings between 15 and 100 percent power was reduced, by the applicant, from 13,200 to 10,300 of loading or unloading.

The applicant stated that it has experienced 62 occurrences of this transient for Unit 1 and 43 occurrences for Unit 2 through July 27, 1989, which are less than 17 percent of the 385 anticipated at that point in the design life. Using the same occurrence rate, the 60-year projected occurrence will be 3,366 events. The applicant stated that this demonstrates a large margin between the analyzed value, 10,300, and the number of projected cycles of 3,366; thus, the analysis is valid for the period of extended operation.

The staff noted that the operating license for Unit 1 was issued on March 22, 1988, and on March 28, 1989, for Unit 2. In addition, LRA Table 4.3-2 provides the "Program Limiting Value" for the unit loading and unloading transients (Transients 5 and 6) of 3,000 for Unit 1 and 10,300 for Unit 2. The staff reviewed the information provided in LRA Section 4.3.2.12. However, it is not clear whether the use of the 16-month (from March 1988 to July 1989) data for Unit 1 and

4-month (March 28, 1989, to July 27, 1989) data for Unit 2 to extrapolate the number of occurrences of unit loading and unloading transients for 60 years is either reasonable or conservative. It is also not clear how the applicant determined that 385 cycles of the loading and unloading transients were anticipated to occur through July 27, 1989. The staff noted that the estimated occurrences of 3,366 cycles for these transients exceeds the "Program Limiting Value" of 3,000, which demonstrates that the applicant's disposition of this TLAA, in accordance with 10 CFR 54.21(c)(1)(i), is not valid.

By letter dated September 22, 2011, the staff issued RAI 4.3-5, requesting that the applicant justify how the 385 cycles of the unit loading and unloading transients that were anticipated to occur through July 27, 1989, was determined for Units 1 and 2. Furthermore, the staff requested that the applicant justify the disposition of the Unit 1 Class 3 feedwater control valves designed to Class 1 methods and provide the CUF contribution for the loading and unloading transients on the feedwater control valves.

In its response dated November 21, 2011, the applicant stated that the total transient count for Unit 1 and Unit 2 for the early period (62 and 43, respectively) contain multiple initial startup operational transients that are not expected to be repeated during the remainder of plant life. Based on recent operating history, this transient would typically be expected to occur only one to three times per 18-month cycle. The applicant clarified that the 385 cycles anticipated to occur during the early operating period were calculated by multiplying the original design basis value of 13,200 cycles (based on a load-following plant design) by the fraction of life that the plant had experienced. The staff noted the applicant's units are operated as base-load plants; therefore, this anticipated number of cycles, which was determined based on a load-following plant, is a conservative estimate. In addition, the staff noted that there is a significant margin between the projected number of cycles through the period of extended operation and the design limit of 10,300 cycles for the feedwater control valves, to account for unexpected occurrences. Based on these two factors, the staff finds that the design limit of 10,300 cycles for the feedwater control valves will not be approached through the period of extended operation.

The applicant clarified that the 3,000 cycle "Program Limiting Value," as noted in UFSAR Table 3.9-8, Footnote 2, pertains only to the Unit 1 BMI half-nozzle repair, and the cycle limiting value of 10,300 is still applicable for the Unit 1 Class 3 feedwater control valves. In addition, the total CUF is 0.999 of which loading and unloading events contribute 0.944 and the other transients contribute 0.055 to the 40-year CUF.

The staff noted a discrepancy between the applicant's response and LRA Section 4.3.2.12, which states that "[t]o obtain acceptable fatigue limits the number of loadings and unloadings between 15 and 100 percent power had to be reduced from 13,200 to 10,300, of loading or unloading for Unit 2. This limit does not apply to design of the Unit 1 feedwater control valves." By letter dated January 30, 2012, the staff issued followup RAI 4.3-5a, requesting that the applicant clarify the reference to LRA Section 4.3.1.12 that was cited in response to RAI 4.3-5. In addition, the staff asked the applicant to clarify the discrepancy between the response to RAI 4.3-5 and the information provided in LRA Section 4.3.2.12 for the limit of the number of loadings and unloadings between 15 and 100 percent power for Unit 1.

In its response to RAI 4.3-5a (followup) dated February 16, 2012, the applicant clarified that the reference to LRA Section 4.3.1.12 cited in response to RAI 4.3-5 dated November 21, 2011, should read 4.3.2.12. In addition, the applicant stated that the 10,300-cycle limit for loadings and unloadings between 15 and 100 percent power is applicable to the Unit 1 feedwater control valves. The applicant explained that the number of Unit 1 loading and unloading events between 15 and 100 percent power is limited to 3,000 because of the RV BMI half-nozzle repairs. The staff noted that the Unit 1 feedwater control valves are qualified for 10,300 events by the analysis and that the applicant revised LRA Section 4.3.2.12 to clarify that the 10,300 cycle limiting value applies to both the Unit 1 and Unit 2 feedwater control valves.

The staff noted that the design of the plant, which included the larger number of cycles for loading and unloading events between 15 and 100 percent power, was intended for a load-following plant design; however, the applicant operates its plant as a base-load plant. Therefore, the staff finds it reasonable that the design cycle limit of 10,300 for the feedwater control valves will not be exceeded based on the way in which the applicant operates its plant. In addition, since the contribution from the loadings and unloadings between 15 and 100 percent power transient is over 94 percent of the calculated CUF value, the staff finds that it is unlikely that the design CUF of 0.999 will be reached and the ASME Code design limit of 1.0 will be exceeded. The applicant does not operate as a load-following plant and, therefore, would not be expected to accumulate the design number of cycles for this transient.

Based on its review, the staff finds the applicant's responses to RAIs 4.3-5 and 4.3-5a acceptable for the following reasons:

- The estimated number of cycles for the loading and unloading events between 15 and 100 percent power considered the accumulated cycles that occurred at a higher rate during initial startup operation, and considered a reasonable scale to project to 60 years from the number of cycles anticipated from 40 years of operation.
- The estimated number of cycles that are expected to occur through the period of extended operation is less than 33 percent of the program limiting value of 10,300 cycles.
- The margin between the expected number of cycles and the design cycle limit of 10,300 for the feedwater control valves is sufficient to account for any unanticipated occurrences through the period of extended operation so that the cycle limit will not be exceeded.

The staff's concerns described in RAI 4.3-5 and RAI 4.3-5a are resolved.

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses for the Units 1 and 2 feedwater control valves remain valid for the period of extended operation. Additionally, the analyses meet the acceptance criteria in SRP-LR Section 4.7.2.1 because the applicant demonstrated that the analyzed number of cycles for 40 years will not be exceeded during the period of extended operation, and that there is sufficient margin to account for any unanticipated occurrence of loading and unloading events that could occur between 15 and 100 percent power transient.

4.3.2.12.3 UFSAR Supplement

LRA Section A3.2.1.12 provides the UFSAR supplement summarizing the metal fatigue TLAA for the Class 1 design of Class 3 feedwater control valves. The staff reviewed LRA Section A3.2.1.12, consistent with the review procedures in SRP-LR Section 4.3.3.2, which state that the reviewer should confirm that the applicant provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA.

Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.3.2.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the metal fatigue TLAA for the Class 1 design of Class 3 feedwater control valves, as required by 10 CFR 54.21(d).

4.3.2.12.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue analyses for the Class 1 design of Class 3 feedwater control valves remain valid for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.3 ASME Code Section III Subsection NG Fatigue Analysis of Reactor Pressure Vessel Internals

4.3.3.1 Summary of Technical Information in the Application

LRA Section 4.3.3 describes the applicant's metal fatigue TLAA for the RPV internals. The applicant stated that the RVIs support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and CRDMs, direct coolant flow past the fuel elements, direct coolant flow to the RPV head, provide gamma and neutron shielding, and guide the incore instrumentation. The applicant also stated that the design and construction of core support structures meet ASME Code Section III, Subsection NG, in full, and other internals are designed and constructed to ensure that their effects on the core support structures remain within the core support structure limits.

The applicant stated that the licensing basis does not describe any time-limited effects for a licensed operating period associated with flow-induced vibration for the RVIs; therefore, there are no TLAAs, in accordance with 10 CFR 54.3(a), Criteria 2 and 3.

In the LRA, the applicant stated it dispositioned the TLAAs for its RVIs in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of fatigue on the RVIs will be adequately managed by the Metal Fatigue of Reactor Coolant Pressure Boundary Program for the period of extended operation.

4.3.3.2 Staff Evaluation

The staff reviewed LRA Section 4.3.3, as amended by letter dated November 21, 2011, and the metal fatigue TLAAs for the RVIs to confirm, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

The staff reviewed the applicant's metal fatigue TLAA for RVIs and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), consistent with the review procedures in SRP-LR Section 4.3.3.1.1.3. These procedures state that the reviewer should confirm the appropriateness of the applicant's program for monitoring and tracking the number of critical thermal and pressure transients for the selected RCS components. The SRP-LR further states that the reviewer should ensure that the applicant's program contains the same program elements that the staff evaluated and relied upon in approving the corresponding generic program in the GALL Report.

The staff's review of the applicant's claim that its licensing basis does not describe any time-limited effects for a licensed operating period associated with flow-induced vibration for the RVIs, and that there are no TLAAs, in accordance with 10 CFR 54.3(a), Criteria 2 and 3, is documented in SER Section 4.1.2.1.2.6.

The staff noted that Westinghouse evaluated the Unit 1 and 2 RVIs for the effect of the 1.4 percent uprating and RSGs. The applicant provided its fatigue CUFs, which resulted in meeting the ASME Code allowable value, in LRA Table 4.3-7. The staff noted that LRA Table 4.3-7 provides the CUF values for the RVI components, which are all less than the ASME Code design limit of 1.0. For the "baffle-former assembly," the limiting 40-year CUF value for Units 1 and 2 is "$< 1_{(test)}$ [i.e., less than 1.0 as verified by testing]."

The metal fatigue TLAA for the RVI components, which include the "baffle-former assembly," was dispositioned in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of fatigue will be managed for the period of extended operation with the Metal Fatigue of Reactor Coolant Pressure Boundary Program. However, the applicant did not describe the details of the test that was performed to determine that the CUF for the "baffle-former assembly" was less than 1.0; therefore, it is not clear how the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program will manage fatigue of the "baffle-former assembly." By letter dated September 22, 2011, the staff issued RAI 4.3-17, requesting that the applicant describe how the CUF for the "baffle-former assembly" for Units 1 and 2 were shown to be less than 1.0 by testing and to identify the sections of the applicable design codes that were used for the fatigue testing. In addition, the staff requested that the applicant describe and justify how the Metal Fatigue of Reactor Coolant Pressure Boundary Program will manage cumulative fatigue damage of the "baffle-former assembly," since the CUF was shown to be less than 1.0 by testing.

In its response dated November 21, 2011, the applicant stated that a test was conducted in accordance with ASME Code Section III Appendix II, Article II-1221, in an arrangement that models the baffle-former-barrel assembly of the top two formers for a width of three baffle-former bolts. The applicant explained that the test was conducted by cyclically displacing the baffle relative to the barrel to the thermal displacement values, and an inspection was done to determine the baffle-former and barrel-former gaps after the test. The applicant stated that all bolts were deemed acceptable and survived cyclical deflection without exhibiting a significant loss of preload or any other characteristic of fatigue failure and that the fatigue test data envelop the number of cycles and the severity of the transients required by the design specification.

The applicant also stated that the fatigue tests were used in lieu of a fatigue analysis; therefore, no CUF existed for these components. The staff noted that ASME Code Section III, Subsection NG-3200, allows the use of fatigue testing in accordance with Appendix II, Article II-1200. Furthermore, Article II-1221 pointed to the provisions in Article II-1500 that require the cyclic testing performed would exceed the cycles and magnitude of the design transients. Thus, the staff found that maintaining those components within specified numbers of design transients and their severities as defined in the design specifications will ensure the tests remain valid. The staff noted that the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program ensures that the number and severity of transients actually experienced during the period of extended operation will remain below the assumed number in the design specification or that corrective actions will be taken. The staff's evaluation of the Metal Fatigue of Reactor Coolant Pressure Boundary Program is documented in SER Section 3.0.3.2.28.

The staff finds the applicant's response acceptable because the baffle-former assemblies were fatigue tested, in accordance with ASME Code Section III, Subsection NG, and Appendix II, which envelop the transients specified in the design specification. Additionally, the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program ensures that the number and severity of transients actually experienced will not exceed the assumptions made in order to qualify this component for fatigue. The staff's concern described in RAI 4.3-17 is resolved.

LRA Sections 4.3.3 and A3.2.2 state that fatigue usage factors for the RVIs do not depend on effects that are time-dependent at steady-state conditions but depend only on effects of normal, upset, and emergency transient events. Furthermore, the Metal Fatigue of Reactor Coolant Pressure Boundary Program ensures that the number of transients actually experienced during the period of extended operation will remain below the assumed number. However, LRA Section 4.3.1.1 states that the ASME Code Section III does not require inclusion of emergency or faulted conditions in fatigue evaluations; therefore, the Metal Fatigue of Reactor Coolant Pressure Boundary Program does not monitor emergency or faulted conditions. The staff reviewed UFSAR Section 3.9.1.1.8 and noted that the small LOCA, small steam line break, and complete loss of flow system transients are considered emergency conditions but are not in LRA Table 4.3-2. By letter dated September 22, 2011, the staff issued RAI 4.3-9, asking the applicant to clarify whether emergency conditions are included in the fatigue analyses of RVI components. If so, the staff requested that the applicant justify whether the Metal Fatigue of Reactor Coolant Pressure Boundary Program monitors emergency transients. RAI 4.3-9 also requested the same information for ASME Code Section III Class 1 piping and nozzles; the evaluation for these components is documented in SER Section 4.3.2.7.2.

In its response dated November 21, 2011, the applicant stated that an editorial error was made in LRA Section 4.3.3 and LRA Appendix A3.3.2. The applicant revised these two sections to remove the discussion of emergency transients for the RVIs. The staff noted that the exclusion of emergency and faulted conditions from the calculation of CUFs is consistent with ASME Code Section III, Subsection NG, for the design of core support structures. The staff finds the applicant's response acceptable because the applicant clarified that its design of the core support structures did not include emergency conditions, consistent with ASME Code Section III, Subsection NG. The staff's concern described in RAI 4.3-9 related to RVIs is resolved.

In the staff's SE dated April 12, 2002 (ADAMS Accession No. ML021130083), which approved a 1.4 percent increase in the reactor core thermal power level from 3,800 MWt to 3,853 MWt for Units 1 and 2, the staff concluded that the resulting stresses and fatigue factors from the 1.4 percent uprating upon the RVIs are within the allowable range (or limits) of the original analysis of record. The staff noted that the SE's approval of the 1.4 percent power uprate was effective after the replacement of the Model 94 SGs, which occurred in 2000 and 2002, for Units 1 and 2, respectively. The staff finds it appropriate that the applicant considered the 1.4 percent uprated conditions and RSGs and their effects on the stresses and fatigue factors for the RVI components.

The staff noted that as long as the number of transients that occurs for each unit remains bounded by the 40-year number of cycles assumed by the analysis, the design basis fatigue evaluation remains valid. The staff noted that the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program ensures that the number of transients actually experienced during the period of extended operation will remain below the assumed number or that corrective actions will be taken. The staff's evaluation of the Metal Fatigue of Reactor Coolant Pressure Boundary Program is documented in SER Section 3.0.3.2.28.

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the intended functions of the RVI components will be adequately managed for the period of extended operation. The staff also finds that the TLAA meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.3 because the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program monitors and tracks the number of design basis transients that will occur through the period of extended operation, and it includes action limits and corrective actions that will ensure that the ASME Code design limit of 1.0 will not be exceeded during the period of extended operation. Additionally, the use of the applicant's program is consistent with the recommendations of GALL Report AMP X.M1.

4.3.3.3 UFSAR Supplement

LRA Section A3.2.2, as amended by letter dated November 21, 2011, provides the UFSAR supplement summarizing the metal fatigue TLAA for the RVI components. The staff reviewed LRA Section A3.2.2, consistent with the review procedures in SRP-LR Section 4.3.3.2, which state that the reviewer should confirm that the applicant provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA.

Based on its review of the amended UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.3.2.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the metal fatigue TLAA for the RVI components, as required by 10 CFR 54.21(d).

4.3.3.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the RVI components will be adequately managed for the period of extended operation. The staff also concludes that the amended UFSAR supplement contains an appropriate summary description of the metal fatigue TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.4 Effects of Reactor Coolant System Environment on Fatigue Life of Piping and Components (Generic Safety Issue 190)

4.3.4.1 Summary of Technical Information in the Application

LRA Section 4.3.4 describes the applicant's evaluation of the effects of reactor coolant environment on component fatigue life for the period of extended operation. The applicant assessed the environmental effects on fatigue at the six sample locations identified by NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," for newer vintage Westinghouse plants.

Three of the NUREG/CR-6260 sample locations in Table 4.3-8 have a 60-year EAF CUF below 1.0, when multiplied by the maximum applicable environmental adjustment factor (F_{en}) for the material, from NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels," for carbon and low-alloy steels. The remaining NUREG/CR-6260 locations have been evaluated using ASME Code Section III, NB-3200, methods to reduce the EAF CUF values. The methods used to reduce the EAF CUF values include the following:

- recalculating the CUF with a more accurate fatigue analysis
- using projected values of the accumulated number of transient events, instead of using the 40-year number of events
- calculating an average F_{en} using strain-rate dependent F_{en} values for load set pairs significant to fatigue; and using the maximum F_{en} for load set pairs not significant to fatigue

The removal of conservatism resulted in the reduction of the accumulator safety injection nozzle and RHR inlet nozzle 60-year EAF CUFs to below 1.0.

The applicant stated that the EAF CUFs for the hot leg surge nozzle and charging nozzles are projected to exceed 1.0 within 60 years of operation. Corrective action for these locations will be required under the Metal Fatigue of Reactor Coolant Pressure Boundary Program in LRA Section B3.1 when the CBF results, including the effects of the reactor coolant environment, indicate that a fatigue based action limit has been reached.

The applicant dispositioned the EAF evaluations for all NUREG/CR-6260 locations in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of reactor coolant environment on fatigue usage will be adequately managed for the period of extended operation.

4.3.4.2 Staff Evaluation

The staff noted that the applicant addressed the effects of the reactor coolant environment on component fatigue life, consistent with the guidance in the SRP-LR and the staff's recommendations for resolving Generic Safety Issue No. 190 (GSI-190), dated December 26, 1999. The staff also noted that, consistent with Commission Memorandum and Order, CLI-10-17, dated July 8, 2010 (ADAMS Accession No. ML101890775), the evaluations associated with the effects of the reactor coolant environment on component fatigue life are not TLAAs in accordance with the definition in 10 CFR 54.3(a) because these evaluations are not in the applicant's CLB. Nevertheless, the applicant credited its Metal Fatigue of Reactor Coolant Pressure Boundary Program to manage the effects of reactor coolant environment on component fatigue life. Therefore, the staff reviewed LRA Section 4.3.4 and the evaluations for EAF to confirm, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of reactor coolant environment on component fatigue life will be adequately managed for the period of extended operation.

The staff reviewed the applicant's EAF evaluations, as presented in the LRA and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.3.3.1.3, which state that the reviewer should confirm that the applicant addressed the effects of the coolant environment on component fatigue life as AMPs are formulated in support of license renewal. This sample of critical components with high-fatigue usage locations should include the locations identified in NUREG/CR-6260, at a minimum, as well as any other alternatives based on plant specific considerations.

The staff reviewed the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program, including an enhancement (Enhancement No. 6 as documented in Commitment No. 34), to develop fatigue usage calculations that consider the effects of the reactor water environment for a set of sample RCS components in SER Section 3.0.3.2.28. This sample set of reactor RCS components will include the locations identified in NUREG/CR-6260 and additional plant-specific component locations in the RCPB if they are found to be more limiting than those considered in NUREG/CR-6260. As described in SER Section 3.0.3.2.28, the staff concludes that applicant's enhancement and Commitment No. 34 are consistent with the recommendations in SRP-LR Section 4.3.2.1.3 and GALL Report AMP X.M1 to consider environmental effects for additional plant-specific locations, if applicable.

The staff noted that LRA Table 4.3-8 contains 10 plant-specific locations, which are based on the six generic components identified in NUREG/CR-6260. LRA Table 4.3-8 also contains the 40-year CUF, the 40-year EAF CUF, and the 60-year EAF CUF for these 10 plant-specific locations. During the AMP audit, the staff noted in documentation onsite that the CUF and EAF CUF values for charging system nozzles (normal line and alternate line) and hot leg surge nozzle were different from those in LRA Table 4.3-8. By letter dated September 22, 2011, the staff issued RAI 4.3-6, requesting that the applicant revise LRA Table 4.3-8 to provide the correct CUF and EAF CUF values for the hot leg surge nozzle and charging system nozzles. The staff also requested that the applicant confirm that the remaining information in LRA Table 4.3-8 is accurate.

In its response dated November 21, 2011, the applicant stated that LRA Table 4.3-8 was revised to provide correct values for the hot leg surge nozzle and charging system nozzles that are consistent with the basis documents. The applicant confirmed that no other changes were identified after reviewing LRA Table 4.3-8. The staff noted that the 60-year design EAF values for these components are currently calculated to exceed 1.0; however, the applicant is managing the environmental effects on fatigue life with its Metal Fatigue of Reactor Coolant Pressure Boundary Program. Therefore, the applicant's program would manage the accumulated fatigue usage of these components to ensure that the actual fatigue usage for the component remains less than the ASME Code design limit of 1.0 during the period of extended operation; otherwise, corrective actions would be taken in accordance with its AMP.

The staff finds the applicant's response acceptable because the applicant revised values in LRA Table 4.3-8 to be consistent with its basis documents, and it is managing the effects of reactor coolant environment on fatigue life for all components in LRA Table 4.3-8 with its Metal Fatigue of Reactor Coolant Pressure Boundary Program during the period of extended operation. The staff's concern described in RAI 4.3-6 is resolved.

The staff noted that LRA Section 4.3.4 describes three methods that were used to reduce the EAF CUF values: (1) recalculating the CUF with a more accurate fatigue analysis, (2) using projected values of the accumulated number of transient events, and (3) calculating an average F_{en} using a strain-rate dependent method for load set pairs significant to fatigue and using the maximum F_{en} for load set pairs not significant to fatigue. Based on the information in the LRA, the staff was not able to determine what constituted a "more accurate fatigue analysis," how it was performed, and what conservatism was removed to obtain the reduced EAF CUF values. The staff also could not identify the locations in LRA Table 4.3-8 that used these three methods to reduce EAF CUF values.

By letter dated September 22, 2011, the staff issued RAI 4.3-7, requesting that the applicant identify the components and the associated methods described above that were used to reduce the EAF CUF values. Furthermore, the staff also requested that the applicant describe and justify the techniques used in performing the "more accurate fatigue analysis" and explain how any conservatism was removed to reduce the EAF CUF.

In its response dated November 21, 2011, the applicant stated that the hot leg surge nozzle, the normal and alternate charging nozzles, the RHR inlet nozzle, and the accumulator safety injection nozzle locations in LRA Table 4.3-8 were evaluated with "more accurate fatigue analyses." The applicant clarified that these evaluations were performed using the ASME Code Section III, NB-3200, methods versus the NB-3600 methods from the original Code calculations. The staff noted that typically the use of NB-3200 methods results in a lower CUF value when compared to the use of the NB-3600 methods that is simpler but more conservative. The staff also noted that these analyses were re-evaluated by using the guidance from Section 4.3 of NUREG/CR-6260, which provided an example of changes to fatigue requirements from the ASME Code edition of record for the design basis calculations to later Code editions. The staff also noted that 10 CFR 50.55a provides the requirements of ASME Code Section III and the endorsement of the Code editions that are acceptable to use.

The applicant also stated that EAF CUFs of the hot leg surge nozzle and the normal and alternate charging nozzles were calculated using the 60-year cycle projections. The staff finds the use of 60-year projections to re-evaluate the EAF CUF reasonable because it provides a more realistic CUF for 60 years of operation, including environmental effects, based on the actual plant operating practices at the applicant's site. The staff has no issue with the use of 60-year projections for EAF CUFs because the Metal Fatigue of Reactor Coolant Pressure Boundary Program manages accumulated fatigue usage of these components to ensure that the design limit of 1.0 is not exceeded during the period of extended operation. Furthermore, the program includes corrective actions if this design limit is approached. In addition, the applicant stated that NUREG-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," was used calculate the F_{en} factor for stainless steel for the hot leg surge nozzle, the normal and alternate charging nozzles, and the accumulator safety injection nozzles. The staff finds the use of the formulae in NUREG-5704 to calculate the F_{en} factor, which is based on the plant-specific information of the dissolved oxygen level, strain rate, and temperature for stainless steel components, acceptable because it is consistent with recommendations of GALL Report AMP X.M1 for methods to address the effects of reactor coolant environment on component fatigue life.

The staff finds the applicant's response acceptable because the refined analyses were performed with staff-accepted methodology in the ASME Code Section III, as endorsed by 10 CFR 50.55a; with the 60-year projections that are based on actual plant operating practices; and in accordance with staff-accepted guidance in NUREG/CR-6260 and NUREG/CR-5704. The staff's concern described in RAI 4.3-7 is resolved.

The staff noted that LRA Table 4.3-8 provides the 60-year EAF CUF of 11.3856 for the hot leg surge nozzle (safe end) and 2.3378 for the charging system nozzles (normal and alternate line). LRA Table 4.3-1 indicates that the stainless steel hot leg surge nozzle and charging system nozzles will be monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program with the CBF monitoring method. In the closure of GSI-190, the staff determined that the risk from fatigue failure of the primary coolant pressure boundary components is very small for a plant life of 40 years. It was not clear to the staff how the applicant will manage metal fatigue with its Metal Fatigue of Reactor Coolant Pressure Boundary Program during the period of extended operation because conservatism has already been removed to calculate the 60-year EAF CUF for these locations in which the values still exceed the ASME Code design limit of 1.0.

By letter dated September 22, 2011, the staff issued RAI 4.3-10, requesting that the applicant describe how the Metal Fatigue of Reactor Coolant Pressure Boundary Program will manage metal fatigue and consider environmental effects for these components for the period of extended operation, considering that conservatism has already been removed to obtain 60-year EAF CUF values.

In its response dated November 21, 2011, the applicant stated that the normal and alternate charging nozzles' EAF CUF is based on the transient severity and the number projected of transients for 60 years. The applicant stated that, by using the current cycle count, the CBF algorithm results in a current EAF CUF of 0.79. Therefore, the charging nozzles will continue to be managed using CBF, and additional corrective actions can be taken when the actual EAF CUF usage approaches 1.0. The applicant also stated that such corrective actions include additional analyses, repair, replacement or implementation of stress based fatigue monitoring consistent with Regulatory Issue Summary 2008-30, "Fatigue Analysis of Nuclear Power Plant Components."

In addition, the applicant stated that the design EAF CUF value is greater than 1.0 using the current cycle count for the hot leg surge nozzle. The staff noted that corrective actions, which include reanalysis, repair, or replacement, will be taken, consistent with the applicant's Metal Fatigue for Reactor Pressure Boundary Program and the UFSAR supplement in LRA Appendix A. The staff reviewed SECY-95-245, "Completion of the Fatigue Action Plan," September 25, 1995 (ADAMS Accession No. ML031480210), and noted that the basis in the Office of Nuclear Regulatory Research study did not support and justify the action of requiring a backfit of the environmental fatigue data to operating plants and concluded that the EAF issues in the Fatigue Action Plan should be evaluated for any proposed extended period of operation for license renewal. Based on the conclusions documented in SECY-95-245, the staff finds it appropriate that the applicant will take corrective actions in accordance with its Metal Fatigue for Reactor Pressure Boundary Program. The staff finds the applicant's response acceptable because the applicant is managing EAF during the period of extended operation, as recommended in SECY-95-245, and the applicant is using its Metal Fatigue for Reactor Pressure Boundary Program consistent with the recommendations of GALL Report AMP X.M1. This program will take corrective actions prior to entering the period of extended operation, consistent with SECY-95-245, to repair, replace, or reanalyze the EAF CUF such that the Code design limit of 1.0 will not be exceeded. The staff's concern described in RAI 4.3-10 is resolved.

LRA Section 4.3.4 states that the RPV wall transition, RPV inlet nozzle, and RPV outlet nozzle have 60-year EAF CUF values less than 1.0 when multiplied by the maximum applicable F_{en} value for low-alloy steels. For these low-alloy steel components, LRA Table 4.3-8 provides a F_{en} value of 2.455, which was determined based on NUREG/CR6583. The staff noted that based on the formulation in NUREG/CR-6583, the F_{en} value depends on sulfur content, temperature, dissolved oxygen, and strain rate at the applicant's site. It was not clear to the staff what assumptions were used by the applicant in determining the F_{en} values for the low-alloy steel components.

By letter dated September 22, 2011, the staff issued RAI 4.3-19 requesting that the applicant clarify how the F_{en} values for the low-alloy steel components were determined and justify any assumptions on the parameters, such as sulfur content, temperature, dissolved oxygen, and strain rate, which were used. Furthermore, the staff requested that the applicant confirm that the dissolved oxygen remained less than 0.05 parts per million (ppm) since initial plant operation. The staff also requested that the applicant justify that the dissolved oxygen content will remain less than 0.05 ppm during the period of extended operation, such that the F_{en} values would remain bounding for the conditions at the plant for the low-alloy steel components.

In its response dated November 21, 2011, the applicant stated strain-rate and sulfur content were assumed to be worst case for the F_{en} value for low-alloy steel components, which the staff finds to be a conservative assumption. The applicant also stated that the dissolved oxygen level was assumed to be less than 0.05 ppm, which corresponds to a low-oxygen environment.

The staff noted that based on equations in NUREG/CR-6583, the dissolved oxygen level only affects the F_{en} calculation when the RCS temperature is greater than 150 °C (302 °F). The applicant stated that the assumption for dissolved oxygen is consistent with its Primary Water Chemistry Program that maintains the dissolved oxygen at less than 0.005 ppm when the temperature is greater than 121 °C (250 °F). The applicant reviewed its primary water chemistry history and identified only one occurrence of short duration (approximately 2 hours) in which the RCS dissolved oxygen exceeded 0.05 ppm while the RCS temperature was greater than 121 °C (250 °F). The staff found that the 2-hour period of time when the dissolved oxygen levels exceeded 0.05 ppm while RCS temperature was greater than 121 °C (250 °F) does not have a significant impact on the overall F_{en} value because the time duration is negligible in comparison to the total amount of time the plant has operated. The applicant also stated that its Primary Chemistry Program maintains the dissolved oxygen at less than 0.005 ppm when the RCS temperature is greater than 121 °C (250 °F), and this program will be continued through the extended period of operation.

The staff finds the applicant's response acceptable for the following reasons:

- The applicant provided adequate justification for the assumptions made in determining F_{en} factors for low-alloy steel components, which the staff confirmed was bounding based on the operating parameters of these components.
- The applicant confirmed that it has historically maintained dissolved oxygen content to less than 0.05 ppm, except as justified above.
- The applicant will continue to maintain its primary water chemistry and dissolved oxygen content to less than 0.05 ppm during the period of extended operation.

The staff's concern described in RAI 4.3-19 is resolved.

Based on its review, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), the effects of reactor coolant environment on component fatigue life will be adequately managed for the period of extended operation. Additionally, it meets the acceptance criteria in SRP-LR Section 4.3.2.1.3 because the applicant demonstrated that the impact of the reactor coolant environment on critical components has been adequately addressed and will be managed by the Metal Fatigue for Reactor Pressure Boundary Program. Therefore, the applicant's EAF evaluations will remain valid, and the ASME Code limit of 1.0 will not be exceeded during the period of extended operation or corrective actions will be taken.

4.3.4.3 UFSAR Supplement

LRA Section A3.2.3 provides the UFSAR supplement summarizing the effects of the reactor coolant environment on fatigue life of piping and components. The staff reviewed LRA Section A3.2.3, consistent with the review procedures in SRP-LR Section 4.3.3.2, which state that the reviewer should confirm that the applicant provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of the effects of reactor coolant environment on fatigue life.

Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.3.2.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the TLAA for the effects of reactor coolant environment on component fatigue life, as required by 10 CFR 54.21(d).

4.3.4.4 Conclusion

Based on its review, the staff concludes that, consistent with Commission Memorandum and Order, CLI-10-17, the applicant's evaluations on the effects of the reactor coolant environment on component fatigue life is not a TLAA, as defined by 10 CFR 54.3(a). However, the staff also concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of reactor coolant environment on component fatigue life will be adequately managed for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the evaluation, as required by 10 CFR 54.21(d).

4.3.5 Assumed Thermal Cycle Count for Allowable Secondary Stress Range Reduction Factor in ANSI B31.1 and ASME Code Section III Class 2 and 3 Piping

4.3.5.1 Summary of Technical Information in the Application

LRA Section 4.3.5 describes the applicant's allowable secondary stress range reduction factor TLAAs for ANSI B31.1 and ASME Code Section III Class 2 and 3 piping. The applicant stated that its non-Class 1 piping was based on the design codes of the 1974 edition, including winter 1975 addenda, of the ASME Code Section III Class 2 and Class 3 and the 1973 edition, including winter 1975 addenda, of the ANSI B31.1. Both codes require a stress range reduction factor to the allowable stress range if the number of equivalent full temperature cycles exceeds 7,000. The applicant compared the 7,000-cycle limit against its 60-year projections for its thermal transients, listed in LRA Table 4.3-2, as applicable to these non-Class 1 components and determined that the 7,000-cycles limit will not be exceeded. The applicant dispositioned the piping analyses with allowable secondary stress range reduction factor in accordance with 10 CFR 54.21(c)(1)(i) to demonstrate that the analyses (stress range reduction factor) for ANSI B31.1 and ASME Code Section III Class 2 and 3 piping remain valid for the period of extended operation.

4.3.5.2 Staff Evaluation

The staff reviewed LRA Section 4.3.5 to confirm, pursuant to TLAA disposition criteria in 10 CFR 54.21(c)(1)(i), that the fatigue analyses for ANSI B31.1 and ASME Code Section III Class 2 and 3 piping remain valid for the period of extended operation.

The staff reviewed the applicant's TLAA and its corresponding disposition, consistent with the review procedures in SRP-LR Section 4.3.3.1.2.1. The SRP-LR states that the staff reviews the relevant information in the TLAA, operating plant transient history, design basis, and CLB (including TS cycle-counting requirements) to confirm that the maximum allowable stress range values for the existing fatigue analysis remain valid for the period of extended operation. It also confirms that the allowable limit for full thermal range transients will not be exceeded during the period of extended operation.

The staff reviewed the applicable design code requirements in UFSAR Tables 3.2.A-1 and 3.2.B-1 for components that are within the scope of license renewal and noted that the TLAAs for non-Class 1 components are based on the criteria in ANSI B31.1 and ASME Code Section III. These design codes required an allowable stress range reduction only if the number of full thermal cycles exceeds the limit of 7,000.

The staff reviewed the applicant's AMR results in the associated LRA Table 2s in LRA Sections 3.2, 3.3 and 3.4, and noted that the applicant did not include applicable AMR items for the TLAAs associated with fatigue of non-Class 1 piping. It is not clear to the staff why the components analyzed for cumulative fatigue damage, as discussed in LRA Section 4.3.5, are not included as AMR items in LRA Sections 3.2, 3.3, and 3.4.

By letter dated September 22, 2011, the staff issued RAI 4.3-18, requesting that the applicant revise the applicable LRA Table 2s in LRA Sections 3.2, 3.3, and 3.4 to include the AMR items that address cumulative fatigue damage for non-Class 1 piping. In its response dated November 21, 2011, the applicant stated that AMR items were inadvertently omitted from the LRA. Therefore, the following LRA tables will be revised to include the omitted AMR items:

- LRA Table 3.3.2-8, Primary Process Sampling
- LRA Table 3.3.2-19, Chemical and Volume Control
- LRA Table 3.3.2-20, Standby Diesel Generator
- LRA Table 3.3.2-21, Nonsafety-related Diesel Generator
- LRA Table 3.3.2-22, Liquid Waste Processing
- LRA Table 3.4.2-1, Main Steam
- LRA Table 3.4.2-2, Auxiliary Steam System and Boilers
- LRA Table 3.4.2-5, Steam Generator Blowdown
- LRA Table 3.4.2-6, Auxiliary Feedwater System

The staff confirmed that the applicant amended the aforementioned LRA tables to include additional AMR items with an aging effect of cumulative fatigue damage. The staff's review of these additional AMR items is documented in SER Sections 3.3.2.2.1 and 3.4.2.2.1.

The staff finds the applicant's response acceptable because the LRA was amended to include those SSCs subject to an AMR in accordance with 10 CFR 54.21(a)(1). The staff's concern described in RAI 4.3-18 is resolved.

The staff also reviewed the projected number of occurrences for plant transients for 60 years of operation, as given in LRA Table 4.3-2, to ensure the full thermal range transient cycle limit of 7,000 will not be exceeded. The staff's review of the applicant's 60-year projection methodology is documented in SER Section 4.3.1.2.

Based on its review, the staff confirmed that the full thermal range transient cycle limit of 7,000—used in the applicant's design basis fatigue evaluations associated with the ANSI B31.1 and ASME Code Section III Class 2 and 3 piping—will not be exceeded during the extended period of operation. Therefore, the maximum allowable stress range values for the existing analyses remain valid.

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the TLAAs of ANSI B31.1 and ASME Code Section III Class 2 and 3 piping fatigue analyses remain valid for the period of extended operation. Additionally, the applicant meets the acceptance criteria in SRP-LR Section 4.3.2.1.2.1 because the projected total number of full thermal range transients over the period of extended operation for ANSI B31.1 and ASME Code Section III Class 2 and 3 piping does not exceed the 7,000-cycle limit.

4.3.5.3 UFSAR Supplement

LRA Section A3.2.4 provides the UFSAR supplement summarizing the TLAA for ANSI B31.1 and ASME Code Section III Class 2 and 3 piping fatigue analyses. The staff reviewed LRA Section A3.2.4, consistent with the review procedures in SRP-LR Section 4.3.3.2, which state that the reviewer should confirm that the applicant provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of the TLAA.

Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.3.2.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the TLAA for ANSI B31.1 and ASME Code Section III Class 2 and 3 piping fatigue analyses, as required by 10 CFR 54.21(d).

4.3.5.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue analyses of ANSI B31.1 and ASME Code Section III Class 2 and 3 piping remain valid for the period of extended operation. The staff also concludes that the UFSAR supplement contains an adequate summary description of the evaluated TLAAs, as required by 10 CFR 54.21(d).

4.3.6 ASME Code Section III Fatigue Analysis of Metal Bellows and Expansion Joints

4.3.6.1 Summary of Technical Information in the Application

LRA Section 4.3.6 describes the applicant's TLAAs for the metal bellows and expansion joints, except for the fuel transfer bellows that are discussed in LRA Section 4.6.2. The applicant stated that a search of its CLB discovered design requirements of the diesel generator cooling water bellows. UFSAR Section 9.5.5, "Diesel Generator Cooling Water System," identifies the design of the diesel generator cooling water bellows as ASME Code Section III, Class 3. In addition, the metal expansion joints design specification requires that these expansion joints be designed in accordance with Section ND of the ASME Code Section III 1977 edition, including summer 1977 addenda, and have a minimum design life of 40 years. The applicant stated that the fatigue analyses for the metal expansion joints confirm the 40-year design requirement for the diesel generator cooling water expansion joints by satisfying ASME Code Section III, Subsection ND-3649.4(d), which limits the component's lifetime cyclical loading.

The applicant dispositioned the TLAA for all but seven of the diesel generator cooling water expansion joints in accordance with 10 CFR 54.21(c)(1)(i) to demonstrate that the analyses remain valid for the period of extended operation. The applicant also dispositioned the TLAA for the seven diesel generator cooling water expansion joints that are projected to exceed the analyzed number of cycles during the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii) and committed (Commitment No. 32) to replace these expansion joints prior to the period of extended operation. The applicant stated that the analyses for the replacement expansion joints will include the period of extended operation.

4.3.6.2 Staff Evaluation

The staff reviewed LRA Section 4.3.6 and the TLAAs for the diesel generator cooling water expansion joints to confirm, pursuant to 10 CFR 54.21(c)(1)(i), that the analysis remains valid for the period of extended operation and pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging will be adequately managed for the period of extended operation.

The staff reviewed the applicant's TLAA for all but seven of the diesel generator cooling water expansion joints and the corresponding disposition of 10 CFR 54.21(c)(1)(i), consistent with the review procedures in SRP-LR Section 4.7.3.1.1. These procedures state that justification provided by the applicant is reviewed to confirm that the existing analyses are valid and are bounding for the period of extended operation.

The staff also reviewed the applicant's TLAA for the seven diesel generator cooling water expansion joints that are projected to exceed the analyzed number of cycles during the period of extended operation and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), consistent with the review procedures in SRP-LR Section 4.7.3.1.3. These procedures state that the reviewer should confirm that the effects of aging on the intended function(s) are adequately managed consistent with the CLB for the period of extended operation.

The applicant stated that the analyzed numbers of cycles for all but seven of the diesel generator cooling water expansion joints are greater than the specified numbers of cycles extrapolated to 60 years; therefore, the analyses are valid for these bellows through the period of extended operation. However, the staff noted that the applicant did not provide the numbers of analyzed cycles and the specified numbers of cycles extrapolated to 60 years to justify the disposition in accordance with 10 CFR 54.21(c)(1)(i) for all but seven of the diesel generator cooling water expansion joints. During its review, the staff also noted that LRA Table 3.3.2-4 provides an AMR item for nickel-alloy expansion joints exposed to raw water and subject to cumulative fatigue damage in the ECW and essential cooling water wash system, which are managed by a TLAA. However, it was not clear which specific TLAA is being credited to manage cumulative fatigue damage for this particular AMR item. By letter dated September 22, 2011, the staff issued RAI 4.3-4, requesting that the applicant provide the analyzed cycles and the "specified number of cycles" extrapolated to 60 years for these diesel generator cooling water expansion joints and justify the associated disposition of this TLAA. In addition, the staff asked the applicant to clarify the fatigue TLAA that is being credited to manage cumulative fatigue damage for the nickel-alloy expansion joints identified by the AMR item in LRA Table 3.3.2-4.

In its response dated November 21, 2011, the applicant provided a table that lists the design analyzed and the design specified numbers of cycles for its metal bellows and expansion joints with an ASME Code Section III fatigue analysis. The staff noted that the applicant extrapolated the specified cycles to 60 years by multiplying it by 1.5, and if the number of design analyzed cycles is greater than the design specified number of cycles projected to 60 years, then the analysis is valid for the period of extended operation. The staff finds the use of this 1.5 factor reasonable for the specified cycles because it provides the ratio of 60 years to 40 years, and the resulting estimated 60-year cycles provide a gauge of how much margin is available before the analyzed cycles are reached. The staff noted that for the seven diesel generator expansion joints in which the design specified cycles exceeded the design analyzed cycles, the applicant will replace them prior to the period of extended operation, as discussed below. Other than these seven expansion joints, the design analyzed number of cycles is greater than the number of cycles specified for 40 years and expected for 60 years of operation.

In addition, the applicant clarified that the nickel-alloy expansion joints identified in LRA Table 3.3.2-4 are the ECW pump expansion joints, 3R281(2)NJX101(201)A/B/C. The applicant revised LRA Section 4.3.6 and Appendix A3.2.5 to include the ECW pump expansion joints identified by the AMR item in LRA Table 3.3.2-4.

Based on its review, the staff finds the applicant's response acceptable because the applicant demonstrated that its specified cycles for 60 years does not exceed the number of analyzed cycles for the analysis of the expansion joints and metal bellows that were dispositioned in accordance with 10 CFR 54.21(c)(1)(i). The staff's concern described in RAI 4.3-4 is resolved.

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses for all but seven of the diesel generator cooling water expansion joints, including the ECW pump expansion joints, remain valid for the period of extended operation. Additionally, it meets the acceptance criteria in SRP-LR Section 4.7.2.1 because the applicant demonstrated that the analyzed number of cycles for 40 years will not be exceeded during the period of extended operation.

The applicant committed (Commitment No. 32) to replace the seven diesel generator cooling water expansion joints that are projected to exceed the analyzed number of cycles during the period of extended operation. Commitment No. 32 also states that the analyses for the replacement expansion joints will include the period of extended operation. The staff noted that the current expansion joints are designed in accordance with Section ND of the ASME Code Section III 1977 Code, including summer 1977 addenda, and have a minimum design life of 40 years. The staff noted that the regulations at 10 CFR 50.55a specify the ASME Code requirements. Specifically, IWA-4000 of the ASME Code Section XI provides the requirements for repair and replacement activities for ASME Code Classes 1, 2, and 3 pressure-retaining components. The staff noted that in order for the applicant to comply with its CLB, the number of cycles for these seven expansion joints cannot exceed the design limit. Furthermore, any repair or replacement activities of these seven expansion joints will be performed in accordance with ASME Code Section XI, which is required by 10 CFR 50.55a. The staff also confirmed in LRA Table 3.3.2-20 that the expansion joints which are subject to a TLAA have a pressure boundary intended function. Since the replacement expansion joints will be installed prior to the period of extended operation and the fatigue analysis for these replacement components will have a minimum design life of 40 years, the staff determined that the fatigue analysis for these seven replacement diesel generator cooling water expansion joints will be beyond the period of extended operation.

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the seven diesel generator cooling water expansion joints that are projected to exceed the analyzed number of cycles during the period of extended operation will be adequately managed for the period of extended operation. The applicant's approach meets the acceptance criteria in SRP-LR Section 4.7.2.1 because the applicant's compliance with its CLB for these seven diesel generator cooling water expansion joints is governed by ASME Code Section XI and 10 CFR 50.55a. In addition, the applicant's Commitment No. 32—to replace these seven diesel generator cooling water expansion joints prior to the period of extended operation—provides a process for the applicant to track the completion of replacing these seven diesel generator cooling water expansion joints.

4.3.6.3 UFSAR Supplement

LRA Section A3.2.5 provides the UFSAR supplement summarizing the TLAA for the ASME Code Section III metal bellows and expansion joints. The staff reviewed LRA Section A3.2.5, consistent with the review procedures in SRP-LR Section 4.7.3.2, which state that the reviewer should confirm that the applicant provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA.

Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.7.2.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the TLAA for the ASME Code Section III metal bellows and expansion joints, as required by 10 CFR 54.21(d).

4.3.6.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses for all but seven of the diesel generator cooling water expansion joints remain valid for the period of extended operation. The staff also concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the seven diesel generator cooling water expansion joints projected to exceed the analyzed number of cycles during the period of extended operation will be adequately managed for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.4 Environmental Qualification of Electrical Equipment

The EQ requirements established by 10 CFR Part 50, Appendix A, Criterion 4, and 10 CFR 50.49 specifically require each applicant to establish a program to qualify electrical equipment so that such equipment, in its end of life condition, will meet its performance specifications during and following design basis accidents. The 10 CFR 50.49 EQ Program is a TLAA for purposes of license renewal. Electrical equipment with a qualified life equal to or greater than the duration of the current operating term is covered by TLAAs. The TLAA for the EQ of electric equipment includes all long-lived, passive, and active electrical and instrumentation and control (I&C) components that are important to safety and are located in a harsh environment. The harsh environment includes those areas subject to environmental effects caused by LOCAs, HELBs, and post-LOCA radiation.

As required by 10 CFR 54.21(c)(1), the applicant must provide a list of TLAAs. In addition, 10 CFR 54.21(c)(1) requires that the applicant demonstrate, for each TLAA, that the analyses remain valid for the period of extended operation, that the analyses have been projected to the end of the period of extended operation, or that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

4.4.1 Summary of Technical Information in the Application

LRA Section 4.4 describes the applicant's TLAA for EQ of electrical equipment. The applicant stated that the scope of equipment requiring qualification is those that automatically perform, that are used by operator action to perform, or whose failure could prevent the performance of:

- emergency reactor shutdown
- containment isolation
- reactor core cooling
- containment and reactor heat removal
- prevention of a significant release of radioactivity to the environment
- certain post-accident monitoring equipment

The applicant also stated that the EQ Program is consistent with the guidance of NUREG-0588, “Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment,” Category I, and the requirements in 10 CFR 50.49, with exemption from the EQ scope for certain low-safety significance (LSS) and non-risk significant (NRS) components.

The applicant dispositioned the EQ of Electric Equipment TLAA in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

4.4.2 Staff Evaluation

The staff reviewed LRA Sections 4.4 and B.3.2, EQ of Electric Equipment TLAA to confirm, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

The staff reviewed the applicant’s TLAA and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.4.3.1.3, which state that the applicant may reference the GALL Report in its LRA, as appropriate. The SRP-LR also states that the reviewer should confirm that the applicant stated that the report is applicable to its plant with respect to its EQ Program.

In LRA Section 4.4, “Environmental Qualification (EQ) of Electric Equipment,” the applicant stated that the EQ program is consistent with the guidance of NUREG-0588, Category I, and the requirements in 10 CFR 50.49, as exempted from environmental scope for certain low safety significance (LSS) and non-risk significant (NRS) components. By letter dated August 3, 2001, the staff granted STP an exemption from special treatment requirements (called the exemption). The NRC letter and associated safety evaluation contained the staff’s analysis and conclusion approving the STP exemption from certain specific requirements based on the applicant’s analysis and identification of non-risk significant (NRS) or low safety significance (LSS) SSCs.

Part 49 of 10 CFR, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,” establishes a program for qualifying the electric equipment (e.g., safety-related electric equipment, nonsafety-related electric equipment, and certain post-accident monitoring equipment). By letters dated July 13, 1999, as supplemented October 14 and 22, 1999; January 26 and August 31, 2000; and January 15, 18, 23, March 19, May 8, and 21, 2001 (hereinafter, the submittal, Adams Accession No. ML011430090), the applicant requested an exemption from 10 CFR Part 49(b), to exclude LSS and NRS components from the scope of electrical equipment important to safety pursuant to 10 CFR 50.49(b).

The staff noted that 10 CFR 54.21(c)(2) states that a list must be provided of plant-specific exemptions granted pursuant to 10 CFR 50.12 and, in effect, that are based on TLAAs, as defined in 10 CFR 54.3. By letter dated September 22, 2011, the staff issued RAI 4.4-1, requesting that the applicant provide an evaluation that justifies the continuation of these exemptions for the period of extended operation.

The staff also noted that 10 CFR 54.4(a)(1) states:

> Plant systems, structures, and components within the scope of this part are safety-related systems, structures, and components which are those relied upon to remain functional during and following design-basis events (as defined in

10 CFR 50.49(b)(1)) to ensure the integrity of the reactor coolant pressure boundary; (and) the capability to shut down the reactor and maintain it in a safe shutdown condition; or the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in § 50.34(a)(1), § 50.67(b)(2), or § 100.11 of this chapter, as applicable.

The applicant did not provide the plant-specific exemptions granted pursuant to 10 CFR 50.12 and, in effect, that are based on TLAAs, as defined in 10 CFR 54.3 and as applied to 10 CFR 50.49(b). Furthermore, the applicant did not provide any evaluation that justifies the continuation of this exemption for the period of extended operation. The staff is concerned that an exemption from 10 CFR 50.49(b) for electric equipment important to safety based on probabilistic risk assessment is inconsistent with the license renewal rule statement of considerations and 10 CFR Part 54.4 scoping, which uses deterministic criteria. Further, the staff is concerned that these exempted LSS and NRS components would not be included in the scope of license renewal; therefore, they are not subject to a TLAA or an associated AMP and, therefore, may not be capable of performing their intended function for the period of extended operation.

By letter dated September 22, 2011, the staff issued RAI 4.4-1, requesting the following from the applicant:

- provide a list of electrical and I&C system SSCs that were excluded from the scope of license renewal (10 CFR 54.4 (a)(1), (a)(2), and (a)(3)) as a result of special treatment requirements exemption of SSCs
- provide a list of electrical and I&C system SSCs that have been exempted from 10 CFR 50.49(b), including SSC replacements, subject to 10 CFR 54.4
- indicate whether the electrical and I&C system components, for which the exemption for 10 CFR 50.49 was granted, are within the scope of license renewal and, if not, provide justification for their exclusion and justify the continuation of these exemptions into the period of extended operation
- describe any subsequent modifications or changes to either plant design or LSS/NRS components that revised LSS/NRS electrical and I&C component environmental conditions or qualification and, if so, describe the modifications or changes incorporated into the aging management of the LSS/NRS electrical and I&C components
- discuss how the specific management program/controls (inspection, tests, and surveillances) are adequate to provide aging management during the period of extended operation such that LSS/NRS electrical and I&C components are capable of performing their intended function under design basis conditions throughout the service life of the component

In its response dated November 21, 2011, the applicant stated the following:

- No components were excluded from the scope of license renewal as a result of special treatment requirements exemption of SSCs (10 CFR 50.69).
- There are no electrical and I&C system SSCs, including SSC replacements that have been exempted from 10 CFR 50.49 qualification requirements. The LSS and NRS EQ components are treated in the same way as non-LSS and non-NRS EQ components

with the exception that the documentation requirements for LSS and NRS components are not as stringent as those for non-LSS and non-NRS EQ components. UFSAR Section 13.7 allows LSS and NRS components to not be qualified per 10 CFR 50.49, but, as stated above, STP has opted to maintain the qualification of the LSS and NRS components.

- The EQ electrical and I&C system components classified as LSS or NRS are within the scope of license renewal. No EQ components were excluded from the scope of license renewal as a result of special treatment requirements exemption of SSCs granted by the staff in a safety evaluation issued on August 3, 2001 (10 CFR 50.69).
- Data loggers were installed in containment at selected locations to determine actual temperatures. This data was then used to determine the qualified life of EQ transmitters at those selected locations. The actual temperatures were lower than the design temperature, which provided margin for extending the qualified life. The data gathered were for extending the qualified life of selected transmitters but did not change the design criteria. Design change packages were prepared with the new qualified lives.
- The components associated with the special treatment requirements are part of the STP EQ Program. They are treated the same way as any other EQ component with the exception that the documentation requirement is not as stringent as that of a normal EQ component. These components would still follow the replacement dates (start of qualified life and replacement due date), as designated under its qualification maintenance database.

The staff found the applicant response acceptable because the applicant will maintain the qualified life of EQ electric equipment in accordance with 10 CFR 54.4 requirements, and no EQ equipment will be excluded from the scope of license renewal.

The staff reviewed LRA Sections 4.4 and B.3.2 and plant-basis documents, and interviewed plant personnel to confirm whether the applicant provided adequate information to meet the requirement in 10 CFR 54.21(c)(1). For the electrical equipment, the applicant uses 10 CFR 54.21(c)(1)(iii) in its TLAA evaluation to demonstrate that the aging effects of EQ equipment will be adequately managed during the period of extended operation. The staff reviewed the applicant's EQ Program to determine whether it will assure that the electrical and I&C components covered under this program will continue to perform their intended functions, consistent with the CLB, for the period of extended operation. Per the GALL Report, plant EQ programs that implement the requirements in 10 CFR 50.49 are considered acceptable AMPs under license renewal (10 CFR 54.21(c)(1)(iii)). GALL Report AMP X.E1, "Environmental Qualification (EQ) of Electric Components," provides a means to meet the requirements in 10 CFR 54.21(c)(1)(iii).

The staff's evaluation of the components' qualification focused on how the EQ Program manages the aging effects to meet the requirements pursuant to 10 CFR 50.49. The staff conducted an audit of the information provided in LRA Sections 4.4 and B.3.2 and program basis documents. LRA Section 4.4 discusses the component reanalysis attributes, including analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions. Based on the AMP audit and as documented in SER Section 3.0.3.1.7, "Environmental Qualification (EQ) of Electrical Components," the staff finds that the EQ program is consistent with the GALL Report. The staff further concludes that the applicant's EQ of Electric Equipment TLAA is implemented per the requirements in 10 CFR 54.21(c)(1)(iii).

Therefore, the staff finds that the applicant's EQ program demonstrates, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions will be adequately managed for the period of extended operation. The applicant's EQ program is, therefore, capable of programmatically managing the qualified life of components within the scope of the program for license renewal. The continued implementation of the EQ program provides assurance that the aging effects will be managed and that components within the scope of the EQ program will continue to perform their intended functions for the period of extended operation.

4.4.3 UFSAR Supplement

LRA Section A. 3.3 provides the UFSAR supplement summarizing the EQ of Electric Equipment TLAA, which manages component thermal, radiation, and cyclical aging using aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, EQ components not qualified for the current license term are to be refurbished or replaced or have their qualification extended before reaching the aging limits established in the evaluation. The staff reviewed LRA Section A.4.4.1, consistent with the review procedures in SRP-LR Section 4.4.3.2, which state that the reviewer should confirm that the applicant provided information to be included in the UFSAR supplement that includes a summary description of the TLAA evaluation of the EQ of electric equipment. The SRP-LR also states that the reviewer should confirm that the applicant provided a UFSAR supplement with information equivalent to that in SRP-LR Table 4.4-2.

Based on its review, the staff finds that the UFSAR supplement meets the acceptance criteria in SRP-LR Section 4.4.2.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the TLAA for the EQ of Electric Equipment Program, which manages component thermal, radiation, and cyclical aging through the use of aging evaluation based on 10 CFR 50.49(f) qualification methods, as required by 10 CFR 54.21(d).

4.4.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that, for the EQ of Electric Equipment TLAA, the effects of aging on the intended functions will be adequately managed for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the EQ of electric equipment TLAA evaluation of the period of extended operation, as required by 10 CFR 54.21(d).

4.5 Concrete Containment Tendon Prestress Analysis

4.5.1 Summary of Technical Information in the Application

LRA Section 4.5 describes the applicant's TLAA for concrete containment tendon prestress analysis. The LRA states that the containment for each unit is a prestressed concrete, hemispherical, dome-on-a-cylinder structure with steel membrane liners and a flat basemat. Post-tensioned tendons compress the concrete and permit the structure to withstand design basis accident internal pressures. The LRA states that the Tendon Surveillance Program is used to ensure that tendons continue to maintain adequate prestress for the period of extended operation. The applicant's Tendon Surveillance Program periodically measures the prestress load on a defined sample of tendons and examines the condition of the tendons and supporting structures, materials, and components. The data collected from the program reconfirm that the expected tendon prestress loads will remain within design limits to at least the next inspection, or, if the relaxation is not acceptable, the program prescribes retensioning or other corrective measures to ensure that at no time will the average prestress in a tendon group fall below the minimum required prestress.

The LRA describes the post-tensioning system of each unit as consisting of two tendon groups. There are 96 vertical, inverted-U-shaped tendons that extend up through the basemat through the full height of the cylindrical walls and over the dome. The vertical tendons are anchored through the bottom of the basemat. There are 133 horizontal circumferential (hoop) tendons located at intervals from the basemat up to approximately the 45-degree elevation of the dome. They are anchored at three exterior buttresses, 120 degrees apart. The total tendon load is carried by a shim stack to steel bearing plates embedded in the structure.

LRA Appendix B, Section B3.3, summarizes the TLAA AMP, “Concrete Containment Tendon Prestress” Program. The inspection program is governed by ASME Code Section XI, Subsection IWL. In accordance with 10 CFR 50.55a(g)(4)(ii), the third ISI Program for ASME Code Section XI, Subsection IWL will be conducted in accordance with the requirements of the 2004 edition (no addenda). The LRA states that the program calculates current trend values for each tendon on an individual basis by regression analysis of the full set of individual tendon lift-off data for each tendon group, consistent with the methodology presented in NRC Information Notice (IN) 99-10, “Degradation of Prestressing Tendon Systems in Prestressed Concrete Containments,” Attachment 3. The LRA further states, “the calculations of predicted force are consistent with NRC RG 1.35.1, ‘Determining Prestressing Forces for Inspection of Prestressed Concrete Containments,’” (July 1990, and reviewed April 2015). The current Tendon Surveillance Program uses the ASME Code Section XI, Subsection IWL acceptance criteria of 95 percent of the predicted force, in lieu of the RG 1.35.1 lower bound.

The LRA states that the surveillance calculation estimates the 40-year loss of prestressing force and lists the predicted and measured lift-off forces for individual tendons selected for surveillance. The LRA further states that the “measured force trend lines,” when projected past 60 years, remain above the minimum required design prestress values. The most recent regression analysis is included in the 2009, 20-year tendon surveillance report. The LRA states that the recent surveillance data for individual tendons have all fallen above the first action limit at 95 percent of the predicted force line, and the regression analysis of surveillance lift-off data has extended the trend lines for both the vertical and horizontal tendons of each unit to 100 years. Finally, the LRA states that the trend lines for horizontal and vertical tendons will remain well above their minimum required values (MRVs) through the period of extended operation.

4.5.2 Staff Evaluation

The staff reviewed LRA Section 4.5 and the concrete containment tendon prestress TLAA to confirm, pursuant to 10 CFR 54.21(c)(1)(iii), that the intended functions will be adequately managed for the period of extended operation.

The staff reviewed the applicant’s TLAA and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.5.3.1.3. These review procedures state that the applicant may reference the GALL Report in its license renewal application provided that a TLAA AMP consistent with the GALL Report AMP X.S1 is in place to manage the effects of aging (i.e., loss of tendon prestress) for the period of extended operation.

The staff noted that LRA Section 4.5 credits the TLAA AMP “Concrete Containment Tendon Prestress” program described in LRA Section B3.3 (evaluated in SER Section 3.0.3.1.8), to manage the loss of tendon prestress for the period of extended operation. The LRA states that the program will confirm that the average lift-off forces of the prestressed tendons remain above their MRVs through the period of extended operation. The staff confirmed in accordance with the review procedures of SRP-LR Section 4.5.3.1.3 that the applicant identified the appropriate TLAA AMP consistent with the GALL Report TLAA AMP X.S1.

The staff also reviewed the tendon regression analysis input data of the measured lift-off forces (LRA Table 4.5-1). The staff verified that Figures 4.5-1 through 4.5-4 of LRA Section 4.5 show the trend lines based on regression analysis of the Unit 1 and Unit 2 vertical and horizontal tendon lift-off force data and confirmed that the lift-off force trend lines for each tendon group are based on individual tendon lift-off forces and not on the average lift-off forces for each tendon group, as discussed in IN 99-10. For both horizontal and vertical tendon groups, the projected prestressing force trend lines remain above their respective MRVs through the period of extended operation.

The review of LRA Section B3.3 indicates that the “Concrete Containment Tendon Prestress” program follows the ASME Code Section XI, Subsection IWL requirements, to manage the loss of tendon prestress aging effect in the post-tensioning system. The LRA states that the containment tendon ISI program was originally in accordance with RG 1.35, “Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures,” and that beginning in year 15, inspections have been in accordance with ASME Code Section IWL. RG 1.35 states that the ISI should be performed at 1, 3 and 5 years after the initial structural integrity test (ISIT) and every 5 years thereafter. ASME Code Subsection IWL-2421, “Sites with Multiple Plants,” states that, following the completion of the ISIT, containment examinations be performed at years 1, 3, and 10 and every 10 years thereafter. The ASME Code also states that, for each subsequent containment identical in design and constructed not more than 2 years apart, examinations should be performed at years 1, 5, and 15 and every 10 years thereafter.

The staff noted that there was no data provided in LRA Table 4.5-1 for a 3-year tendon inspection of either containment; therefore, in Part 2 of RAI B3.3-2 (by letter dated August 15, 2011), the staff requested that the applicant describe the tendon surveillance intervals for both containments. In its response dated October 10, 2011, the applicant stated that the plant was originally licensed for a containment inspection program that was in accordance with RG 1.35 (April 1979, proposed revision 3). The applicant stated that the schedule in the proposed RG 1.35 did not call for inspection at year 3, but rather at year 5, and that the actual liftoff testing for Unit 1 was performed at year 5 in accordance with that schedule. The staff determined this to be inconsistent with the applicant’s program basis documentation, specifically CC-5207, Revision 8, “RCB Tendon Surveillance,” approved on August 12, 2004, which contains a list of tendons examined in year-3 surveillances for both units, as well as the methodology for determining the year-3 sample population. It was unclear to the staff whether the year-3 surveillances mentioned in the program basis documentation were associated with lift-off testing.

In a teleconference on January 4, 2012 (ADAMS Accession No. ML12011A008), the staff requested that the applicant clarify the apparent discrepancy between the schedule of tendon inspections and the year-3 surveillance activities referenced in the Tendon Surveillance Program basis documents that the staff reviewed onsite. The applicant explained that its inspection program takes credit for Regulatory Position 1.5 in RG 1.35, Revision 3. Regulatory Position 1.5 states:

> [T]he liftoff force comparison [may be modified from the 1, 3 and 5 year schedule stated in Regulatory Position 1.3 to perform the second tendon surveillance lift-off test at year 5 instead of year 3] if any two containments at the same site are shown to satisfy all three of the following conditions: (a) the containments are identical in all aspects such as size, tendon system, design, materials of construction, and method of construction; (b) their ISITs were performed within two years of each other; and (c) there is no unique situation that may subject either containment to a different potential for structural or tendon deterioration.

The applicant clarified that the information in the program basis documents regarding a year-3 surveillance referred to a visual examination only, and that no lift-off testing was scheduled or performed then. The staff noted that this clarification of the applicant's RAI response was not inconsistent with program basis document information reviewed by the staff onsite, and determined that the schedule of inspections performed was consistent with Regulatory Position 1.5 in RG 1.35. The staff finds the applicant's response acceptable because it confirmed that the program basis documents apply the schedule provisions of Regulatory Position 1.5 of RG 1.35. The staff's concern discussed in RAI B3.3-2 is resolved.

RG 1.35, Section 2.4, states that the tendons to be inspected should be randomly selected from each tendon group, with the groups defined in Section 2.1 as vertical, hoop, dome, and inverted U, as applicable. Sections 2.1 through 2.3 of RG 1.35 recommend and Table IWL-2521-1 of the ASME Code Section XI requires that a defined minimum number of tendons of each type (e.g., hoop, vertical, dome, and inverted U) be examined at every inspection interval. The staff noted LRA Section 4.5 provides the minimum required lift-off force for three groups of tendons—inverted U-shaped vertical tendons, horizontal dome tendons, and horizontal wall tendons. The staff also noted that although LRA Table 4.5-1 uses the same grouping, there are no data for the examination of horizontal dome tendons for Unit 1 for year 20 or Unit 2 for years 5 or 15. The staff was unclear as to whether the applicant considers the horizontal dome tendons as a separate tendon group from the horizontal cylinder tendons and, if so, why dome tendons are not consistently inspected at each interval. The staff also noted that four Unit 2 horizontal dome tendons were surveyed in year 10 even though the inspection schedule for Unit 2 is for year 5 and year 15, not year 10.

In a teleconference on January 4, 2012, the staff requested that the applicant explain, given the requirements to examine a minimum number of tendons of each type, why there are no surveillance data for horizontal dome tendons for the inspection intervals listed above. Subsequently, the staff issued a request for additional information (RAI 4.5-1), dated February 15, 2012. By letter dated March 12, 2012, the applicant responded that the horizontal tendons in the dome are grouped with the horizontal tendons in the cylinder wall. For both the horizontal and vertical tendons, the tendons scheduled to be inspected during each interval are selected at random. There is one control tendon in each group that is inspected at each interval; otherwise, the sample of tendons inspected is always different. Since the horizontal tendons are considered one group and randomly selected for inspection, it is possible that a dome tendon will not be selected. The staff finds the applicant's response acceptable because the applicant clarified that the applicant's post-tensioned containment horizontal tendons are considered as one group, and the controlling minimum required value (MRV) for both dome and cylinder hoop tendons is the prestress force of cylinder hoop tendons, which is greater than that required for the dome hoop tendons, as shown in Figures 4.5-2 and 4.5-4. Further, because the tendons to be inspected are randomly selected, dome tendons may not be included in the inspection sample at every interval.

In response to the staff's request to explain why there was a year-10 surveillance done outside of the originally defined inspection schedule, the applicant stated that the inspection was done as a result of the discovery that errors had been made in the methodology for determining the prescribed lower limit of the tendon prestressing forces. One tendon in each unit and the adjacent tendons were retested during the year-10 interval. This issue is discussed further in SER Section 3.0.3.1.8, "Concrete Containment Tendon Prestress."

The staff noted that, in the applicant's Concrete Containment Tendon Prestress Program, one tendon of each type is designated as the control tendon, to be examined during every inspection interval, in accordance with RG 1.35 and ASME Code Section XI, Subsection IWL. The expectation is that prestressing tendons will lose their prestressing forces with time due to creep and shrinkage of concrete and relaxation of the prestressing steel. In its review of the tendon regression analysis input data in LRA Table 4.5-1, the staff noted that for Unit 1, the lift-off force for the vertical U-shaped control tendon V126 increased from 1,340 kips (shop end) and 1,380 kips (field end) at the year-10 inspection to 1,363 kips (shop end) and 1,389 kips (field end) at the year-20 inspection. There was also an increase in prestressing force between the year-10 and year-20 interval inspection of the Unit 1 horizontal cylinder (wall) control tendon 1H091. Examinations of vertical and horizontal control tendon lift-off measurement results for Unit 2 did not result in any increases in lift-off forces. In the January 4, 2012, teleconference, the staff requested that the applicant explain these anomalies. The applicant responded that it used a different vendor for the testing machinery in the year-10 and year-20 inspection intervals, and that there may have been slight inaccuracies in calibration. The applicant stated that there is no other reason why larger forces were measured in the two tendons. The applicant justified the results by citing the provisions of ASME Code Section XI, Subsection IWL-2522(b), which states that "equipment used to measure tendon force shall be calibrated prior to the first tendon force measurement and following the final tendon force measurement of the inspection period." ASME Code Section XI, Subsection IWL-2522(b) also states that the "accuracy of the calibration shall be within 1.5 [percent] of the specified minimum ultimate strength of the tendon." The applicant stated that, for all instances in which the year-20 lift-off force was larger than the year-10 lift-off force, the largest discrepancy is at the shop end of tendon V126, where the tendon was measured to gain 23 kips of prestressing force rather than the predicted loss of 10 kips. The applicant stated that the delta of 33 kips is within the acceptance criteria of 1.5 percent allowed by ASME Code Section IWL-2522. The staff was unclear as to how the applicant applied provisions of IWL-2522(b) to tendon liftoff forces over successive intervals, when IWL-2522(b) applies to calibration of the hydraulic lift-off jack over the same inspection interval. The staff determined that it needed more information to complete its review.

By letter dated February 15, 2012, the staff issued RAI 4.5-1, requesting that the applicant explain how it applied the provisions of IWL-2522(b) to the condition of the surveillance measuring an increase in lift-off force when the tendon was predicted to relax. The staff requested that the applicant explain its basis for applying IWL-2522(b) to the lift-off results for individual tendons and to provide details of the calibration measurements of the jacking equipment used to perform the tendon surveillance. In its letter dated March 12, 2012, the applicant responded, stating that surveillances performed 10 years apart by different vendors using different equipment cannot be assumed to produce results that are more accurate than the calibration tolerance specified in the ASME Code. ASME Code Section XI, Subsection IWL-2522(b) allows accuracy of the tendon calibration to be within 1.5 percent of the specified minimum ultimate strength of the tendon. The applicant stated that its tendons each have 186 wires of one-quarter inch diameter, for a total cross section of 9.13 square inches. The material ultimate strength is 240 ksi. Therefore, 1.5 percent of the specified minimum ultimate strength of one tendon is 33 kips. The largest discrepancy between the measured and predicted tendon forces was in tendon V126 (shop end) between the 10-year and 20-year inspection interval, where the tendon was predicted to lose 10 kips of prestressing force but measured an increase in prestress of 23 kips instead, for a total difference of 33 kips (which is the upper limit allowed by the ASME Code for each measurement). The staff finds the applicant's response acceptable because the apparent increase in the three noted tendon liftoff forces between years 10 and 20 can be attributed to measurement and equipment calibration, and the error remains within the ASME Code allowable 1.5 percent calibration tolerance of tendon liftoff forces. The staff's concern discussed in RAI 4.5-1 is resolved.

The applicant's program meets the acceptance criteria in SRP-LR Section 4.5.2.1.3 because it assesses the concrete containment tendon prestressing forces, and the staff has determined that the AMP is acceptable to address concrete containment tendon prestress in accordance with 10 CFR 54.21(c)(1)(iii), except for operating experience. The staff reviewed the applicant's operating experience related to the containment tendon prestressing surveillances. The results of the review are documented in the staff evaluation of the Concrete Containment Tendon Prestress Program in SER Section 3.0.3.1.8. The results show that the applicant's program has adequately considered plant-specific operating experience.

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions of the concrete containment prestressed tendons will be adequately managed for the period of extended operation. Additionally, it meets the acceptance criteria in SRP-LR Section 4.5.2.1.3 because the Containment Tendon Prestress Program assesses the concrete containment tendon prestressing forces, and the staff has determined that the program is an acceptable way to manage the effects of aging of the containment tendon prestressing system.

The staff's review of the concrete containment tendon prestress TLAA, as described above, was conducted and documented prior to withdrawal of RG 1.35. In August 2015, the staff withdrew RG 1.35 (see 80 FR 52067, dated August 27, 2015). The NRC's Basis for Withdrawal related to RG 1.35 (ADAMS Accession No. ML15040A665) states that the RG 1.35 guidance has been incorporated into ASME Section XI, Subsection IWL, which, with specified modifications and limitations, is now mandated by 10 CFR 50.55a. As a result, the use of RG 1.35 to assist licensees in meeting the requirements in Appendix A to 10 CFR Part 50 is superfluous. The *Federal Register* notice (80 FR 52067) states that the withdrawal does not affect the licensing bases of current licensees approved to use RG 1.35. The staff confirmed that the portions of the program that credit RG 1.35 are addressed in or bounded by the requirements of the ASME Code Subsection IWL that are now mandated by 10 CFR 50.55a. Therefore, the withdrawal of RG 1.35 does not affect the staff's conclusion that the containment prestressed tendons can be adequately managed using the applicant's "Concrete Containment Tendon Prestress" AMP (LRA Section B3.3), pursuant to 10 CFR 54.21(c)(1)(iii).

4.5.3 UFSAR Supplement

LRA Section A3.4 provides the UFSAR supplement summarizing the Containment Tendon Prestress TLAA. The staff reviewed LRA Section A3.4, consistent with the review procedures in SRP-LR Section 4.5.3.2, which state that the reviewer verifies that the applicant has provided an UFSAR supplement that includes a summary description of the evaluation of tendon prestress TLAA with information equivalent to that in SRP-LR Table 4.5-1.

Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.5.3.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the TLAA for concrete containment prestress, as required by 10 CFR 54.21(d).

4.5.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions of the containment prestressing system will be adequately managed for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.6 Containment Liner Plate, Metal Containments, and Penetrations Fatigue Analysis

4.6.1 Containment Liner Plate, Containment Equipment Hatches, and Containment Polar Crane Brackets

4.6.1.1 Summary of Technical Information in the Application

LRA Section 4.6 describes the applicant's TLAA of the containment structure liner plate, metal containments, and penetration fatigue analysis. The LRA states:

> [t]he STP containment building design report cites Bechtel Topical Report BC-TOP 1, "Containment Building Liner Plate Design Report," and Bechtel Topical Report BC-TOP 5A, "Prestressed Concrete Nuclear Reactor Containment Structures" for design of the reactor building containment and the containment liner. The LRA further states that the containment structure was primarily designed in accordance with the proposed ACI 359-ASME Code Section III, Division 2, issued for trial use and comments in 1973, including subsequent addenda 1 through 6.

The applicant stated that a review of the penetration specification, liner specification, containment building design report, and design calculations found the application of cyclic limits to the design to be time-dependent only for design of the personnel and emergency airlocks and some of the process penetrations.

Design Criteria and Design Codes. The applicant stated that the post-tensioned concrete containment vessels were poured against steel membrane liners. The applicant also stated, "no credit is taken for the liner for the pressure design of the containment vessel, but the liner and penetrations ensure the vessel is leak-tight, and its electrical, process, personnel airlock, and equipment hatch penetrations are part of the containment pressure boundary."

The LRA states that the liner fatigue evaluation was performed in accordance with the design and construction specifications of Subsections NE-3222.4 and NE-3131(d) of the ASME Code Section III. The LRA also states that subparagraph NE-3222.4 of the ASME Code Section III Division I, Subsection NE, 1974 and later, provides rules for fatigue analysis of metal containment (MC) components subject to operating condition cyclic loads and thermal conditions, where part NE-3222.4(d) specifically addresses waivers to such an analysis. The LRA further states that the reference is to ASME Code Subsection NE; any TLAAs arising from its use would apply only to the containment liner, penetrations, airlocks, and hatches.

Containment Liner Plate. The LRA states that the containment liner and penetrations were designed to BC-TOP-1 and ASME Code Section III, Division 2, issued for trial comment in 1973, including addenda 1-6. The applicant stated that it performed a thorough search of the CLB, including the liner specification and the containment building design report and found no indication of any fatigue analysis or design reference for a stated number of cyclic loads for the containment liner plate.

The applicant stated that the containment liner plate fatigue is not a TLAA by Criterion 6 in 10 CFR 54.3(a). This criterion states that TLAAs for the purposes of this part are those applicant calculations and analyses that "are contained or incorporated by reference in the CLB."

Equipment Hatches. The applicant stated that the Unit 1 and 2 equipment hatches were designed to ASME Code Section III, 1971 edition, winter 1973 addenda. The applicant further stated that the design report exhibits no design for a stated number of load cycles or any other evidence of a TLAA.

The applicant stated that equipment hatch fatigue is not a TLAA by Criterion 3 in 10 CFR 54.3(a). This criterion states that TLAAs for the purposes of this part "involve time-limited assumptions defined by the current operating term, for example, 40 years."

Personnel and Emergency (Auxiliary) Airlocks. The applicant stated that the personnel and emergency (auxiliary) airlocks were specified to ASME Code Section III Division 1, Subsection NE, Class MC components, 1974 edition, winter 1974 addenda, but were analyzed to the winter 1975 addenda. The applicant also stated that an NB-3222.4(d) fatigue waiver for each depends, in part, on the assumed number of load cycles and is, therefore, a TLAA. The fatigue waiver TLAA is described in LRA Section 4.6.1.

Polar Crane Brackets. The applicant stated that the polar crane is supported on a system of girders, which are supported by a series of brackets that are attached to the containment shell. The applicant also stated that the design of the polar crane brackets neither reports nor specifies a fatigue analysis. The applicant further stated that the applicable design code specifies that no evaluation of fatigue resistance is required if the number of cycles of application of live load (lifts or load cycles) is less than 20,000, which is greater than the revised expected number of 3,416 lifts (see ADAMS Accession No. ML11291A152) for the polar crane. Therefore, the applicant stated that polar crane bracket fatigue is not a TLAA because it does not meet Criterion 3 in 10 CFR 54.3(a). LRA Section 4.7.1 discusses the design of the polar crane itself.

Penetrations. The applicant stated that the design of a number of containment penetrations includes a fatigue analysis. The containment penetrations fatigue TLAA is described in LRA Section 4.6.2.

4.6.1.2 Staff Evaluation

The staff reviewed the applicant's TLAA in LRA Section 4.6 on the absence of TLAAs related to fatigue of the containment liner, equipment hatches, and polar crane brackets.

Design Criteria and Design Codes. The staff reviewed the STP UFSAR and confirmed that the proposed ACI 359-ASME Code Section III, Division 2, and BC-TOP-5A are referenced. The staff's review of the UFSAR also confirmed that it contains the referenced sections and editions of the ASME Codes. The applicability of these codes are reviewed and discussed in the appropriate sections for each of the staff's evaluations below.

Containment Liner Plate Analysis. The staff reviewed the applicant's claim that the lack of any CLB information related to containment liner plate cyclic loading or any fatigue analysis excludes the liner plate from TLAA consideration, per TLAA Criterion 6 in 10 CFR 54.3(a). This criterion states that TLAAs for the purposes of this part are those applicant calculations and analyses that "are contained or incorporated by reference in the CLB." The staff also noted that UFSAR Section 3.8.2 and Table 3.2.A-1 state that the liner plate does not require ASME (certification) N-stamp.

To verify the applicant's claim of absence of a TLAA due to its absence in the CLB (see also SER Section 4.1.2.1.2 "Evaluations, Analyses, and Calculations in the CLB That Do Not Conform to TLAA Criteria, or Absence of a TLAA Due to Absence in the CLB"), the staff examined the UFSAR for a specific entry on fatigue or cycles of loading and noted that UFSAR Section 3.8.1.5.9 states, "[t]he effect of cycled stresses and strains in the liner is considered by performing a fatigue analysis, in accordance with Section 3.8.1.5.6, which includes the reactor shutdown-startup cycles." The staff also reviewed UFSAR Sections 3.8.1.2 and 3.8.1.5.6, which state that the allowable stresses and strains in the liner plate should be in accordance with the proposed ACI 359-ASME Code Section III, Division 2, Concrete Reactor Vessels and Containments. Section CC-3760 of the proposed Code states that the design of liners is not considered to be fatigue-controlled because the stress and strain changes would occur only a small number of times and produce minor stress-strain fluctuations. Furthermore, for strains due to earthquakes and design basis accidents, the Code states that these are too infrequent and with too few cycles to be controlling. Nevertheless, the staff noted that the Code holds the designer responsible to meet the design specifications for cyclic loading and thermal conditions.

The staff also noted that the requirements for cyclic loading are stated in UFSAR Section 3.8.2.5.5.3, which references the ASME Code Section III, Division 1, Sections NE-3131(d) and NE-3222.4. The ASME Code NE-3131(d) (1974 editions or later) rules out consideration for earthquake transients unless they impact designated liner locations recognized in the specifications. ASME Code NE-3222.4(d), "Analysis for Cyclic Operations, Vessels Not Requiring Analysis for Cyclic Operation," provides for a relief from fatigue analysis when certain cyclic loading criteria are met. The staff further reviewed the UFSAR and Bechtel Topical Report BC-TOP-1, "Containment Building Liner Plate Design Report, Part I: Liner Plate and Anchorage System," and other available topical reports and specifications for applicable cyclic loads or calculations that consider the number of cycles satisfying the exclusion criteria of NE-3222.4(d). The staff confirmed that the aforementioned documents had no entries for cyclic loading calculations and did not consider fatigue analysis of the liner plate.

The staff noted that there was an apparent inconsistency or gap with regard to the information that was provided by the applicant on the design requirements for the containment liners. Therefore, by letter dated September 22, 2011, the staff issued RAI 4.1-2, requesting that the applicant clarify if subparagraph NE-3222.4 in the 1974 edition of the ASME Code was used for the containment liners. The staff also asked the applicant to justify why fatigue analyses for the containment liner plate were not performed in accordance with subparagraph NE-3222.4 of the 1974 ASME Code, or to clarify if the liner had been exempted (waived) from fatigue analysis under provisions of NE-3222.4(d). If the liner plate was waived from fatigue analysis under NE-3222.4(d), the staff requested clarification on why the fatigue waiver analysis would not need to be identified as a TLAA for the LRA in the manner that the fatigue waiver analysis for the personnel and emergency (auxiliary) air locks was identified as a TLAA in the LRA.

The applicant responded to RAI 4.1-2 by letter dated November 21, 2011. In its response, the applicant stated that the containment liner was not designed to the ASME Code Section III, subarticle NE-3000, design requirements. The applicant stated that UFSAR Section 3.8.1.2 identifies that the containment liner was designed to the 1973 edition of the ASME Code Section III, Division 2, including addenda 1 through 6. The applicant also stated that the NRC approved Bechtel Specification BC-TOP-5-A as an acceptable means of meeting the ASME Code design criteria for the liner plate and that specification BC-TOP-5-A references the methodology in Bechtel Specification BC-TOP-1. The applicant further stated that this design method compares the stresses in BC-TOP-1, which are independent of the number of load cycles and have no fatigue analyses.

The staff confirmed the accuracy of the information in the applicant's response to RAI 4.1-2 through an audit of the applicant's design specification for the containment liner, penetrations, airlocks, and equipment hatches, with the exception of one matter that needed clarification by the applicant. Specifically, the staff noted that the design specification states that the "requirements for an 'analysis of cyclical loading' will be investigated in accordance with Section NE-3222.4 and NE-3121 of the ASME Code Section III." However, the staff noted that the design specification did not identify which of the containment components in the design specification were within the scope of the design specification's fatigue analysis statement.

By letter dated February 15, 2012, the staff issued RAI 4.1-2a, requesting additional clarification on whether the fatigue analysis statement in the containment liner design specification was only applicable to those components in the specification that were designed to ASME Code Section III, Division 1, requirements (e.g., the containment penetrations) or if it also applied to the containment liner plate, which was designed to ASME Code Section III, Division 2, requirements.

The applicant responded to RAI 4.1-2a by letter dated March 29, 2012. In its response, the applicant stated that, upon review of the design specification for the containment structures, it confirmed that the design specification did not require the containment liners to be analyzed to ASME Code Section III, Division 1, requirements because they were not qualified as pressure-retaining components for the containment structures. The applicant stated that the containment liners were only analyzed to ASME Code Section III, Division 2, requirements, which did not require the liners to be the subject of a CUF-based fatigue analysis. Based on this review, the staff finds that the applicant resolved the issue on whether the design specification for the containment liners required the liners to be analyzed with a fatigue analysis. Additionally, based on the response to RAI 4.1-2a, the staff finds that the LRA does not need to include a fatigue analysis-based TLAA for the containment liners because the containment liners are containment pressure boundary components that were not analyzed to ASME Code Section III, Division 1, requirements. The staff's concerns in RAIs 4.1-2 and 4.1-2a are resolved.

The staff finds that the applicant demonstrated that liner plate fatigue is not a TLAA because it does not meet Criterion 6 of 10 CFR 54.3(a).

<u>Equipment Hatches Analysis</u>. The staff reviewed the applicant's claim that the design of the equipment hatches is not a TLAA and confirmed that the lack of any CLB information related to "Containment Equipment Hatches" cyclic loading or its fatigue analysis excludes the equipment hatches from TLAA consideration, per TLAA Criterion 3 in 10 CFR 54.3(a). This criterion states that TLAAs for the purposes of this part "involve time-limited assumptions defined by the current operating term, for example, 40 years." Criterion 3 is discussed in the review procedure in

SRP-LR, Revision 1 (and also Revision 2), Section 4.1.3, which states, "[t]he defined operating term should be explicit in the analysis. Simply asserting that a component is designed for a service life or plant life is not sufficient. The assertion is supported by calculations or other analyses that explicitly include a time limit."

To verify the applicant's statement of absence of a TLAA because of lack of time limited assumptions defined in the CLB, the staff reviewed UFSAR Section 3.8.1.5.9 and confirmed that thermal cycling and startups and shutdowns are considered over a 40-year plant life. The staff also reviewed applicant topical and vendor reports to locate calculations and analyses that could demonstrate the fatigue life of equipment hatches to be limited by reactor startup and shutdown cyclic loading of 40 years and did not locate any calculations. The staff noted that UFSAR Section 3.8.2, "Steel Containment System (ASME [Code] Class MC Components)," indicates that the equipment hatches are designed, fabricated, and installed in accordance with ASME Code Section III, Class MC components and that the equipment hatches are not stamped because they are an integral part of an unstamped containment vessel. The staff noted that BC-TOP-5A, which addresses the design of the reactor building containment, includes the equipment hatch openings as part of the structure. Through review of the referenced topical reports in the LRA, resolution of RAIs 4.1-2 and 4.1-2a, discussed and resolved above, including the audit of the applicant's design specification, the staff confirmed that the equipment hatches have no cyclic loading requirements.

The staff finds that the applicant demonstrated that equipment hatch fatigue is not a TLAA because it does not meet Criterion 3 of 10 CFR 54.3(a).

<u>Personnel and Emergency (Auxiliary) Airlocks Analysis</u>. The staff's evaluation of the applicant's personnel and emergency (auxiliary) airlocks TLAA is described in SER Section 4.6.2.

<u>Polar Crane Brackets Analysis</u>. The staff reviewed the applicant's claim that the polar crane brackets are not TLAAs and confirmed that the lack of any CLB information related to polar crane brackets cyclic loading or its fatigue analysis excludes the polar crane brackets from TLAA consideration, pursuant to the review procedure delineated in SRP-LR Section 4.1.3. This section states that "[t]he defined operating term should be explicit in the analysis. Simply asserting that a component is designed for a service life or plant life is not sufficient. The assertion is supported by calculations or other analyses that explicitly include a time limit."

To verify the applicant's statement of absence of a TLAA because of lack of time-limited assumptions defined in the CLB, the staff reviewed Bechtel Topical Report BC-TOP-1, the UFSAR Section 3.8.1.2.1-referenced "American Institute of Steel Construction (AISC) Specifications for Structural Steel Buildings, 1969," supplements 1, 2, and 3, and other vendor topical reports to identify any calculations that involve time-limited assumptions defined by the current operating term of 40 years. The staff noted that, in accordance with UFSAR Section 3.8.3.2, "Applicable Codes, Standards and Specifications," the code of record for cranes is the Crane Manufacturers Association of America (CMAA) Specification 70. The staff also noted that the Bechtel Topical Reports, BC-TOP-1 and BC-TOP-5A, do not contain cyclic loadings or report fatigue analyses calculations for the polar crane brackets. The staff further reviewed the latest AISC Load and Resistance Factor Design Specifications and confirmed the applicant's claim that the specifications do not consider fatigue to be applicable when the number of cycles of live loads for the life of the crane is less than 20,000.

The staff finds that the applicant demonstrated that polar crane bracket fatigue is not a TLAA because it does not satisfy Criterion 3 of 10 CFR 54.3(a).

Penetrations Analysis. The staff's evaluation of the applicant's penetrations TLAAs is discussed in SER Section 4.6.3.

4.6.1.3 UFSAR Supplement

The staff concludes that no UFSAR supplement is required because containment liner plate, equipment hatch, and polar crane bracket fatigue are not TLAAs.

4.6.1.4 Conclusion

Based on its review, the staff concludes that containment liner plate, equipment hatch, and polar crane bracket fatigue are not TLAAs.

4.6.2 Fatigue Waivers for the Personnel Airlocks and Emergency (Auxiliary) Airlocks

4.6.2.1 Summary of Technical Information in the Application

LRA Section 4.6.1 describes the applicant's fatigue waiver analysis for the personnel and emergency (auxiliary) airlocks. It states that the design of the personnel and emergency airlocks included an ASME Code Section III NE-3222.4(d) fatigue waiver analysis, which confirmed that a fatigue analysis was not required. The LRA also states that the fatigue waiver analyses depend on the number of assumed load cycles, and are therefore TLAAs.

Analysis of Fatigue Waiver for the Personnel Airlocks. The applicant stated that the fatigue waiver for the personnel airlocks applied values from the reactor containment structures specification to determine if the six criteria of ASME Code Section III NE-3222.4(d) are met. The applicant also stated that the fatigue waiver analysis demonstrated that the specified maximum allowable 1,900 startup and shutdown cycles satisfies the ASME Code NE-3222.4(d) criteria. This allowable number of cycles, however, is much higher than the assumed 120 cycles.

The applicant dispositioned the personnel airlocks TLAA in accordance with 10 CFR 54.21(c)(1)(ii) to demonstrate that the analyses have been projected to the end of the period of extended operation.

Analysis of Fatigue Waiver for the Emergency (Auxiliary) Airlocks. The applicant stated that the fatigue waiver for the emergency (auxiliary) airlocks assumed three values not supplied by the reactor containment structures specification to determine if the six criteria of ASME Code Section III NE-3222.4(d) are met. The applicant reported the following loading cycles: test temperature and pressure (10 cycles), operating temperature (300 cycles), and operating-basis earthquake (OBE) (500 cycles).

The applicant also stated that for this fatigue waiver, analyses of the emergency (auxiliary) airlocks Criteria 4 and 6 of Section NE-3222.4(d) of the ASME Code are time-dependent. The fatigue waiver analysis demonstrated that the assumed conservative operating temperature range was within the limit determined for the assumed number of cycles by ASME Code NE-3222.4(d), Criterion 4, and will remain so even if the assumed number of cycles is increased from 300 to 450 to account for the period of extended operation. The analysis also demonstrated that the stress range allowed by Criterion 6 for the expected number of mechanical cycles would not be exceeded if the assumed number of cycles were increased from 500 to 750 to account for the period of extended operation.

The applicant dispositioned the emergency (auxiliary) airlocks TLAA in accordance with 10 CFR 54.21(c)(1)(ii) to demonstrate that the analyses have been projected to the end of the period of extended operation.

4.6.2.2 Staff Evaluation

Fatigue Waiver for Personnel Airlocks. The staff reviewed LRA Section 4.6.1 regarding the fatigue waiver of personnel airlocks TLAA to confirm pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis has been projected to the end of the period of extended operation.

The staff reviewed the applicant's TLAA and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.6.3.1.1.2, which state that the operating transient experience and the increased number of assumed cyclic loads projected to the end of the period of extended operation are to be reviewed to ensure that the cyclic load projection is adequate. The SRP-LR also states that, for the re-evaluation, either the code of record remains the same or the applicant may update it to a later edition pursuant to the requirements in 10 CFR 50.55a.

For the personnel airlocks, the staff reviewed the LRA, the applicant's UFSAR, and the applicant-provided vendor information. The staff confirmed in UFSAR Section 3.8.2.2.2 that the applicant's code of record is the ASME Code Section III, Division I, 1974 edition, including winter 1975 addenda. The staff also reviewed UFSAR Section 3.8.2.1, item 3, which states that the personnel airlocks' penetrations are designed to accommodate thermal and mechanical stresses encountered in normal and other modes of operation and testing. The staff confirmed the applicant's claim that temperature, pressure, and OBE are the loading conditions for which the personnel airlocks need to be evaluated. The staff noted that UFSAR Sections 3.8.1.1.6 and 3.8.1.6.4.1 state that personnel airlocks' penetrations are part of the containment pressure boundary and are double door welded-steel assemblies made of SA-516 Grade 70 or SA-537 Class 1 steel per ASME Code Section III, Division 1, Subsection NE, Class MC component criteria. The staff also noted that UFSAR Section 3.8.2 states that the personnel airlocks are tested and receive a nameplate with an N symbol stamp, which indicates their conformance to the ASME Code.

The staff reviewed UFSAR Sections 3.8.2.2.2 and 3.8.2.4 and confirmed that the Class MC items and components (i.e., airlocks) are analyzed and designed in accordance with the applicable requirements of Section NE-3131(d) of the ASME Code Section III, Division I, 1974 edition. Section NE-3131(d), however, requires further evaluation per Section NE-3222.4(d), of the Code. The staff reviewed the requirements of ASME Code Section NE-3222.4(d), "Analysis for Cyclic Operation, (d) Vessels Not Requiring Analysis for Cyclic Operation" and noted that the following operating conditions must be analyzed for a fatigue waiver:

- atmospheric-to-operating pressure cycles
- normal operation pressure fluctuation
- temperature difference—startup and shutdown
- temperature difference—normal operation
- temperature difference—dissimilar materials
- mechanical loads

The staff also noted that UFSAR Section 3.8.2.4 discusses the analysis and design of the personnel airlocks performed by a selected vendor. The referenced calculations are not included in the applicant's UFSAR. However, per 10 CFR 54.3, Criterion 6, and Section 4.1 of the SRP-LR, these calculations and analyses are part of the TLAA acceptance criteria; therefore, the analyses are incorporated by reference in the CLB. The staff audited applicant-provided code of record reference (vendor) calculations and noted that the TLAA has been addressed. The staff noted that UFSAR Section 3.9.1.1.6.10 addresses the applicant's assumed atmospheric-to-operating pressure cycles to be 80, for the life of the plant initially set at 40 years. The staff noted that LRA Section 4.6.2 states that that one thermal cycle occurs during each refueling operation; hence, there are 80 thermal cycles for 40 years or 120 for 60 years, which includes the period of extended operation.

The staff independently performed confirmatory calculations for SA-516 Grade 70 steel (note: SA-537 Class 1 steel has higher minimum tensile and yield strengths) used in the fabrication of the personnel airlocks per UFSAR Section 3.8.1.1.3. This section states "[a]n increased plate thickness up to 2 in. is provided around all penetrations." The staff confirmed the validity of the applicant's claim for the maximum code allowable 1,900 startup and shutdown cycles. The airlock meets the following applicable conditions identified by the NE-3222.4(d) of the ASME Code:

- Atmospheric-to-Operating Pressure Cycle—Three times the design stress intensity (S_m) value for a ferrous material (SA-516 Grade 70) at operating temperatures corresponds to an allowable stress value (Sa) of 69.9 ksi, which yields 1,900 cycles from the fatigue curve of Figure I-9.0 of the ASME Code.
- Normal Operation Pressure Fluctuation—Maintaining the limit of 1,900 cycles corresponds to an allowable stress intensity (S_a) of 69.9 ksi, which yields the calculated design pressure of 56.5 psig discussed in UFSAR Section 2.5.4.10.4.1.5 and UFSAR Table 6.2.1.1-3.
- Temperature Difference—Startup and Shutdown—The temperature difference between any two adjacent points of the containment boundary for the limit of 1,900 cycles is below the roughly 190 °F temperature difference at which fatigue would become noteworthy. In accordance with UFSAR Table 6.2.1.1-3, temperature difference during operation does not exceed 114 °F.
- Temperature Difference—Normal Operation—It remains within the bounds of difference between the design temperature of 286 °F and the operating temperature of 114 °F per UFSAR Table 6.2.1.1-3. Specifically, UFSAR Section 3.9.1.1.6.1 specifies 400 heatup-cooldown operations over 40 years. The LRA redefines these in Table 4.3-2 for 60 years to be 171 and 154 cycles for Unit 1 and Unit 2, respectively. Considering the average of the operating and design temperatures to be about 200 °F, this condition yields for about 171 cycles an S_a of 175 ksi and an allowable temperature difference for normal operation of about 450 °F, which is above the 172 °F temperature difference of operating and design temperatures.
- Temperature Difference—Dissimilar Materials—The staff further reviewed the UFSAR for dissimilar materials that may have been used in the fabrication of the personnel airlocks and found none. The staff also noted that this evaluation is in accordance with the audited applicant's vendor provided calculations.
- Mechanical loads—These were determined to be not applicable per staff review of applicant-provided vendor information called for by the code of record referenced in the assessment.

Based on the above, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses for the personnel airlocks have been projected to the end of the period of extended operation.

Fatigue Waiver for Emergency (Auxiliary) Airlocks. The staff reviewed LRA Section 4.6.1 regarding the fatigue waiver of the emergency (auxiliary) airlocks TLAA to confirm pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis has been projected to the end of the period of extended operation.

The staff reviewed the applicant's TLAA and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.6.3.1.1.2, which state that the operating transients experienced and the increased number of assumed cyclic loads projected to the end of the period of extended operation are to be reviewed to ensure that the cyclic load projection is adequate. The SRP-LR also states that for the re-evaluation, either the code of record remains the same or the applicant may update it to a later edition pursuant to the requirements in 10 CFR 50.55a.

For the emergency (auxiliary) airlocks, the staff reviewed the LRA, the applicant's UFSAR, and vendor information provided by the applicant. The staff noted in UFSAR Section 3.8.1.6.4.1 that the emergency (auxiliary) airlocks are also made of SA-516 Grade 70 steel or SA-537, ASME Code Section III, Division 1, Class 1 steel. The staff independently performed confirmatory calculations for a plate thickness of 2 inches made with the lower tensile and yield strength SA-516 Grade 70 steel.

The staff noted that conditions 1, 2 and 3 of NE-3222.4(d) of the ASME Code, as discussed in the section above, "Fatigue Waiver for the Personnel Airlocks," are equally applicable to the emergency (auxiliary) airlocks because the two airlock types (personnel and the emergency/auxiliary) are addressed and referenced within the same UFSAR sections. Therefore, the calculations for the atmospheric-to-operating pressure cycle, normal operation pressure fluctuation, and temperature difference—startup and shutdown are the same for the emergency (auxiliary) airlocks. The staff further noted that the design cycles for test transients are limited to 10 cycles and are independent of any other transients (e.g., see UFSAR Sections 3.9.1.1.10.1 and 3.9.1.1.10.2). The staff also noted that LRA Table 4.3-2, "STP Units 1 and 2 Transient Cycle Count 60-year Projections," for test conditions, limits the transients to one for each unit.

The emergency (auxiliary) airlock meets the remaining conditions identified by NE-3222.4(d) of the ASME Code, as follows:

- Temperature Difference: Normal Operation—An assumed number of 300 cycles, when increased by 1.5 times to 450 cycles and considering the average of the operating and design temperatures to be about 200 °F, yields an S_a of 110 ksi and an allowable temperature difference of about 285 °F, which is greater than the 172 °F difference of operating and design temperatures.
- Temperature Difference: Dissimilar Materials—The staff reviewed the UFSAR for dissimilar materials that may have been used in the fabrication of the emergency (auxiliary) airlocks and found none.
- Mechanical Loads: LRA Section 4.6.1 lists 500 cycles for OBE, which is far in excess of those reported in UFSAR 3.7.3A.2. The UFSAR defines the total number of earthquake cycles for the design of seismic Category 1 SSCs to be 10 for safe shutdown

earthquakes (SSEs) (one event) and 50 for OBEs (five events). These values are in accordance with LRA Table 4.3-2. As noted previously, UFSAR Section 3.8.2.4 discusses the analysis and design of the personnel airlocks performed by a selected vendor. The referenced calculations are not included in the applicant's UFSAR. However, per 10 CFR 54.3, Criterion 6, and Section 4.1 of the SRP-LR, these calculations and analyses are part of the TLAA acceptance criteria; therefore, the analyses are incorporated by reference in the CLB. The staff audited the applicant-provided code of record referenced (vendor) calculations and noted that the TLAA has been addressed and that the 500 cycles listed in the LRA are due to a range of mechanical loads that include earthquake loading. The staff then independently performed confirmatory calculations, increasing the 500 cycles by 1.5 times to 750 cycles, which yielded an S_a range of 95 ksi. This is higher than the maximum allowable factored overload stress (38 ksi times 1.33) if indeed all the stresses and all the cycles were due to seismic loads, and there were no potential crack initiators. The staff also noted that the applicant-provided vendor information called for by the code of record has a calculated stress intensity of 60 ksi.

Based on the above, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses for the emergency (auxiliary) airlock have been projected to the end of the period of extended operation.

4.6.2.3 UFSAR Supplement

LRA Section A3.5.1 provides the UFSAR supplement summarizing the personnel and emergency airlock fatigue waiver analysis per the code of record and design calculations and documents called for by that code of record. The staff reviewed LRA Section A3.5.1, consistent with the review procedures in the SRP-LR. SRP-LR Section 4.1.3 states that if a code of record is in the UFSAR for particular group of structures or components, reference material includes all calculations called for by that code of record for those structures and components. SRP-LR Section 4.6.3.1.1.2 states that the operating transients experienced and the increased number of assumed cyclic loads projected to the end of the period of extended operation are to be reviewed to ensure that the cyclic load projection is adequate and that the fatigue waiver will remain valid for the period of extended operation.

Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Sections 4.1.3 and 4.6.3.1.1.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the fatigue waivers for the personnel airlocks and emergency (auxiliary) airlocks, as required by 10 CFR 54.21(d).

4.6.2.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses for the fatigue waivers for the personnel airlocks and emergency airlocks have been projected to the end of the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.6.3 Fatigue of Containment Penetrations

4.6.3.1 Summary of Technical Information in the Application

Containment Penetrations Other Than Fuel Transfer Bellows. LRA Section 4.6.2 states that a thorough search of the licensing basis and design documents identified all containment penetrations whose design is supported by a fatigue or cyclic load analysis. The LRA further states that these analyses are TLAAs.

The LRA states that the applicant evaluated the criteria in ASME Code Section III NC-3219.2(a) for fatigue analyses of penetrations. The LRA also states:

> The calculation determined that fatigue analyses are necessary for main steam (M-1 through M-4), feedwater (M-5 through M-8), auxiliary feedwater (M-83, M-84, M-94, and M-95), and steam generator blowdown (M-62 through M-65) penetrations. Further examination of the design reports and calculations for each penetration type identified an additional fatigue analysis for sample line penetrations M-85 and M-86. [LRA] Table 4.6-1 summarizes the result of this document review. The penetration fatigue analyses were calculated in accordance with ASME Code Section NC-3200…
>
> The fatigue analyses of the containment penetration pressure boundaries are dependent on the assumed 40-year number of transient cycles. The Metal Fatigue of Reactor Coolant Pressure Boundary program described in LRA Sections 4.3.1 and B3.1 ensures that the numbers of transients actually experienced during the period of extended operation remain below the assumed number; or that appropriate corrective actions maintain the design and licensing basis by other acceptable means. The effects of fatigue will therefore be managed for the period of extended operation.

The applicant dispositioned the TLAA for containment penetrations in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

Fuel Transfer Tube Bellows. The applicant stated that the fuel transfer tube penetration connects the refueling canal (inside the reactor containment building (RCB)) to the spent fuel pool (inside the fuel handling building) and consists of a stainless steel pipe inside of a carbon steel sleeve. The applicant further stated:

> Stainless steel casing pipes with expansion bellows are welded to both ends of the sleeve. These bellows allow differential movement between the buildings on the outside of the containment wall and between the containment liner and the refueling cavity concrete on the inside. The casing pipe and the bellows in the fuel handling building perform a leakage boundary intended function and are within the scope of license renewal. The applicant further stated that the casing pipe and the bellows inside the containment building are part of the containment pressure boundary and are within the scope of license renewal with a structural pressure boundary intended function. Each of these bellows is designed for 1,000 cycles of expansion and contraction; therefore, these design analyses are TLAAs requiring evaluation for the period of extended operation.
>
> In order to determine if the design analyses remain valid for 60 years of operation, the number of cycles for 60 years has been conservatively projected. For each of these

> components, one thermal cycle occurs during each refueling operation. The design number of refueling operations is 80 cycles (120 cycles when multiplied by 1.5 for 60 years). In addition to these cycles, the fuel transfer canal penetration assembly is exposed to pressurization cycles during integrated leak rate tests [ILRTs], conservatively projected to occur once every 5 years. This contributes 12 cycles in 60 years. These penetrations would also be exposed to up to one Safe Shutdown Earthquake [SSE] cycle. Therefore, the total cycles projected for 60 years are a fraction of the design cycles analyzed for these bellows.

The applicant dispositioned the fuel transfer tube bellows as a TLAA in accordance with 10 CFR 10 CFR 54.21(c)(1)(i) to demonstrate that the analysis remains valid for the period of extended operation.

4.6.3.2 Staff Evaluation

Containment Penetrations Other Than Fuel Transfer Bellows. The staff reviewed LRA Section 4.6.2 regarding the fatigue design of the containment penetrations TLAA to confirm, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

The staff reviewed the applicant's TLAA and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.6.3.1.1.3, which state that the applicant's proposed AMP to ensure that the effects of aging on the intended function(s) of the penetrations are adequately managed for the period of extended operation, is reviewed.

For the containment penetrations, the staff reviewed the applicant's disposition of the TLAA in the LRA, which states that the fatigue analyses of the containment penetration pressure boundaries are dependent on the assumed 40-year number of transient cycles and are based on the existing "Metal Fatigue of Reactor Coolant Pressure Boundary" program (evaluated in SER Section 3.0.3.2.28). This program, when enhanced, will be consistent with GALL Report AMP X.M1. The staff noted that the program, as amended by the applicant in a letter dated January 26, 2012, is an existing program. This program ensures that the number of transients experienced during the period of extended operation remains below their design cycles or that appropriate corrective actions are taken that may include repair, replacement, or more rigorous analyses of the pressure boundary containment components. The staff also noted that for the containment penetration assemblies, the program manages fatigue based on one of its two available methods: (1) the cycle-counting or (2) the CBF management method. All penetrations are monitored using the cycle-counting method, except containment penetrations M-62 through M-65, listed in LRA Table 4.3-1, "Summary of CBF Monitored Locations in the STP Fatigue Management." These four penetrations are CBF-monitored and managed to ensure that the CUF remains below the ASME Code allowable fatigue limit of 1.0. UFSAR Table 3.9-8 contains transients that are also tabulated in LRA Table 4.3-2, "STP Units 1 and 2 Transient Cycle Count 60-year Projections." The LRA states that the most limiting number of cycles for each transient is used as the limiting values for the program.

The staff reviewed LRA Table 4.6-1, "Containment Penetration Assemblies," and noted that the 40-year CUFs for M-1 through M-8, M-62 through M-65, M-83 through M-86, and M-94 and M-95 are all less than 1.0. The staff also reviewed LRA Table 4.3-2 and noted the conservatism involved when comparing for each specific transient the design cycles (UFSAR design), the actual cycles to the year 2008 (baseline events), and the 60 years of operation cycles projected (projected events). The projections provided in LRA Table 4.3-2, demonstrate that the 40-year design basis numbers of events are sufficient for 60 years. The staff multiplied the 40-year CUF based on the assumed number of transients by 1.5 to obtain a 60-year projected CUF. The 1.5 multiplier is based on the linear increase of the total projected number of cycles from 40 to 60 years. The calculated CUF for all listed penetrations were less than 1.0. For the feedwater penetrations M-5 through M-8, seismic anchor movement, Condition A of ASME Code Section III, Division 1, NC-3219.2, states that fatigue analysis is not mandatory for materials having a specified minimum tensile strength not exceeding 80 ksi, when the total expected number of cycles is less than 1,000. The staff further noted that UFSAR Section 3.7.3A.2 defines the total number of earthquake cycles for the design of seismic Category 1 SSCs to be 10 for SSEs (one event) and 50 for OBEs (five events).

The staff then reviewed the applicant's response to RAIs 4.3.2.4-2, 4.3.2.4-6, and 4.3.2.11-3, which further describe the methodology that the applicant's procedures will follow in screening the experienced transients. These RAIs are discussed in SER Section 4.3, "Fatigue of ASME Code Class 1 Components." The applicant's procedures require the control room to complete daily screening data sheets and identify if there were any transients. If a transient occurs, a transient-specific datasheet is completed to record the plant's conditions during the event. The applicant will assess these by interpreting the collected data, and identifying the transients of importance through a software application for the period of extended operation. At least once per refueling cycle, the information will be validated to ensure that an accurate transient count exists and that the actual transient severity remains within the design basis. The procedures indicate that the cycle counts are compared to the action limits specified in the procedures, and corrective actions are initiated when a transient exceeds 80 percent of its design limit.

Based on the above, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions of the containment penetrations will be adequately managed for the period of extended operation.

Additionally, the analysis meets the acceptance criteria in SRP-LR Section 4.6.3.1.1.3. The program will be enhanced prior to the period of extended operation, as indicated in the amended AMP with its UFSAR supplement A2.1, "Metal Fatigue of Reactor Coolant Pressure Boundary." This supplement was updated by the applicant in a letter dated November 21, 2011, which states:

> [t]he program ensures that actual plant experience remains bounded by the transients assumed in the design calculations, or that appropriate corrective actions maintain the design and licensing basis by other acceptable means. If a cycle count or CUF value increases to a program action limit, corrective actions include fatigue reanalysis, repair, or replacement… Action limits permit completion of corrective actions before the design basis number of events is exceeded.

Based on this information, the program will ensure that the effects of aging on the containment penetrations intended function(s) will be adequately managed for the period of extended operation.

<u>Fuel Transfer Tube Bellows</u>. The staff reviewed LRA Section 4.6.2, "Fuel Transfer Tube Bellows," regarding the fatigue design of the containment fuel transfer tube bellows TLAA to confirm, pursuant to 10 CFR 54.21(c)(1)(i), that the analysis remains valid for the period of extended operation.

The staff reviewed the applicant's TLAA and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.6.3.1.1.1, which state that the number of assumed transients used in the existing CUF calculations for the current operating term is compared to the extrapolation to 60 years of operation of the number of operating transients experienced to date. The comparison confirms that the number of transients in the existing analyses will not be exceeded during the period of extended operation.

The staff reviewed LRA Section 4.6.2 for the fuel transfer tube bellows and noted that the bellows were designed for 1,000 cycles of expansion and contraction. The applicant also stated that a thermal cycle occurs during each refueling operation. The design number of refueling operations is 80 cycles for 40 years or 120 cycles for 60 years of operation. In addition to these cycles, the fuel transfer canal penetration assembly is exposed to pressurization cycles during ILRTs, very conservatively projected to occur once every 5 years. This contributes 12 cycles in 60 years. These penetrations would also be exposed to up to one SSE cycle.

The staff confirmed that the fuel transfer tube bellows were designed for 1,000 cycles when it audited the applicant's vendor records. For those cycles, the bellows combined stress to failure was extracted from a best-fit curve of meridional stress value versus cycle life based on fatigue test data of series of bellows. The staff noted that the total number of cycles to be experienced by the bellows are far less than their design cycles. The staff also noted that UFSAR Section 3.8.1.1.6 identifies the assembly of transfer tube and bellows to consist of a stainless steel pipe inside a carbon steel sleeve, where the inner pipe acts as a transfer tube with the outer tube welded to the containment liner. Bellows expansion joints are provided to permit differential movement. The staff further noted that NUREG/CR-6726, "Aging Management and Performance of Stainless Steel Bellows in Nuclear Power Plants," in its "Operating Experience from Nuclear Plant Reliability Data System Data" subchapter, states that fuel transfer tube bellows failures have not occurred in the bellows but on their gasket subcomponents. Because of such recorded failures, and even though the bellows are designed in excess of the anticipated thermal, refueling, pressurization, and earthquake based cycles, the applicant, in its response to RAI 3.5.2.2.1.7-1, by letter dated November 21, 2011, revised the LRA (see SER Section 3.5.2.2.1, item 7). Accordingly, the applicant instituted a bellows inspection, based on its ASME Code Section XI, Subsection IWE and 10 CFR Part 50, Appendix J programs to manage any potential aging effects.

Based on the above, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the analysis for the fuel transfer tube bellows remains valid for the period of extended operation.

Additionally, the analysis for the fuel transfer tube bellows meets the acceptance criteria in SRP-LR Section 4.6.3.1.1.1 because the bellows have been designed based on actual tests to sustain far more cycles of operation than a projected number; therefore, the analysis is valid for the period of extended operation.

4.6.3.3 UFSAR Supplement

LRA Section A3.5.2 provides the UFSAR supplement summarizing the fatigue design of containment penetrations that includes the fuel transfer tube bellows. The staff reviewed LRA Section A3.5.2, "Fatigue Design of Containment Penetrations," consistent with the review procedures in SRP-LR Section 4.6.3.1.1.3. These procedures state that, for the fatigue of the containment penetrations (other than the fuel transfer bellows), the applicant's proposed AMP needs to be reviewed on a case-by-case basis to ensure that the effects of aging on the intended function(s) of the components are adequately managed for the period of extended operation. For the case of the fuel transfer tubes fatigue, SRP-LR Section 4.6.3.1.1.1 states that the number of assumed transients used in the existing CUF calculations for the current operating term has been compared to the extrapolation to 60 years of operation of the number of operating transients experienced to date. The comparison confirmed that the number of transients in the existing analyses would not be exceeded during the period of extended operation.

Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.6.3.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the TLAA for the fatigue design of the containment penetrations, as required by 10 CFR 54.21(d).

4.6.3.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the TLAA for the fuel transfer tube bellows remains valid for the period of extended operation. The staff also finds, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions of the containment penetrations will be adequately managed for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7 Other Plant-Specific Time-Limited Aging Analyses

4.7.1 Load Cycle Limits of Cranes, Lifts, and Fuel Handling Equipment Designed to CMAA-70

4.7.1.1 Summary of Technical Information in the Application

LRA Section 4.7.1 describes the applicant's TLAA for load cycle limits of cranes, lifts, and fuel handling equipment designed to CMAA-70. LRA Table 4.7-1, as revised by letter dated October 10, 2011, summarizes the estimated maximum number of significant crane lifts (i.e., lifts that approach or equal the crane design load) for each machine or system. The applicant stated that the number of significant lifts for each machine per RFO is estimated from the UFSAR Section 9.1.4.2.2 description of refueling operations. The estimated number of lifts is multiplied by a factor of 1.5 to account for non-refueling lifts. The applicant further stated that based on an 18-month refueling cycle, approximately 27 refuel cycles are expected over a 40-year plant design life, or about 40 refuel cycles in a 60-year design life.

New Fuel Handling Area Overhead Crane. The applicant stated that the new fuel handling area overhead crane is designed to handle fuel assemblies and their shipping containers in the new fuel handling area.

Cask Handling Overhead Crane. The applicant stated that the cask handling overhead crane is designed to three primary operations: (1) transfer the spent fuel cask from the bed of the transport vehicle to the cask decontamination area, (2) lower the spent fuel cask into the dry cask handling system transporter tank following inspection or walkdown, and (3) return the spent fuel cask to the transport vehicle following fuel loading operations.

Fuel Handling Building Overhead Crane. The applicant stated that the fuel handling building overhead crane is designed to five primary operations: (1) transfer the new fuel shipping containers from the transport vehicle to the new fuel handling area, (2) transfer the new fuel assemblies from the new fuel handling area to the new fuel storage area or to the new fuel elevator, (3) transfer the spent fuel shipping cask head from the cask to its storage shelf in the cask loading pool, and to lower the head onto the cask, (4) replace the safety injection and containment spray pumps, and (5) perform general service and maintenance operations as required.

Containment Polar Crane. The applicant stated that the containment polar crane is evaluated to refueling and fuel handing operations. It is also used for construction, maintenance, and repair operations as needed. The applicant also stated that this crane is classified as non-nuclear safety class since it neither provides nor supports any system safety function.

Refueling Machine. The applicant stated that the refueling machine is designed to transfer fuel from one location to another.

Fuel Handling Machine. The applicant stated that the fuel handling machine is designed to handle fuel assemblies and core components in the spent fuel pool by means of handling tools suspended from the hoist. The applicant also stated that the fuel handling machine has a two-step magnetic control for the bridge and hoist.

New Fuel Elevator. The applicant stated that the new fuel elevator is designed to lower a new fuel assembly into the fuel transfer canal and can be used to raise a new or spent fuel assembly.

Fuel Transfer System. The applicant stated that the fuel transfer system is designed to transfer fuel between the RCB and the fuel handling building. The applicant also stated that a hydraulically actuated lifting arm (upender) at each end of the transfer tube is used to take the fuel from a vertical position to a horizontal position to pass through the transfer tube and then back into the vertical position for placement.

Disposition. The applicant dispositioned the load cycle limits of cranes, lifts, and fuel handling equipment designed to CMAA-70 TLAA in accordance with 10 CFR 54.21(c)(1)(i) to demonstrate that the analyses remain valid for the period of extended operation.

4.7.1.2 Staff Evaluation

The staff reviewed LRA Section 4.7.1 and the load cycle limits of cranes, lifts, and fuel handling equipment designed to CMAA-70 TLAA to confirm, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses remain valid for the period of extended operation.

The staff reviewed the applicant's TLAA and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.7.3.1.1, which state that the existing analyses should be shown to be bounding even during the period of extended operation. The SRP-LR also states that the applicant describes the TLAA with respect to the objectives of the analysis; assumptions used in the analysis; and conditions, acceptance criteria, relevant aging effects, and intended functions. The applicant shows that conditions and assumptions used in the analysis already address the relevant aging effects for the period of extended operation, and acceptance criteria are maintained to provide assurance that the intended functions are maintained for renewal.

New Fuel Handling Area Overhead Crane. The staff reviewed LRA Section 4.7.1, UFSAR Section 9.1.4.2, and UFSAR Table 3.2.A-1 and found that the new fuel handling area overhead crane is a 5-ton crane designed to CMAA-70, Class A1. LRA Table 4.7-1 indicates that the new fuel handling area overhead crane is designed to 100,000 lifts (or load cycles).

The estimated maximum number of significant crane lifts for the new fuel handling area overhead crane projected for 40 years, based on 27 RFOs, was 5,346. The estimated maximum number of significant crane lifts projected for 60 years, based on 40 RFOs, is 8,019. This number of lifts is significantly less than the 100,000 allowable design cycles and, therefore, is acceptable.

Cask Handling Overhead Crane. The staff reviewed LRA Section 4.7.1, UFSAR Section 9.1.4.2, and UFSAR Table 3.2.A-1 and found that the cask handling overhead crane is a 150-ton crane designed to CMAA-70, Class A1. LRA Table 4.7-1 indicates that the cask handling overhead crane is designed to 100,000 lifts (or load cycles).

LRA Table 4.7-1 shows the estimated maximum number of significant crane lifts for the cask handling overhead crane to be 420 for 40 years based on 10 refuels and 740 for 60 years based on 20 refuels. It is unclear to the staff how these numbers were calculated and why the calculations were based on 10 refuels and 20 refuels for the 40-year and 60-year cycles, respectively. Therefore, in a letter dated August 15, 2011, the staff issued RAI 4.7.1-1, requesting that the applicant provide the basis for the estimated number of significant crane lifts for both a 40- and 60-year design life. The staff also asked the applicant to explain why the number of refuel cycles being used in this calculation differs from 27 refuel cycles expected over a 40-year design life and 40 refuel cycles expected over a 60-year design life based on an 18-month refuel cycle, as stated in LRA Section 4.7.1.

In its response dated October 10, 2011, the applicant stated that the number of lifts for the cask handling overhead crane is based on three lifts per cask and seven casks per RFO, which equals 21 lifts per unit per RFO. In addition to the RFO lifts, the 40-year and 60-year cycles include an estimated 100 construction lifts.

In the applicant’s response regarding the number of refuels used in the cask handling overhead crane calculation, the applicant stated that the number of RFOs differs because cask loading is assumed to begin in year 30 of plant operation. The staff determined that additional explanation was needed; therefore, the staff participated in a teleconference with the applicant on November 17, 2011, to clarify the response. Based on the clarification call, the applicant agreed to revise its response to RAI 4.7.1-1.

In its revised response dated December 7, 2011, the applicant clarified that once spent fuel cask loading begins, the number of fuel assemblies moved to dry cask storage is equal to the number of new fuel assemblies received each RFO. Therefore, the number of casks loaded and, hence, the number of cask handling crane lifts is dependent on the number of RFOs. The applicant stated that the calculated number for each outage was multiplied by 1.5 for conservatism, resulting in the estimated 32 significant lifts per RFO.

The applicant further clarified that the number of RFOs assumed in the lift cycle estimate for the cask handling overhead crane differs from the 27 refueling cycles expected over a 40-year design life and the 40 refueling lifts (or load cycles) expected over a 60-year design life assumed in the estimate for the other cranes because cask loading is assumed to begin in year 30 of plant operation. The applicant also clarified that the 40-year estimate was based on a rounded up number of 10 RFOs, from the actual 6.67 RFOs, to simplify the calculation.

The staff finds the applicant's response acceptable because the estimated maximum number of significant crane lifts for the cask handling overhead crane does not exceed the design lifts for the crane. The estimated maximum number of significant crane lifts for the cask handling overhead crane projected for 40 years, based on 10 RFOs, was 420. The estimated maximum number of significant crane lifts projected for 60 years, based on 20 RFOs, is 740. This number of lifts (or load cycles) is significantly less than the 100,000 allowable design cycles; therefore, it is acceptable. The staff's concern discussed in RAI 4.7.1-1 is resolved.

<u>Fuel Handling Building Overhead Crane</u>. The staff reviewed LRA Section 4.7.1, UFSAR Sections 9.1.4.2 and 9.1.4.3, and UFSAR Table 3.2.A-1 and found that the fuel handling building overhead crane is a 15/2-ton (15-ton main hook and 2-ton auxiliary hook) crane designed to CMAA-70, Class A1. LRA Table 4.7-1 indicates that the fuel handling building overhead crane is designed to 100,000 cycles.

The estimated maximum number of significant crane lifts for the fuel handling building overhead crane projected for 40 years, based on 27 RFOs, was 12,636. The estimated maximum number of significant crane lifts projected for 60 years, based on 40 RFOs, is 18,954. This number of lifts (or load cycles) is significantly less than the 100,000 allowable design cycles; therefore, it is acceptable.

<u>Containment Polar Crane</u>. The staff reviewed LRA Section 4.7.1, UFSAR Section 9.1.4.2, and UFSAR Table 3.2.A-1 and found that the containment polar crane is a 310/15-ton (310-ton main hook and 15-ton auxiliary hook) crane designed to CMAA-70. The LRA Table 4.7-1 indicates that the containment polar crane is designed to 200,000 cycles.

LRA Table 4.7-1 shows the estimated maximum number of significant crane lifts for the containment polar crane to be 2,411 for 40 years, and 3,542 for 60 years based on an 18-month refuel cycle. It is unclear to the staff how these numbers were calculated; therefore, by letter dated August 15, 2011, the staff issued RAI 4.7.1-2, requesting that the applicant show how the estimated maximum number of significant crane lifts for the 40-year and 60-year cycles were calculated, based on the estimated 54 lifts per refuel.

In its response dated October 10, 2011, the applicant stated that the number of lifts for the polar crane is based on the following refueling lifts: reactor head (2 lifts per refueling), reactor upper internals (2 lifts per refueling), and maintenance and repair operations (50 lifts per refueling). The 40-year and 60-year estimates also include 9 and 13 lower internals lifts, respectively (once every three refuelings), and an additional 150 construction lifts. The applicant further stated that, while reviewing this RAI, a calculation error was found in LRA Table 4.7-1 for the number of polar crane lifts. This correction does not change the disposition of the crane TLAA evaluation.

The staff finds the applicant's response acceptable because the estimated maximum number of significant crane lifts for the polar crane does not exceed the design lifts for the crane. The estimated maximum number of significant crane lifts projected for 40 years, based on 27 RFOs, was updated to 2,355, and the estimated maximum number of significant crane lifts projected for 60 years, based on 40 RFOs, to 3,416. These lifts (or load cycles) are less than those reported in the LRA and significantly less than the 200,000 permissible cycles; therefore, it is acceptable. The staff's concern discussed in RAI 4.7.1-2 is resolved.

Refueling Machine. The staff reviewed LRA Section 4.7.1, UFSAR Sections 9.1.4.2 and 9.1.4.3, and UFSAR Table 3.2.A-1 and found that the refueling machine is a rectilinear bridge and trolley crane with a vertical mast extending down into the refueling cavity. In general, the crane structure is considered in the Class A1, "Standby Service," as defined by CMAA-70. LRA Table 4.7-1 indicates that the refueling machine is designed to 100,000 cycles.

The estimated maximum number of significant crane lifts for the new fuel handling area overhead crane projected for 40 years, based on 27 RFOs, was 17,658. The estimated maximum number of significant crane lifts projected for 60 years, based on 40 RFOs, is 26,487. This number of lifts (or load cycles) is significantly less than the 100,000 allowable design cycles; therefore, it is acceptable.

Fuel Handling Machine. The staff reviewed LRA Section 4.7.1, UFSAR Sections 9.1.4.2 and 9.1.4.3, and UFSAR Table 3.2.A-1 and found that the fuel handling machine consists of an electric monorail hoist carried on a wheel-mounted bridge. In general, the crane structure is considered in the Class A1, "Standby Service," as defined by CMAA-70. LRA Table 4.7-1 indicates that the fuel handling machine is designed to 100,000 cycles.

The estimated maximum number of significant crane lifts for the new fuel handling area overhead crane projected for 40 years, based on 27 RFOs, was 30,186. The estimated maximum number of significant crane lifts projected for 60 years, based on 40 RFOs, is 45,279. This number of lifts (or load cycles) is less than the 100,000 allowable design cycles; therefore, it is acceptable.

New Fuel Elevator. The staff reviewed LRA Section 4.7.1 and UFSAR Section 9.1.4.2 and found that the new fuel elevator consists of a box-shaped elevator assembly with its top end open, designed to meet the requirements of CMAA-70. LRA Table 4.7-1 indicates that the new fuel elevator is designed to 100,000 cycles.

The estimated maximum number of significant crane lifts for the new fuel handling area overhead crane projected for 40 years, based on 27 RFOs, was 2,673. The estimated maximum number of significant crane lifts projected for 60 years, based on 40 RFOs, is 4,010. This number of lifts (or load cycles) is less than the 100,000 allowable design cycles; therefore, it is acceptable.

Fuel Transfer System. The staff reviewed LRA Section 4.7.1 and UFSAR Section 9.1.4.2 and found that the fuel transfer system designed to CMAA-70 includes an underwater, electric-motor-driven transfer car that runs on tracks extending from the refueling canal in the RCB, through the fuel transfer tube, and into the fuel transfer canal in the FHB. LRA Table 4.7-1 indicates that the fuel handling machine is designed to 100,000 cycles.

The estimated maximum number of significant load cycles for the fuel transfer system projected for 40 years, based on 27 RFOs, was 17,658. The estimated maximum number of significant load cycles projected for 60 years, based on 40 RFOs, is 26,487. This number of load cycles is less than the 100,000 allowable design cycles; therefore, it is acceptable.

Summary. Based on its review, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses for the load cycle limits of cranes, lifts, and fuel handling equipment designed to CMAA-70 remain valid for the period of extended operation.

Additionally, the analyses for the load cycle limits of cranes, lifts, and fuel handling equipment meet the acceptance criteria in SRP-LR Section 4.7.2 because the applicant demonstrated that the analyses for load cycle limits of cranes, lifts, and fuel handling equipment designed to CMAA-70 remain valid for the period of extended operation pursuant to 10 CFR 54.21(c)(i).

4.7.1.3 UFSAR Supplement

LRA Section A3.6.1 provides the UFSAR supplement summarizing the load cycle limits of cranes, lifts, and fuel handling equipment designed to CMAA-70. The staff reviewed LRA Section A3.6.1, consistent with the review procedures in SRP-LR Section 4.7.3.2, which states that the applicant provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of each TLAA. SRP-LR Section 4.7.3.2 also states that each summary description is reviewed to confirm that it is appropriate, such that later changes can be controlled by 10 CFR 50.59 and that the description should contain information that the TLAAs have been dispositioned for the period of extended operation.

Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.7.3.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the TLAA for load cycle limits of cranes, lifts, and fuel handling equipment designed to CMAA-70, as required by 10 CFR 54.21(d).

4.7.1.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses for the load cycle limits of cranes, lifts, and fuel handling equipment designed to CMAA-70 remain valid for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.2 Inservice Flaw Growth Analyses That Demonstrate Structural Stability for 40 years

4.7.2.1 Summary of Technical Information in the Application

LRA Section 4.7.2 states that inservice flaw growth is identified in NUREG-1800 as a potential TLAA. The applicant searched the CLB and did not identify any flaws evaluated for the remaining life of the plant other than those discussed elsewhere in the LRA, such as the flaw growth analysis of the half-nozzle repair on the Unit 1 BMI nozzles (this is a TLAA, which will remain valid for the period of extended operation and is dispositioned in accordance with 10 CFR 54.21(c)(1)(i), as discussed in LRA Section 4.3.2.1) and the pressurizer SWOL repairs and mitigations performed on Unit 1 and 2 pressurizer nozzles. The flaw growth analysis for the pressurizer nozzles is discussed in LRA Section 4.3.2.4. The flaw growth analysis related to the pressurizer SWOL repairs does not qualify cracks for the life of the plant but only the 10-year inspection interval. Therefore, this analysis is not a TLAA in accordance with 10 CFR 54.3(a), Criterion 3, as discussed in LRA Section 4.3.2.4.

4.7.2.2 Staff Evaluation

The staff reviewed LRA Section 4.7.2 to confirm that the TLAA for inservice flaw growth analyses will meet 10 CFR 54.21(c)(1). The staff reviewed the applicant’s TLAA and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.7.3, which states that the review of the TLAA provides assurance that the aging effect is properly addressed through the period of extended operation. The staff's review on the flaw growth analysis of the half-nozzle repair for the Unit 1 BMI nozzle is discussed in SER Section 4.3.2.1. The staff's review of the flaw growth analysis of the overlaid Alloy 82/182 welds at the pressurizer surge line nozzles is discussed in SER Section 4.3.2.4.

In RAI 4.7.2-1 (April 14, 2011), the staff asked the applicant to discuss the sources that have been searched to obtain the information on the flaw growth analyses. The staff also asked the applicant to discuss whether there are recordable indications or flaws that have remained inservice in the piping without a flaw evaluation for pipes within the scope of LRA and discuss how these flaws will be monitored to the end of 60 years. By letter dated May 12, 2011, the applicant responded that, to identify flaws in the components, it searched the UFSAR, TS, the NRC SERs for the original operating licenses, subsequent NRC SEs, and STPNOC and NRC-docketed licensing correspondence.

Based on its search and provided to the staff in its response to RAI 4.7.2-1, the applicant stated that besides the flaws discussed above, it identified a flaw of a small active leak at the top of the shell to base plate weld in the Unit 1 RWST. The applicant submitted for NRC review and approval Relief Request RR-ENG-33 to allow the flaw to remain in service for one fuel cycle in a letter dated February 22, 2000 (ADAMS Accession No. ML003686976.) The applicant determined that the fatigue flaw growth was insignificant (growth of 1 inch for 100,000 fill/drain cycles). The NRC authorized Relief Request RR-ENG-33, in letter dated June 22, 2000, to allow Unit 1 to operate with the flaw in place for one fuel cycle until the tank could be inspected (ADAMS Accession No. ML003725735).

Subsequently, the applicant inspected the RWST and found no evidence of base plate or sidewall cracking inside the tank. Based on those inspection results and a large allowable flaw length of 63.6 inches, the staff concluded that Unit 1 can continue to be operated, subject to future inspections as required by ASME Code Section XI, which will monitor the leak to the end of 60 years. The NRC's SE is documented in a letter dated December 14, 2001 (ADAMS Accession No. ML013460299).

The applicant stated that the safety evaluation of RR-ENG-33 found that the fatigue crack growth analysis for the flaw identified at the RWST is not required to be considered in the final safety determination; thus, it is not a TLAA in accordance with 10 CFR 54.3(a) Criterion 4. The staff finds that the RWST and associated flaw will be periodically inspected in accordance with ASME Code Section XI. As such, the staff finds that the flaw in the RWST does not have to be considered a TLAA, in accordance with 10 CFR 54.3(a), because any potential aging effect on the RWST will be monitored by the periodic inspections.

4.7.2.3 UFSAR Supplement

LRA Section A3.6.2 provides the UFSAR supplement summarizing description of this TLAA for the flaw growth analyses of piping in the scope of the LRA. The staff reviewed LRA Section A3.6.2, consistent with the review procedures in SRP-LR Section 4.7.3.2, which states that the staff confirms that the UFSAR supplement includes a summary description of the evaluation of each TLAA. Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.7.2.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the TLAA for the flaw growth analyses of piping in the scope of the LRA, as required by 10 CFR 54.21(d).

4.7.2.4 Conclusion

The staff's conclusion on the flaw growth analysis of the half nozzle repair on the Unit 1 BMI nozzle is discussed in SER Section 4.3.2.1. The staff's conclusion on the flaw growth analysis of the overlaid Alloy 82/182 welds at the pressurizer surge line nozzles is discussed in SER Section 4.3.2.4. The staff concludes that the flaw in the RWST is not a TLAA, in accordance with 10 CFR 54.3(a), because the RWST will be inspected periodically in accordance with the ASME Code Section XI. The inspection will monitor the flaw growth and monitor the aging effects on the RWST. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation of the flaw growth calculations of piping in the scope of the LRA, as required by 10 CFR 54.21(d).

4.7.3 TLAA for the Corrosion Effects in the Essential Cooling Water System

4.7.3.1 Summary of Technical Information in the Application

LRA Section 4.7.3 describes the applicant's analyses of corrosion rate in the ECW system. The applicant's revised response to NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," dated June 23, 1992, stated that the corrosion rate in the ECW system was 0.6 mil/year, which would result in a wall thickness loss less than the design limit of 40 mils during the 40 years of plant operation.

The applicant dispositioned the corrosion effects in the ECW system TLAA in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

4.7.3.2 Staff Evaluation

The staff reviewed LRA Section 4.7.3 and the corrosion effects in the ECW system TLAA to confirm, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

The staff reviewed the applicant's TLAA and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.7.3.1.3, which states that the applicant is to adequately manage the effects of aging on the intended functions with an AMP, consistent with the CLB for the period of extended operation.

The staff noted that corrosion in the ECW system was to be managed with the Open-Cycle Cooling Water System Program. The staff also noted that this program, as originally described in the LRA, consisted of visual inspections to detect loss of material in the ECW system. It was unclear to the staff how the visual inspection techniques in the Open-Cycle Cooling Water System Program would be capable of monitoring component wall thickness. By letter dated September 21, 2011, the staff issued RAI 4.7.3-1 requesting that the applicant state how the visual inspections would be capable of ensuring that the corrosion in the ECW system will not exceed the 40-mil design limit in the period of extended operation or propose an alternate methodology for ensuring the design limit is not exceeded.

In its response dated November 21, 2011, the applicant stated that when visual inspections identify corrosion, thickness measurements are taken as part of the Corrective Action Program. The staff found the response unacceptable because it lacked sufficient information to conclude that visual inspections alone would be capable of prompting followup thickness measurements.

By letter dated December 14, 2011, the staff issued followup RAI 4.7.3-2 requesting that the applicant state how visual inspections would be capable of detecting a 40-mil corrosion loss, or alternatively, state what augmented inspection techniques will be used to detect loss of material. A teleconference was held with the applicant on January 4, 2012, to clarify the staff's concerns in the followup RAI.

In its response dated February 6, 2012, the applicant stated that the 0.6-mil/year corrosion rate was not used in a plant analysis for making a safety determination for the ECW system; thus, the corrosion effects in the ECW system were incorrectly identified as a TLAA in the LRA. The applicant also stated that Section 4.7.3 would be deleted from the LRA. A teleconference was held with the applicant on February 9, 2012, to discuss how the applicant concluded that the corrosion rate analysis was not used in a safety determination, given that the analysis was included in the applicant's revised response to NRC Generic Letter 89-13 to provide justification for discontinuing the use of corrosion inhibitors in the ECW system. In response to the discussion, the applicant stated that it would provide a revised response to RAI 4.7.3-2.

In its response dated March 5, 2012, the applicant re-evaluated the TLAA and determined that it remains valid for the period of extended operation. The applicant stated that the aging effects would be managed using volumetric inspections in the Open-Cycle Cooling Water System Program, dispositioning the TLAA in accordance with 10 CFR 54.21(c)(1)(iii). The applicant also stated that wall thinning would be monitored at a minimum of 25 locations, in areas considered to have the highest corrosion rate, prior to the period of extended operation. The applicant further stated that subsequent inspections would be scheduled prior to the piping reaching minimum wall thickness, at which point an engineering analysis would be performed to determine if acceptable safety margin exists for continued operation. If an acceptable safety margin does not exist, the pipe would be isolated, repaired, or replaced. The applicant revised LRA Section A1.9, Section B2.1.9, and Commitment No. 4 to account for the changes in the Open-Cycle Cooling Water System Program. The applicant also added an item to LRA Table 3.3.2-4 for disposition of the TLAA.

The staff finds the applicant response acceptable because the volumetric wall thickness measurements in the Open-Cycle Cooling Water System Program, performed at a minimum of 25 locations in areas considered to have the highest corrosion rate, are capable of detecting wall thinning prior to reaching the minimum wall thickness. The staff noted that, as described in the applicant's RAI response dated February 6, 2012, the 40-mil design limit was originally added to the minimum pipe wall thickness to account for potential reductions in wall thickness due to factors such as erosion and corrosion. The staff also noted that the proposed volumetric inspections will be capable of directly monitoring such wall thickness reductions as the minimum wall thickness is approached; thus, the inspections are capable of detecting degradation before loss of intended function. The staff's concerns described in RAIs 4.7.3-1 and 4.7.3-2 are resolved.

Based on its review, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions of the corrosion effects in the ECW system TLAA will be adequately managed for the period of extended operation.

4.7.3.3 UFSAR Supplement

LRA Section A3.6.3 provides the UFSAR supplement summarizing the ECW system corrosion rate analysis and the disposition of this TLAA to manage corrosion with the Open-Cycle Cooling

Water System Program. The staff reviewed LRA Section A3.6.3, consistent with the review procedures in SRP-LR Section 4.7.3.2, which states that the applicant should provide a summary description of each TLAA that contains information on how the TLAA was dispositioned for the period of extended operation.

Based on its review of the UFSAR supplement, as modified by RAI response dated March 5, 2012, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.7.3.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the TLAA for corrosion effects in the ECW system, as required by 10 CFR 54.21(d).

4.7.3.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of corrosion on the intended functions of the ECW system TLAA will be adequately managed for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.4 Reactor Vessel Underclad Cracking Analysis

4.7.4.1 Summary of Technical Information in the Application

LRA Section 4.7.4, as amended by letter dated March 29, 2012, describes the applicant's TLAA for underclad cracking of the RPV components fabricated from SA-508, Class 2, forging materials. The applicant stated that the phenomenon of underclad cracking was originally addressed in the design basis though the implementation of welding practices that conformed to the crack-mitigation strategy and position in NRC RG 1.43, "Control of Stainless Steel Weld Cladding of Low Alloy Steel Components" (March 2011).

The applicant stated that, in Topical Report No. WCAP-15338-A, Westinghouse evaluated and demonstrated that the vessel integrity is maintained in the presence of underclad cracks. The phenomenon of underclad cracking is only applicable to RPV alloy steel components that were fabricated from SA-508, Class 2, alloy steel forging materials that were manufactured to a coarse grain practice and clad by a high-heat-input submerged arc welding process. The only RPV alloy steel components that are fabricated from SA-508, Class 2, forging materials are the RPV nozzles and the RPV flanges. The applicant stated that the generic fatigue flaw growth analysis in WCAP-15338-A is a time-dependent analysis that meets the definition of a TLAA.

The LRA states that the applicant dispositioned the TLAA for underclad cracking of RPV components in accordance with 10 CFR 54.21(c)(1)(i) to demonstrate that the analyses remain valid for the period of extended operation.

4.7.4.2 Staff Evaluation

The staff reviewed LRA Section 4.7.4 and the TLAA for underclad cracking of RPV components to confirm, pursuant to 10 CFR 54.21(c)(1)(i), that the analysis remains valid for the period of extended operation.

The staff reviewed the applicant's TLAAs for underclad cracking of RPV components and the corresponding disposition of 10 CFR 54.21(c)(1)(i), consistent with the review procedures in

SRP-LR Section 4.3.3.1.5.1. These procedures state that the operating cyclic experiences, and a list of the assumed cycles used in the existing analyses, are reviewed to ensure that the number of assumed cycles would not be exceeded during the period of extended operation.

The staff noted that non-proprietary Westinghouse Report No. WCAP-15338 provides a fracture toughness and flaw growth analysis for underclad cracks that are postulated in the internal surface of SA-508 Class 2 alloy steel components in Westinghouse design RPVs. The staff noted that the flaw growth analysis is based on ASME Code Section XI, Appendix A, which involves fatigue flaw growth methods that evaluate potential RPV underclad flaws over a 60-year licensed operating period. The staff's review of the fracture toughness and flaw growth analyses in WCAP-15338 is documented in an SE to the Westinghouse Owners Group dated October 15, 2001.

The staff noted that WCAP-15338-A is applicable to two-loop and four-loop Westinghouse reactor designs; therefore, WCAP-15338-A is applicable to the applicant's reactor design, which is a four-loop Westinghouse Electric-designed PWR. The generic safety and flaw analysis in WCAP-15338 evaluated the impact of 60 years of operation on the growth of postulated underclad cracks initiated in the internal cladding of Westinghouse-designed RPV components made from SA-508 Class 2 alloy steel forged materials. In the staff's SE on WCAP-15338-A, two renewal applicant action items for PWR applicants that reference WCAP-15338-A in the LRA were identified. The first renewal action item states that for applicants with Westinghouse two-loop and four-loop designed PWRs, the license renewal applicant should demonstrate that the transients for normal, upset, emergency, faulted, and PTS conditions assessed in WCAP-15338 are bounding for the plant-specific transients for these conditions; otherwise, the applicant will perform similar ASME Code Section XI flaw evaluations using its plant-specific transients to demonstrate that the RPVs with underclad cracks are acceptable though 60 years of licensed operation. The second renewal action item states that license renewal applicants referencing WCAP-15338-A should provide a summary description of the TLAA evaluation in the UFSAR supplement. By letter dated March 29, 2012, the LRA was revised to include LRA Section A.3.6.5, which is the UFSAR supplement for the TLAA related to underclad cracking of RPV components. The staff's review of LRA Section A.3.6.5 is documented in SER Section 4.7.4.3.

The staff noted that, in Section 5.4 of WCAP-15338-A, Westinghouse evaluated the fatigue-induced crack growth that would occur in postulated flaws that have 2:1, 6:1, and 100:1 length to depth aspect ratios. In addition, it was noted that Westinghouse considered the entire set of design basis transients for Westinghouse-designed plants to assess the impact of each design basis transient on the postulated flaw sizes in the analysis. The staff confirmed that Westinghouse calculated the crack growth associated with limiting number of cycles for each Westinghouse design basis transient over 60 years of operation by adding the crack growth increment to the original postulated flaw size and then repeating the process until all transient cycles have been accounted for in the final analyzed flaw size.

The staff also confirmed that the design basis transients for the applicant are described in LRA Table 4.3-2 and that the number of cycles for design transients analyzed for in WCAP-15338-A are bounding for the number of cycles projected for the applicant's units through 60 years of operation. Since the Westinghouse analysis incorporates the entire set of design basis transients for a four-loop Westinghouse-designed nuclear reactor, the staff finds that the applicant demonstrated that the generic fatigue flaw growth analysis bounds the set of design basis transients for the applicant's units through 60 years of operation.

Based on its review, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses for underclad cracking of the RPV components remain valid for the period of extended operation. Additionally, these analyses meet the acceptance criteria in SRP-LR Section 4.3.3.1.5.1 because the applicant and the Westinghouse flaw growth analysis for underclad cracks in RPV components made with SA-508 Class 2 forging materials demonstrated that the full set of design transients for 60 years of operation were considered and will not be exceeded during the period of extended operation.

4.7.4.3 UFSAR Supplement

LRA Section A3.6.5, as amended by letter dated March 29, 2012, provides the UFSAR supplement summarizing the TLAA for underclad cracking of the RPV components. The staff reviewed LRA Section A3.6.5, consistent with the review procedures in SRP-LR Section 4.3.3.2, which state that the reviewer should confirm that the applicant provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA.

Based on its review of the amended UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.3.2.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the TLAA for underclad cracking of the RPV components, as required by 10 CFR 54.21(d).

4.7.4.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses for underclad cracking of the RPV components fabricated from SA 508, Class 2, forging materials remain valid for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.5 Reactor Coolant Pump Flywheel Crack Growth Analysis

4.7.5.1 Summary of Technical Information in the Application

LRA Section 4.7.5 describes the applicant’s TLAA for RCP flywheel fatigue crack growth analyses. The applicant stated that UFSAR Section 5.4.1.5.2 describes RCP flywheel design and its compliance with RG 1.14, Revision 1, “Reactor Coolant Pump Flywheel Integrity,” and that RCP flywheel inspections are included in the STP ISI Program and are required by STP TS 4.4.10.

To reduce the inspection frequency and scope, the applicant amended its initial compliance with RG 1.14 by implementing Westinghouse Topical Report WCAP-14535-A, “Reactor Coolant Pump Motor Flywheel Inspection Elimination,” which supports the relaxation of inspections required by RG 1.14, Positions C.4.b(1) and (2).

The applicant stated that the topical report, Westinghouse Topical Report WCAP-14535-A, “Reactor Coolant Pump Flywheel Inspection Elimination,” provided an engineering basis for elimination of RCP flywheel ISI requirements for all operating Westinghouse plants and certain Babcock and Wilcox plants. Fatigue crack growth analyses that are included in the WCAP-14535-A report have been identified as a TLAA. The applicant stated that WCAP-14535-A performed a Monte-Carlo simulation to evaluate the probability of failure over the period of extended operation for all operating Westinghouse plants. It demonstrated that the flywheel design has a high structural reliability with a very high flaw tolerance and negligible flaw crack extension over a 60-year service life (assumed 6,000 pump starts). Therefore, any potential crack growth from an existing flaw would be minimal, and the analysis in the WCAP-14535-A report remains valid for the period of extended operation.

The LRA stated that the applicant dispositioned the flywheel TLAA in accordance with 10 CFR 54.21(c)(1)(i) to demonstrate that the analysis remains valid for the period of extended operation.

4.7.5.2 Staff Evaluation

SRP-LR Section 4 does not list RCP flywheel fatigue crack growth analyses as TLAAs that are generic to industry LRAs. As a result, the staff reviewed LRA Section 4.7.5 against the acceptance guidance in SRP-LR Section 4.7.5.1 for disposition of a plant-specific TLAA in accordance with 10 CFR 54.21(c)(1)(i). The staff reviewed LRA Section 4.7.5 to confirm, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses remain valid for the period of extended operation.

The staff noted that RG 1.14 provides the staff's recommended acceptance criteria for material and minimum fracture toughness properties of SA 508, Classes 2 and 3, materials and SA 533 Grade B, Class 2, materials used in the fabrication of U.S. RCP flywheels. RG 1.14, Revision 1, also provides guidelines for performing structural integrity assessments of the RCP flywheels in U.S. light-water reactors (LWRs), including assessments for ensuring the integrity of the flywheels against unacceptable fatigue-induced crack growth failures.

The staff noted that the applicant is relying on the flaw growth analysis in WCAP-14535-A (ADAMS Accession No. 9601290393) as the TLAA for the RCP flywheels. The staff confirmed that the NRC endorsed the methodology and results in this WCAP report in an SE dated September 12, 1996 (ADAMS Accession No. 9609230010). However, in the SE conclusion section (Section 4.0), the staff concluded that that the inspections of the RCP flywheels should be performed even if all of the recommendations in RG 1.14, Revision 1, were met and that the inspections of the RCP flywheels should not be eliminated.

The staff issued RAI 4.7.5-1 (ADAMS Accession No. ML12039A240), requesting that the applicant describe the past examinations for the RCP flywheels and explain how those results justify the use of WCAP-14535-A. The staff also asked the applicant to clarify whether the safety basis in the TLAA for the RCP flywheels is being used to justify elimination of the RCP flywheel examinations altogether or whether the applicant intends to continue the inspection of the RCP flywheels, consistent with the NRC's SE on WCAP-14535-A, dated September 12, 1996. If inspection will be performed during the period of extended operation, the staff also asked the applicant to justify what type of examinations will be performed on the RCP flywheels during the period of extended operation and note the frequency that will be used for the examinations. Otherwise, the staff requested that the applicant justify its basis for discontinuing inspection of the RCP flywheels if ISIs will be discontinued during the period of extended operation.

The applicant's March 12, 2012, response indicated that STP, Unit 1, RCP flywheels have been inspected four times, and STP, Unit 2, RCP flywheels have been inspected five times. The most recent UT examinations were conducted in fall 2009 and fall 2008 for STP, Units 1 and 2, respectively.

The applicant stated that no unacceptable indications have been found in any of the required inspections. In addition, the applicant stated that during the period of extended operation, the applicant will continue the surface and volumetric inspections of the RCP flywheels on the required interval.

In summary, the staff finds the applicant's response to RAI 4.7.5-1, and the applicant's claim that the RCP flywheels will maintain their structural integrity during the period of the extended operation, acceptable for the following reasons:

- WCAP-14535-A performed a Monte-Carlo simulation to evaluate the probability of failure over the period of extended operation for all operating Westinghouse plants, demonstrating that the RCP flywheel design has a high structural reliability with a very high flaw tolerance and negligible flaw crack extension over a 60-year service life (assumed 6,000 pump starts).
- WCAP-14535-A has been endorsed for use in the staff's SE of September 12, 1996.
- Future inspections will be performed once every 10 years.
- In accordance with 10 CFR 54.21(c)(1)(i), the current analysis has been demonstrated to remain valid for the period of extended operation.

The staff's concerns described in RAI 4.7.5-1 are resolved.

4.7.5.3 UFSAR Supplement

LRA Section A3.6.4 provides the UFSAR supplement summary description of the applicant's TLAA evaluation of the RCP flywheel fatigue crack growth analysis. The staff reviewed LRA Section A3.6.4, consistent with the review procedures in SRP-LR Section 4.7.3.2, which state that the reviewer should confirm that the applicant provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of each TLAA. Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.7.2.2. Additionally, the staff finds that the applicant provided an adequate summary description of its actions to address the TLAA for RCP flywheel fatigue crack analysis, as required by 10 CFR 54.21(d).

4.7.5.4 Conclusion

Based on its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that for RCP flywheel fatigue crack analyses from WCAP-14535-A remains valid for the period of extended operation and applicable to STP, Units 1 and 2. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.8 Conclusion for Time-Limited Aging Analyses

The staff reviewed the information in LRA Section 4, "Time-Limited Aging Analyses." Based on its review, the staff concludes that the applicant provided an adequate list of TLAAs, as defined in 10 CFR 54.3. Furthermore, the staff concludes that the applicant demonstrated that the TLAAs will remain valid for the period of extended operation, as required by 10 CFR 54.21(c)(1)(i); that the TLAAs have been projected to the end of the period of extended operation, as required by 10 CFR 54.21(c)(1)(ii); or that the effects of aging on the intended functions will be adequately managed for the period of extended operation, as required by

10 CFR 54.21(c)(1)(iii). The staff also reviewed the UFSAR supplement for the TLAAs and found that the UFSAR supplement contains descriptions of the TLAAs sufficient to satisfy the requirements in 10 CFR 54.21(d). In addition, the staff concludes that one plant-specific exemption (see Section 4.4) is in effect that is based on TLAAs and that the applicant provided an adequate evaluation that justifies the continuation of this exemption for the period of extended operation, as required by 10 CFR 54.21(c)(2).

With regard to these matters, the staff concludes that the activities authorized by the renewed license will continue to be conducted in accordance with the CLB and that any changes made to the CLB, in order to comply with 10 CFR 54.21(c), are in accordance with the Atomic Energy Act of 1954 and NRC regulations.

SECTION 5

REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

In accordance with Title 10 of the *Code of Federal Regulations,* Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants" (10 CFR Part 54), the Advisory Committee on Reactor Safeguards (ACRS) will review the license renewal application (LRA) for South Texas Project (STP), Units 1 and 2. The U.S. Nuclear Regulatory Commission (NRC) staff (the staff) presented its safety evaluation report (SER) with open items to the ACRS Subcommittee on Plant License Renewal in a public meeting on November 17, 2016. The ACRS Subcommittee on Plant License Renewal will continue its detailed review of the LRA after this SER is issued. STP Nuclear Operating Company (the applicant) and the NRC staff will meet with the full committee to discuss issues associated with the LRA review.

After the staff issues its final SER and the ACRS completes its review of the LRA and SER, the ACRS will issue a letter discussing the results of its review. An update to this SER will include the ACRS letter and the staff's response to any issues and concerns reported.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

July 26, 2017

The Honorable Kristine L. Svinicki
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR SOUTH TEXAS PROJECT, UNITS 1 AND 2

Dear Chairman:

During the 645th meeting of the Advisory Committee on Reactor Safeguards (ACRS), July 12 - 14, 2017, we completed our review of the license renewal application for South Texas Project, Units 1 and 2 (STP 1 & 2) and the final safety evaluation report (SER) prepared by the NRC staff. Our subcommittee on Plant License Renewal reviewed this matter during a meeting on November 17, 2016. During these reviews, we had the benefit of discussions with representatives of the staff and the South Texas Project Nuclear Operating Company (STPNOC, or the applicant). We also had the benefit of the referenced documents. This report fulfills the requirement of 10 CFR Part 54.25 that the ACRS review and report on all license renewal applications.

CONCLUSION AND RECOMMENDATION

1. The programs established and committed to by STPNOC to manage age-related degradation provide reasonable assurance that STP 1 & 2 can be operated in accordance with its current licensing bases for the period of extended operation without undue risk to the health and safety of the public.

2. STPNOC's application for renewal of the operating licenses for STP 1 & 2 should be approved.

BACKGROUND

STP is located near the town of Matagorda in Matagorda County, Texas. The construction permits for STP 1 & 2 were issued on December 22, 1975 and the operating licenses were issued on August 20, 1987 (Unit 1) and December 15, 1988 (Unit 2). Each unit's nuclear steam supply system consists of a 4-loop pressurized water reactor (PWR) designed by Westinghouse Electric Corporation. The primary containment for each unit is a large dry design. The balance of plant was designed and constructed by Bechtel Corporation. Each unit operates at a licensed thermal power of 3,853 MWt, with an electrical output of approximately 1,250 MWe.

In this application, STPNOC requests renewal of Facility Operating Licenses DPR-76 and DPR-80 for a period of 20 years beyond the current expiration dates of August 20, 2027 (Unit 1) and December 15, 2028 (Unit 2).

DISCUSSION

In its final SER dated June 2017, the staff documented its review of the license renewal application and other information consisting of staff audits and inspections at the plant site. The staff reviewed the completeness of the identification of structures, systems, and components (SSCs) that are within the scope of license renewal. The staff also reviewed the integrated plant assessment process; the identification of plausible aging mechanisms associated with passive, long-lived components; the adequacy of the Aging Management Programs (AMPs); and the identification and assessment of Time-Limited Aging Analyses (TLAAs) requiring review.

The license renewal application identified the SSCs that fall within the scope of license renewal. The application demonstrates consistency with the Generic Aging Lessons Learned (GALL) Report (NUREG-1801, Revision 2) and documents and justifies deviations to the specified approaches in that report. STPNOC will implement 41 AMPs for license renewal, comprised of 33 existing programs and 8 new programs. Seven of the 41 AMPs are consistent with the GALL Report without enhancements or exceptions. Thirteen AMPs have been enhanced so they are consistent with GALL. Four AMPs take allowed exceptions to GALL. Thirteen AMPs are consistent with enhancements and allowed exceptions. Four AMPs, Protective Coating Monitoring and Maintenance Program, Nickel-Alloy Aging Management Program, PWR Reactor Internals, and Selective Leaching of Aluminum Bronze, are plant-specific.

The license renewal application includes seventeen programs with allowed exceptions to the GALL Report. We conclude that the exceptions are acceptable.

The staff conducted license renewal audits and performed a license renewal inspection at STP 1 & 2. The audits verified the appropriateness of the scoping and screening methodology for AMPs, the appropriateness of the aging management review, and the acceptability of the TLAAs. The license renewal inspection verified that the license renewal requirements are implemented appropriately. Both the inspection and the report of that inspection are thorough.

Based on the audits, the inspection, and the staff reviews related to this license renewal application, the staff concluded that STPNOC has demonstrated that the effects of aging at STP 1 & 2 will be adequately managed so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation, as required by 10 CFR Part 54.21(a)(3).

The single open item that remained in the SER was resolved between our subcommittee meeting on November 17, 2016 and our final review on July 12, 2017. Resolution of this item is as follows:

Selective Leaching of Aluminum Bronze Aging Management Program

STP has experienced degradation of its service water and other piping systems due to selective leaching of aluminum bronze components and welds in those systems. The Selective Leaching of Aluminum Bronze Program is intended to manage this issue. This program manages loss of material due to selective leaching for aluminum bronze components and welds exposed to raw water within the scope of license renewal. The open item identified and addressed issues that were identified in the staff's evaluation of the program for selective leaching of aluminum bronze.

Based on the staff's interaction with the applicant during public meetings and supplemental audits, and the final revised plant-specific AMP and its associated updated final safety analysis report supplement, the open issues have been resolved through the applicant's commitment to replace susceptible castings and by improvements to the aging management program for welds.

All aluminum bronze castings susceptible to selective leaching, including attachment welds related to the castings and aluminum bronze root valve adapter socket welds, will be replaced prior to the period of extended operation with material that is not susceptible to selective leaching. Samples of susceptible welds in wrought copper alloy piping will be inspected and other samples will be subjected to destructive examinations. In the event that unacceptable indications are detected, additional examinations will be conducted. All defective welds will be replaced. We concur with the staff's acceptance of the resolution for this open item.

We agree with the staff that there are no issues related to the matters described in 10 CFR Parts 54.29(a)(1) and (a)(2) that preclude renewal of the operating license for STP 1 & 2. The programs established and committed to by STPNOC provide reasonable assurance that STP 1 & 2 can be operated in accordance with their current licensing basis for the period of extended operation without undue risk to the health and safety of the public. The STPNOC application for renewal of the operating licenses for STP 1 & 2 should be approved.

Sincerely,

/RA/

Dennis C. Bley
Chairman

REFERENCES

1. South Texas Nuclear Operating Company, South Texas Project, Units 1 and 2, "License Renewal Application," October 25, 2010 (ML103010262).

2. South Texas Nuclear Operating Company, South Texas Project, Units 1 and 2, "Supplement to the South Texas Project License Renewal Application (CAC Nos. ME4936 and ME4937)," March 30, 2017 (ML17102B415).

3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of South Texas Project, Units 1 and 2, June 2017 (ML17146B242 & ML17146B224).

4. U.S. Nuclear Regulatory Commission, "Scoping and Screening Audit Report Regarding the South Texas Project, Units 1 and 2 (TAC NOS. ME4936 AND ME4937)," September 6, 2011 (ML11230A003).

5. U.S. Nuclear Regulatory Commission, “Aging Management Programs Audit Report Regarding the South Texas Project, Units 1 and 2 (TAC NOS. ME4936 AND ME4937),” September 22, 2011 (ML11224A265).

6. U.S. Nuclear Regulatory Commission, “South Texas Project Electric Generating Station, Units 1 and 2 NRC License Renewal Inspection Report 05000498/2011007 and 05000499/2011007,” October 7, 2011 (ML112800109).

7. U.S. Nuclear Regulatory Commission, NUREG-1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” Revision 2, December 2010 (ML103490036).

8. U.S. Nuclear Regulatory Commission, NUREG 1801, “Generic Aging Lessons Learned (GALL) Report,” Revision 2, December 2010 (ML103490041).

9. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.188, “Standard Format and Content for Application to Renew Nuclear Power Plant Operating Licenses,” Revision 1, September 2005 (ML082950585).

10. U.S. Nuclear Regulatory Commission, “Selective Leaching of Aluminum Bronze Aging Management Program Audit Report Regarding the South Texas Project, Units 1 and 2 (CAC NOS. ME4936 AND ME4937),” May 16, 2017 (ML17107A319).

5. U.S. Nuclear Regulatory Commission, "Aging Management Programs Audit Report Regarding the South Texas Project, Units 1 and 2 (TAC NOS. ME4936 AND ME4937)," September 22, 2011 (ML11224A265).

6. U.S. Nuclear Regulatory Commission, "South Texas Project Electric Generating Station, Units 1 and 2 NRC License Renewal Inspection Report 05000498/2011007 and 05000499/2011007," October 7, 2011 (ML112800109).

7. U.S. Nuclear Regulatory Commission, NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," Revision 2, December 2010 (ML103490036).

8. U.S. Nuclear Regulatory Commission, NUREG 1801, "Generic Aging Lessons Learned (GALL) Report," Revision 2, December 2010 (ML103490041).

9. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.188, "Standard Format and Content for Application to Renew Nuclear Power Plant Operating Licenses," Revision 1, September 2005 (ML082950585).

10. U.S. Nuclear Regulatory Commission, "Selective Leaching of Aluminum Bronze Aging Management Program Audit Report Regarding the South Texas Project, Units 1 and 2 (CAC NOS. ME4936 AND ME4937)," May 16, 2017 (ML17107A319).)

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DATE	7/26/2017	7/26/2017	7/26/2017	7/26/2017	7/26/2017

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SECTION 6

CONCLUSION

The U.S. Nuclear Regulatory Commission (NRC) staff (the staff) reviewed the license renewal application (LRA) for South Texas Project (STP), Units 1 and 2, in accordance with NRC regulations and NUREG-1800, Revision 2, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” dated December 2010. Title 10 of the *Code of Federal Regulations* 54.29, “Standards for Issuance of a Renewal License” (10 CFR 54.29) sets the standards for issuance of a renewed license.

Based on its review of the LRA, the staff finds that the requirements of 10 CFR 54.29(a) have been met.

The staff noted that any requirements of 10 CFR Part 51, “Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions,” Subpart A, “National Environmental Policy Act – Regulations Implementing Section 102(2),” are documented in a plant-specific supplement to NUREG-1437, “Generic Environmental Impact Statement for License Renewal of Nuclear Plants (GEIS),” Supplement 48, “Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants, Supplement 48, Regarding South Texas Project, Units 1 and 2 - Final Report,” dated November 2013.

APPENDIX A

COMMITMENTS FOR LICENSE RENEWAL OF SOUTH TEXAS PROJECT, UNITS 1 AND 2

A. Commitments for License Renewal of South Texas Project, Units 1 and 2

During the review of the South Texas Project (STP), Units 1 and 2, license renewal application (LRA) by the staff of the U.S. Nuclear Regulatory Commission (NRC or the staff), STP Nuclear Operating Company (STPNOC or the applicant) made commitments related to aging management programs (AMPs) and time-limited aging analyses (TLAAs) to manage the aging effects of structures and components (SCs) prior to the period of extended operation. LRA Section A0, “Appendix A Introduction,” states that “[LRA] Section A4 [as revised by supplements and RAI responses] contains summary descriptions of license renewal commitments,” and that “license renewal commitments will be incorporated in the STP UFSAR [updated final safety analysis report] Update following the issuance of the renewed license in accordance with 10 CFR 50.71(e) [Title 10 of the *Code of Federal Regulations* 50.71(e)].” The following table lists these commitments, along with the respective implementation schedules and sources of the commitment.

No.	Commitment	Implementation Schedule	LRA Section	Reference Letter & Date
1	Enhance the Water Chemistry Program procedures by doing the following: (NOC-AE-10002607) • Include a statement that the sampling frequency for the primary and secondary water systems is temporarily increased whenever corrective actions are taken to address an abnormal chemistry condition for action level parameters. • Explain that this increased sampling is used to verify that the desired condition has been achieved, and, when it is achieved, the sampling frequencies are returned to the Electric Power Research Institute (EPRI)-recommended frequencies.	Completed	B2.1.2	NOC-AE-10002607, October 25, 2010 NOC-AE-14003141, June 3, 2014
2	Enhance the Boric Acid Corrosion Program procedures by doing the following: (NOC-AE-10002607) • State that susceptible components adjacent to potential leakage sources include electrical components and connectors. • State that it is applicable to other materials (such as aluminum and copper alloy) that are susceptible to boric acid corrosion.	Completed	B2.1.4	NOC-AE-10002607, October 25, 2010 NOC-AE-14003141, June 3, 2014
3	Enhance the Bolting Integrity Program procedures by doing the following: • Conform to the guidance contained in EPRI TR-104213 (NOC-AE-11002750). • Evaluate loss of preload of the joint connection, including bolt stress, gasket stress, flange alignment, and operating condition to determine the corrective actions consistent with EPRI TR-104213 (NOC-AE-10002607). • Require a leak check of ASME pressure boundary bolted connections where the internal environment consists of dry gas, or compressed air using a method that detects leakage such as a visual inspection for discoloration, monitoring and trending for pressure decay, leak fluid detection, or when the temperature of the system is higher than ambient conditions thermography testing. (NOC-AE-17003459). • ASME pressure boundary bolted connections where the internal environment consists of air at atmospheric pressure are checked for tightness prior to the period of extended operation and once every six years thereafter. (NOC-AE-17003459).	No later than six months prior to the period of extended operation	B2.1.7	NOC-AE-10002607, October 25, 2010 NOC-AE-11002750, November 4, 2011 NOC-AE-14003141, June 3, 2014 NOC-AE-17003459 April 19, 2017
4	Enhance the Open-Cycle Cooling Water System Program procedures by doing the following: • Include visual inspection of the strainer inlet area and the interior surfaces of the adjacent upstream and downstream piping to identify material wastage, dimensional change, discoloration, and discontinuities in surface texture. These inspections will provide visual evidence of loss of material and fouling in the essential cooling water (ECW) system and serve as an indicator of the condition of the interior of ECW system piping components otherwise inaccessible for visual inspection (NOC-AE-10002607). • Include the acceptance criteria for this visual inspection (NOC-AE-10002607).	No later than six months prior to the period of extended operation Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	B2.1.9	NOC-AE-10002607, October 25, 2010 NOC-AE-12002809, March 5, 2012 NOC-AE-12002825, March 29, 2012 NOC-AE-12002874, July 5, 2012

No.	Commitment	Implementation Schedule	LRA Section	Reference Letter & Date
	• Require that a minimum of 25 ECW piping locations be measured for wall thickness prior to the period of extended operation. Selected areas will include locations considered to have the highest corrosion rates, such as areas with stagnant flow (NOC-AE-12002809). • Require an engineering evaluation after each inspection of the aluminum-bronze piping inserted inside the slip-on flange downstream of the component cooling water heat exchanger: (NOC-AE-12002874) – Require that the evaluation will calculate projected wear over the next inspection interval, including a margin of 4 years of wear at the most recent actual yearly wear rate. – Require that repair or replacement, in accordance with the Corrective Action Program, be initiated if the projected wear (which includes a margin of 4 years of wear at the most recent actual yearly wear rate) indicates that the aluminum-bronze piping wall will reduce to a thickness of less than minimum wall thickness. • Require loss of material in piping and protective coating failures be documented in the Corrective Action Program (NOC-AE-12002825). • Require an engineering evaluation be performed when loss of material in piping or protective coating failures is identified (NOC-AE-12002825).			NOC-AE-12002942, December 19, 2012 NOC-AE-14003078 February 27, 2014 NOC-AE-14003141, June 3, 2014 NOC-AE-15003260, June 11, 2015 NOC-AE-15003303, November 12, 2015
	Enhance the Open-Cycle Cooling Water System program procedures to: • Inspect every six years and test after 12 years of service at a six year frequency 100 percent of the coating applied on the essential chiller water box covers, standby diesel generator (SDG) jacket water coolers, SDG lube oil coolers, SDG intercoolers, and interconnection piping. The coating test performed are low voltage holiday test per ASTM D5162-08, dry film thickness test per ASTM D7091-13 and Steel Structures Painting Council (SSPC) PA-2 January 2015 and pull off adhesion test per ASTM D4541-09. • Require coating inspections and tests be performed by a qualified Nuclear Coating Specialist (NCS) as defined by ASTM D7108 endorsed in RG 1.54. • Require monitoring and trending of coatings installed on the internals of in-scope components. • Require coatings specialist prepare a post-inspection report that includes a list and location of all areas of deterioration that were remediated. • Specify the acceptance criteria for coatings as no blistering, cracking erosion, cavitation erosion, flaking, peeling, delamination, rusting or physical damage of the coatings installed on the internals of in-scope components is observed. • Require coatings not meeting these criteria be considered degraded and a condition report be initiated to document and resolve the concern.	Complete no later than the date the renewed operating license is issued		

No.	Commitment	Implementation Schedule	LRA Section	Reference Letter & Date
	• Require degraded coating be removed to sound material and replaced with new coating. • Require physical testing where physically possible be performed in conjunction with repair or replacement of coatings.			
5	Enhance the Closed-Cycle Cooling Water System Program procedures by doing the following: • Include visual inspection of representative samples of each combination of material and water treatment program at least every 10 years and opportunistically (NOC-AE-10002607) (NOC-AE-11002681) (NOC-AE-11002750). • Include acceptance criteria (NOC-AE-10002607).	No later than six month prior to the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	B2.1.10	NOC-AE-10002607, October 25, 2010 NOC-AE-11002681, June 16, 2011 NOC-AE-11002750, November 4, 2011 NOC-AE-14003141, June 3, 2014
6	Enhance the Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program procedures by doing the following: • Inspect crane structural members for loss of material due to corrosion and rail wear. (NOC-AE-10002607).	Completed	B2.1.11	NOC-AE-10002607, October 25, 2010 NOC-AE-14003141, June 3, 2014
7	Enhance the Fire Protection Program procedures by doing the following: (NOC-AE-10002607) • Provide visual inspection for corrosion and mechanical damage on Halon system components at least once every 6 months. • Provide inspections to detect the following penetration seal deficiencies: signs of degradation such as cracking, seal separation from walls and components, separation of layers of material, rupture and puncture of seals. • Include qualification criteria for individuals performing inspections of fire doors, fire barrier penetration seals, fire barrier walls, ceilings and floors in accordance with NUREG-1801. • Include the following fire barrier inspection acceptance criteria: no cracks, spalling, or loss of material that would prevent the barrier from performing its design function. • Provide visual inspection for degradation, corrosion, and mechanical damage on Halon system components at least once every 6 months.	No later than six month prior to the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	B2.1.12	NOC-AE-10002607, October 25, 2010 NOC-AE-14003141, June 3, 2014
8	Enhance the Fire Water System program procedures to perform periodic inspections, testing, and cleaning on the following: • include volumetric examinations or direct measurement on representative locations of the fire water system to determine pipe wall thickness,	Complete no later than six months prior to the period of extended operation. Inspections to be	B2.1.13	NOC-AE-10002607, October 25, 2010

No.	Commitment	Implementation Schedule	LRA Section	Reference Letter & Date
	• replace sprinklers prior to 50 years in service or field service test a representative sample and test every 10 years thereafter to ensure signs of degradation are detected in a timely manner, and • trending of fire water piping flow parameters recorded during fire water flow tests. • Sprinkler inspections every 18 months per NFPA 25, 2011 Edition Section 5.2.1.1, • 50-year sprinkler replacement or testing per NFPA 25, 2011 Edition Section 5.3.1, • Standpipe and hose systems flow tests every 3 years per NFPA 25, 2011 Edition Section 6.3.1, • Underground and exposed piping flow tests every 3 years per NFPA 25, 2011 Edition Section 7.3.1, • Hydrants flow testing and visually inspection annually per NFPA 25, 2011 Edition Section 7.3.2, • Fire pumps suction screens cleaning and inspections per NFPA 25, 2011 Edition Section 8.3.3.7, • Fire water storage tank exterior inspections annually per NFPA 25, 2011 Edition Section 9.2.5.5, • Fire water storage tank coated interior surfaces are inspected every 5 years per NFPA 25, 2011 Edition Section 9.2.6. Testing is performed in accordance with NFPA 25, 2011 Edition Section 9.2.7 whenever there is evidence of pitting and corrosion below nominal wall depth or failure of tank coatings. Additionally, bottom thickness ultrasonic tests are done at least once every 10 years. • Main drain testing every 18 months per NFPA 25, 2011 Edition Section 13.2.5, • Deluge Valve testing annually per NFPA 25, 2011 Edition Sections 13.4.3.2.2 through 13.4.3.2.5, • Water Spray Fixed System strainers cleaning and inspections per NFPA 25, 2011 Edition Section 10.2.1.6, 10.2.1.7, 10.2.7, • Spray/sprinkler nozzles full flow test every 18 months per NFPA 25, 2011 Edition Section 10.3.4.3, • Foam water sprinkler systems spray nozzle strainers per NFPA 25, 2011 Edition Section 11.2.7.1, • Foam water sprinkler systems operational test discharge patterns annually per NFPA 25, 2011 Edition Section 11.3.2.6, • Foam water sprinkler systems storage tank visual inspection for internal corrosion once every 10 years, and	complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.		NOC-AE-14003141, June 3, 2014 NOC-AE-1 5003260, June 11, 2015 NOC-AE-15003303, November 12, 2015

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	• Internal surface of piping and branch lines obstruction inspections every 5 years per NFPA 25, 2011 Edition Sections 14.2 and 14.3. Procedures will be enhanced to: • perform follow-up volumetric wall thickness examinations when surface irregularities are detected; • perform either flow testing or flushing sufficient to detect flow blockage or 100 percent visual inspection in each 5-year interval, beginning 5 years prior to the period of extended operation on portions of water-based fire protection components that have been wetted but are normally dry or piping segments that cannot be drained or segments that allow water to collect; • perform volumetric wall thickness inspection on 20 percent of the length of piping segments that cannot be drained or piping segments that allow water to collect in each 5-year interval prior to the period of extended operation. The 20 percent of piping inspected in each 5-year interval shall be in different location than previously inspected piping; • monitor and trend fire water piping flow parameters recorded during fire water flow tests; • specify the acceptance criteria to be: – Minimum design fire water piping wall thickness is maintained. – Fouling shall not be observed during inspections of sprinklers and associated piping in the sprinkler system that could cause flow blockage. – Sprinklers that show signs of leakage or corrosion shall be replaced. If any sprinklers fail the representative sample testing required for sprinkler in service for 50 years, all sprinklers within the area represented by the sample will be replaced. – Sufficient foreign organic or inorganic material obstructing pipe or sprinklers is removed and its source is determined and corrected; • manage coatings installed on the internals of in-scope fire water components for loss of coating integrity; • visually inspect the coatings on fire water storage tank every 5 years as outlined by NFPA-25, 2011 Edition: • inspect 100 percent of the coatings installed on the internals of non-tank in-scope fire water components every six years, and tested after 12 years of service at a six-year frequency. Replaced coatings are inspected every 4 years until there are three consecutive inspections with no change in the coating condition. Following three consecutive inspections with no change in the coating condition the 6 year inspection interval can be restored. The coating tests performed are low voltage holiday test per ASTM D5162-08, dry film thickness test per ASTM D7091 -13 and Steel Structures			

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	Painting Council, (SSPC) PA-2 January 2015, and pull off adhesion test per ASTM D4541-09. Coating inspections and tests are performed by a qualified Nuclear Coating Specialist (NCS) as defined by ASTM D7108 endorsed in RG 1.54; • monitor and trend coatings installed on the internals of in-scope fire water components; • require coatings specialist prepare a post-inspection report that includes a list and location of all areas of deterioration that were remediated; • specify the acceptance criteria for coatings as no blistering, cracking, erosion, cavitation erosion, flaking, peeling, delamination, rusting or physical damage of the coatings installed on the internals of in-scope fire water components is observed; • require coatings not meeting the acceptance criteria be considered degraded and a condition report be initiated to document and resolve the concern and, • require degraded coating be removed to sound material and replaced with new coating; • require physical testing where physically possible in conjunction with repair or replacement of coatings.			
9	Enhance the Fuel Oil Chemistry Program procedures by doing the following: • Extend the scope of the program to include the standby diesel generator (SDG) fuel oil drain tanks (NOC-AE-10002607). • Check and remove the accumulated water from the fuel oil drain tanks, day tanks, and storage tanks associated with the SDG, balance of plant (BOP), lighting diesel generator, and fire water pump diesel generators. Include a minimum frequency of water removal from the fuel oil tanks in the procedure (NOC-AE-10002607) (NOC-AE-11002758). • Include 10-year periodic draining, cleaning, and inspection for corrosion of the SDG fuel oil drain tanks, lighting diesel generator fuel oil tank, BOP diesel generator fuel oil day tanks, and diesel fire pump fuel oil storage tanks (NOC-AE-10002607) (NOC-AE-11002758) (NOC-AE-11002763). • Require periodic testing of the lighting diesel generator fuel oil tank and the SDG and diesel fire pump fuel oil storage tanks for microbiological organisms (NOC-AE-10002607) (NOC-AE-11002758). • Require analysis for water, biological activity, sediment, and particulate contamination of the diesel fire pump fuel oil storage tanks, lighting diesel generator fuel oil tank, and the BOP diesel generator fuel oil day tanks on a quarterly basis (NOC-AE-10002607) (NOC-AE-11002681) (NOC-AE-11002758).	Complete no later than six months prior to the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	B2.1.14	NOC-AE-10002607, October 25, 2010 NOC-AE-11002681, June 16, 2011 NOC-AE-11002758, November 30, 2011 NOC-AE-11002763, December 6, 2011 NOC-AE-14003141, June 3, 2014

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	• Conduct ultrasonic testing or pulsed eddy current thickness examination to detect corrosion-related wall thinning once on the tank bottoms for the SDG and diesel fire pump and the BOP diesel generator fuel oil day tanks (NOC-AE-10002607). • Incorporate the sampling and testing of the diesel fire pump fuel oil storage tanks for particulate contamination and water and to incorporate the trending of water, particulate contamination, and microbiological activity in the SDG and diesel fire pump fuel oil storage tanks, lighting diesel generator fuel oil tank, and the BOP diesel generator fuel oil day tanks (NOC-AE-10002607) (NOC-AE-11002758).			
10	Enhance the Reactor Vessel Surveillance Program procedures by doing the following: (NOC-AE-10002607) • Include the withdrawal schedule and analysis of the ex-vessel dosimetry chain. • Demonstrate that the reactor vessel inlet and outlet nozzles are exposed to a fluence of less than 10^{17} n/cm^2, or incorporate the adjusted reference temperature (ART) for the inlet and outlet nozzles with bounding chemistry and fluence values into the pressure-temperature (P-T) limit curves. • Enhance the program to include the Unit 2 bottom head torus in the Reactor Vessel Surveillance Program.	Complete no later than six months prior to the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	B2.1.15	NOC-AE-10002607, October 25, 2010 NOC-AE-14003141, June 3, 2014
11	Implement the One-Time Inspection Program, as described in LRA Section B2.1.16 (NOC-AE-10002607)	Start implementation during the 10 years prior to the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	B2.1.16	NOC-AE-10002607, October 25, 2010 NOC-AE-14003141, June 3, 2014
12	Implement the Selective Leaching of Materials Program, as described in LRA Section B2.1.17 (NOC-AE-10002607) (NOC-AE-12002789)	Start implementation during the 5 years prior to the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	B2.1.17	NOC-AE-10002607, October 25, 2010 NOC-AE-12002789, January 26, 2012 NOC-AE-14003141, June 3, 2014

No.	Commitment	Implementation Schedule	LRA Section	Reference Letter & Date
13	Enhance plant specifications to: • Lower coated piping carefully into a trench to avoid external coating damage. • Use proper storage and handling practices to prevent damage to pipe coating prior to installation. These practices include padded storage, use of proper slings for installation, and ultraviolet light resistant topcoats. • Over excavate trenches, use qualified backfill for bedding piping, and take care during backfilling to prevent rocks and debris from striking and damaging the pipe coating. • Include the coating used for copper-alloy buried piping in the coating database; the coating system shall be in accordance with National Association of Corrosion Engineers (NACE) SP0169-2007 and will be used for repair or for new coatings of the buried copper-alloy piping in the ECW system. • Coat the portion of the ECW system copper-alloy piping directly embedded in backfill or directly encased in concrete, extending the coating 2 feet or more above grade. Enhance the Buried Piping and Tanks Inspection Program procedures to include the following: • Specify that in lieu of visual inspections of the fire protection system (FP), this program credits flow testing of the fire mains as described in Section 7.3 of NFPA 25, 2011 Edition. • Consider backfill located within 6 inches of the pipe, and consistent with American Society for Testing and Materials (ASTM) D 448-08 size number 67, acceptable. Backfill quality is determined through examination during the inspections conducted by the program. Backfill that does not meet the ASTM criteria, during the initial and subsequent inspections of the program, is considered acceptable if the inspections of buried piping do not reveal evidence of mechanical damage to the pipe coatings due to the backfill. • Ensure the cathodic protection system survey is performed annually. • Monitor the output of the cathodic protection system rectifiers every 2 months. Record the measured current at each rectifier and compare it against a target value. Following the completion of the plant yard cathodic protection system annual survey, record the current of the rectifier used to achieve an acceptable pipe/soil potential. That current will be the target current for the rectifier until the next annual survey. If the current measured at the rectifier during the bimonthly monitoring deviates significantly from the target value, create a condition report. The rectifier current should be adjusted to an acceptable value. The results of the survey will be documented and trended to identify degrading conditions. When degraded rectifier performance is identified, documentation is required in accordance with the Corrective Action Program. The system should not be operated outside of established acceptable limits for longer than 90 days.	Start implementation during the 10 years prior to the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	B2.1.18	NOC-AE-11002681, June 16, 2011 NOC-AE-14003141, June 3, 2014 NOC-AE-14003149, June 26, 2014 NOC-AE-15003260, June 11, 2015 NOC-AE-16003380, May 19, 2016 NOC-AE-16003385 June 28, 2016

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	• Recommend increased monitoring of the cathodic protection system and/or additional inspections, or both, if adverse indications are discovered during the monitoring of the cathodic protection system. • Evaluate the effectiveness of isolating fittings, continuity bonds, and casing isolation during the plant yard cathodic protection system annual survey. This may be accomplished through electrical measurements. • The personnel performing the plant yard cathodic protection system annual survey must be National Association of Corrosion Engineers (NACE)-certified, certified by a site-approved training procedure consistent with the NACE requirements, or supervised by a NACE-certified inspector. • Use of excessive cathodic protection polarized potential on coated piping should be avoided. The limiting critical potential should not be more negative than 1200 mV relative to a copper/copper sulfate reference electrode (CSE). • Visually inspect buried piping and, if significant indications of degradation are observed, supplement the visual inspections by surface or volumetric (or both) non-destructive testing. • Specify uncoated stainless steel piping and coated stainless steel piping where the coating is not well-adhered be inspected using a surface examination or other method capable of detecting cracking. Coatings that are intact, well-adhered, and otherwise sound for the remaining inspection interval, and coatings exhibiting small blisters that are few in number and completely surrounded by sound coating bonded to the substrate do not have to be removed. • Define the inspection interval for the program-directed inspections as every 10 years, beginning the 10-year interval prior to the beginning of the period of extended operation. • Select the buried and underground piping inspection locations based on risk, considering susceptibility to degradation and consequences of failure. • The risk ranking for buried piping should consider characteristics such as coating type, coating condition, cathodic protection efficiency, backfill characteristics, soil resistivity, pipe contents, and pipe function. • The risk ranking for underground piping should consider characteristics such as coating type, coating condition, exact external environment, pipe contents, pipe function, and flow characteristics within the pipe. • The risk ranking should generally give piping systems that are backfilled using compacted aggregate a higher inspection priority than comparable systems that are completely backfilled using controlled low strength material. • Consider the External Corrosion Direct Assessment, as described in NACE Standard Practice SP0502-2010, for use in identifying inspection locations.			

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	• Credit opportunistic examinations of non-leaking pipes toward required examinations only if they meet the risk ranking selection criteria. • Use guided wave ultrasonic or other advanced inspection techniques, if practical, for the purpose of determining piping locations that should be inspected. These inspections may not be used as substitutes for inspections required by the program. • Credit an inspection of piping shared between Units 1 and 2 toward the required inspections and inspections are distributed evenly among the units. • Examine any piping, valves, and closure bolting exposed during inspections. • Examine bolting for loss of material and loose or missing fasteners. • Include two alternatives to directed inspections of the buried or underground piping that is within the scope of license renewal. – The first alternative is to hydrostatically test 25 percent of the subject piping to 110 percent of the design pressure of any component within the boundary with test pressure being held for eight hours on an interval not to exceed 5 years. – The second alternative is an internal inspection of 25 percent of the subject piping by a method capable of accurately determining pipe wall thickness on an interval of every 10 years. • Flow testing of the fire mains, as described in National Fire Protection Act (NFPA) 25, 2011 Edition, to detect degradation of the buried pipe in lieu of visual inspections of the fire protection system buried and underground piping. • Specify that each inspection will examine either the entire length of a run of pipe, or a minimum of 10 feet. If the entire run of pipe of that material type is less than 10 feet in total length, then the entire run of pipe should be inspected. The inspection consists of a 100 percent visual inspection of the exposed pipe. • Specify that if a transition from Category C to Category E or from Category E to Category F occurs in the latter half of the current 10-year interval, the timing of the additional examinations is based on the severity of the degradation identified and is commensurate with the consequences of a leak or loss of function. In all cases, the examinations are completed within 4 years after the end of the particular 10-year interval. These additional inspections conducted in an inspection interval cannot be credited towards the base number of inspections required for the 10-year interval. • Specify where steel or copper alloy piping has been coated with the same coating system and the backfill has the same requirements, the total inspections for this piping may be combined to satisfy the recommended inspection quantity. For example, for Category F, 10 percent of the total of the associated steel or copper alloy is inspected; or 9 10-foot segments of steel or copper alloy piping are inspected. • Specify that Category C inspections be used when the external cathodic protection system for buried steel or copper alloy pipe meets the acceptance criteria. Category			

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	C inspections are 0.5 percent Not-to-Exceed (NTE) two inspections of that piping per inspection period performed.
	• Specify that Category E inspections be used when the cathodic protection system has been installed but the portions of the piping covered by that system fail to meet the acceptance criteria. Category E inspections are 5 percent, NTE 15. The following condition must be present.
	  ○ Coatings and backfill are provided in accordance with STP backfill specification.
	  ○ There have been no leaks in buried piping due to external corrosion and no significant coating degradation or metal loss in more than 10 percent of inspections conducted.
	  ○ Soil has been demonstrated to be not corrosive for the material type using the following.
	    – A minimum of three sets of soil samples will be obtained in the vicinity where the cathodic protection system fails to meet the acceptance criteria.
	    – The soil will be tested for soil resistivity, corrosion accelerating bacteria, pH, moisture, chlorides, sulfates, and redox potential.
	    – The potential soil corrosivity will be determined for each material type of buried in-scope piping in the vicinity of the failed cathodic protection system. In addition to evaluating each individual parameter, the overall soil corrosivity will be determined.
	    – If portions of the installed cathodic protection system fail to meet the acceptance criteria, soil testing will be conducted at a minimum of once in each 10-year period starting at the time when it was determined that the cathodic protection system failed to meet the acceptance.
	• Specify that inspection scope for piping that does not meet Category C or E inspection schedule requirements is 10 percent, NTE 9 inspections.
	• Specify that the AF system underground uncoated stainless steel piping located in a vault and buried coated stainless steel piping will undergo two inspections each 10-year inspection period.
	• Specify that the OW system underground piping will undergo 2% NTE 3 inspection each 10-year inspection period.
	• Include acceptance criteria for the cathodic protection to be operational (available) at least 85 percent of the time since either 10 years prior to the period of extended operation or since installation or refurbishment, whichever is shorter.
	• Include acceptance criteria for the cathodic protection system to provide protection for buried piping at least 80 percent of the time since either 10 years prior to the period of extended operation or since installation or refurbishment, whichever is shorter.			

No.	Commitment	Implementation Schedule	LRA Section	Reference Letter & Date
	• Include examples of adverse indications discovered during piping inspections. • Repair or replace the affected component when adverse indications fail to meet the acceptance criteria described in the program are discovered. • Specify that if adverse indications are detected, an expansion of the sample size is conducted. The number of inspections within the affected piping categories is doubled or increased by 5, whichever is smaller. If adverse indications are found in the expanded sample, an analysis is conducted to determine the extent of condition and extent of cause. The size of the follow-on inspections will be determined based on the extent of condition and extent of cause. The timing of the additional examinations should be based on the severity of the degradation identified and should be commensurate with the consequences of a leak or loss of function. However, in all cases, the expanded sample inspections should be completed within the 10-year interval in which the original inspection was conducted or, if identified in the latter half of the current 10-year interval, within 4 years after the end of the 10-year interval. If adverse conditions are extensive within the 10-year interval in which the inspections were conducted or, if identified in the latter half of the current 10-year interval within 4 years after the end of the 10-year interval, inspections may be halted in an area of concern that is planned for replacement, provided continued operation does not pose a significant hazard. Expansion of sample size may be limited to the piping subject to the observed degradation mechanism. • Observe for brittle failure at flanges, connections, and joints due to frost heaving, soil stresses, or groundwater effects during inspection of buried piping. • Require trending cathodic protection system annual surveys results. • Where wall thickness measurements are conducted, the results should be trended if follow-up examinations are conducted. • Specify that the cathodic protection system pipe-to-soil potential when using a saturated copper/copper sulfate reference electrode must be at least -850 mV relative to a CSE, instant off for steel piping. A minimum polarization value of 100 mV is required for copper alloy piping. • Specify that If the cathodic protection system fails to meet the acceptance criteria of at least -850 mV relative to a CSE instant off for steel components the following alternatives may be used: – 100 mV minimum polarization – -750 mV relative to a CSE. Instant off where soil resistivity is greater than 10.000 ohm-cm to less than 100,000 ohm-cm. – -650 mV relative to a CSE. Instant off where soil resistivity is greater than 100,000 ohm-cm. – Verify less than 1 mil/year (mpy) loss of material.			

No.	Commitment	Implementation Schedule	LRA Section	Reference Letter & Date
	• Specify means to verify the effectiveness of the protection of the most anodic metal when alternatives are used are incorporated into the program. The external loss of material rate is verified by: – Every year when verifying the effectiveness of the cathodic protection system by measuring the loss of material rate. – Every 2 years when using the 100 mV minimum polarization. – Every 5 years when using the -750 mV or -650 mV criteria associated with higher resistivity soils. The soil resistivity is verified every 5 years. • Specify where electrical resistance corrosion rate probes are used the installation locations of the probes and the methods of use will be determined by qualified NACE CP4 Cathodic Protection Specialist. • Require the impact of significant site features (e.g., large cathodic protection current collectors, shielding due to large objects located in the vicinity of the protected piping) and local soil conditions be factored into placement of the probes and use of probe data. • For coated piping, indicate that there should be no evidence of coating degradation. If coating degradation is present, it may be considered acceptable if it is determined to be insignificant by an individual possessing a NACE Coating Inspector Program Level 2 or 3 inspector qualification, or an individual has attended the Electric Power Research Institute (EPRI) Comprehensive Coatings Course and completed the EPRI Buried Pipe Condition Assessment and Repair Training Computer Based Training Course. • Specify where damage to the coating has been evaluated as significant and the damage was caused by non-conforming backfill, an extent of condition evaluation should be conducted to ensure that the as-left condition of backfill in the vicinity of observed damage will not lead to further degradation. • Specify that backfill is acceptable if the inspections do not reveal evidence that the backfill caused damage to the component's coatings or the surface of the component. • Indicate that for any hydrostatic tests credited by the program, the condition acceptance criteria is no visible indications of leakage and no drop in pressure within the isolated portion of the piping that is not accounted for by a temperature change in the test media or quantified leakage across test boundary valves. • Specify that if coated or uncoated metallic piping shows evidence of corrosion, the remaining wall thickness in the affected area is determined to ensure that the minimum wall thickness is maintained. • Indicate that wall thickness will be extrapolated to next inspection for that pipe section or to the end of the period of extended operation in order for the component to meet acceptance criteria and to not conduct expanded inspections.			

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	• Specify where wall thickness meets minimum wall thickness requirements, recommendations for expansion of sample size does not apply. • Require unacceptable cathodic protection survey results be entered into the plant corrective action program. • Specify that sources of leakage detected during pressure tests be identified and corrected. • Specify that indications of cracking are evaluated in accordance with applicable codes and plant-specific design criteria.			
14	Implement the One-Time Inspection of American Society of Mechanical Engineers (ASME) Code Class 1 Small-Bore Piping Program, as described in LRA Section B2.1.19 (NOC-AE-10002607) (NOC-AE-11002681).	Start implementation during the 6 years prior to the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	B2.1.19	NOC-AE-10002607, October 25, 2010 NOC-AE-11002681, June 16, 2011 NOC-AE-14003141, June 3, 2014
15	Implement the External Surfaces Monitoring Program, as described in LRA Section B2.1.20 (NOC-AE-10002607). Existing plant procedures will be enhanced to include the following: • Require a leak check of non-ASME pressure boundary bolted connections where the internal environment consists of dry gas, compressed air, or diesel exhaust using a method that detects leakage such as a visual inspection for discoloration, monitoring and trending for pressure decay, leak fluid detection, or when the temperature of the system is higher than ambient conditions thermography testing. • Require bolted connections where the internal environment consists of air at atmospheric pressure be checked for tightness prior to the period of extended operation and once every six years thereafter.	Complete no later than six months prior to the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	B2.1.20	NOC-AE-10002607, October 25, 2010 NOC-AE-14003141, June 3, 2014 NOC-AE-17003459 April 17, 2017
16	Enhance the Flux Thimble Tube Inspection Program by generating a new procedure that includes the following provisions: (NOC-AE-10002607) • Perform a wall thickness eddy current inspection of all flux thimble tubes that form part of the reactor coolant system pressure boundary. Schedule the inspections for each outage. An inspection may be deferred by using an evaluation that considers the actual wear rate.	Complete no later than six months prior to the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to	B2.1.21	NOC-AE-10002607, October 25, 2010 NOC-AE-14003141, June 3, 2014

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	• Design engineering personnel will evaluate flux thimble tube wear. Perform corrective actions based on evaluation results after each inspection. • Design engineering personnel will trend wall thickness measurements and calculate wear rates after each inspection. • Take corrective actions to reposition, cap, or replace the tube if the predicted wear (as a measure of percent through-wall) for a given flux thimble tube is projected to exceed the established acceptance criterion prior to the next outage. • Include a description of the testing and analysis methodology and percent through-wall acceptance criteria of a maximum of 80 percent through-wall loss. • Remove flux thimbles from service to ensure the integrity of the reactor coolant system pressure boundary for flux thimble tubes that cannot be inspected over the tube length, that are subject to wear due to restriction or other defect, and that cannot be shown by analysis to be satisfactory for continued service.	the PEO, whichever occurs later.		
17	Implement the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program, as described in LRA Section B2.1.22. (NOC-AE-10002607) (NOC-AE-11002764)	Start implementation during the 5 year period prior to the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	B2.1.22	NOC-AE-10002607, October 25, 2010 NOC-AE-11002764, December 15, 2011 NOC-AE-14003141, June 3, 2014
18	Enhance the Lubricating Oil Analysis Program procedures by doing the following: (NOC-AE-10002607) • Require analysis for particle count of the lubricating oil for the centrifugal charging pump. • Require that sample analysis data results, for which no acceptance criteria is specified, be evaluated and trended against baseline data and data from previous samples to determine the acceptability of oil for continued use.	Complete no later than six month prior to the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	B2.1.23	NOC-AE-10002607, October 25, 2010 NOC-AE-14003141, June 3, 2014
19	Implement the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program as described in LRA Section B2.1.24. (NOC-AE-10002607)	Complete no later than six month prior to the period of extended operation. Inspections to be complete no later than six	B2.1.24	NOC-AE-10002607, October 25, 2010 NOC-AE-14003141, June 3, 2014

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		months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.		
20	Enhance the Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program procedures by doing the following: • Identify the cables, manholes, and trenches that are within the scope of the program (NOC-AE-11002607) (NOC-AE-11002732). • Require all in-scope non-EQ inaccessible medium and low voltage power cables (>400 volts) exposed to significant moisture be tested at least once every 6 years, with the first test being completed prior to period of extended operation. (NOC-AE-10002607) (NOC-AE-11002681) (NOC-AE-11002732) (NOC-AE-12002789). • Require that the acceptance criteria be defined prior to each test for the specific type of test performed and the specific cable tested. (NOC-AE-10002607). • Require an engineering evaluation that considers the age and operating environment of the cable be performed when the test acceptance criteria are not met. The engineering evaluation shall consider the significance of the test or inspection results, the operability of the component, the reportability of the event, the extent of the concern, the potential root causes for not meeting the test or inspection acceptance criteria, the corrective actions required, and the likelihood of recurrence. (NOC-AE-10002607) (NOC-AE-11002732). • Inspect in-scope manholes and trenches based on plant-specific operating experience with water accumulation. (NOC-AE-11002732) • Require inspections be conducted at least annually. (NOC-AE-11002732) • Include performance of event-driven inspections of in-scope manholes as an on-demand activity based on actual plant experience. (NOC-AE-11002769) • Perform direct observation that cables are not wetted or submerged. (NOC-AE-11002732) • Remove collected water and confirm sump pump operability. (NOC-AE-11002732) • Initiate a corrective action if wetted cables or inoperable sump pumps are found. (NOC-AE-11002732) • Inspect cables/splices and cable support structures if wetted cables are found. (NOC-AE-11002732) • Take corrective actions to keep cables dry. (NOC-AE-11002732)	Complete no later than six month prior to the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	B2.1.25	NOC-AE-10002607, October 25, 2010 NOC-AE-11002681, June 16, 2011 NOC-AE-11002732, October 10, 2011 NOC-AE-11002769, December 7, 2011 NOC-AE-11002772, January 5, 2012 NOC-AE-12002789, January 26, 2012 NOC-AE-14003141, June 3, 2014

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	• Evaluate manhole inspection results based on actual plant experience, with the inspection frequency increased based on experience with water accumulation. (NOC-AE-11002732) (NOC-AE-11002772) • Test in-scope inaccessible medium and low voltage (>400 volts) power cables exposed to significant moisture using a test capable of detecting reduced insulation resistance. (NOC-AE-11002732) • Trend inspection and test results to provide additional information on the rate of cable insulation degradation. (NOC-AE-11002732) • Test frequency may be adjusted based on test results or operating experience. (NOC-AE-11002772) • Require that the acceptance criterion for manhole and trench be cables/splices and support structures is that they are not submerged or immersed in water. (NOC-AE-11002732 • Require an extent of condition when an unacceptable condition or situation is identified. (NOC-AE-11002732)			
21	Enhance the Metal-Enclosed Bus Program procedures by doing the following: (NOC-AE-11002732) • Identify the metal enclosed buses (MEBs) that are within the scope of the program. • Inspect internal portions of all MEBs for cracks, corrosion, foreign debris, excessive dust buildup, and evidence of water intrusion every 10 years. • Inspect non-segregated phase bus insulation and isolated phase bus insulators for signs of embrittlement, cracking, melting, swelling, or discoloration every 10 years. • Inspect internal bus supports for structural integrity and signs of cracks every 10 years. • Inspect bus enclosure assemblies for loss of material due to corrosion and hardening of boots and gaskets every 10 years. • Inspect 20 percent of the population of non-segregated phase bus accessible bolted connections insulation material (with a maximum sample size of 25) for surface anomalies every 5 years. • Perform the first inspection of all portions of in-scope MEBs prior to the period of extended operation. • Identify acceptance criteria for non-segregated phase bus insulation and isolated phase bus insulators as no unacceptable visual indications of surface anomalies. • Identify acceptance criteria for non-segregated phase bus sections and internal portions of isolated phase bus as no unacceptable indications of corrosion, cracks,	No later than six months prior to the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	B2.1.26	NOC-AE-11002732, October 10, 2011 NOC-AE-14003141, June 3, 2014

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	foreign debris, excessive dust buildup, loss of material, hardening, or evidence of water intrusion. • Identify acceptance criteria for the exterior of MEBs as no unacceptable indications of general corrosion. • Identify acceptance criteria for boots and gaskets as no unacceptable indications of cracking, checkering, or discoloration. • Identify acceptance criteria for accessible bolted connection insulation material as no unacceptable evidence of embrittlement, cracking, melting, discoloration, swelling, or surface contamination. • Require an engineering evaluation when acceptance criteria are not met, to include a determination of corrective actions. • Require an engineering evaluation to determine whether the unacceptable conditions may be applicable to other accessible or inaccessible MEBs.			
22	Enhance the ASME Code, Section XI, Subsection IWL Program procedures by doing the following: (NOC-AE-10002607) • Incorporate the 2004 Edition of ASME Code, Section XI, Subsection IWL (no addenda), supplemented with the applicable requirements of 10 CFR 50.55a(b)(2).	Completed	B2.1.28	NOC-AE-10002607, October 25, 2010 NOC-AE-14003141 June 3, 2014
23	Enhance the ASME Code, Section XI, Subsection IWF Program procedures by doing the following: • Incorporate the 2004 Edition of ASME Code, Section XI, Subsection IWF (with no addenda). (NOC-AE-10002607) • Specify the preventive actions for storage, protection, and lubricants recommended in Section 2 of Research Council for Structural Connections publication, “Specification for Structural Joints Using ASTM A325 or A490 Bolts,” for ASTM A325, ASTM F1852, or ASTM 490 bolts. (NOC-AE-11002732) • Specify that visual examinations are augmented with volumetric examinations, in accordance with ASME Code, Section XI, Table IWB-2500-1, Examination Category B-G-1, to detect stress corrosion cracking (SCC) for 20 percent (25 bolts maximum per unit) of high strength bolts greater than 1-inch nominal diameter and with an actual yield strength greater than or equal to 150 ksi. (NOC-AE-11002772)	Complete no later than six months prior to the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	B2.1.29	NOC-AE-10002607, October 25, 2010 NOC-AE-11002732, October 10, 2011 NOC-AE-11002772, January 5, 2012 NOC-AE-14003141, June 3, 2014
24	Enhance the 10 CFR Part 50 Appendix J Program procedures by doing the following: (NOC-AE-10002607) • Specify a surveillance frequency of 15 years following a successful Type A test. (NOC-AE-16003385)	Complete no later than six months prior to the period of extended operation	B2.1.30	NOC-AE-10002607, October 25, 2010 NOC-AE-14003141, June 3, 2014 NOC-AE-16003385, June 28, 2016

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25	Enhance the Structures Monitoring Program procedures by doing the following: • Include the switchyard control building into the scope of the Structures Monitoring Program. (NOC-AE-11002759) • Specify inspections of seismic gaps, caulking and sealants, duct banks and manholes, valve pits and access vaults, doors, electrical conduits, raceways, cable trays, electrical cabinets/enclosures and associated anchorage. (NOC-AE-10002607) • Monitor at least two groundwater samples every 5 years for pH, sulfates, and chloride concentrations. (NOC-AE-10002607) • Specify that the inspection frequency for structures within the scope of license renewal will be in accordance with American Concrete Institute (ACI) 349.3R, Table 6.1, as follows: (NOC-AE-11002732) – For below-grade structures and structures in controlled interior environment (except inside primary containment), all accessible areas of both units will be inspected every 10 years. – For all other structures (including inside primary containment), all accessible areas of both units will be inspected every 5 years. • Specify inspector qualifications in accordance with ACI 349.3R-96. (NOC-AE-10002607) • Perform periodic visual inspection of the accessible sections of the spent fuel pool and transfer canal tell-tale drain lines for blockage every 5 years. The first inspection will be performed within the 5 years before entering the period of extended operation. (NOC-AE-11002732) (NOC-AE-11002772) (NOC-AE-12002789) • Specify ACI 349.3R-96 and ACI 201.1R-68 as the basis for defining quantitative acceptance criteria. (NOC-AE-11002732) • Specify the preventive actions for storage, protection, and lubricants recommended in Section 2 of Research Council for Structural Connections publication “Specification for Structural Joints Using ASTM A325 or A490 Bolts” for ASTM A325, ASTM F1852, or ASTM 490 bolts. (NOC-AE-1102732) • Enhance procedures to perform opportunistic inspections of exposed portions of the below-grade concrete when excavated for any reason. (NOC-AE-11002737) • Enhance procedures to require an evaluation in cases where groundwater is determined to be aggressive or inspections of accessible concrete structural elements identify degradation. The evaluation will include determination of the appropriate actions necessary to assure that the affected structures will continue to perform their intended functions. These actions may include increased visual inspections or other examination techniques. (NOC-AE-11002737) • Specify that visual examinations will be augmented with volumetric examinations, in accordance with ASME Code, Section XI, Table IWB-2500-1, Examination	Complete no later than six months prior to the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	B2.1.32	NOC-AE-10002607, October 25, 2010 NOC-AE-11002732, October 10, 2011 NOC-AE-11002737, October 18, 2011 NOC-AE-11002759, November 21, 2011 NOC-AE-11002772, January 5, 2012 NOC-AE-12002789, January 26, 2012 NOC-AE-14003141, June 3, 2014

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	Category B-G-1, to detect SCC for 20 percent (25 bolts maximum) of high-strength bolts greater than 1-inch nominal diameter and with an actual yield strength greater than or equal to 150 ksi. (NOC-AE-11002772)			
26	Enhance Regulatory Guide (RG) 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants Program procedures by doing the following: • Specify inspections of the essential cooling pond and ECW Intake and Discharge structures at intervals not to exceed 5 years or to immediately follow significant natural phenomena. (NOC-AE-16003403) • Specify essential cooling pond sediment monitoring be performed every ten years using soundings. (NOC-AE-16003403) • Specify the preventive actions for storage, protection, and lubricants recommended in Section 2 of Research Council for Structural Connections publication, "Specification for Structural Joints Using ASTM A325 or A490 Bolts" for ASTM A325, ASTM F1852, or ASTM 490 bolts. (NOC-AE-11002732) • Specify ACI 349.3R-96 and ACI 201.1R-68 as the basis for defining quantitative acceptance criteria. (NOC-AE-11002732) • Specify the essential cooling pond seepage rate evaluation be performed not less than once every 5 years. (NOC-AE-16003403) • Specify visual inspection of the essential cooling pond embankment lining for signs of erosion, loss of form as in degradation of slope protection features. (NOC-AE-16003403)	Complete no later than six months prior to the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	B2.1.33	NOC-AE-10002607, October 25, 2010 NOC-AE-1002732, October 10, 2011 NOC-AE-11002758, November 30, 2011 NOC-AE-14003141, June 3, 2014 NOC-AE-16003403 September 28, 2016
27	Implement the PWR Reactor Internals Program as described in LRA Section B2.1.35. (NOC-AE-10002607) (NOC-AE-12002797)	Completed	B2.1.35	NOC-AE-10002607, October 25, 2010 NOC-AE-12002797, February 27, 2012 NOC-AE-1 3003041, October 28, 2013 NOC-AE-1 5003270 June 30, 2015 NOC-AE-1 5003320 December 17, 2015
28	Implement the Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program as described in LRA Section B2.1.36. (NOC-AE-10002607)	Complete no later than six months prior to the period of extended operation. Inspections to be complete no later than six	B2.1.36	NOC-AE-10002607, October 25, 2010 NOC-AE-14003141, June 3, 2014

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		months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.		
29	As additional industry and plant-specific aging-related operating experience becomes available, evaluate and incorporate it into applicable AMPs or develop new AMPs, as necessary, to provide assurance that the effects of aging will be managed during the period of extended operation. (NOC-AE-11002683)	Start implementation within 10 years prior to entering the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	B2.1.16 B2.1.17 B2.1.19 B2.1.20 B2.1.22 B2.1.24 B2.1.35 B2.1.36 B1.4	NOC-AE-11002683, June 23, 2011 NOC-AE-14003141, June 3, 2014
30	Enhance the Metal Fatigue of Reactor Coolant Pressure Boundary Program procedures by doing the following: • Include additional locations necessary to ensure accurate calculations of fatigue. (NOC-AE-10002607) • Include additional transients that contribute significantly to fatigue usage. (NOC-AE-10002607) • Include counting of the transients used in the fatigue crack growth analyses, which support the leak-before-break analyses and ASME Code, Section XI evaluations, to ensure the analyses remain valid. (NOC-AE-11002672) • Include additional transients necessary to ensure accurate calculations of fatigue and fatigue usage monitoring at specified locations, and specify the frequency and process of periodic reviews of the results of the monitored cycle count and CUF data at least once per fuel cycle. (NOC-AE-10002607) • Include additional cycle count and fatigue usage action limits, which will invoke appropriate corrective actions if a component approaches a cycle count action limit or a fatigue usage action limit. The acceptance criteria associated with the NUREG/CR-6260 sample locations for a newer vintage Westinghouse plant will account for environmental effects on fatigue locations in the reactor coolant pressure boundary, and reactor vessel internals locations with fatigue usage calculations, and (NOC-AE-5003320) • Include appropriate corrective actions to be invoked if a component approaches a cycle count action limit or a fatigue usage action limit. Acceptable corrective actions include fatigue reanalysis, repair, or replacement. Reanalysis of a fatigue crack	Complete no later than six months prior to the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	B3.1	NOC-AE-10002607, October 25, 2010 NOC-AE-11002672, May 12, 2011 NOC-AE-11002759, November 21, 2011 NOC-AE-14003141, June 3, 2014 NOC-AE-1 5003320, December 17, 2015

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	growth analysis must be consistent with or reconciled to the originally submitted analysis and receive the same level of regulatory review as the original analysis. (NOC-AE-10002607) (NOC-AE-11002672) (NOC-AE-11002759)			
31	STPNOC commits to the following: (NOC-AE-10002607) • for reactor coolant system nickel-alloy pressure boundary components: – Implement applicable NRC orders, bulletins, and generic letters associated with nickel-alloys. – Implement staff-accepted industry guidelines. – Participate in the industry initiatives, such as owners group programs and the EPRI Materials Reliability Program, for managing aging effects associated with nickel-alloys. – Submit an inspection plan for reactor coolant system nickel-alloy pressure boundary components to the NRC for review and approval upon completion of these programs, but not less than 24 months before entering the period of extended operation. • for reactor vessel internals: – Participate in the industry programs for investigating and managing aging effects on reactor internals. – Evaluate and implement the results of the industry programs as applicable to the reactor internals. – Submit an inspection plan for reactor internals to the NRC for review and approval upon completion of these programs, but not less than 24 months before entering the period of extended operation.	Concurrent with industry initiatives and upon completion submit an inspection plan and not less than 24 months before entering the period of extended operation.	3.1	NOC-AE-10002607, October 25, 2010
32	Replace the seven diesel generator cooling water expansion joints that are projected to exceed the analyzed number of cycles during the period of extended operation. The analyses for the replacement expansion joints will include the period of extended operation. (NOC-AE-10002607)	Complete no later than six months prior to the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	4.3.6	NOC-AE-10002607, October 25, 2010 NOC-AE-14003141, June 3, 2014
33	Periodic inspection of a sample of transmission conductor connections for loose connections using thermography is currently performed as part of the preventive maintenance activities. The periodic thermography will continue into the period of extended operation. (NOC-AE-10002607)	Continued into the period of extended operation	3.6.2.2.3	NOC-AE-10002607, October 25, 2010 NOC-AE-14003141, June 3, 2014

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				NOC-AE-14003180, October 22, 2014 NOC-AE-14003180, October 22, 2014
34	Prior to the period of extended operation, STP will perform a review of design basis ASME Code Class 1 component fatigue evaluations to determine whether the NUREG/CR-6260-based components that have been evaluated for the effects of the reactor coolant environment on fatigue usage are the limiting components for the STP configuration. If more limiting components are identified, the most limiting component will be evaluated for the effects of the reactor coolant environment on fatigue usage. If the limiting location consists of nickel alloy, the methodology for nickel alloy in NUREG/CR-6909 will be used to perform the environmentally-assisted fatigue calculation. The additional evaluation will be performed through the Metal Fatigue of Reactor Coolant Pressure Boundary Program in accordance with 10 CFR 54.21(c)(1)(iii). (NOC-AE-11002731)	Complete no later than six month prior to the period of extended operation.	B3.1	NOC-AE-11002731, September 15, 2011 NOC-AE-14003141, June 3, 2014
35	Enhance the ASME Code, Section XI, Subsection IWE Program procedures by doing the following: (NOC-AE-11002732) • Specify the preventive actions for storage, protection, and lubricants recommended in Section 2 of Research Council for Structural Connections publication, “Specification for Structural Joints Using ASTM A325 or A490 Bolts,” for ASTM A325, ASTM F1852, or ASTM 490 bolts.	Complete no later than six months prior to the period of extended operation.	B2.1.27	NOC-AE-11002732, October 10, 2011 NOC-AE-14003141, June 3, 2014
36	Enhance the Masonry Wall Program procedures by specifying that the inspection frequency for structures within the scope of license renewal will be in accordance with ACI 349.3R, Table 6.1, as follows: (NOC-AE-11002732) • For below-grade structures and structures in controlled interior environment (except inside primary containment), inspect all accessible areas of both units every 10 years. • For all other structures (including inside primary containment), inspect all accessible areas of both units every 5 years.	Complete no later than six months prior to the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	B2.1.31	NOC-AE-11002732, October 10, 2011 NOC-AE-14003141, June 3, 2014
37	Take groundwater samples at multiple locations around the site every 3 months for at least 24 consecutive months. The samples will analyze for pH, sulfates, and chlorides. This sampling plan will begin no later than September 2012. (NOC-AE-11002737)	Completed	B2.1.32	NOC-AE-11002737, October 18, 2011 NOC-AE-1 3003041, October 28, 2013
38	Enhance the Reactor Head Closure Studs Program procedures by doing the following:	Complete no later than six months prior to the period	B2.1.3	NOC-AE-11002750, November 4, 2011

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	• Preclude the future use of replacement closure stud assemblies fabricated from material with an actual measured yield strength greater than or equal to 150 ksi. The use of currently installed components and any spare components that are currently on site is allowed. (NOC-AE-11002750)	of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.		NOC-AE-11002764, December 15, 2011 NOC-AE-12002830, April 17, 2012 NOC-AE-14003141, June 3, 2014
39	Enhance the Selective Leaching of Aluminum Bronze procedures to: • Visually examine aluminum bronze material exposed during inspection of the buried essential cooling water piping for evidence of leakage coating degradation and • If degradation is identified near a weld, a volumetric examination will be performed to determine if cracking of the weld is occurring. • If a leak from buried aluminum bronze welds is discovered by surface water monitoring or during a buried ECW piping inspection, a section of each leaking weld will be removed for destructive metallurgical examination.	No later than the date the renewed operating licenses are issued.	B2.1.37	NOC-AE-11002766, December 8, 2011 NOC-AE-12002853, May 31, 2012 NOC-AE-12002889, October 4, 2012 NOC-AE-14003135, July 31, 2014 NOC-AE-1 5003278, July 29, 2015 NOC-AE-16003365, May 4, 2016 NOC-AE-16003394 July 28, 2016
40	Enhance the Protective Coating Monitoring and Maintenance Program procedures by doing the following: (NOC-AE-12002797) • Specify parameters monitored or inspected include any visible defects, such as blistering, cracking, flaking, peeling, rusting, and physical damage, as specified in ASTM D 5163-08. • Specify that inspection frequencies, personnel qualifications, inspection plans, inspection methods, and inspection equipment meet the requirements of ASTM D 5163-08. • Perform a pre-inspection review of the previous two monitoring reports and, based on inspection report results, prioritize repair areas as either needing repair during the same outage, needing repair during the next available outage, or needing re-evaluation in the next available outage. • Develop a standardized coating condition assessment report form that will include both the identification of coatings found intact with no defects identified, and the	Complete no later than six months prior to the period of extended operation. Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.	B2.1.39	NOC-AE-12002797, February 27, 2012 NOC-AE-14003141, June 3, 2014

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	identification of coatings that were not inspected along with the reason why the inspection could not be conducted. • Develop a standardized coating condition assessment report that will include written or photographic documentation, or both, of coating inspection areas, failures, and defects. • Perform destructive/non-destructive tests by individuals trained in the applicable referenced standards of Guide D5498 on an as-needed basis as determined by the Nuclear Coatings Specialist.			
41	Enhance the STP Operating Experience Program (OEP) and Corrective Action Program for managing the effects of aging by doing the following: (NOC-AE-12002797) • Add license renewal interim staff guidance and revisions to the GALL Report to the OEP procedure as sources of information within the scope of this program. • Revise the OEP procedure to include “aging effects” to the list of characteristics for determining applicability of an operating experience document that may require further evaluation. A screened-in evaluation should consider: (a) systems, structures, or components, (b) materials, (c) environments, (d) aging effects, (e) aging mechanisms, and (f) AMPs. • Review the Corrective Action Program event codes to determine if additional codes are needed to ensure age-related degradation effects are identified. • Perform a training “needs analysis” for those plant personnel, including AMP owners, who screen, assign, evaluate, implement, and submit plant-specific and industry operating experience information for age-related effects. Include in the analysis: – A requirement that individuals complete training before performing tasks, and – A determination of the periodicity of the training • Revise the OEP procedure to provide criteria for reporting plant-specific operating experience of age-related degradation.	No later than the date the renewed operating licenses are issued	A1	NOC-AE-12002797, February 27, 2012 NOC-AE-12002870, June 14, 2012 NOC-AE-12002907, December 6, 2012
42	Enhance the Reactor Head Closure Studs Program procedures by doing the following: • Perform a remote VT-1 of stud insert #30 (Unit 2 only) concurrent with the volumetric examination once every 10 years to confirm no additional loss of bearing surface area.	Starting with the current (Third Interval) 10-year ASME Code Section XI inspection interval	B2.1.3	NOC-AE-12002830, April 17, 2012 NOC-AE-1 3003041, October 28, 2013
43	The seal cap enclosures from Unit 2 safety injection system check valve SI0010A and from Unit 1 and Unit 2 chemical volume control system check valves CV0001, CV0002, CV0004, and CV0005 will be permanently removed. After removal of the seal cap enclosures, the component bolting will be replaced or inspected for intergranular SCC.	Completed (Unit 1) Completed (Unit 2)	B2.1.7	NOC-AE-12002855, May 14, 2012 NOC-AE-1 3003041, October 28, 2013

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				NOC-AE-14003141, June 3, 2014 NOC-AE-14003180, October 22, 2014
44	The Selective Leaching of Aluminum Bronze program will: • Replace all aluminum bronze castings susceptible to selective leaching, including attachment welds related to the castings with material that is not susceptible to selective leaching. • Replace aluminum bronze root valve adapter socket welds with material that is not susceptible to selective leaching. • Replace extruded piping tees with aluminum bronze weld repairs where the repair size is such that failure of the repair would affect the structural integrity of the component. Enhance the Selective Leaching of Aluminum Bronze procedure to: • Specify loss of material due to selective leaching is monitored through system walkdowns and destructive examinations. • Specify cracking associated with selective leaching is monitored through volumetric examination and destructive examination. • Specify phase distribution to verify the potential for continuous selective leaching is monitored through destructive examination. • Verify the management of cracking of the above ground weld population with no backing rings by performing a one-time volumetric examination on 20 percent with a maximum of 25 welds prior to the period of extended operation. • Specify, if a weld indication that does not meet the acceptance criteria is found during the one-time inspection of welds with no backing rings, periodic volumetric examinations of 20 percent with a maximum of 25 welds will be performed every 10 years thereafter. • Verify, the management of cracking of the above ground weld population with backing rings by performing periodic volumetric examinations on 20 percent with a maximum of 25 welds prior to the period of extended operation and every 10 years thereafter. • Specify, the samples for volumetric examination be selected from the total population of above ground welds, considering construction and size distributions. • Verify, the management of loss of material due to selective leaching and microstructure phase distribution of the above ground weld population with and without backing rings by performing a one-time destructive examination on 20 percent	Replacements and inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later. Procedure changes no later than the date the renewed operating licenses are issued.	B2.1.37	NOC-AE-1 2002942, December 19, 2012 NOC-AE-12002853, May 31, 2012 NOC-AE-12002889, October 4, 2012 NOC-AE-14003141, June 3, 2014 NOC-AE-14003135, July 31, 2014 NOC-AE-14003180, October 22, 2014 NOC-AE-1 5003278, July 29, 2015 NOC-AE-16003365, May 4, 2016 NOC-AE-16003394 July 28, 2016 NOC-AE-16003428 January 12, 2017 NOC-AE-17003454 March 30, 2017 NOC-AE-17003475 May 2, 2017

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	with a maximum of 25 welds with backing rings and 20 percent with a maximum of 25 welds without backing rings prior to the period of extended operation. • Require the sample population for destructive examinations be selected from the total population of welds with and without backing rings, construction and size distributions. • Require a weld which does not meet the acceptance criteria or has through wall leakage, be removed and destructively examined to determine extent of cracking, extent of selective leaching, and the microstructure phase distribution. • Require a weld which does not meet the acceptance criteria or has through wall leakage, be documented in the corrective action program, and a structural integrity analysis be performed to confirm that the load carrying capacity of the installed welds remain adequate to support the intended function of the ECW system through the period of extended operation. • Require an external surface examination capable of detecting selective leaching will be performed on the buried ECW piping welds in the vicinity of degraded coatings to detect loss of material due to selective leaching. • Require that the history of the volumetric, TOFD UT, and destructive examinations results be maintained and a review be performed to identify potential adverse trends or other indications requiring action. • Specify, the acceptance criterion for volumetric examination of aluminum bronze welds is no detected planar indication that is surface connected (exposed to the ECW environment) unless the depth of the indication is contained within the 80 percent of the weld root pass region. An indication not connected to the surface (not exposed to the ECW environment) is acceptable. • Specify, the acceptance criterion for visual inspection of the aluminum bronze welds and adjacent copper alloy piping during the walkdowns is no through wall leakage. • Specify, the acceptance criterion for destructive examinations is no loss of material due to selective leaching penetrating 80 percent of the root-pass region and non-propagating (surrounded by a resistant phase distribution). The microstructure of the weld root region exhibits a non-continuous phase distribution consistent with the metallurgical technical basis report. • Specify, the acceptance criterion for destructive examinations is: – No loss of material due to selective leaching penetrating 80% of the root-pass region. – Found selective leaching is non-propagating (surrounded by resistant phase distribution). – The microstructure of the weld root region shall exhibit a resistant phase distribution consistent with the metallurgical technical basis report.			

No.	Commitment	Implementation Schedule	LRA Section	Reference Letter & Date
	• Specify, the acceptance criterion for TOFD UT examination is no loss of material due to selective leaching resulting in not meeting ASME Section XI Code required margins imposed by ASME Section XI structural factors for normal/upset and emergency/faulted conditions. • Specify, discovery of selective leaching or continuous microstructure phase distribution that do not meet the acceptance criteria but the welds meet structural integrity requires performing the following: − Five TOFD UT examinations within 60 days for each weld not meeting acceptance criteria until no additional weld not meeting the acceptance criteria is found. Welds for examination will be selected from the total population of above ground welds associated with the weld type (with or without backing ring) considering variability of construction, size distributions, structural integrity margins, and consequence of failure. − Periodic TOFD UT monitoring every 5 years of any welds not removed and previously found to not meet acceptance criterion but met structural integrity capability. These welds shall be monitored until 3 consecutive examinations identify no additional propagation of the selective leaching. − Periodic TOFD examinations of an additional 10% sample of the remaining above ground weld types every 5 years. The sample will be selected from the total population of above ground welds associated with the weld type (with or without backing ring) not meeting acceptance criteria, considering construction, size distributions, structural integrity margins, and consequence of failure. − A structural integrity evaluation on a weld not meeting acceptance criteria to confirm that the load carrying capacity of the installed welds remain adequate to support the intended function of the ECW system through the period of extended operation. − An AMP effectiveness evaluation to determine program changes required to manage the aging. • Specify, discovery of loss of material due to selective leaching resulting in a weld not meeting ASME Section XI Code required margins with the weld declared operable per station Operability, Functionality, and Reportability procedure − An extent of condition evaluation to identify other locations requiring examination. These additional examinations will focus on stress margin locations less than or equal to that of the structurally unacceptable weld. − Monthly walkdowns of above ground aluminum bronze welds. − Monthly yard walkdowns to verify no through-wall leakage is occurring. − Performing TOFD UT examinations on the remaining above ground weld population using a sample with a 95/95 confidence until no additional weld			

No.	Commitment	Implementation Schedule	LRA Section	Reference Letter & Date
	indication not meeting the TOFD UT examination acceptance criteria and within structural integrity is found. The weld population used to determine the 95/95 confidence sample will be based on the above ground weld types (with or without backing rings) and locations that would not meet code allowable margins when evaluated against the failed components' degraded load carrying capability. The TOFD UT examinations will be prioritized by examining the weld locations with the least structural integrity margin and with the highest consequence of failure first. Planning and preparations for performing TOFD UT extent of condition examinations will commence upon discovery of the condition. The examinations will commence at the next ECW train outage and will sequence through all the ECW trains during each ECW train outage with at least 20% of the examinations being completed within 30 days and all examinations completed within 180 days. This allows for timely planning and execution of sequenced train by train examinations during first available train work windows. – If a second weld is found that does not meet TOFD UT examination acceptance criteria structural integrity: o Develop examination plan, schedule, and bases for the examination of the remaining above ground welds. o Perform TOFD UT examinations on 100 percent of the remaining above ground welds to determine extent of condition with at least 20% of the examinations being completed within 30 days and all examinations completed within 180 days of finding the second weld. o Perform an evaluation of the below ground weld margins to identify locations requiring inspection. The evaluation will focus on below ground locations where structural integrity could be challenged based on the relative stress margins and the inspection results obtained on the above ground structurally unacceptable weld(s). – Performing periodic 95/95 confidence sample TOFD UT examinations every 5 years on the remaining welds which have not been TOFD UT examined. The population used to determine the 95/95 confidence sample will be based on the above ground weld types (with or without backing rings). The sample will be selected from the total population of above ground welds associated with the weld type (with or without backing ring), considering variability of construction, size distributions, structural integrity margins, and consequence of failure. – Repair or replacement program of the susceptible welds within the STP Technical Specification requirements based on the cause of the structural integrity evaluation failure, results of the additional volumetric examinations, and the extent of condition.			

No.	Commitment	Implementation Schedule	LRA Section	Reference Letter & Date
	• Specify, discovery of a weld not meeting ASME Section XI Code required margins with the weld declared inoperable per station Operability, Functionality, and Reportability procedure requires: – If the weld has been removed from service for examination, then the examination results will be used to determine past operability and reportability. – An extent of condition evaluation to determine the cause of the structural integrity evaluation failure and identify weld population requiring examination. – Performing TOFD UT examinations on 100% of the remaining above ground weld population. The TOFD UT examinations will be prioritized by examining the weld locations with the least structural integrity margin and with the highest consequence of failure first. Planning and preparations for performing TOFD UT extent of condition examinations will commence upon discovery of the condition. The examinations will commence at the next ECW train outage and will sequence through all the ECW trains during each ECW train outage with at least 20% of the examinations being completed within 30 days and all examinations completed within 180 days. This allows for timely planning and execution of sequenced train by train examinations during first available train work windows. – An evaluation of the below ground weld margins to identify locations requiring inspection. The evaluation will focus on below ground locations where structural integrity could be challenged based on the relative stress margins and the inspection results obtained on the above ground structurally unacceptable weld(s). All below ground welds where the evaluation shows that the structural integrity could challenge operability will be examined using TOFD UT during the next scheduled refueling outage. – Twice a month above ground walkdowns of the aluminum bronze welds. – Twice a month yard walkdowns to verify no through-wall leakage is occurring. – Repair or replacement of the susceptible weld(s) based on the cause of the structural integrity evaluation failure, results of the additional TOFD UT examinations, and the extent of condition. • Specify, the acceptance criterion for extent of loss of material on the external surface of buried aluminum bronze piping with coating degradation is that upon removal of the selective leaching the minimum wall thickness is maintained. • Specify, corrective action for selective leaching found under depredated ECW buried piping coatings such as surface conditioning is performed until no selective leaching is detected. If unacceptable wall thickness following surface conditioning is found, the buried ECW piping is repaired or replaced.			

No.	Commitment	Implementation Schedule	LRA Section	Reference Letter & Date
45	• Deleted			NOC-AE-1 2002942, December 19, 2012 NOC-AE-12002889, October 4, 2012 NOC-AE-14003135, July 31, 2014
46	Leak rates that could occur upstream of any individual component supplied by the ECW system will be determined to validate the maximum size flaw for which piping can still perform its intended function. • A summary of the results of these leak rates will be provided to the NRC for review.	Completed	N/A	NOC-AE-12002889, October 4, 2012 NOC-AE-1 2002942, December 19, 2012 NOC-AE-14003141, June 3, 2014 NOC-AE-14003135, July 14, 2014 NOC-AE-14003135, July 31, 2014 NOC-AE-14003180, October 22, 2014
47	Unit 1 RWST only: Perform a one time internal tank bottom and side weld inspection to confirm the effectiveness of the corrective actions to repair the leaking tank floor 5 years prior to entering the period of extended operation. The inspection will include VTW: PT: and Vacuum Box (VB) Leak Test of susceptible locations of the floor bottom and side welds to ensure no leaks.	Five years prior to the Period of Extended Operation	B2.1.20	NOC-AE-14003079, February 18, 2014
48	Enhance the Steam Generator Tube Integrity Program procedures to: • Specify perform visual inspections of the steam generator head internal areas (head interior surfaces, divider plate assemblies, tubesheets (primary side) and tube-to-tubesheet welds) for signs of cracking or loss of material. • Specify the frequency of the visual inspections be at least every 72 effective full power months or every third refueling outage whichever results in more frequent inspections. • Procedures will be revised to evaluate the acceptability of any degraded conditions of the divider plate assemblies, tubesheets (primary side), tube to tubesheet welds, and primary head (interior surfaces) on a case-by-case basis.	Complete no later than six months prior to the period of extended operation	B2.1.8	NOC-AE-16003429 January 5, 2017

APPENDIX B

CHRONOLOGY

B. Chronology

This appendix contains a chronological listing of the routine correspondence between the staff of the U.S. Nuclear Regulatory Commission (NRC) (the staff) and the South Texas Project Nuclear Operating Company (STPNOC) (the applicant) and other correspondence regarding the staff's review of the South Texas Project (STP), Units 1 and 2, license renewal application (LRA), Docket Numbers 50-498 and 50-499.

Document Date	Title
10/25/2010	Letter from Powell, G.T., STPNOC, "South Texas Project, Units 1 and 2, Transmittal of License Renewal Application" (Agencywide Document Access and Management System (ADAMS) Accession No. ML103010257)
10/25/2010	STPNOC, "South Texas Project, Units 1 and 2, License Renewal Application" (ADAMS Accession No. ML103010262)
10/28/2010	Letter from Harrison, A.W., STPNOC, "South Texas Project, Units 1 and 2, License Renewal Application Scoping Drawings" (ADAMS Accession No. ML103270165)
11/4/2010	Press Release: "Press Release-10-202: NRC Announces Availability of License Renewal Application for South Texas Project Nuclear Power Plant" (ADAMS Accession No. ML103081029)
11/23/2010	Letter to Powell, G.T., STPNOC, "Receipt and Availability of the License Renewal Application for the South Texas Project Electric Generating Station Units One and Two (LTR)" (ADAMS Accession No. ML103020399)
11/23/2010	*Federal Register* Notice: "FRN: General Notice. Notice of Receipt and Availability of Application for Renewal of South Texas Project, Units 1 and 2" (ADAMS Accession No. ML103020406)
11/23/2010	Letter to Moore, A., Bay City, TX, Public Library, "Maintenance of Reference Materials at the Bay City Public Library Related to the Review of South Texas Project, Units 1 and 2, License Renewal Application" (ADAMS Accession No. ML103090389)
11/23/2010	*Federal Register* Notice: "FRN: General Notice. USNRC STP Nuclear Operating Co. Notice of Receipt and Availability of Application for Renewal of South Texas Project Units 1 and 2" (ADAMS Accession No. ML103360179)
12/3/2010	Email from Daily, J., NRC, to Taplett, K., STPNOC, "South Texas Project LRA – RAI 4.1-1 on missing exemption under 10 CFR 50.12" (ADAMS Accession No. ML103400343)
12/9/2010	Letter to Powell, G.T., STPNOC, "Project Manager Change for the License Renewal of South Texas Project, Units 1 and 2 (TAC No. ME4936)" (ADAMS Accession No. ML103410524)
12/9/2010	Letter from Powell, G.T., STPNOC, "South Texas Project, Units 1 and 2, Response to Request for Additional Information for the South Texas Project License Renewal Application" (ADAMS Accession No. ML103540235)
12/10/2010	Email from Daily, J., NRC, to Taplett, K., STPNOC, "South Texas Project RAI 1.1.4-1 concerning foreign ownership or control, STPEGS license renewal application acceptance review" (ADAMS Accession No. ML103490335)
12/21/2010	Letter from Powell, G.T., STPNOC, "South Texas Project, Units 1 and 2, Response to Request for Additional Information Related to Part 1, Administrative Information, License Renewal Application" (ADAMS Accession No. ML103570142)
1/7/2011	Letter to Powell, G.T., STPNOC, "Determination of Acceptability & Sufficiency for Docketing, Proposed Review Schedule, and Opportunity for a Hearing Regarding the Application from STP Nuclear Operating Company for Renewal of the Operating Licenses for South Texas Project Electric Gene" (ADAMS Accession No. ML103420531)
1/7/2011	*Federal Register* Notice: "Notice of Acceptance for Docketing of the Application and Notice of Opportunity for Hearing Regarding Renewal of Facility Operating License Numbers NPF-76 and NPF-80 for an Additional 20-year Period STP Nuclear Operating Company, South Texas Project" (ADAMS Accession No. ML103420650)
2/17/2011	Letter from Harrison, A.W., STPNOC, "South Texas Project, Units 1 and 2, License Renewal Application Online Reference Portal" (ADAMS Accession No. ML110610201)

Document Date	Title
3/17/2011	Letter to Powell, G.T., STPNOC, “South Texas Project, Units 1 and 2, License Renewal Application Online Reference Portal” (ADAMS Accession No. ML110620203)
4/5/2011	Letter to Powell, G.T., STPNOC, “Project Manager Change for the License Renewal of South Texas Project, Units 1 and 2 (TAC No. ME4938)” (ADAMS Accession No. ML110872079)
4/14/2011	Letter to Powell, G.T., STPNOC, “Request for Additional Information for the Review of the South Texas Project, License Renewal Application – Section 2.4, ‘Structural’ (TAC Nos. ME4936, ME4937)” (ADAMS Accession No. ML110820579)
4/14/2011	Letter to Powell, G.T., STPNOC, “Request for Additional Information for the Review of the South Texas Project, License Renewal Application – Electrical Branch Scoping” (ADAMS Accession No. ML110890764)
4/4/2011	Letter to Powell, G.T., STPNOC, “Requests for Additional Information for the Review of the South Texas Project, License Renewal Application – Fire Protection and Component Integrity” (ADAMS Accession No. ML110830978)
4/22/2011	Meeting Summary, Daily J.W., “March 29, 2011 and March 31, 2011 Summary of Telephone Conference Calls Held Between NRC and STP Nuclear Operating Company, Concerning Requests for Additional Information Pertaining to the South Texas Project, Units 1 and 2, License Renewal Application” (ADAMS Accession No. ML110940477)
5/5/2011	Letter from Powell, G.T., STPNOC, “South Texas Project, Units 1 and 2, Response to Request for Additional Information for License Renewal Application” (ADAMS Accession No. ML11130A026)
5/5/2011	Letter from Powell, G.T., STPNOC, “South Texas Project, Units 1 and 2, Response to Request for Additional Information for License Renewal Application” (ADAMS Accession No. ML11130A061)
5/12/2011	Letter from Powell, G.T., STPNOC, “South Texas Project, Units 1 and 2, Response to Request for Additional Information for License Renewal Application” (ADAMS Accession No. ML11145A090)
5/24/2011	Letter to Powell, G.T., STPNOC, “Request for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application Future Consideration of Operating Experience” (ADAMS Accession No. ML11137A092)
6/6/2011	Letter to Powell, G.T., STPNOC, “Plan for the Aging Management Program Regulatory Audit Regarding the South Texas Project, Units 1 and 2, License Renewal Application Review (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML11140A163)
6/16/2011	Letter from Powell, G.T., STPNOC, “South Texas Project, Units 1 and 2, Amendment 2 to the License Renewal Application” (ADAMS Accession No. ML11172A096)
6/23/2011	Letter from Powell, G.T., STPNOC, “South Texas Project, Units 1 and 2, Response to Request for Additional Information for License Renewal Application (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML11181A037)
7/5/2011	Letter from Powell, G.T., STPNOC, “South Texas Project, Units 1 and 2, Response to Request for Additional Information for the South Texas Project License Renewal Application” (ADAMS Accession No. ML11193A016)
7/5/2011	Letter from Powell, G.T., STPNOC, “South Texas Project, Units 1 and 2, Response to Request for Additional Information for the Review of the License Renewal Application” (ADAMS Accession No. ML11193A074)
7/6/2011	Meeting Summary, Daily, J.W., STPNOC, “Summary of Telephone Conference Call Held on May 23, 2011, Between the NRC and STPNOC, Concerning Requests for Additional Information Pertaining to the South Texas Project, LRA – Future Consideration of Operating Experience (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML11154A013)
7/12/2011	Letter to Powell, G.T., STPNOC, “Letter re: Request for Additional Information for South Texas Project Electric Generating Station, Units 1 and 2 License Renewal Application – Scoping and Screening Balance of Plant (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML11166A239)
7/28/2011	Letter to Powell, G.T., STPNOC, “Requests for Additional Information for the Review of the South Texas Project, License Renewal Application – Scoping and Screening Audit (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML11201A055)

Document Date	Title
8/4/2011	Letter to Powell, G.T., STPNOC, “Requests for Additional Information for the Review of the South Texas Project, License Renewal Application” (ADAMS Accession No. ML11201A062)
8/9/2011	Letter from Powell, G.T., STPNOC, “South Texas Project, Units 1 and 2, Response to Request for Additional Information for the South Texas Project License Renewal Application (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML11234A045)
8/9/2011	Letter from Powell, G.T., STPNOC, “South Texas Project, Units 1 and 2, Response to Request for Additional Information for the Review of the South Texas Project License Renewal Application” (ADAMS Accession No. ML11245A101)
8/15/2011	Letter to Powell, G.T., STPNOC, “Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2 License Renewal Application – Aging Management Programs Audit, Structures/Electrical (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML11214A005)
8/15/2011	Letter to Powell, G.T., STPNOC, “Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2 License Renewal Application – Aging Management Programs Audit, Reactor Systems (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML11214A027)
8/15/2011	Letter to Powell, G.T., STPNOC, “Request for Additional Information for the Review of the South Texas Project, Units 1 and 2 License Renewal Application – Aging Management Programs Audit, Plant Systems (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML11214A088)
8/16/2011	Meeting Summary, Daily, J.W., STPNOC, “Summary of Telephone Conference Call Held on August 8, 2011, Between the U.S. Nuclear Regulatory Commission and STP Nuclear Operating Company, Concerning Requests for Additional Information Pertaining to the South Texas Project, License Renewal Application” (ADAMS Accession No. ML11222A001)
8/16/2011	Memo to File, Daily, J.W., “Summary of Telephone Conference Call Held on August 9, 2011, Between the U.S. NRC and STP Nuclear Operating Company, Concerning Requests for Additional Information Pertaining to the South Texas Project, License Renewal Application” (ADAMS Accession No. ML11222A002)
8/18/2011	Email to Aldridge, A.J., Taplett, K., STPNOC, “South Texas Project, Units 1 and 2, License Renewal – Errata in AMP RAI Package” (ADAMS Accession No. ML112300016)
8/18/2011	Letter from Powell, G.T., STPNOC, “South Texas Project, Units 1 and 2, Response to Request for Additional Information for the South Texas Project License Renewal Application (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML11238A071)
8/23/2011	Letter from Powell, G.T., STPNOC, “South Texas Project, Units 1 and 2, Response to Request for Additional Information for the South Texas Project License Renewal Application (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML11238A072)
8/23/2011	Letter from Powell, G.T., STPNOC, “South Texas Project, Units 1 and 2, Response to Request for Additional Information for License Renewal Application” (ADAMS Accession No. ML11250A067)
8/31/2011	Letter from Harrison, A.W., STPNOC, “South Texas Project, Units 1 and 2, Transmittal of Documents to Support Review of the South Texas Project License Renewal Application” (ADAMS Accession No. ML11256A056)
8/31/2011	Letter from Harrison, A.W., STPNOC, “Documents to Support Review of the South Texas Project License Renewal Application, List of Transmitted Documents Including Copy of Each Document, Enclosure to NOC-AE-11002720” (ADAMS Accession No. ML11256A057)
9/6/2011	Letter to Powell, G.T., STPNOC, “Scoping and Screening Audit Report Regarding the South Texas Project, Units 1 and 2 (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML11230A003)
9/6/2011	Letter from Powell, G.T., STPNOC, “South Texas Project, Units 1 and 2, Response to Request for Additional Information for the License Renewal Application” (ADAMS Accession No. ML11255A211)
9/12/2011	Letter from Powell, G.T., STPNOC, “South Texas Project, Units 1 and 2, Response to Requests for Additional Information for the License Renewal Application” (ADAMS Accession No. ML11259A014)

Document Date	Title
9/12/2011	Letter from Powell, G.T., STPNOC, "South Texas Project, Units 1 and 2, Transmittal of Document to Support Review of the License Renewal Application" (ADAMS Accession No. ML11259A031)
9/15/2011	Letter from Powell, G.T., STPNOC, "South Texas Project, Units 1 and 2, Response to Request for Additional Information for License Renewal Application (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML11266A019)
9/15/2011	Letter from Powell, G.T., STPNOC, "South Texas Project, Units 1 and 2, Response to Requests for Additional Information for License Renewal Application (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML11266A020)
9/21/2011	Letter to Powell, G.T., STPNOC, "Request for Additional Information for the Review of the South Texas Project, Units 1 and 2 License Renewal Application – Aging Management Review, Set 2 (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML112440201)
9/22/2011	Letter to Powell, G.T., STPNOC, "Aging Management Programs Audit Report Regarding the South Texas Project, Units 1 and 2, Station License Renewal Application (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML11224A265)
9/22/2011	Letter to Powell, G.T., STPNOC, "Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Aging Management Review, Set 1 (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML11250A043)
9/22/2011	Letter to Powell, G.T., STPNOC, "Request for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Aging Management Review, Set 3 (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML11258A161)
10/4/2011	Meeting Summary, Daily, J.W., "Summary of Telephone Conference Call Held on September 28, 2011, Between the NRC and STP Nuclear Operating Company, Concerning Request for Additional Information Pertaining to the South Texas Project, LRA (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML11272A165)
10/10/2011	Letter from Powell, G.T., STPNOC, "South Texas Project, Units 1 and 2, Response to Requests for Additional Information for the License Renewal Application" (ADAMS Accession No. ML11291A152)
10/11/2011	Letter to Powell, G.T., STPNOC, "Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Aging Management Review, Set 4 (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML11273A008)
10/11/2011	Letter to Powell, G.T., STPNOC, "Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Aging Management Review, Set 5 (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML11273A017)
10/14/2011	Meeting Summary, Daily, J.W., "Summary of Telephone Conference Call Held on October 11, 2011, Between the USNRC and STP Nuclear Operating Company, Concerning Request for Additional Information Pertaining to the South Texas Project, License Renewal Application" (ADAMS Accession No. ML11286A002)
10/18/2011	Letter to Powell, G.T., STPNOC, "Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Aging Management Program, Set 6" (ADAMS Accession No. ML11277A047)
10/18/2011	Meeting Summary, Daily, J.W., "Summary of Teleconference Call Held on October 6, 2011, Between the USNRC and STP Nuclear Operating Company, Concerning Request for Additional Information Pertaining to the STP, License Renewal Application" (ADAMS Accession No. ML11286A001)
10/18/2011	Letter from Powell, G.T., STPNOC, "South Texas Project, Units 1 and 2, Supplement to License Renewal Application (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML11298A082)
10/18/2011	Letter from Powell, G.T., STPNOC, "South Texas Project, Units 1 and 2, Response to Requests for Additional Information for License Renewal Application (TAC Nos. ME4938 and ME5122)" (ADAMS Accession No. ML11298A085)
10/25/2011	Meeting Summary, Daily, J.W., "Summary of Teleconference Call Held on September 7, 2011, Between the USNRC and STP Nuclear Operating Co., Concerning Requests for Additional Information Pertaining to the South Texas Project, License Renewal Application" (ADAMS Accession No. ML11250A129)

Document Date	Title
10/25/2011	Letter from Rencurrel, D.W., STPNOC, "South Texas Project, Units 1 and 2, Response to Requests for Additional Information for License Renewal Application (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML11305A076)
10/26/2011	Letter from Harrison, A.W., STPNOC, "South Texas Project, Units 1 and 2, Contact Information Change, License Renewal Application (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML11305A075)
11/3/2011	Letter to Rencurrel, D.W., STPNOC, "Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Aging Management Program, Set 7" (ADAMS Accession No. ML11299A105)
11/3/2011	Letter from Harrison, A.W., STPNOC, "South Texas Project, Units 1 and 2, License Renewal Application Revised Scoping Drawings" (ADAMS Accession No. ML11318A121)
11/4/2011	Letter from Rencurrel, D.W., STPNOC, "South Texas Project, Units 1 and 2, Supplement to the License Renewal Application" (ADAMS Accession No. ML11319A026)
11/4/2011	Letter from Rencurrel, D.W., STPNOC, "South Texas Project, Units 1 and 2, Response to Requests for Additional Information for the Renewal Application" (ADAMS Accession No. ML11325A192)
11/9/2011	Meeting Summary, Daily, J.W., "Summary of Teleconference Call Held on October 31, 2011, Between USNRC and STP Nuclear Operating Company, Concerning Requests for Additional Information Pertaining to the South Texas Project, License Renewal Application, Set 7" (ADAMS Accession No. ML11307A202)
11/15/2011	Letter to Rencurrel, D.W., STPNOC, "Requests for Additional Information for the Review of South Texas Project, Units 1 and 2, License Renewal Application – Aging Management Program, Set 8" (ADAMS Accession No. ML11306A155)
11/17/2011	Letter from Rencurrel, D.W., STPNOC, "South Texas Project Units 1 and 2, Response to Requests for Additional Information for the License Renewal Application, Set 6 (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML11333A093)
11/21/2011	Letter from Rencurrel, D.W., STPNOC, "South Texas Project Units 1 and 2, Response to Requests for Additional Information for the South Texas Project License Renewal Application – Aging Management Review, Set 2 (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML11333A095)
11/21/2011	Letter from Rencurrel, D.W., STPNOC, "South Texas Project, Units 1 and 2, Response to Requests for Additional Information (Set 5) for the License Renewal Application" (ADAMS Accession No. ML11334A047)
11/21/2011	Letter from Rencurrel, D.W., STPNOC, "South Texas Project, Units 1 and 2, Response to Requests for Additional Information for the License Renewal Application" (ADAMS Accession No. ML11335A131)
11/30/2011	Letter from Rencurrel, D.W., STPNOC, "South Texas Project, Units 1 and 2—Annual Update to the South Texas Project License Renewal Application" (ADAMS Accession No. ML11335A140)
12/6/2011	Letter to Rencurrel, D.W., STPNOC, "Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Aging Management, Set 9 (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML11312A176)
12/6/2011	Meeting Summary, Daily, J.W., "Summary of Telephone Conference Call Held on November 17, 2011, Between the U.S. Nuclear Regulatory Commission and STP Nuclear Operating Company, Concerning Clarifications to Some Responses to Requests for Additional Information – South Texas Project" (ADAMS Accession No. ML11335A076)
12/6/2011	Letter from Rencurrel, D.W., STPNOC, "South Texas Project, Units 1 and 2, Response to Requests for Additional Information for the License Renewal Application Aging Management Program, Set 7 (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML11346A012)
12/7/2011	Letter from Rencurrel, D.W., STPNOC, "South Texas Project Units 1 and 2, Supplement to the South Texas Project License Renewal Application (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML11347A365)
12/8/2011	Letter from Rencurrel, D.W., STPNOC, "South Texas Project, Units 1 and 2, Supplement to the License Renewal Application" (ADAMS Accession No. ML11354A087)

Document Date	Title
12/14/2011	Letter to Rencurrel, D.W., STPNOC, "Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Aging Management, Set 10 (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML11332A100)
12/15/2011	Letter from Rencurrel, D.W., STPNOC, "South Texas Project, Units 1 and 2, Response to Requests for Additional Information for the Renewal Application" (ADAMS Accession No. ML11362A080)
12/15/2011	Letter from Rencurrel, D.W., STPNOC, "South Texas Project, Units 1 and 2, Response to Request for Additional Information to the License Renewal Application" (ADAMS Accession No. ML11362A081)
1/5/2012	Letter from Powell, G.T., STPNOC, "South Texas Project, Units 1 and 2, Response to Requests for Additional Information for License Renewal Application Aging Management, Set 9" (ADAMS Accession No. ML12013A206)
1/10/2012	Letter from Harrison, A.W., STPNOC, "South Texas Project, Units 1 and 2, Clarification of Information in Support of the Review of the License Renewal Application" (ADAMS Accession No. ML12011A188)
1/10/2012	Letter from Harrison, A.W., STPNOC, "South Texas Project, Units 1 and 2, License Renewal Application, Revised Scoping Drawing" (ADAMS Accession No. ML120470225)
1/18/2012	Letter from Rencurrel, D.W., STPNOC, "South Texas Project, Units 1 and 2, Response to Requests for Additional Information for the License Renewal Application, Aging Management Program, Set 10" (ADAMS Accession No. ML12020A072)
1/26/2012	Letter from Rencurrel, D.W., STPNOC, "South Texas Project, Units 1 and 2, Transmittal of Errata Associated with the South Texas Project License Renewal Application" (ADAMS Accession No. ML12033A155)
1/30/2012	Letter to Rencurrel, D.W., STPNOC, "Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Aging Management, Set 11 (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML12030A164)
2/6/2012	Letter from Rencurrel, D.W., STPNOC, "South Texas Project, Units 1 and 2, Response to Requests for Additional Information for License Renewal Application, Aging Management Program, Set 10 (RAI 4.7.3-2) (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML12041A170)
2/8/2012	Letter to Rencurrel, D.W., STPNOC, "Request for Additional Information for the Review of the South Texas Project, License Renewal Application – Aging Management, Set 12 (TAC Nos. ME4936 and ME4937) STP RAI, Set 12 Draft" (ADAMS Accession No. ML12009A117)
2/9/2012	Meeting Summary, Daily, J.W., "STP – Record of Conference Call January 10, 2012, Regarding Aluminum-Bronze RAI Responses.docx" (ADAMS Accession No. ML12011A009)
2/15/2012	Letter to Rencurrel, D.W., STPNOC, "Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Aging Management, Set 13 (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML12039A240)
2/16/2012	Meeting Summary, Daily, J.W., "Summary of Telephone Conference Call Held on January 4, 2012, Between the NRC and STP Nuclear Operating Company, Regarding Clarifications on Containment Tendon Prestress and One Request for Additional Information, for the South Texas Project, Units 1 and 2" (ADAMS Accession No. ML12011A008)
2/16/2012	Letter to Rencurrel, D.W., STPNOC, "Plan for the Aluminum Bronze Aging Management Program Regulatory Audit Regarding the South Texas Project, Units 1 and 2, License Renewal Application Review (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML12039A054)
2/16/2012	Letter from Rencurrel, D.W., STPNOC, "South Texas Project, Units 1 and 2, Response to Requests for Additional Information for the South Texas Project License Renewal Application – Aging Management Program, Set 11" (ADAMS Accession No. ML12053A258)
2/27/2012	Letter from Rencurrel, D.W., STPNOC, "South Texas Project, Units 1 and 2, Response to Requests for Additional Information for the South Texas Project License Renewal Application – Aging Management Program, Set 12 (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML12069A024)

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2/28/2012	Letter to Rencurrel, D.W., STPNOC, “Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Aging Management, Set 14 (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12053A430)
3/5/2012	Letter from Rencurrel, D.W., STPNOC, “South Texas Project, Units 1 and 2, Revised Response to Requests for Additional Information for the South Texas Project License Renewal Application – Aging Management Program, Set 10 (RAI 4.7.3-2) (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12073A106)
3/12/2012	Letter from Powell, G.T., STPNOC, “South Texas Project, Units 1 and 2, Response to Requests for Additional Information for the Review of the South Texas Project License Renewal Application” (ADAMS Accession No. ML12079A014)
3/12/2012	Letter from Powell, G.T., STPNOC, “South Texas Project, Units 1 and 2, Response to Requests for Additional Information for the License Renewal Application – Aging Management Program, Set 13” (ADAMS Accession No. ML12079A015)
3/21/2012	Letter to Rencurrel, D.W., STPNOC, “Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Aging Management, Set 15 (TAC Nos. ME4936 and ME4937) STP-RAIs-Set 15-letter” (ADAMS Accession No. ML12065A201)
3/28/2012	Letter from Rencurrel, D.W., STPNOC, “Supplement to the Response to Requests for Additional Information for the South Texas Project License Renewal Application Aging Management Program, Set 12” (ADAMS Accession No. ML12097A063)
3/28/2012	Letter from Rencurrel, D.W., STPNOC, “South Texas Project, Units 1 and 2, Response to Requests for Additional Information for the South Texas Project License Renewal Application – Aging Management Program, Set 14” (ADAMS Accession No. ML12097A064)
3/29/2012	Letter from Rencurrel, D.W., STPNOC, “South Texas Project, Units 1 and 2, Supplemental Response to Requests for Additional Information for the South Texas Project License Renewal Application – Aging Management Program, Set 13 and Set 14” (ADAMS Accession No. ML12097A065)
4/3/2012	Meeting Summary, Daily, J.W., “Summary of Telephone Conference Call Held on January 30, 2012, Between the U.S. Nuclear Regulatory Commission and STP Nuclear Operating Company, Regarding Requests for Information on Flow-Accelerated Corrosion and Others, for the South Texas Project” (ADAMS Accession No. ML12067A243)
4/11/2012	Meeting Summary, Daily, J.W., “STP – Record of Conference Call Held on February 16, 2012, Regarding RAIs-RVIs-31180-etc.” (ADAMS Accession No. ML12080A040)
4/11/2012	Meeting Summary, Daily, J.W., “STP – Record of Conference Call Held on February 9, 2012, Regarding RAIs – TLAAs” (ADAMS Accession No. ML12080A044)
4/11/2012	Meeting Summary, Daily, J.W., “STP – Record of Conference Call Held on March 1, 2012, Regarding RAIs – RV Beltline” (ADAMS Accession No. ML12080A049)
4/11/2012	Meeting Summary, Daily, J.W., “STP – Record of Conference Call Held on March 8, 2012, Regarding RAIs-SGTI-RxHeadStudClosures” (ADAMS Accession No. ML12080A055)
4/12/2012	Letter to Rencurrel, D.W., STPNOC, “Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Aging Management, Set 16 (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12087A141)
4/17/2012	Letter from Rencurrel, D.W., STPNOC, “Response to Requests for Additional Information for the South Texas Project License Renewal Application – Aging Management Program, Set 13 (Supplemental) and Set 15 (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12114A231)
4/19/2012	Meeting Summary, Daily, J.W., “Summary of Telephone Conference Call Held January 18, 2012, Between the U.S. Nuclear Regulatory Commission and STP Nuclear Operating Company, Regarding Applicant Response Related to High-Strength Bolts, for the South Texas Project, Units 1 and 2” (ADAMS Accession No. ML12067A127)

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4/19/2012	Meeting Summary, Daily, J.W., “Summary of Telephone Conference Call Held on February 22, 2012, Between the U.S. Nuclear Regulatory Commission and STP Nuclear Operating Company, Regarding Requests for Additional Information, Set 14” (ADAMS Accession No. ML12080A035)
4/19/2012	Meeting Summary, Daily, J.W., “STP – Record of Conference Call Held on February 6, 2012, Regarding NRC Management Concerns on Open Items” (ADAMS Accession No. ML12080A042)
4/26/2012	Letter from Rencurrel, D.W., STPNOC, “South Texas Project, Units 1 and 2, Supplemental Response to Request for Additional Information Item B2.1.30-1 for License Renewal Application (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12124A227)
5/3/2012	Letter from Richards, K.D., STPNOC, “Response to Requests for Additional Information (RAI) 3.3.2.2.12.2-1 for the South Texas Project License Renewal Application” (ADAMS Accession No. ML12135A224)
5/10/2012	Meeting Summary, Daily, J.W., “STP – Record of Conference Call Held on April 16, 2012, Regarding Possible Confirmatory Items” (ADAMS Accession No. ML12115A272)
5/10/2012	Letter from Rencurrel, D.W., STPNOC, “Response to Requests for Additional Information (RAI) B2.1.9-3 (Supplement) for the South Texas Project License Renewal Application” (ADAMS Accession No. ML12138A065)
5/14/2012	Letter to Rencurrel, D.W., STPNOC, “Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Aging Management, Set 18 (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12124A094)
5/14/2012	Letter from Rencurrel, D.W., STPNOC, “Response to Requests for Additional Information (RAI) B3.2.2.1-1a for the South Texas Project License Renewal Application” (ADAMS Accession No. ML12139A131)
5/16/2012	Meeting Summary, Daily, J.W., “Summary of Telephone Conference Call Held on April 9, 2012, Concerning Request for Additional Information Pertaining to the South Texas Project License Renewal Application” (ADAMS Accession No. ML12101A027)
5/16/2012	Meeting Summary, Daily, J.W., “Record of Conference Call Held on April 24, 2012, Regarding HX Fouling from Upstream Internal Lining Degradation” (ADAMS Accession No. ML12122A002)
5/22/2012	Letter to Rencurrel, D.W., STPNOC, “Schedule Revision for the Safety Review of the South Texas Project, Units 1 and 2, License Renewal Application (TAC ME4936 and ME4937)” (ADAMS Accession No. ML12129A060)
5/22/2012	Letter to Rencurrel, D.W., STPNOC, “Request for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Aging Management, Set 19 (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12143A031)
5/31/2012	Letter from Rencurrel, D.W., STPNOC, “South Texas Project, Units 1 and 2 – Response to Requests for Additional Information for the License Renewal Application Aging Management Program, Set 16” (ADAMS Accession No. ML12160A068)
5/31/2012	Letter from Rencurrel, D.W., STPNOC, “South Texas Project, Units 1 and 2, Response to Requests for Additional Information B2.1.9-1a License Renewal Application (TAC Nos. ME4936 and ME49371)” (ADAMS Accession No. ML12160A069)
5/31/2012	Letter from Rencurrel, D.W., STPNOC, “Response to Requests for Additional Information for the South Texas Project License Renewal Application, Aging Management Program, Set 19 (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12160A073)
5/31/2012	Letter from Rencurrel, D.W., STPNOC, “South Texas Project, Units 1 and 2 – Response to Requests for Additional Information (RAI) B2.1.9-1 for the License Renewal Application” (ADAMS Accession No. ML12163A332)
5/31/2012	Letter from Rencurrel, D.W., STPNOC, “South Texas Project, Units 1 and 2, Response to Requests for Additional Information for the License Renewal Application, Aging Management Program, Set 16” (ADAMS Accession No. ML12163A333)
5/31/2012	Letter from Rencurrel, D.W., STPNOC, “South Texas Project, Units 1 and 2, Response to Requests for Additional Information for the License Renewal Application, Aging Management Program, Set 19” (ADAMS Accession No. ML12163A334)

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6/7/2012	Letter from Rencurrel, D.W., STPNOC, “South Texas Project, Units 1 and 2 – Supplemental Response to Requests for Additional Information for License Renewal Application” (ADAMS Accession No. ML12167A263)
6/14/2012	Letter to Rencurrel, D.W., STPNOC, “STP RAIs, Set 21 RAI for the review of the South Texas Project, Units 1 & 2, License Renewal Application – Aging Management” (ADAMS Accession No. ML12157A227)
6/14/2012	Letter from Rencurrel, D.W., STPNOC, “Response to Requests for Additional Information (RAI) B1.4-3 for the South Texas Project License Renewal Application” (ADAMS Accession No. ML12174A340)
6/15/2012	Letter to Rencurrel, D.W., STPNOC, “Report Regarding the Follow-up Audit of the Selective Leaching of Aluminum Bronze Aging Management Program for the South Texas Project, Units 1 and 2, License Renewal Application (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12165A239)
6/25/2012	Letter to Rencurrel, D.W., STPNOC, “Request For Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Aging Management, Set 20 (TAC Nos. ME4939 and ME4937)” (ADAMS Accession No. ML12144A443)
6/27/2012	Letter from Rencurrel, D.W., STPNOC, “South Texas Project, Units 1 and 2, Supplemental Response to Requests for Additional Information for the South Texas Project License Renewal Application Aging Management Program, Set 19” (ADAMS Accession No. ML12200A035)
7/5/2012	Letter from Rencurrel, D.W., STPNOC, “South Texas Project, Units 1 and 2, Response to Requests for Additional Information (RAI) B2.1.9-4b for the South Texas Project License Renewal Application” (ADAMS Accession No. ML12200A034)
7/9/2012	Meeting Summary, Daily, J.W., “Summary of Telephone Conference Call Held on June 6, 2012, Between the U.S. Nuclear Regulatory Commission and STP Nuclear Operating Company, Concerning Future Use of Operating Experience – Followup, Pertaining to the South Texas Project, Units 1 and 2, LRA” (ADAMS Accession No. ML12165A608)
7/12/2012	Letter to Rencurrel, D.W., STPNOC, “Request for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Aging Management, Set 22 (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12185A031)
7/17/2012	Letter from Rencurrel, D.W., STPNOC, “Response to Requests for Additional Information for the South Texas Project License Renewal Application Aging Management Program, Set 20 (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12208A095)
7/26/2012	Letter to Rencurrel, D.W., STPNOC, “RAI for the Review of the STP, Units 1 and 2, LRA – Aluminum Bronze, Set 23 (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12201B541)
7/31/2012	Letter from Rencurrel, D.W., STPNOC, “South Texas Project, Units 1 and 2, Amendment to License Renewal Application in Response to LR-ISG-2011-01” (ADAMS Accession No. ML12222A010)
8/7/2012	Meeting Summary, Daily, J.W., “Summary of Telephone Conference Call Held on July 3, 2012, Between the U.S. NRC and STP Nuclear Operating Company, Concerning Debris from Coating Failures – Followup, Pertaining to the South Texas Project, Units 1 and 2, LRA (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12206A303)
8/7/2012	Meeting Summary, Daily, J.W., “Summary of Telephone Conference Call Held on July 24, 2012, Between the US NRC and STP Nuclear Operating Company, Concerning Selective Leaching of Aluminum Bronze – Followup, Pertaining to the South Texas Project, Units 1 and 2, LRA (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12208A020)
8/15/2012	Meeting Notice, Daily, J.W., “Forthcoming Meeting with STP Nuclear Operating Company Regarding License Renewal for the South Texas Project, Units 1 and 2” (ADAMS Accession No. ML12226A455)
8/21/2012	Letter from Rencurrel, D.W., STPNOC, “South Texas Project, Units 1 and 2, Response to Requests for Additional Information for License Renewal Application Aging Management Program, Set 22 (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12248A148)

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9/12/2012	Meeting Summary, Daily, J.W., “Summary of Telephone Conference Call Held on August 9, 2012, Between NRC and STPNOC, Concerning Clarification of One Part of Response to RAI 4.3-13, Pertaining to the STP, Units 1 and 2, LRA (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12227A560)
9/27/2012	Letter to Rencurrel, D.W., STPNOC, “Schedule Revision for the Safety Review of the South Texas Project, Units 1 and 2, License Renewal Application (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12254A057)
10/3/2012	Letter to Rencurrel, D.W., STPNOC, “Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – RWST Cracking, Set 24 (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12256B049)
10/4/2012	Letter from Rencurrel, D.W., STPNOC, “South Texas Project, Units 1 and 2, Response to Requests for Additional Information for Review of License Renewal Application – Aluminum Bronze, Set 23 (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML122920722)
10/23/2012	Meeting Summary, Daily, J.W., “August 27, 2012, Summary of Meeting Held on Between NRC and STPNOC Representatives to Discuss the STP, LRA (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12270A469)
10/29/2012	Letter from Rencurrel, D.W., STPNOC, “South Texas Project, Units 1 and 2 – Annual Update to the South Texas Project License Renewal Application” (ADAMS Accession No. ML12313A011)
11/19/2012	Letter to Rencurrel, D.W., STPNOC, “Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Set 25 (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12311A438)
12/6/2012	Letter from Rencurrel, D.W., STPNOC, “South Texas Project, Units 1 and 2, Partial Response to Requests for Additional Information for the Review of the License Renewal Application – Set 25” (ADAMS Accession No. ML12359A063)
12/11/2012	Letter from Rencurrel, D.W., STPNOC, “Clarification to Requests for Additional Information for the South Texas Project License Renewal Application Aging Management Program, Set 13 (Supplemental) (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12361A024)
12/11/2012	Letter from Rencurrel, D.W., STPNOC, “South Texas Project Units 1 & 2, Request for NRC Staff to Suspend Safety Review of the South Texas Project License Renewal Application (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12359A062)
12/18/2012	Letter from Daily, J., to Rencurrel, D.W., “RAI for the Review of the STP, Units 1 and 2, LRA – Set 26 (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12333A227)
12/19/2012	Letter from Powell, G.T., STPNOC, “South Texas Project Units 1 and 2, Supplement 1 to Request for NRC Staff to Suspend Safety Review of South Texas Project License Renewal Application (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML12363A102)
1/10/2013	Letter from Rencurrel, D.W., STPNOC, “South Texas Project, Units 1 and 2, Supplement 2 to Request for NRC Staff to Suspend Safety Review of License Renewal Application” (ADAMS Accession No. ML13024A413)
2/15/2013	Letter from Lubinski, J.W., NRC, “Safety Evaluation Report Related to the License Renewal of South Texas Project, Units 1 and 2” (ADAMS Accession No. ML13044A115)
2/26/2013	Letter from Lubinski, J.W., NRC “Requested Schedule Suspension for the Safety Review of the South Texas Project, Units 1 and 2, License Renewal Application (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML13052A382)
10/28/2013	Letter from Powell, G.T., “South Texas Project, Units 1 and 2, Docket Nos. STN 50-498, STN 50-499, 2013 Annual Update to the South Texas Project License Renewal Application (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML13310A311)
12/17/2013	Letter from Murray, M.P., “South Texas Project, Units 1 and 2, Docket Nos. STN 50-498, STN 50-499, Review of License Renewal Application Safety Evaluation with Open Items (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML14002A157)
12/31/2013	Letter from Powell, G.T., “South Texas Project, Units 1 and 2, Docket Nos. STN 50-498, STN 50-499, Review of License Renewal Application Safety Evaluation with Open Items – Proposed License Conditions (TAC Nos. ME4936 and ME4937)” (ADAMS Accession No. ML14015A313)

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2/18/2014	Letter from Powell, G.T., "South Texas Project, Units 1 and 2, Docket Nos. STN 50-498, STN 50-499, Response to Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – RWST Cracking, Set 24 (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML14069A169)
2/27/2014	Letter from Powell, G.T., "South Texas Project, Units 1 and 2, Docket Nos. STN 50-498, STN 50-499, Response to Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Set 25 (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML14073A456)
3/6/2014	Letter from Daily, J.W., "Request for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Set 27 (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML14050A172)
4/8/2014	Meeting Summary, Daily, J.W., "Summary of Meeting held on November 13, 2013, between the U.S. Nuclear Regulatory Commission Staff and STP Nuclear Operation Company Representatives to Discuss the South Texas Project, License Renewal Application (TAC Nos. ME4936, ME4937)" (ADAMS Accession No. ML14010A014)
6/3/2014	Letter from Powell, G.T., "South Texas Project, Units 1 and 2, Docket Nos. STN 50-498, STN 50-499, Response to Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Set 27 (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML14163A020)
6/26/2014	Letter from Powell, G.T., "South Texas Project, Units 1 and 2, Docket Nos. STN 50-498, STN 50-499, Review of Final License Renewal Interim Staff Guidance LR-ISG-2011-03, 'Changes to the Generic Aging Lessons Learned (GALL) Report Revision 2 Aging Management Program XI.M41,' 'Buried and Underground Piping and Tanks,' and update to License Renewal Application (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML14191A632)
7/16/2014	Letter from Daily, J.W., "Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Set 28 (TAC Nos. ME4936 and ME4937)," (ADAMS Accession No. ML14183B719)
7/31/2014	Letter from Powell, G.T., "South Texas Project, Units 1 and 2, Docket Nos. STN 50-498, STN 50-499, Response to Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Set 26 (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML14224A151)
9/30/2014	Letter from Powell, G.T., "South Texas Project, Units 1 and 2, Docket Nos. STN 50-498, STN 50-499, Response to Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Set 28 (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML14302A069)
10/22/2014	Letter from Powell, G.T., "South Texas Project, Units 1 and 2, Docket Nos. STN 50-498, STN 50-499, 2014 Annual Update to the South Texas Project License Renewal Application (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML14308A073)
2/5/2015	Meeting Summary, Daily, J.W., "Summary of Telephone Conference Call Held on January 8, 2015, between the U.S. Nuclear Regulatory Commission and STP Nuclear Operating Company, Concerning RAI Set 29 for the South Texas Project, Units 1 and 2, License Renewal Application (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML15020A349)
2/23/2015	Letter from Daily, J.W., "Plan for the 2015 Selective Leaching of Aluminum-Bronze Aging Management Program Regulatory Audit Regarding the South Texas Project, Units 1 and 2, License Renewal Application Review (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML15026A110)
2/25/2015	Meeting Summary, Daily, J.W., "Summary of Telephone Conference Call (Call No. 2) Held on February 3, 2015, Between the U.S. Nuclear Regulatory Commission and STP Nuclear Operating Company, Concerning RAI Set 29 for the South Texas Project, Units 1 and 2, License Renewal Application (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML15035A280)

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3/24/2015	Letter from Powell, G.T., "South Texas Project, Units 1 and 2, Docket Nos. STN 50-498, STN 50-499, Reference Correction to Request for Additional Information Set 27 – South Texas Project License Renewal Application (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML15097A013)
5/27/2015	Letter from Daily, J.W., "Request for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Set 30 (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML15131A219)
5/28/2015	Letter from Daily, J.W., "Request for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Set 31 (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML15131A272)
6/4/2015	Letter from Daily, J.W., "Selective Leaching of Aluminum Bronze Aging Management Program 2015 Audit Report for the South Texas Project, Units 1 and 2, License Renewal Application (TAC Nos. ME4936 and ME4937)" (ADAMS Accession No. ML15131A145)
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APPENDIX C

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APPENDIX D

REFERENCES

D. References

This appendix contains a listing of the references used in the preparation of the safety evaluation report (SER) prepared during the review of the license renewal application (LRA) for South Texas Project (STP), Units 1 and 2, Docket Numbers 50-498 and 50-499.

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NRC FORM 335 (12-2010) NRCMD 3.7 **BIBLIOGRAPHIC DATA SHEET** (See instructions on the reverse)	U.S. NUCLEAR REGULATORY COMMISSION	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG-2223 Volume 2
2. TITLE AND SUBTITLE Safety Evaluation Report Related to the License Renewal of South Texas Project, Units 1 and 2, Volume 2		3. DATE REPORT PUBLISHED MONTH: November YEAR: 2019
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5. AUTHOR(S) See Appendix C of Manuscript		6. TYPE OF REPORT Technical
		7. PERIOD COVERED (Inclusive Dates)

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same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This safety evaluation report (SER) documents the technical review of the South Texas Project (STP), Units 1 and 2, license renewal application (LRA) by the U.S. Nuclear Regulatory Commission (NRC) staff(the stafi). By letter dated October 25, 2010, South Texas Nuclear Operating Company (STPNOC) submitted the LRA in accordance with Title 10, Part 54, of the Code of Federal Regulations, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants" (10 CFR Part 54). STPNOC requests renewal of the STP operating licenses (Facility Operating License Numbers DPR 76 and DPR 80, respectively) for a period of 20 years beyond the current license periods ending August 20, 2027 (Unit 1), and December 15, 2028 (Unit 2).

Unless otherwise indicated, this SER presents the status of the staffs review of information submitted through May 2, 2017, the cutoff date for consideration in this SER. The open item previously identified in the SER with Open Items, issued October 2016, has been closed (see Section 1.5); therefore, no open items remain to be resolved before the final determination is reached by the staff.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	
Safety Evaluation Report, SER License Renewal, LR License Renewal Application, LRA South Texas Project, Units 1 and 2 South Texas Project STP South Texas Project Nuclear Operating Company, STPNOC STP Nuclear Operating Company	13. AVAILABILITY STATEMENT unlimited 14. SECURITY CLASSIFICATION (This Page) unclassified (This Report) unclassified 15. NUMBER OF PAGES 16. PRICE

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