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NUREG-2193
Volume 2

Safety Evaluation Report

Related to the License Renewal of Davis-Besse Nuclear Power Station

Docket Number 50-346

FirstEnergy Nuclear Operating Company

Sections 4 to 6
Appendices

Office of Nuclear Reactor Regulation

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ABSTRACT

This safety evaluation report (SER) documents the technical review of the Davis-Besse Nuclear Power Station (Davis-Besse) license renewal application (LRA) by the U.S. Nuclear Regulatory Commission (NRC or the staff). By letter dated August 27, 2010, FirstEnergy Nuclear Operating Company (FENOC 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." FENOC requests renewal of Davis-Besse operating license (Facility Operating License Number NPF-3) for a period of 20 years beyond the current license date of April 22, 2017.

Davis-Besse is located approximately 20 miles east of Toledo, Ohio. The NRC issued the construction permit on March 24, 1971. The NRC issued the operating license on April 22, 1977. The unit is a pressurized-water reactor design with a dry ambient containment. Babcock and Wilcox Corporation supplied the nuclear steam supply system, and Bechtel designed and constructed the balance of the plant. The licensed power output of the unit is 2,817 megawatt thermal with a gross electrical output of approximately 908 megawatt electric.

This SER presents the status of the staff's review of information submitted through June 4, 2013, the cutoff date for consideration in the SER. The staff has resolved all issues associated with requests for additional information and closed all open items since publishing the SER with Open Items. The staff did not identify any new open items that must be resolved before any final determination can be made on the LRA.

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ABBREVIATIONS

AC	alternating current
ACAR	aluminum core alloy reinforced
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
ACSR	aluminum core steel reinforced
ADAMS	Agencywide Document Access and Management System
AERM	aging effect requiring management
AFW	auxiliary feedwater
ALARA	as low as is reasonable achievable
AMP	aging management program
AMR	aging management review
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOTC	allowable operating transient cycles
APCSB	Auxiliary and Power Conversion Systems Branch
ART	adjusted reference temperature
ARTS	anticipatory reactor trip system
ASCE	American Society of Civil Engineers
ASM	American Society for Metals
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
AWWA	American Water Works Association
B&W	Babcock and Wilcox
B&WOG	Babcock and Wilcox Owners Group
B-10	Boron-10
BAA	boric acid addition
BTP	branch technical position
BWR	boiling water reactor
BWST	borated water storage tank
C	Celsius
CAL	confirmatory action letter number
CASS	cast austenitic stainless steel
CCW	component cooling water
CEA	control element assembly

CF	chemistry factor
CFR	U.S. Code of Federal Regulations
CLB	current licensing basis
CMAA	Crane Manufacturers Association of America
CMTR	certified material test report
CR	condition report
CRD	control rod drive
CRGT	control rod guide tube
CSA	core support assembly
CSS	core support shield
CST	condensate storage tank
Cu	copper
CUF	cumulative usage factor
CUF_{en}	environmentally-assisted fatigue cumulative usage factor
Davis-Besse	Davis-Besse Nuclear Power Station
DBA	design-basis accident
DBE	design-basis event
DG	diesel generator
DHR	decay heat removal
DLR	Division of License Renewal
DMW	dissimilar metal weld
DWS	demineralized water storage
EAF	environmentally-assisted fatigue
ECCS	emergency core cooling system
ECP	engineering change package
EDG	emergency diesel generator
EFPY	effective full power year
EMA	equivalent margins analysis
EPDM	ethylene-propylene-diene
EPRI	Electrical Power Research Institute
EQ	environmental qualification
ESF	engineered safety features
ESOMS	shift operations management system
F	Fahrenheit
F_{en}	environmentally-assisted fatigue correction factor
FENOC	FirstEnergy Nuclear Operating Company
FERC	Federal Energy Regulatory Commission
FHAR	fire hazards analysis report

FIV	flow-induced vibration
FR	Federal Register
FSAR	final safety analysis report
ft	foot
ft-lb	foot-pound
g/cm^2	grams per square centimeter
GALL	generic aging lessons learned
GCR	general corrosion rate
GEIS	generic environmental impact statement
GL	generic letter
HAZ	heat-affected zone
HELB	high-energy line break
HPI	high-pressure injection
HPSI	high-pressure safety injection
HU/CD	heatup and cooldown
HVAC	heating, ventilation, and air conditioning
I&C	instrumentation and control
IASCC	irradiation-assisted stress corrosion cracking
IGA	intergranular attack
IGSCC	intergranular stress corrosion cracking
ILRT	integrated leak rate tests
IMI	incore monitoring instrumentation
IN	information notice
in.	inch
INPO	Institute of Nuclear Power Operations
IPA	integrated plant assessment
ISA	Instrument Society of America
ISG	interim staff guidance
ISI	inservice inspection
ksi	kips per square inch
kV	kilovolt
LBB	leak-before-break
LCO	limiting condition for operation
LER	licensee event report
LOCA	loss-of-coolant accident
LPI	low-pressure injection

LRA	license renewal application
LTOP	low-temperature overpressure protection
LWR	light-water reactor
M	margin term
MDFP	motor-driven feedwater pump
MEB	metal-enclosed bus
MeV	megaelectron volt
MFW	main feedwater
mg/l	milligrams per liter
mi	mile
MIC	microbiologically-influenced corrosion
MIRVP	Master Integrated Reactor Vessel Program
MIRVSP	Master Integrated Reactor Vessel Surveillance Program
MoS^2	molybdenum disulfide
MRP	Materials Reliability Program
MRPM	Maintenance Rule Program Manual
MUR	measurement uncertainty recapture
MWe	megawatt electric
MWt	megawatt thermal
MWT	Makeup Water Treatment
n/cm^2	Newton per square centimeter
NACE	National Association of Corrosion Engineers
NDE	nondestructive examination
NDT	nondestructive testing
NEI	Nuclear Energy Institute
NEPA	National Environmental Protection Agency
NFPA	National Fire Protection Association
NPS	nominal pipe size
NRC	U.S. Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OE	operating experience
OEM	original equipment manufacturer
OI	open item
OPA	Office of Public Affairs
OTSG	once-through steam generator
P&ID	piping and instrumentation diagram
PM	preventive maintenance

PORV	pilot-operated relief valve
ppb	parts per billion
ppm	parts per million
psi	pounds per square inch
psia	pounds per square inch absolute
psig	pounds per square inch gauge
P-T	pressure-temperature
PTLR	pressure-temperature limits report
PTS	pressurized thermal shock
PWR	pressurized-water reactor
PWSCC	primary water stress corrosion cracking
QA	quality assurance
QAPM	Quality Assurance Program Manual
RAI	request for additional information
RCCA	rod cluster control assembly
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RCSC	Research Council for Structural Connections
RFO	refueling outage
RG	regulatory guide
RIS	regulatory issue summary
RP	regulatory position
RPV	reactor pressure vessel
RT	radiographic testing
RT_{NDT}	nil-ductility reference temperature
RT_{PTS}	reference temperature for pressurized thermal shock
RV	reactor vessel
RVI	reactor vessel internals
RVID	Reactor Vessel Integrity Database
SBO	station blackout
SBODG	station blackout diesel generator
SC	structure and component
SCC	stress corrosion cracking
SE	safety evaluation
SER	safety evaluation report
SFP	spent fuel pool
SG	steam generator

SOC	statement of consideration
SOER	supplemental operating experience report
SRP	Standard Review Plan
SRP-LR	standard review plan-license renewal
SSC	system, structure, and component
SUFP	startup feed pump
TLAA	time-limited aging analysis
TPCW	turbine plant cooling water
TR	technical report
TS	technical specifications
U_{en}	environmentally-adjusted cumulative usage factor
UFSAR	updated final safety analysis report
USACE	U.S. Army Corps of Engineers
USAR	updated safety analysis report
USE	upper-shelf energy
UT	ultrasonic thickness
V	volt
Zn	zinc

SECTION 4

TIME-LIMITED AGING ANALYSES

4.1 Identification of Time-Limited Aging Analyses

This section of the safety evaluation report (SER) addresses the identification of time-limited aging analyses (TLAAs). In Sections 4.2–4.7 of the license renewal application (LRA), FirstEnergy Nuclear Operating Company (FENOC or the applicant) addressed the TLAAs for Davis-Besse Nuclear Power Station, Unit 1 (Davis-Besse). SER Sections 4.2–4.7 document the review of the TLAAs conducted by the staff of the U.S. Nuclear Regulatory Commission (NRC or the staff).

TLAAs are certain plant-specific safety analyses that involve time-limited assumptions defined by the current operating term. Pursuant to Title 10, Section 54.21(c)(1), of the *Code of Federal Regulations* (10 CFR 54.21(c)(1)), applicants must list TLAAs as defined in 10 CFR 54.3, "Definitions."

In addition, pursuant to 10 CFR 54.21(c)(2), applicants must list existing plant-specific exemptions granted in accordance with 10 CFR 50.12, "Specific Exemptions," based on TLAAs. For any such exemptions, the applicant must evaluate and justify the continuation of the exemptions for the period of extended operation.

4.1.1 Summary of Technical Information in the Application

LRA Section 4.1.1 gives the basis for identifying those analyses that need to be evaluated as TLAAs in accordance with 10 CFR 54.21(c)(1). The applicant stated that, for the purpose of meeting this requirement, it evaluated those calculations that met the six criteria for defining an analysis as a TLAA, as specified in 10 CFR 54.3. The applicant stated that its review of the current licensing basis (CLB) included the following documents:

- updated safety analysis report (USAR)
- fire hazards analysis report (incorporated by reference in the USAR)
- Quality Assurance (QA) Program
- Inservice Inspection (ISI) Program
- Inservice Testing Program
- operating license (including technical specifications (TSs))
- exemptions and inspection relief requests
- docketed licensing correspondence
- design calculations and reports (incorporated in the CLB, (e.g., by reference))

In LRA Table 4.1-1, "Time-Limited Aging Analyses," the applicant listed the following TLAA categories:

- reactor vessel (RV) neutron embrittlement
- metal fatigue
- environmental qualification (EQ) of electrical equipment
- concrete containment tendon prestress

- containment fatigue
- other plant-specific TLAAs

LRA Section 4.1.2 provides the applicant's basis for identifying those plant-specific exemptions based on TLAAs, in accordance with 10 CFR 54.21(c)(2). Pursuant to 10 CFR 54.21(c)(2), the applicant stated that it did not identify any exemptions granted as required 10 CFR 50.12 and in effect, or related to 10 CFR Part 50 Appendix R.

4.1.2 Staff Evaluation

LRA Section 4.1.1 documents the applicant's methodology for identifying applicable TLAAs, and LRA Table 4.1-1 provides a list of the TLAAs that are applicable. The staff reviewed the information to determine if the applicant provided sufficient information, pursuant to 10 CFR 54.21(c)(1) and 10 CFR 54.21(c)(2).

As defined in 10 CFR 54.3, TLAAs meet the following six criteria:

(1) involve systems, structures, and components (SSCs) within the period of extended operation, pursuant to 10 CFR 54.4(a)

(2) consider the effects of aging

(3) involve time-limited assumptions defined by the current operating term (for example, 40 years)

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

(5) involve conclusions, or provide the basis for conclusions, related to the capability of the SSC to perform its intended functions, pursuant to 10 CFR 54.4(b)

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

The applicant reviewed the list of potential TLAAs from NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," (SRP-LR) dated September 2005. The applicant listed those potential TLAAs applicable to Davis-Besse in LRA Table 4.1-2, "Review of Generic TLAAs listed in NUREG-1800," and indicated whether the TLAA was applicable to Davis-Besse or not.

The staff noted that the applicant's list of potential TLAAs was assembled using the following regulatory and industry documents and experience:

- NUREG-1800
- Nuclear Energy Institute (NEI) 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR 54—The License Renewal Rule," Revision 6
- Electric Power Research Institute (EPRI) Report TR-105090, "Guidelines to Implement the License Renewal Technical Requirements of 10 CFR 54 for Integrated Plant Assessments and Time-Limited Aging Analyses"
- LRAs for Babcock and Wilcox (B&W) pressurized-water reactor (PWR) designs, other PWR designs that use B&W RVs, and the associated SERs
- recent LRAs for PWRs

The staff finds the applicant’s use of these documents to compile a list of potential TLAAs reasonable because the applicant has used available resources from the NRC, NEI, and past LRAs.

Using the documents listed above, the applicant performed a review of its CLB to determine if the design or analysis feature of each potential TLAA, in fact, exists at Davis-Besse; to ascertain if the feature is included in its licensing basis; and to identify additional potential plant-specific TLAAs. In accordance with 10 CFR 54.21(c), the potential TLAAs that meet all six criteria of a TLAA, as defined in 10 CFR 54.3(a), are actual TLAAs and require a disposition. The applicant reviewed the six criteria based on information in the CLB source documents (as listed above).

The staff finds the applicant’s approach for determining TLAAs reasonable because the applicant performed a comprehensive search through its CLB, based on available staff and industry guidance and experience, and reviewed the potential TLAAs against the six criteria of a TLAA, as defined in 10 CFR 54.3(a).

The staff confirmed that the applicant’s LRA includes the TLAAs that are normally applicable to PWR applications, including the following TLAAs:

- RV neutron embrittlement: upper-shelf energy (USE), pressurized thermal shock (PTS) limits, and pressure-temperature (P-T) limits (neutron fluence is discussed in the LRA as it applies to RV neutron embrittlement TLAAs but it is identified as “not a TLAA”)
- metal fatigue of American Society of Mechanical Engineers (ASME) Code Class 1 components and non-Class 1 components, including effects of reactor water environment on fatigue
- EQ of electrical equipment
- fatigue of the reactor containment vessel

The staff finds the applicant’s identification of these TLAAs acceptable because they are consistent with the TLAAs identified in SRP-LR Sections 4.2, 4.3, 4.4, and 4.6 as being applicable to PWR LRAs.

The staff also confirmed that the LRA included the following additional plant-specific TLAAs:

- leak-before-break (LBB)
- metal corrosion allowance for pressurizer instrument nozzles
- RV thermal shock due to borated water storage tank (BWST) water injection
- high-pressure injection (HPI) and makeup nozzle thermal sleeves
- ISI-fracture mechanics analyses (reactor coolant system Loop 1 cold leg drain line weld overlay repair, once-through steam generator (OTSG) 1-2 flaw evaluations)
- ASME Code Case N-481 evaluation (added by LRA Amendment 13 on August 17, 2011)
- crane load cycles (added by LRA Amendment 19 on October 7, 2011)

The staff confirmed that the applicant’s identification of these additional TLAAs satisfies the recommendation in SRP-LR Section 4.7, which states that the applicant identify any additional analyses for the facilities that meet the definition of a TLAA, in accordance with the requirements of 10 CFR 54.3. The staff did not identify any omissions of TLAAs for this LRA.

Based on its review, the staff concludes that the applicant satisfied the requirements of 10 CFR 54.3 to identify the TLAAs that are applicable to the LRA because the applicant satisfied the TLAA identification guidance and recommendations in SRP-LR Sections 4.2 through 4.7.

The staff confirmed that the TLAAs identified by the applicant as being applicable to the LRA have been evaluated by the applicant against the provisions and criteria of 10 CFR 54.21(c)(1). The staff's evaluations of these TLAAs are summarized in SER Sections 4.2 through 4.7.

As required by 10 CFR 54.21(c)(2), the applicant must list all exemptions granted in accordance with 10 CFR 50.12, based on TLAAs, and evaluate and justify continuation through the period of extended operation. The LRA states that each active exemption was reviewed to determine if it was based on a TLAA. The applicant did not identify any TLAA-based exemptions applicable to the period of extended operation. Based on the information provided by the applicant regarding the process used to identify these exemptions and its results, the staff concludes, in accordance with 10 CFR 54.21(c)(2), that there are no TLAA-based exemptions justified for continuation through period of extended operation.

4.1.2.1 Evaluation of the Applicant's Identification of Time-Limited Aging Analyses

The staff's Statement of Considerations (SOC) on 10 CFR Part 54, as given in *Federal Register* Notice, Volume 60, Number 88, Section III.g.(i), (FRN Volume 60, No. 88, dated May 8, 1995), clarifies when an analysis in the CLB needs to be identified as a TLAA in accordance with the rule. SRP-LR Section 4.1 provides additional guidance as to when an analysis in the CLB needs to be identified as a TLAA in accordance with 10 CFR 54.3.

For each of the TLAAs identified in LRA Table 4.1-1, the staff evaluated the applicant's basis for disposition of these TLAAs, in accordance with 10 CFR 54.21(c)(1)(i), 10 CFR 54.21(c)(1)(ii), or 10 CFR 54.21(c)(1)(iii). The staff's evaluation of the applicant's disposition are documented in SER Sections 4.2, 4.3, 4.4, 4.5, 4.6, and 4.7, and their applicable subsections.

For those analyses in LRA Tables 4.1-1 and 4.1-2 identified as "not TLAAs," the staff reviewed the applicant's basis for its conclusion. Specifically, the staff confirmed that either an existing analysis in the CLB would not need to be identified as a TLAA, or a specific, generic TLAA mentioned in either SRP-LR Table 4.1-2 or Table 4.1-3 was not applicable. If the analysis is addressed in detail in the LRA, such as LRA Section 4.2.1 on Neutron Fluence, then the staff's evaluation is provided in the analogous section of this SER, such as SER Section 4.2.1, "Neutron Fluence." Otherwise, the staff's evaluation is provided in the following subsections.

4.1.2.1.1 Reactor Coolant Pump Flywheel

LRA Table 4.1-2 states that the fatigue analysis of the reactor coolant pump (RCP) flywheels in the SRP-LR is not applicable to the applicant's CLB because it did not identify any applicable time-dependent analysis for the RCP flywheels that conforms to the criteria for a TLAA in 10 CFR 54.3.

The staff noted that SRP-LR Table 4.1-3 identifies the fatigue analysis of RCP flywheels as a potential, plant-specific TLAA. The staff reviewed relevant information in SRP-LR Section 4.1, NUREG-0800 (SRP), SRP Section 5.4.1.1, and the applicant's USAR as the basis for determining if the applicant's basis was valid.

The staff noted that the applicant's bases for ensuring the structural integrity of the RCP flywheel against the consequences of a non-ductile fast fracture or a postulated fracture of the flywheel during a sudden seizure of the RCP flywheel rotor are documented in USAR Section 5A. The staff reviewed USAR Section 5A, which states that the applicant applies the criteria from SRP Section 5.4.1.1, "Pump Flywheel Integrity," and NRC Safety Guide 14 as the basis for ensuring the integrity of the RCP flywheels during these types of postulated events and the integrity of the RCP casing against the generation of postulated RCP flywheel missiles. The staff noted that this is required to conform to the missile generation protection criteria in 10 CFR Part 50, Appendix A, General Design Criterion 4, "Dynamic Effects." The staff also noted that USAR Section 5A established the applicant's basis for using a value of 40 degrees Fahrenheit (°F) as a conservative estimate of the nil-ductility reference temperature (RT_{NDT}) for the SA-533, Grade B ferritic materials that were used to fabricate the RCP flywheel discs. USAR Section 5A also set a 120 °F operating temperature as the basis for meeting a 150 ksi-in$^{1/2}$ linear-elastic fracture toughness criterion (K_{Ic}), which is used as the basis in SRP Section 5.4.1.1 for protecting against the generation of postulated RCP flywheel missiles.

The staff compared the applicant's basis in USAR Section 5A against the design overspeed criteria (sudden seizure protection criteria) in SRP Section 5.4.1.1, to determine if the sudden seizure basis in USAR Section 5A should be identified as a TLAA. The staff noted that the NRC's bases in SRP Section 5.4.1.1 do not use an analysis as the basis for protecting RCP flywheels against design overspeed (sudden seizure) events. Instead, the SRP section recommends that specific RCP flywheel design overspeed considerations be met as the basis for protecting against RCP flywheel overspeed sudden seizure events. Thus, based on this review, the staff confirmed that the design overspeed basis in USAR Section 5A does not rely on an analysis that has a time dependency; therefore, it does not involve a TLAA for protecting against a design overspeed sudden seizure event.

The staff also reviewed the applicant's basis in USAR Section 5A against the fracture toughness criteria (non-ductile failure criteria) in SRP Section 5.4.1.1, to determine if the non-ductile failure analysis basis in USAR Section 5A would need to be identified as a TLAA. The staff also performed a review of the RCP flywheel design and the reactor coolant system (RCS) operating criteria to determine if the RT_{NDT} value methodology, which is a part of the non-ductile failure basis, would need to include a time-dependent neutron fluence consideration and adjustment.

Specifically, the staff noted that the non-ductile failure analysis referenced in SRP Section 5.4.1.1 recommends that the RCP flywheel materials made from ferritic steel materials meet a minimum allowable fracture toughness (K_{Ic}) of 150 ksi-in$^{1/2}$ to demonstrate that the RCP flywheel rotors and discs will be protected from a postulated non-ductile failure. The staff also noted that Appendix A in the ASME Code Section XI provides an acceptable basis for relating the K_{Ic} fracture toughness property for a ferritic material (i.e., SA-533, Grade B Class ferritic plate materials, or SA-508, Class 2 or 3 ferritic forging materials) to the operating temperature of a component that is made from one of these materials. The staff noted that ASME Code Appendix A permits users applying the appendix to use Figure A-4200-1 in the appendix or, alternatively, the following equation, to establish this K_{Ic}-operating temperature relationship:

$$K_{Ic} = 33.2+20.734*\exp[0.02*(T-RT_{NDT})]$$

From this equation, the K_{Ic} for SA-533, Grade B steel will be maintained above the acceptance criterion of 150 ksi-in$^{1/2}$ if $T-RT_{NDT}$ is greater than or equal to 80 °F. The staff noted that the applicant's licensing basis of 40 °F for RT_{NDT} and minimum 120 °F flywheel operating temperature would indicate that the 150 ksi-in$^{1/2}$ K_{Ic} criterion in SRP Section 5.4.1.1 is met.

The RCP flywheels are not exposed to an operating environment that could cause time-dependent changes in the fracture toughness, such as exposure to a high-energy neutron environment (i.e., neutrons with energies in excess of 1.0 MeV). Therefore, the non-ductile failure analysis of RCP flywheels in USAR Appendix 5A does not need to be identified as a TLAA because it does not include a time-dependent assumption defined by the life of the plant, and thus does not conform to Criterion 3 of 10 CFR 54.3(a).

Therefore, the staff finds acceptable the applicant's conclusion that there are no TLAAs associated with the RCP flywheels.

4.1.2.1.2 Inservice Local Metal Containment Corrosion Analysis

LRA Tables 4.1-1 and 4.1-2 state that the CLB does not include any inservice local metal containment corrosion analyses which conform to the definition of a TLAA in 10 CFR 54.3, that need to be identified as a TLAA in accordance with the TLAA identification requirement in 10 CFR 54.21.

The staff reviewed the Davis-Besse USAR for relevant information. The staff confirmed that the USAR does not make any reference to a corrosion analysis for the steel containment vessel. Based on this review, the staff finds that the applicant provided an acceptable basis for concluding that the generic inservice local metal containment corrosion allowance TLAA, mentioned in SRP-LR Table 4.1-2, does not need to be applied as a TLAA for the LRA because the CLB does use this type of analysis to justify management of corrosion in the applicant's steel containment vessel. Instead, the staff confirmed that the applicant uses its ISI Program—IWE as the basis for managing loss of material due to corrosion in the steel containment vessel. The staff's evaluation of the applicant's ISI Program—IWE is documented in SER Section 3.0.3.1.10.

4.1.2.2 Evaluation of the Applicant's Identification of those Exemptions in the CLB that are Based on TLAA-s

As required by 10 CFR 54.21(c)(2), the applicant must list all exemptions granted pursuant to 10 CFR 50.12, which are based on TLAAs, and evaluate them for continuation through the period of extended operation. LRA Section 4.1.2 states that the USAR, fire hazards analysis report, operating license (including TSs), initial Davis-Besse SER, and docketed licensing correspondence were searched to identify exemptions that were granted pursuant to 10 CFR 50.12, as well as those related to 10 CFR Part 50, Appendix R. From this document review, the applicant determined that there are no exemptions identified as being granted pursuant to 10 CFR 50.12 and based on a TLAA.

The staff reviewed the LRA and applicant's documents in the CLB to verify whether there were any exemptions granted in accordance with 10 CFR 50.12 and based on a TLAA. The staff's search included those exemptions that may have been requested pursuant to 10 CFR 50.60(b) to deviate from the requirements for applicable USE or P-T limit assessments in 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," and that may have been approved under the exemption acceptance provisions of 10 CFR 50.12.

The staff noted that the applicant's CLB includes approved exemptions from complying with the applicable 10 CFR 50.46 and 10 CFR Part 50, Appendix K, requirements for operability of emergency core cooling systems and with 10 CFR Part 50, Appendix R, requirements for ensuring adequate fire protection. The staff noted that these exemptions were not based on a time-dependent analysis that involved an assessment of either a detected or postulated aging effect. Therefore, based on its review, the staff finds these exemptions did not need to be identified as exemptions that were based on a TLAA, in accordance with 10 CFR 54.21(c)(2).

The staff noted that the applicant's CLB included an exemption request, in accordance with 10 CFR 50.60(b), related to the proposed use of an alternative methodology in Areva Report BAW-2308-A to deviate from the applicable requirements for generation of P-T limits in 10 CFR Part 50, Appendix G, and from the PTS analysis requirements in 10 CFR 50.61. The NRC approved this exemption to use the alternative methods in Report No. BAW-2380-A in a safety evaluation (SE) dated December 14, 2010. The staff noted that this exemption relates to the use of the following alternative technologies:

- alternative technology for establishing the initial RT_{NDT} values for the Linde 80 weld materials using the master curve test data and technology, as related to generation of the adjusted reference temperatures used in the P-T limit and PTS analyses
- use of ASME Code Case N-588 to establish alternative primary stress intensity factors for RV beltline material circumferential welds, as used in the generation of the applicant's 10 CFR Part 50, Appendix G, P-T limit curves for the facility
- use of ASME Code Case N-640 as the basis using a K_{Ic} stress intensity factor acceptance criterion (in lieu of using the K_{Ia} stress intensity factor acceptance criterion that would be required by Appendix G of the ASME Code), as used in the generation of the applicant's 10 CFR Part 50, Appendix G, P-T limit curves for the facility

The staff noted that, although this exemption relates to the applicant's P-T limit and PTS TLAAs, it is not based on a time-dependent analysis that needs to be identified as a TLAA. The staff confirmed that this exemption, to apply the alternative methods in BAW-2380-A, is based on the analysis of alternative master curve surveillance program data and the use of alternative stress intensity factor criteria in ASME Codes N-588 and N-640, which do not involve a time-dependent analysis. The staff noted that Regulatory Issue Summary (RIS) 2004-04 permits applicants to use the methods of analysis in NRC-approved ASME Code Cases N-588, N-640, or N-641 for the development of their plant's P-T limit curves without the need for a regulatory exemption on the licensing basis. Thus, the staff determined that the exemption and alternative methodologies in Report BAW-2380-A do not need to be identified as an exemption for the LRA because they are not based on a TLAA.

Based on its review, the staff finds that the applicant provided an acceptable basis for concluding that the LRA does not need to list any exemptions, in accordance with exemption identification requirements in 10 CFR 54.21(c)(2). The staff confirmed that those exemptions, which were previously granted under the provisions of 10 CFR 50.12, were not based on a TLAA.

4.1.3 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable list of TLAAs, as required by 10 CFR 54.21(c)(1). The staff confirmed, as required by 10 CFR 54.21(c)(2), that no exemptions exist in the CLB that have been granted under the requirements in 10 CFR 50.12 and that were based on a TLAA.

4.2 Reactor Vessel Neutron Embrittlement

During plant service, neutron irradiation reduces the fracture toughness of ferritic steel in the beltline region of the RV (as defined in 10 CFR Part 50, Appendix G) and stainless steel reactor vessel internals (RVI) for light-water nuclear power reactors. Areas of review to ensure that the ferritic RV beltline materials and stainless steel RVI have adequate fracture resistance during both normal and off-normal operating conditions (e.g., upset, emergency, and faulted conditions) include the following:

- RV neutron fluence
- RV materials' USE reduction due to neutron embrittlement
- RV materials' resistance to PTS
- adjusted RT_{NDT} for RV materials due to neutron embrittlement
- operating P-T limits for heatup and cooldown (HU/CD) operations, as well as hydrostatic and leak-testing conditions
- RCS low-temperature overpressure protection (LTOP) system limits for protection of the RV against brittle fracture
- protection of stainless-steel-clad SA-508, Class 2 RV forgings against underclad cracking
- reduction in fracture toughness for RVI

Appendix G of 10 CFR Part 50 specifies fracture toughness requirements for ferritic pressure-retaining components that make up the reactor coolant pressure boundary (RCPB) of light water nuclear reactors. This rule states that RV beltline material properties, including the RT_{NDT} values and Charpy USE values, must account for the effects of neutron radiation. The adjusted RT_{NDT} (ART) value is defined as the sum of the initial RT_{NDT} value for the material in the unirradiated condition, the shift in the RT_{NDT} value caused by irradiation (ΔRT_{NDT}), and a margin term (M). The ART value forms the basis for determining the P-T limits. ΔRT_{NDT} is a function of the material's copper (Cu) and nickel (Ni) content and the neutron fluence to which the material is exposed. ΔRT_{NDT} is calculated as the product of a chemistry factor (CF) and a fluence factor, based on the NRC staff guidelines for radiation embrittlement calculations in Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988. The CF is dependent upon the amount of Cu and Ni in the material and may be determined from tables in RG 1.99, Revision 2, or from surveillance data. The fluence factor is exclusively dependent upon the neutron fluence and may be calculated using the formula specified in RG 1.99, Revision 2. The M term is dependent upon whether the initial RT_{NDT} value is a plant-specific value or a generic value and whether the CF value was determined using the tables in RG 1.99, Revision 2, or surveillance data. The M term accounts for uncertainties in the values of the initial RT_{NDT}, the Cu and Ni contents, the fluence, and the calculation methods. RG 1.99, Revision 2, describes the methodology for calculating the M term.

RG 1.99, Revision 2, also specifies methods for determining the projected percentage decrease in USE as a function of Cu content and neutron fluence, including methods for adjusting the percentage USE decrease using credible USE surveillance data.

10 CFR 50.61, the PTS Rule, provides requirements for ensuring the resistance of RV beltline materials against PTS events, as applicable only to PWR plants. The PTS Rule characterizes

the toughness of RV beltline materials by the reference temperature for PTS, RT_{PTS}, which is defined as the RT_{NDT} value evaluated for the projected end-of-license fluence. The PTS Rule requires that RT_{PTS} values be determined for all RV beltline materials using the procedures specified in paragraph C of the rule. Procedures for calculating RT_{PTS} are the same as those used for calculating RT_{NDT}. RT_{PTS} values for all RV beltline materials shall not exceed the screening criteria specified in 10 CFR 50.61(b)(2), except as provided in 10 CFR 50.61(b)(3)–10 CFR 50.61(b)(7).

The USE, ART, and RT_{PTS} calculations meet the criteria of 10 CFR 54.3(a), and thus they are TLAAs. The TLAAs of the USE, ART, and RT_{PTS} for RV beltline materials are based on projected neutron fluence inputs at specific locations in the RV wall. The USE and ART values are calculated using neutron fluence values at a depth equal to one-quarter of the vessel wall thickness (1/4T). The RT_{PTS} values are calculated using neutron fluence values at the clad-to-base metal interface of the RV wall, as required by 10 CFR 50.61. RG 1.99, Revision 2, quantifies fluence attenuation through the RV wall, based on a known wetted surface fluence, for calculating fluence at various depths from the wetted surface.

4.2.1 Neutron Fluence

4.2.1.1 Summary of Technical Information in the Application

LRA Section 4.2.1 states that the fast neutron fluence values were conservatively estimated at 52 effective full power years (EFPY) of reactor operation, as given in LRA Table 4.2-1. The applicant stated that these fluence values are based on the original licensed thermal power of 2,772 megawatt thermal (MWt) through 2008 and 100 percent power of 2,817 MWt from 2008 to end-of-life, uprated through a licensed measurement uncertainty recapture (MUR) power uprate. The applicant stated that, based on actual reactor core power histories to date and conservative estimates of future core designs, extended plant operation to 60 years will be bounded by 52 EFPY of facility operation. According to the applicant, reaching 52 EFPY at the end of 60 years would require an average plant capacity factor greater than 98.5 percent from 2008 to the end of the period of extended operation.

LRA Section 4.2.1 states that neutron fluence values were calculated for the Davis-Besse RV for the extended 60-year licensed operating period (52 EFPY) using the fluence methodology of Areva Document BAW-2241P-A, “Fluence and Uncertainty Methodologies,” April 1999. The applicant stated that the neutron fluence values were calculated in accordance with RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence.” According to the applicant, neutron fluence results were calculated for Cycles 13-14 irradiation using a computer model that extends from below the core to the RV mating surface. The applicant also stated that the sum of the end of cycle 12 and Cycles 13-14 neutron fluence results in the end of cycle 14 cumulative neutron fluence, and this data was benchmarked against cavity dosimetry data for Cycles 13-14. To extrapolate the neutron fluence values to the end of life, the applicant indicated that Cycle 15 design information was used to develop flux projections at each location. The applicant stated that these Cycle 15 flux values were used to extrapolate the end of Cycle 14 fluence to 52 EFPY assuming 100 percent power at 2817 MWt and a partial low leakage core design whereby high thermal performance fuel assemblies were introduced on the periphery. A summary of the 52 EFPY fluence values for the RV beltline materials, including forgings and welds, is provided in Table 4.2-1 of the LRA. The fluence values in LRA Table 4.2-1 were calculated at the inside wetted surface of the RV.

LRA Tables 4.2-2 through 4.2.4 provide USE, ART, and RT_{PTS} values for the beltline materials, which includes all ferritic materials with projected neutron fluence exposures greater than 1×10^{17} n/cm^2. The applicant identified that the limiting weld and forging materials (with regard to their embrittlement analyses for 60 years of operation) are WF-182-1 and BCC 241, respectively. These limiting materials are the same as for the case of 40 years operation and are included in the RV Surveillance Program so that no additional materials are required for irradiation and testing. The LRA states that neutron fluence is not a TLAA, that it "is an assumption used in various neutron embrittlement TLAAs."

4.2.1.2 Staff Evaluation

The staff reviewed LRA Section 4.2.1 on neutron fluence to evaluate the applicant's determination that neutron fluence is not a TLAA in accordance with 10 CFR 54.21(c)(1). The staff also reviewed this section for technical adequacy of the neutron fluence values used by the applicant in its determinations on the TLAAs in LRA Section 4.2.

SRP-LR Section 4.2.3.1 does not identify specific review procedures for a TLAA related to the neutron fluence. However, neutron fluence is a time-dependent parameter used by the applicant for determining RV beltline materials requiring neutron embrittlement analyses in LRA Sections 4.2.2, 4.2.3, and 4.2.4. LRA Section 4.2.1 documents the applicant's determination of RV beltline materials, based on the projected neutron fluence. Since projected neutron fluence is a time-dependent parameter used by the applicant for determining the RV beltline materials subject to neutron embrittlement analysis, neutron fluence is a TLAA. Therefore, the staff stated in a conference call held on February 9, 2012, that its position is that neutron fluence is a TLAA and asked the applicant to explain why it does not consider neutron fluence to be a TLAA. The applicant stated that it agrees with the staff position and that it will revise LRA Section 4.2.1 to identify neutron fluence as a TLAA and select an appropriate disposition for this TLAA as required by 10 CFR 54.21(c)(1).

By letter dated March 9, 2012, the applicant provided LRA Amendment 24 to identify neutron fluence as a TLAA. The applicant dispositioned neutron fluence as a TLAA in accordance with 10 CFR 54.21(c)(1)(ii) based on the fact that RV neutron fluence values have been projected to the end of the period of extended operation and were used in LRA Section 4.2.1 as the basis for determining the RV beltline materials subject to neutron embrittlement evaluation. Therefore, the staff determined that its concern regarding the identification of neutron fluence as a TLAA and the disposition of this TLAA as required by 10 CFR 54.21(c)(1) is resolved.

In accordance with 10 CFR Part 50, Appendix G, and 10 CFR 50.61, ferritic materials for all RV beltline components shall be evaluated for neutron radiation embrittlement for the duration of the facility operating license. The RV beltline region is defined in 10 CFR Part 50, Appendix G, and 10 CFR 50.61 as the region of the RV that surrounds the effective height of the active core and adjacent regions of the RV that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with respect to neutron radiation damage. A fluence threshold for identifying the RV beltline components is defined in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Revision 2, December 2010, as 1×10^{17} n/cm^2 (E > 1.0 MeV) at the end of the period of extended operation. This fluence threshold is based on the RV Materials Surveillance Program requirements of 10 CFR Part 50, Appendix H, which requires monitoring the changes in fracture toughness (i.e., neutron embrittlement) parameters for all RV materials projected to experience neutron fluence greater than 10^{17} n/cm^2 (E > 1.0 MeV) at the expiration of the facility operating license. Therefore, if projected high-energy neutron fluence for ferritic RV materials at the clad/base metal interface is greater than a threshold value of 1 x 10^{17} n/cm^2 (E > 1.0 MeV) at the end of the period of extended operation, these materials shall be evaluated for neutron embrittlement. LRA Table 4.2-1 lists the 52 EFPY fluence values at the inside wetted surface of the RV. All ferritic RV materials with fluence values at the inside wetted surface greater than 1 x 10^{17} n/cm^2 (E > 1.0 MeV) at the end of the period of extended operation are identified in LRA Section 4.2.1 as RV beltline components requiring neutron embrittlement evaluation in accordance with 10 CFR Part 50, Appendix G. The staff noted that fluence values at the inside wetted surface of the RV are slightly greater than those at the clad/base metal interface by a factor of approximately 1.006–1.026 due to neutron fluence attenuation through the approximately 1/8 in. thick stainless steel cladding. Therefore, the staff found that the applicant's identification of RV beltline materials, based on a projected 52 EFPY fluence at the inside wetted surface of the RV, is acceptable.

The staff independently reviewed the 52 EFPY fluence values provided by the applicant in LRA Table 4.2-1 and determined that these fluence values were appropriately calculated using the BAW-2241NP-A fluence methodology. This fluence methodology was previously reviewed and approved by the staff using the generic methodology and the staff's SE authorizing the use of this methodology is provided in an attachment to the BAW-2241NP-A report. The BAW-2241NP-A fluence methodology appropriately follows the guidance of RG 1.190. Although the revision of BAW-2241 cited by the applicant was approved prior to the initial issuance of RG 1.190, the revision had been found adherent to Draft Guide 1053, on which RG 1.190 was based. The applicable guidance in both regulatory documents is effectively the same; therefore, the calculational framework used by the applicant is acceptable. Furthermore, the staff confirmed that the 52 EFPY fluence values account for the effects of the applicant's 2008 measurement uncertainty recapture power uprate. Therefore, the staff finds that the 52 EFPY neutron fluence values provided in LRA Table 4.2-1 are acceptable for use as inputs for the neutron embrittlement TLAAs provided in LRA Sections 4.2.2, 4.2.3, and 4.2.4.

Because the fluence calculation methodology is approved by the staff and adherent to RG 1.190, the staff finds that the calculated fluence values in LRA Table 4.2.1-1 are acceptable for use as inputs for the neutron embrittlement TLAAs provided in LRA Sections 4.2.2, 4.2.3, and 4.2.4.

Based on the revision to LRA Section 4.2.1 provided in LRA Amendment 24, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the RV neutron fluence values and the determination of RV beltline materials subject to neutron embrittlement analysis, have been projected for the period of extended operation and, therefore, are acceptable.

4.2.1.3 USAR Supplement

LRA Section A.2.2.1 as amended (LRA Amendment 24) by letter dated March 9, 2012, provides the USAR supplement for the RV neutron fluence and beltline analysis TLAA evaluation. Based on its review of the USAR supplement, the staff concludes that the information in the USAR supplement is an adequate summary description of the evaluation, as required by 10 CFR 54.21(d), and is consistent with SRP-LR Section 4.2.3.2.

4.2.1.4 Conclusion

On the basis of 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 determining the neutron fluence values and the RV beltline materials subject to neutron embrittlement evaluation have been projected to the end of the period of extended operation. The staff also concludes that the USAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d), and, therefore, is acceptable.

4.2.2 Upper-Shelf Energy

4.2.2.1 Summary of Technical Information in the Application

LRA Section 4.2.2 describes the applicant's USE TLAA. The applicant used initial (unirradiated) USE values for the Davis-Besse RV beltline forgings from USAR Table 5.2-15. As stated in LRA Section 4.2.2, no initial USE data is available for the beltline welds (Linde 80 submerged-arc welds). Therefore, according to the applicant, weld acceptability for 32 EFPY was justified based on an equivalent margins analysis (EMA) performed using elastic plastic fracture mechanics (EPFM) analysis methods. The applicant stated that, for the Linde 80 welds, 32 EFPY EMAs are documented in topical reports BAW-2192P-A, "Low Upper-Shelf Toughness Fracture Analysis of Reactor Vessels of B&W Owners Group Reactor Vessel Working Group for Load Level A & B Conditions," April 1994, and BAW-2178P-A, "Low Upper-Shelf Toughness Fracture Analysis of Reactor Vessels of B&W Owners Group Reactor Vessel Working Group for Level C & D Service Loads," April 1994.

The applicant stated that a subsequent EMA was performed for the limiting weld, upper shell forging to lower shell forging circumferential Weld WF-182-1, for MUR power uprate conditions. This EMA demonstrated limiting weld acceptability through 32 EFPY.

The applicant stated that USE values are projected to 52 EFPY based on the methods in RG 1.99, Revision 2. The applicant stated that surveillance data was also used for weld WF-182-1 and lower shell forging BCC 241, in accordance with Position 2.2 of the RG. The applicant also stated that all RV beltline locations are projected to maintain USE greater than 50 foot-pound (ft-lb) with the exception of weld WF-182-1. For the RV inlet nozzle forging and attachment weld, the RV outlet nozzle forging and attachment weld, the dutchman forging, and the weld that connects the lower shell forging to the dutchman forging, the applicant stated that the projected USE is conservatively calculated based on a 1/4T fluence of 1.0×10^{18} n/cm^2 (E > 1.0 MeV), which is the lowest fluence for the curves plotted in RG 1.99, Revision 2, Figure 2, for determining projected percent decrease in USE. For all other beltline materials, 52 EFPY USE values are based on fluence values from LRA Table 4.2-1, adjusted for attenuation to a 1/4T depth. The 52 EFPY USE data for all RV beltline materials are presented in LRA Table 4.2-2.

The applicant reported that the limiting RV beltline weld, WF-182-1, is the only beltline material with a projected USE value below 50 ft-lb at 60 years (52 EFPY). The LRA states that the EPFM evaluation (EMA) of Weld WF-182-1 at Davis-Besse was extended from 32 EFPY to 52 EFPY based on the projected 52 EFPY neutron fluence values. The applicant stated that this analysis demonstrates that the limiting RV beltline weld at Davis-Besse satisfies the requirements of the ASME Code, Section XI, Appendix K, analysis for ductile flaw extension and tensile stability using projected USE value for the weld material at 52 EFPY. The LRA states that the 52 EFPY EPFM analysis addresses ASME Code, Section III, Levels A, B, C, and D service loadings and was performed using the procedures and acceptance criteria in Appendix K of the ASME Code, Section XI.

The applicant evaluated the effect of 52 EFPY fluence on the J-integral resistance (J-R) of the material, as determined from the J-R fracture resistance curve. The J-R curve is a plot of J-R versus the postulated crack extension. The applicant stated that the neutron fluence at the postulated crack tip was calculated using the methods in RG 1.99, Revision 2.

The applicant stated that the analytical methodology and the applied loadings for the EMA have not changed. The applicant also noted that the initial RT_{NDT} value for weld WF-182-1 was revised from 2 °F to negative 80.2 °F, and the margin term for this weld was revised from 56 °F to 59 °F. The applicant stated that all other mechanical properties are unchanged. The applicant further stated that the existing transition region fracture toughness curve, which is used to define the beginning of the upper shelf region, is indexed to the initial RT_{NDT} value. The applicant determined that the existing transition region fracture toughness curve evaluation remains conservative for 52 EFPY since the initial RT_{NDT} value decreased.

The applicant determined that the hot leg large break loss-of-coolant accident (LOCA) is the limiting transient at 32 EFPY and 52 EFPY since it most closely approaches the K_{Jc} limit (plane-strain fracture toughness in units of J) of the weld. The applicant stated that, in the USE toughness range, the applied stress intensity factor (K_I) curve is closest to the lower bound K_{Jc} curve at 5.60 minutes into the transient. The applicant also stated that this time would be used as the critical time in the transient at which to perform the flaw evaluation for Levels C and D service loadings.

As a summary of EMA results for Level A, B, C, and D service loadings at 52 EFPY, the applicant provided the following data demonstrating that the ASME Code, Section XI, Appendix K acceptance criteria are satisfied for Levels A and B service loadings:

- With factors of safety of 1.15 on pressure and 1.0 on thermal loading, the applied J-integral (J) at a flaw extension of 0.10 in. (J_1) is less than the J-R curve at a ductile flaw extension of 0.10 in. ($J_{0.1}$). The ratio $J_{0.1}/J_1$ = 3.69 is significantly greater than the minimum required value of 1.0.
- With factors of safety of 1.25 on pressure and 1.0 on thermal loading, flaw extensions are ductile and stable because the derivative (with respect to flaw extension) of the applied J curve is less than the derivative of the lower bound J-R curve, at the point where the two curves intersect.

The applicant provided the following data to demonstrate that the ASME Code, Section XI, Appendix K acceptance criteria are satisfied for Levels C and D service loadings:

- With a factor of safety of 1.0 on loading, the ratio $J_{0.1}/J_1$ = 2.16 is significantly greater than the minimum required value of 1.0.
- With a factor of safety of 1.0 on loading, flaw extensions are ductile and stable because the derivative of the applied J curve is less than the derivatives of both the lower bound and mean J-R curves at the points of intersection.
- Flaw growth is stable at much less than 75 percent of the vessel wall thickness. It was also shown that the remaining ligament is sufficient to preclude tensile instability by a large margin.

The applicant also stated that the limiting beltline weld at Davis-Besse satisfies the requirements of the ASME Code, Section XI, Appendix K, for ductile flaw extension and tensile stability using projected Charpy USE values for the weld material at 32 EFPY and 52 EFPY.

Based on the above, the applicant concluded that the USE values and EMA for the RV beltline materials have been projected to remain acceptable during the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

4.2.2.2 Staff Evaluation

The staff reviewed LRA Section 4.2.2 on USE to confirm, pursuant to 54.21(c)(1)(ii), that the USE and EMA for the RV beltline materials have been projected to the end of the period of extended operation.

The staff reviewed the applicant's TLAA 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 or EMA 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. The staff used the applicant's 60-year projected neutron fluence values for the RV beltline materials as the basis for determining either whether the beltline materials would maintain acceptable levels of USE during the period of extended operation or whether the EMA would meet the acceptance criteria for the period of extended operation. The staff reviewed the applicant's 60-year projected USE values against the USE criteria in 10 CFR Part 50, Appendix G, which establish the lower limits on acceptable values of USE. The staff also reviewed the applicant's 60-year projected EMA against the EPFM acceptance criteria of the ASME Code, Section XI, Appendix K.

Section IV.A.1.a of Appendix G to 10 CFR Part 50 states, in part, that RV beltline materials must maintain Charpy USE values in the transverse direction for base metal and along the weld for weld material of no less than 50 ft-lb, throughout the life of the RV, unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy USE will ensure margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. For RV shell materials with low projected end-of-license USE values or unknown initial USE values, analyses to demonstrate margins of safety against fracture, equivalent to those required by Appendix G of Section XI of the ASME Code can be performed using the ASME Code, Section XI, Appendix K methodology (i.e., EMAs).

In accordance with RG 1.99, Revision 2, the predicted decrease in USE due to neutron embrittlement during plant operation is dependent upon the amount of Cu in the material and the projected neutron fluence for the material. Regulatory Position 1.2 of the RG specifies methods for calculating the predicted percentage decrease in USE for materials that do not have sufficient credible surveillance data. The applicant provided calculations of the projected USE values at 52 EFPY for all RV beltline forgings in LRA Table 4.2-2, based on the initial (unirradiated) USE values for these materials. The staff determined that the applicant correctly used Regulatory Position 1.2 of RG 1.99, Revision 2 (Figure 2 from the RG), for calculating the projected percentage decrease in USE at 52 EFPY for these RV beltline forgings. The staff confirmed that the LRA Table 4.2-2 values for initial USE and Cu content are consistent with those listed in the staff's RV Integrity Database (RVID). The staff found that the applicant correctly determined the projected 52 EFPY USE values for the RV beltline forgings by applying the predicted percentage decrease in USE, as determined using Figure 2 of RG 1.99, Revision 2, to the initial USE values. All of the 52 EFPY USE values for the RV beltline forgings, as listed in LRA Table 4.2-2, are projected to remain greater than the 50 ft-lb minimum USE requirement specified in 10 CFR Part 50, Appendix G. Therefore, the staff determined that the applicant's USE analysis for the RV beltline forgings was acceptable for the period of extended operation.

In LRA Section 4.2.2, the applicant stated that no initial USE data is available for the Linde 80 beltline welds; thus, operation for 32 EFPY was justified based on an EMA. However, the applicant listed a generic initial USE value of 70 ft-lb in LRA Table 4.2-2 for the Linde 80 beltline welds. The applicant calculated projected 52 EFPY USE values for the Linde 80 beltline welds based on predicted percentage decrease in USE, using Figure 2 of RG 1.99, Revision 2, and the 70 ft-lb initial USE value. All of the 52 EFPY USE values for the Linde 80 beltline welds, as listed in LRA Table 4.2-2, are projected to remain greater than the 50 ft-lb, with the exception of weld WF-182-1. Accordingly, the applicant determined that an EMA would be required to demonstrate that this weld will remain in compliance with 10 CFR Part 50, Appendix G, requirements through 52 EFPY.

The applicant provided a discussion of an EMA for the limiting beltline weld, WF-182-1, to demonstrate that the weld will remain in compliance with 10 CFR Part 50, Appendix G, through the period of extended operation. The applicant provided specific information in LRA Section 4.2.2 demonstrating that weld WF-182-1 would remain bounded by the ASME Code, Section XI, Appendix K, acceptance criteria, which are based on the weld's J-R curve and the applied J curve. The applicant's evaluation, as reported in LRA Section 4.2.2, demonstrated that weld WF-182-1 would satisfy the Appendix K acceptance criteria for Levels A, B, C, and D service loads through 52 EFPY.

The applicant referenced the B&W Owners Group (B&WOG) EMAs documented in BAW-2192P-A (Service Levels A and B) and BAW-2178P-A (Service Levels C and D). These EMA reports were previously approved by the NRC for use in determining the 40-year acceptability of B&WOG plants' RV beltline materials with low projected end-of-life USE (less than 50 ft-lb) or unknown heat-specific initial USE values. The applicant stated that these EMAs were used as the basis for acceptance of all Davis-Besse Linde 80 beltline welds through 32 EFPY. Based on the NRC-approved responses to Generic Letter (GL) 92-01, Revision 1, "Reactor Vessel Structural Integrity," the staff confirmed that the 32 EFPY EMAs documented in BAW-2178P-A and BAW-2192P-A were used as the basis for acceptance of the Linde 80 beltline welds at most B&W plants, including all Linde 80 beltline welds at Davis-Besse.

However, the staff identified several issues with the EMA for the Linde 80 beltline welds that required clarification and, by letter dated March 17, 2011, issued request for additional information (RAI) 4.2.2-1.

In RAI 4.2.2-1, the staff requested that the applicant discuss whether the EMA methods and minimum Charpy USE acceptance criteria developed in BAW-2178P-A and BAW-2192P-A are valid for demonstrating Linde 80 beltline weld acceptability through 52 EFPY, based on the calculations of the projected percentage decrease in USE for 52 EFPY, as listed in LRA Table 4.2-2. If the validity cannot be established, the staff requested that the applicant provide the reports documenting the EMA calculations for demonstrating that all RV beltline welds, including the limiting beltline weld (WF-182-1), will satisfy the requirements of 10 CFR Part 50, Appendix G, for equivalent margins against ductile fracture through the period of extended operation.

In its response dated April 15, 2011, the applicant stated that the EMA methods and acceptance criteria developed in BAW-2178P-A and BAW-2192P-A are valid for demonstrating Linde 80 beltline weld acceptability through 52 EFPY. The applicant also stated that these methodologies were based on ASME Code Case N-512, which was later incorporated into Appendix K of the ASME Code, Section XI. The applicant stated that the BAW-2178P-A and BAW-2192P-A methodologies require a comparison of J_1 to $J_{0.1}$ and a comparison of derivatives of the applied J and J-R curves at the point of their intersection. The applicant further stated that the applied J and J-R curves for the Linde 80 welds are dependent on the change in material properties but independent of the calculated 52 EFPY USE value reported in LRA Table 4.2-2. This is because the Linde 80 weld's J-R curve is calculated using the Cu content, the projected fluence at the postulated crack tip, the metal temperature at the postulated crack tip, and the specimen thickness.

In its response to the second part of RAI 4.2.2-1, the applicant clarified the following:

- The analysis for the limiting weld was extended to 52 EFPY based on the Appendix K methods.
- The limiting RV beltine weld, WF-182-1, is the only RV beltline material with a projected 52 EFPY USE value less than 50 ft-lb, thusly requiring an EMA for the period of extended operation.
- The limiting weld's 32 EFPY EMA was previously updated to account for the MUR power uprate conditions, and the updated EMA was extended to 52 EFPY.

The staff reviewed the applicant's response to RAI 4.2.2-1 and found it acceptable because the applicant provided the information necessary for the staff to determine that the 52 EFPY EMA for weld WF-182-1 is consistent with the previously approved methods documented in BAW-2178P-A and BAW-2192P-A. Specifically, the staff confirmed that the ASME Code, Section XI, Appendix K, methods and acceptance criteria remained essentially unchanged compared to those specified in ASME Code Case N-512, which were used as the basis for the BAW-2178P-A and BAW-2192P-A EMAs for 32 EFPY. Additionally, based on the applicant's RAI response, the staff also confirmed that the methods used to establish the 32 EFPY J-R curves for Linde 80 beltline welds, as documented in BAW-2178P-A and BAW-2192P-A, are consistent with those used to establish the 52 EFPY J-R curve for weld WF-182-1 for the 52 EFPY EMA of this weld. Finally, the staff confirmed that 52 EFPY J-R values were determined using 52 EFPY fluence values at the postulated crack tip, consistent with the staff-approved EMA methods in BAW-2178P-A, BAW-2192P-A, and the ASME Code, Section XI, Appendix K. The staff's concern described in RAI 4.2.2-1 is resolved.

The staff confirmed that for Linde 80 welds, EMAs are performed using J-R curves based on the Cu-Fluence model from NUREG/CR-5729, "Multivariable Modeling of Pressure Vessel and Piping J-R Data," May 1991. Consistent with the applicant's response to RAI 4.2.2-1, the staff confirmed that for the Linde 80 welds, J-R curves are established based on Cu content, projected neutron fluence at the postulated crack tip, metal temperature at the postulated crack tip, and specimen thickness. The specimen thickness is set equal to 0.8 in., as discussed in the applicant's GL 92-01 response. This approach for specimen thickness is conservative because, as discussed in RG 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper Shelf Energy Less Than 50 ft-lb," June 1995, the use of specimen thickness terms less than the RG 1.161 recommended value of 1.0 in. produce lower J-R curve values. The J-R models presented in NUREG/CR-5729 also reflect this trend. The J-R curve is evaluated against the applied J values, per the Appendix K acceptance criteria. Applied J values from BAW-2178P-A and BAW-2192-A were used by the applicant for the 52 EFPY EMA of the limiting weld. The applied J values are not time-dependent parameters. Therefore, the applicant's use of applied J values from BAW-2178P-A and BAW-2192-A for the 52 EFPY EMA of the limiting weld is acceptable.

As stated in NUREG/CR-5729, the Cu-fluence model is recommended only for those cases where initial Charpy USE values for the material are not available and only for Linde 80 weld material. Based on the information included in the applicant's response to GL 92-01, Revision 1, for the USE, as well as the applicant's response to RAI 4.2.2-1, the staff determined that heat-specific initial Charpy USE values are not available for the Linde 80 RV beltline weld materials at Davis-Besse. Therefore, the applicant's use of the Cu-fluence model from NUREG/CR-5729 for determining the Linde 80 weld WF-182-1 J-R curve is appropriate.

The staff also noted that RG 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper Shelf Energy Less Than 50 ft-lb," June 1995, includes a brief discussion of material properties, as characterized by the material's J-R curve. RG 1.161 also recommends the Cu fluence model from NUREG/CR-5729 for Linde 80 weld material.

The staff evaluated the applicant's statement in LRA Section 4.2.2 regarding the revision to the initial RT_{NDT} value and the margin term for weld WF-182-1. The staff confirmed that the changes in the initial RT_{NDT} value from 2 °F to negative 80.2 °F, and the margin term from 56 °F to 59 °F, based on a staff-approved exemption to use alternative methods for determining these parameters (discussed in SER Section 4.2.3), do not affect the EMA for the limiting weld, WF-182-1. This is because EMAs use EPFM methods, which are based on the assumption that the material is operating at temperatures in the upper shelf region of the ductile-to-brittle transition curve. The large decrease in the initial RT_{NDT} and slight increase in the margin term has the net effect of decreasing the adjusted RT_{NDT} for weld WF-182-1 for a given level of embrittlement, resulting only in the extension of the upper shelf region to lower temperatures. The initial RT_{NDT} and margin terms do not enter into any of the equations for calculating applied J values or J-R values, which are the basis for EMAs using the current staff-approved methods.

In summary, the staff reviewed the applicant's response to RAI 4.2.2-1, the EMA methodology described in Appendix K, the applicant's responses to GL 92-01 (pertaining to the Linde 80 weld EMAs for 32 EFPY), and the original 32 EFPY EMAs documented in BAW-2178P-A and BAW-2192P-A. Based on its review, the staff determined that the applicant demonstrated that the limiting weld, WF-182-1, will maintain the necessary equivalent margins against fracture through 52 EFPY, as specified in 10 CFR Part 50, Appendix G.

As discussed above, the applicant stated, in LRA Section 4.2.2, that initial USE values are not available for the Linde 80 beltline welds. However, LRA Table 4.2-2 lists an initial USE value of 70 ft-lb for all Linde 80 beltline welds. Therefore, by letter dated March 17, 2011, the staff issued RAI 4.2.2-2 requesting that the applicant explain the technical basis for the RV beltline welds' initial USE value of 70 ft-lb, including the underlying statistics.

In its response dated April 15, 2011, the applicant indicated that the discussion in LRA Section 4.2.2 regarding the Linde 80 beltline welds requires a revision. The applicant stated that the 70 ft-lb initial USE value was based on an assessment from the B&WOG Master Integrated Reactor Vessel Program (MIRVP) of available unirradiated Charpy USE data for Linde 80 weld material. The applicant stated that the MIRVP established a generic mean value for all Linde 80 welds using measured unirradiated Charpy USE data from archived specimens designated with plant-specific capsules from each of the participating MIRVP plants. The applicant also stated that the statistical analysis of the unirradiated Charpy USE data, reported in B&WOG Topical Report BAW-1803, "Correlations for Predicting the Effects of Neutron Radiation on Linde 80 Submerged-Arc Welds," Revision 1, May 1991, yielded a mean initial USE value of 69.7 ft-lb. The applicant further stated that the 69.7 ft-lb value, rounded to 70 ft-lb, was established as the generic initial USE value for the Linde 80 welds at all participating MIRVP plants.

The staff reviewed the applicant's response to RAI 4.2.2-2 and determined that the applicant adequately explained how it obtained the 70 ft-lb initial USE value for the Linde 80 RV beltline welds. The staff noted that the 70 ft-lb initial USE value for the Linde 80 welds is based, approximately, on the generic mean value of the available measured Charpy USE data from archived Linde 80 weld specimens from the B&WOG MIRVP. The 70 ft-lb initial USE value was used by the applicant for calculating the projected USE at 52 EFPY for the Linde 80 welds. The 52 EFPY projected USE values for the non-limiting Linde 80 welds, WF-232 and WF-233, were determined to be acceptable by the applicant because these values were projected to be greater than the 50 ft-lb minimum USE requirement specified in 10 CFR Part 50, Appendix G.

However, the staff has concerns with the use of a generic initial USE of 70 ft-lb for Linde 80 welds, for implementation in direct projections of end-of-license USE for the period of extended operation (52 EFPY), for the following reasons:

- The mean value from a database has generally not been acceptable to the staff for establishing a generic initial USE value for a specification, class, or type of RV material because generic mean values are not statistically defensible for embrittlement calculations. In the past, the staff has generally only accepted generic initial USE values if they are based on a statistically-conservative position, such as the mean value minus two standard deviations, or the lowest value in the database.
- The BAW-1803 initial USE database has not been reviewed and approved by the staff as a statistical basis for the selection of any generic initial USE value for Linde 80 welds.

Therefore, by letter dated May 15, 2012, the staff issued RAI 4.2.2-4, requesting that the applicant demonstrate an acceptable USE evaluation for the RV beltline weld materials, WF-232 and WF-233, by providing a response to either (a) or (b) below:

(a) Provide a direct projection of USE through 52 EFPY based on either (i) measured heat-specific initial USE values from certified material test reports, or (ii) a statistically-based conservative generic initial USE value, along with a technical justification for the value.

(b) Provide EMAs for weld materials WF-232 and WF-233 in the shell region of the RV, which may use the existing methods developed in B&WOG Topical Reports BAW-2191P-A and BAW-2178P-A, or the ASME Code, Section XI, Appendix K, accounting for neutron embrittlement through 52 EFPY. EMAs for non-shell welds must use applied J-integral values based on the specific weld geometry.

In its response dated June 14, 2012, the applicant elected to provide 52 EFPY EMAs for weld materials WF-232 and WF-233, as specified in option (b) above. The applicant's response listed the RV beltline weld components according to their location in the RV as shown in Table 4.2-1.

Table 4.2-1. RV Beltline Weld Components Material Identification

Weld no.:	Weld component description:	Weld material identification:
1	Nozzle belt forging to bottom of RV inlet nozzle forging welds	WF-232/WF-233

2	Nozzle belt forging to bottom of RV outlet nozzle forging welds	WF-233
3	Upper shell forging to lower shell forging circumferential weld	WF-182-1
4	Nozzle belt forging to upper shell forging circumferential weld	WF-232/WF-233
5	Lower shell forging to dutchman forging circumferential weld	WF-232/WF-233

Each weld material identifier (i.e., WF-232, WF-182-1, etc.) corresponds to a specific heat of weld wire. The heat numbers were identified by the applicant in LRA Section 4.2.1.

The applicant stated that weld 3, which is the limiting beltline weld with respect to Cu content and 52 EFPY neutron fluence, has been found acceptable for the period of extended operation, based on the EMA documented in LRA Section 4.2.2. The staff's evaluation of the EMA for this weld is documented above. The applicant noted that weld 3 has historically been treated as the limiting weld in the RV based on material properties alone. However, since this circumferential seam weld is remote from structural discontinuities, the applicant acknowledged that other locations may potentially control due to higher stresses, even with higher toughness values for the weld material.

The applicant stated that the EMA for welds 1 and 2 above, which connect the nozzle belt forging to the bottom of the inlet and outlet nozzles, is documented in the AREVA NP, Inc. (AREVA) proprietary calculation report, Calculation 32-9110426-000, "DB-1 EMA of RPV Inlet & Outlet Nozzle-to-Shell Welds for 60 Years," dated May 2010. The applicant provided AREVA Calculation 32-9110426-000 in Enclosure B of its June 14, 2012, response letter to RAI 4.2.2-4. According to the applicant, these welds were analyzed using EPFM techniques based on the ASME Code, Section XI, Appendix K. The applicant stated that welds 1 and 2 are full penetration nozzle attachment welds that are located in the 12 in. thick nozzle belt forging section of the RV. The applicant stated that bounding stresses from the connected nozzles due to piping loads were considered in addition to pressure and thermal stresses for these welds. Regarding the 52 EFPY fluence values for the welds, the applicant stated that the higher fluence for weld 2 was selected for the J-R curve calculation for welds 1 and 2 due to the closer proximity of the larger diameter outlet nozzle weld to the reactor core. The applicant also stated that the EMA for welds 1 and 2 satisfies the acceptance criteria of the ASME Code, Section XI, Appendix K for Level A, B, C, and D service loads at 52 EFPY.

The applicant stated that welds 4 and 5 are located at thickness transitions above and below the beltline shell, respectively. Therefore, these welds require an EMA using local stresses and material properties specific to these weld locations. In its RAI response, the applicant committed (regulatory commitment) to complete the following on or before September 14, 2012:

- submit an EMA for welds 4 and 5
- revise LRA Section 4.2.2, LRA Table 4.2-2, and the corresponding USAR supplement section in LRA Section A.2.2.2 for the USE evaluation to address the EMA results for welds 1, 2, 4, and 5

By letter dated September 7, 2012, the applicant provided EMAs for welds 4 and 5 and revised the LRA sections for the USE evaluation, as committed to above. The applicant stated that the EMA for welds 4 and 5 is documented in an AREVA proprietary calculation report, Calculation 32-9184568-000, "Equivalent Margins Assessment of Davis-Besse Transition Welds for 52 EFPY," dated August 30, 2012. AREVA Calculation 32-9184568-000 was provided in Enclosure C to the September 7, 2012, RAI response. The applicant stated that this calculation demonstrates that welds 4 and 5 at Davis-Besse satisfy the requirements of the ASME Code, Section XI, Appendix K for 52 EFPY. LRA Amendment 34, provided in Enclosure A of the applicant's September 7, 2012, RAI response, revised LRA Sections 4.2.1, 4.2.2, 4.8, and A.2.2.2; and Table 4.2-2, to address the EMA results for welds 1, 2, 4, and 5.

On the basis of its review, the staff finds the applicant's response to RAI 4.2.2-4 acceptable because the applicant provided EMAs for weld materials WF-232 and WF-233 (used for welds 1, 2, 4, and 5) to demonstrate that the welds will maintain the required margins against ductile fracture through 52 EFPY, as required by 10 CFR Part 50, Appendix G. The staff reviewed the EMAs documented in the AREVA proprietary calculation reports, Calculation 32-9110426-000 and Calculation 32-9184568-000, for welds 1, 2, 4, and 5. Non-proprietary publicly available versions of the AREVA calculation reports reviewed by the staff were provided by the applicant by letters dated December 11 and 12, 2012. Based on its review, the staff determined that the EMAs for welds 1, 2, 4, and 5 demonstrate the required margins against ductile fracture for the period of extended operation, and the applicant correctly implemented the EPFM methods and acceptance criteria of the ASME Code, Section XI, Appendix K for demonstrating weld acceptability. The staff's determination is based on the findings below.

The staff finds that the fracture resistance for the welds, as characterized by the welds' J-R curve, was established using the staff-approved Cu-fluence model for Linde 80 welds, as discussed above for the limiting beltline shell weld, WF-182-1. The staff finds this appropriate because the J-R curve for the material is not dependent on the local configuration of the component, including whether the material is at or near a structural discontinuity, such as a nozzle. The staff confirmed that the Cu contents and 52 EFPY neutron fluence values used to calculate the J R curves for the welds are consistent with those provided in LRA Table 4.2-2 and, therefore, acceptable. Welds 1, 4, and 5 are fabricated from two different heats of weld wire (e.g., the inner 12 percent of weld 5 is fabricated from WF-232, and the outer 88 percent is fabricated from WF-233). The staff confirmed that the Cu content and 52 EFPY neutron fluence values used for the Level A, B, C, and D service load evaluations correspond to the weld material and fluence at the location of the postulated crack tip specified in the ASME Code, Section XI, Appendix K—the 1/4T location for Levels A and B and the 1/10T location for Levels C and D. Since the J-R curves for the welds were determined based on an NRC-approved J-R model, appropriate Cu contents, and 52 EFPY neutron fluence values, the staff finds that the applicant's J-R calculations for welds 1, 2, 4, 5 are acceptable for the period of extended operation.

For calculating the crack driving force parameter, applied J, the applicant noted that the ASME Code, Section XI, Appendix K, provides detailed procedures for calculating applied J values for materials in the cylindrical shell portion of the RV, remote from discontinuities. The Appendix K procedures specify that the applied J value is calculated as a function of applied K_I inputs for each loading condition. The specific K_I formulations prescribed in Appendix K are applicable only to the shell region, remote from structural discontinuities. The applicant stated that the applied J values for welds 1, 2, 4, and 5 were calculated by augmenting the Appendix K methods to account for the weld structural discontinuities. This approach involved performing a finite element analysis for each weld to determine the applied K_I values, based on the specific weld geometry. Based on its review of these analyses, the staff determined that the applicant's K_I inputs for determining the applied J values were appropriately calculated taking into consideration the structural discontinuities associated with the RV thickness transitions for welds 4 and 5, above, and the nozzle-to-shell transitions for welds 1 and 2, above. Applied stress intensity factors at these locations were calculated based on the appropriate internal pressures, thermal transients, and piping reactions for Level A, B, C, and D service loadings. The staff finds this approach acceptable because it is consistent with the EPFM methods of the ASME Code, Section XI, Appendix K, appropriately modified to account for the structural discontinuities at the above weld locations.

The staff finds that the applicant correctly applied the ASME Code, Section XI, Appendix K acceptance criteria for demonstrating the required margins against ductile failure. Specifically, the applicant demonstrated that the ratio, $J_{0.1}/J_1$, is significantly greater than the minimum required value of 1.0 for Level A, B, C, and D service loadings, using the appropriate safety factors on applied loads, as specified in Appendix K for each service condition. The applicant also demonstrated that flaw extensions are ductile and stable for all service conditions, based on a comparison of the derivatives of the applied J and J-R curves at their points of intersection. Therefore, the staff finds that the EMAs for welds 1, 2, 4, and 5 adequately demonstrated the required margins against ductile fracture, in accordance with 10 CFR Part 50, Appendix G.

The staff finds that LRA Amendment 34 includes the appropriate revisions to LRA Sections 4.2.1, 4.2.2, 4.8, and A.2.2.2; and LRA Table 4.2-2 to address the EMA results for welds 1, 2, 4, and 5. Specifically, LRA Sections 4.2.1 and A.2.2.2 were appropriately revised to state that 52 EFPY USE values were conservatively assumed to be below 50 ft-lb for all RV beltline welds (based on the fact the measured initial USE values are unknown); therefore, the welds were evaluated for 52 EFPY based on an EMA. LRA Section 4.2.2 was supplemented to include a discussion of the EMA results for welds 1, 2, 4, and 5. The LRA Section 4.2.2 discussion now includes a summary of EMA results for demonstrating that each weld meets the ASME Code, Section XI, Appendix K acceptance criteria for the ratio, $J_{0.1}/J_1$, and ductile flaw stability for Level A, B, C and D service loadings. All line entries for the RV beltline welds in LRA Table 4.2-2 were revised to delete the results of the 52 EFPY USE calculation, replacing them with a notation stating that the 52 EFPY USE values were conservatively assumed to be below 50 ft-lb for all RV beltline welds. Therefore, the welds were evaluated for 52 EFPY based on an EMA. Finally, LRA Sections 4.2.2 and 4.8 were supplemented to include references to the EMA calculation reports, AREVA Calculations 32-9110426-000 and 32-9184568-000, for welds 1, 2, 4, and 5. The staff finds that these LRA revisions are consistent with the applicant's response to RAI 4.2.2-4 and the EMAs performed for all RV beltline welds. Accordingly, the staff finds the revisions implemented by LRA Amendment 34 acceptable.

Based on the staff's determination regarding the acceptability of the EMAs for welds 1, 2, 4, and 5 and the revisions to LRA Sections 4.2.1, 4.2.2, 4.8, and A.2.2.2; and Table 4.2-2, the staff's concern described in RAI 4.2.2-4 is resolved, and open item (OI) 4.2-1 is closed.

LRA Section 4.2.2.2 states that Position 2.2 of RG 1.99, Revision 2, was used to calculate 52 EFPY USE values for weld WF-182-1 and forging BCC-241 using surveillance data. By letter dated March 17, 2011, the staff issued RAI 4.2.2-3 requesting that the applicant state whether the 52 EFPY USE values for these materials in LRA Table 4.2-2, based on Position 2.2 of RG 1.99, Revision 2, were calculated using at least two credible sets of USE surveillance data for these materials.

In its response dated April 15, 2011, the applicant stated that four credible sets of surveillance data were used to calculate the Position 2.2 USE value for weld WF-182-1, and five credible sets of surveillance data were used to calculate the Position 2.2 USE value for forging BCC-241. The staff reviewed the applicant's response to RAI 4.2.2-3 and determined that the applicant's use of surveillance data for the 52 EFPY USE projections for these materials is acceptable because it is consistent with Position 2.2 of RG 1.99, Revision 2, in that more than two credible sets of surveillance data were used to determine the 52 EFPY USE values for these materials. The staff's concern described in RAI 4.2.2-3 is resolved.

Based on the above evaluation, the staff determined that all Davis-Besse RV beltline forgings are projected to maintain USE values greater than 50 ft-lb through 52 EFPY. The staff also determined that the EMAs for all RV beltline welds are projected to remain acceptable through 52 EFPY based on the ASME Code, Section XI, Appendix K, acceptance criteria. Therefore, the staff determined that the applicant demonstrated that all RV beltline materials are projected to remain in compliance with the USE requirements of 10 CFR Part 50, Appendix G, during the period of extended operation.

Based on its evaluation, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis of the USE and the EMAs for the Davis-Besse RV beltline region have been projected to the end of the period of extended operation and, therefore, are acceptable. Additionally, the staff finds that the applicant's TLAA meets the acceptance criterion in SRP-LR Section 4.2.2.1.1.2 because the applicant's USE analysis and EMAs have been accurately projected to the end of the period of extended operation.

4.2.2.3 USAR Supplement

LRA Section A.2.2.2 provides the USAR supplement summary description for the USE TLAA. Based on its review of the USAR supplement, as revised by LRA Amendment 34, the staff concludes that the information in the USAR supplement is an adequate summary description of the evaluation, as required by 10 CFR 54.21(d), and is consistent with SRP-LR Section 4.2.3.2.

4.2.2.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis of the USE and EMAs have been projected to the end of the period of extended operation. The staff also concludes that the USAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d); therefore, it is acceptable.

4.2.3 Pressurized Thermal Shock

4.2.3.1 Summary of Technical Information in the Application

LRA Section 4.2.3 describes the applicant's evaluation of 52 EFPY RT_{PTS} values for the RV beltline materials. In accordance with 10 CFR 50.61, the applicant calculated RT_{PTS} values by adding the initial RT_{NDT} to the predicted radiation-induced ΔRT_{NDT}, plus a margin to cover uncertainties, as prescribed by RG 1.99, Revision 2. The applicant's projected ΔRT_{NDT} values were calculated using the 52 EFPY neutron fluence at the clad-low alloy steel interface. LRA Table 4.2-3 includes 52 EFPY RT_{PTS} values for the beltline materials based on Position 1.1 from RG 1.99, Revision 2. The applicant stated that surveillance data was not used because two credible sets of RT_{PTS} surveillance data are not available for any of the RV beltline materials. The applicant stated that initial RT_{NDT} and margin values for upper shell forging to lower shell forging circumferential weld WF-182-1 and nozzle belt forging to upper shell forging circumferential weld WF-232 were obtained from BAW-2308, "Initial RT_{NDT} of Linde 80 Weld Materials," Revision 1-A, August 2005. The applicant also stated that, using the RG 1.99, Revision 2, tabulated CF values, the CF for WF-232 is 157.3. The applicant further stated that when initial RT_{NDT} values from BAW-2308, Revision 1-A are used, the CF cannot be less than

167.0. Thus, the applicant's CF in LRA Table 4.2-3 for nozzle belt forging to upper shell forging circumferential weld WF-232 is 167.0. However, for lower shell forging to Dutchman forging circumferential weld WF-232, the applicant choose to use a CF value of 157.3, per RG 1.99, Revision 2, criteria and a much higher initial RT_{NDT} and M from the RVID to conservatively determine the RT_{PTS} value for this weld.

According to the applicant, the RT_{PTS} values for all RV beltline materials are projected to remain below the PTS screening criteria at 60 years. The applicant stated that weld WF-182-1 is the limiting RV beltline material, with an RT_{PTS} value of 182.2 °F, as compared to an acceptance criterion of 300 °F for circumferential welds.

Based on the information above, the applicant concluded that the RT_{PTS} values for the RV beltline materials have been projected to remain acceptable to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

4.2.3.2 Staff Evaluation

The staff reviewed LRA Section 4.2.3 on PTS to confirm, pursuant to 10 CFR 54.21(c)(1)(ii), that the PTS analysis for the RV beltline materials has been projected to the end of the period of extended operation.

The staff reviewed the applicant's TLAA 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 staff used the applicant's 60-year projected neutron fluence values for the RV beltline materials as the basis for determining whether the RV beltline materials would have acceptable RT_{PTS} values for the period of extended operation. The staff reviewed the applicant's 60-year RT_{PTS} values against the RT_{PTS} screening criteria in 10 CFR 50.61, which establish the upper limits on acceptable values of RT_{PTS}.

The staff's review of LRA Section 4.2.3 covered the applicant's PTS methodology and RT_{PTS} calculations for the end of the period of extended operation, considering the effects of neutron embrittlement. In 10 CFR 50.61, the required methodology for calculating these RT_{PTS} values is provided, which is similar to the calculation methodology described in RG 1.99, Revision 2, for determining the ART values used for P-T limits calculations. Pursuant to 10 CFR 50.61, RT_{PTS} calculations account for the effects of neutron embrittlement and incorporate any relevant RV surveillance capsule data as part of the applicant's implementation of its RV Materials Surveillance Program. Also in 10 CFR 50.61, RT_{PTS} is defined as the RT_{NDT} value evaluated at the clad/base metal interface using the projected end-of-license fluence. RT_{PTS} values for all RV beltline materials shall not exceed the screening criteria specified in 10 CFR 50.61(b)(2), except as provided in 10 CFR 50.61(b)(3)–10 CFR 50.61(b)(7). The PTS screening criteria are 270 °F for RV plates, forgings, and axial welds and 300 °F for circumferential welds.

LRA Table 4.2-3 lists the projected 52 EFPY RT_{PTS} values for all RV beltline materials, including all input data used in the calculations. The staff confirmed that all 52 EFPY RT_{PTS} values are correctly calculated using the methods specified in 10 CFR 50.61 (as modified by an NRC-approved exemption discussed below) and RG 1.99, Revision 2, Position 1.1. At the clad/low alloy steel interface of the RV wall, 52 EFPY fluence values were appropriately used for the RT_{PTS} calculations. All Cu and Ni content values listed in LRA Table 4.2-3 are consistent with those in the RVID. With respect to the initial RT_{NDT} values and the M values for welds WF-182-1 and WF-232, the staff confirmed that these values were determined based on the methods described in BAW-2308, Revision 1-A. The staff confirmed that the applicant received NRC approval, in a letter dated December 14, 2010, for an exemption to use the BAW-2308, Revision 1-A, methods as alternatives to 10 CFR Part 50, Appendix G, and 10 CFR 50.61 requirements in determining the initial RT_{NDT} and σ_i terms. The staff also confirmed that, in a letter dated January 28, 2011, License Amendment No. 282 authorized the TS changes for referencing the BAW-2308, Revision 1-A and 2-A methods in the TS 5.6.4 requirements for the P-T limits report (PTLR) methodology. The initial RT_{NDT} values and M values for all other RV beltline materials are consistent with those listed in the RVID. Accordingly, the staff found the applicant's 52 EFPY RT_{PTS} values acceptable because all RV beltline materials are projected to maintain RT_{PTS} values less than the applicable screening criteria specified in 10 CFR 50.61 at the end of the period of extended operation.

Based on its review of the applicant's RT_{PTS} calculations, the staff found that the applicant accurately calculated the 52 EFPY RT_{PTS} values for all Davis-Besse RV beltline materials. Accordingly, the staff determined that the applicant demonstrated that all RV beltline materials at Davis-Besse are projected to remain in compliance with the PTS screening requirements of 10 CFR 50.61 through the end of the period of extended operation.

Based on its evaluation, as discussed above, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis of PTS for the Davis-Besse RV has been projected to the end of the period of extended operation, and all of the RT_{PTS} values remain in compliance with the PTS screening requirements of 10 CFR 50.61 through the end of the period of extended operation. Additionally, the staff finds that the applicant's TLAA meets the acceptance criterion in SRP-LR Section 4.2.2.1.2.2 because the applicant's PTS analysis has been accurately projected to the end of the period of extended operation.

4.2.3.3 USAR Supplement

LRA Section A.2.2.3 provides the USAR supplement summary description for the PTS TLAA evaluation. Based on its review of the USAR supplement, the staff concludes that the information in the USAR supplement is an adequate summary description of the evaluation, as required by 10 CFR 54.21(d), and is consistent with SRP-LR Section 4.2.3.2.

4.2.3.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis of PTS has been projected to the end of the period of extended operation. The staff also concludes that the USAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d), and, therefore, is acceptable.

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 P-T limits. The 52 EFPY ART values at the one-quarter of the vessel thickness (1/4T) and three-quarters of the vessel wall thickness (3/4T) locations for all RV beltline materials were provided in LRA Table 4.2-4. These ART values are based on the use of RG 1.99, Revision 2, Position 1.1. The applicant calculated the 1/4T and 3/4T fluence values for the Davis-Besse RV beltline shell materials in accordance with RG 1.99, Revision 2, using the stainless steel cladding thickness, low allow steel vessel wall depths, and the neutron fluence values at the inner wetted surface of the RV listed in LRA Table 4.2-1. According to the applicant, fluence values at the 1/4T and 3/4T locations for the RV inlet and outlet nozzle and associated welds that connect the nozzles to the nozzle belt forging were obtained by adding the attenuation from both the inside and outside surface. Position 2.1 from RG 1.99, Revision 2, was not used since two sets of credible ART surveillance data were not available. Initial RT_{NDT} values and M values for welds WF-182-1 and WF-233 were obtained from BAW-2308, Revision 1-A. The applicant's P-T limit curves and supporting ART calculations are established in the Davis-Besse PTLR. P-T limits are valid only for the operating period (corresponding to a specific RV neutron fluence) specified in the PTLR, and the P-T limits expire upon reaching the fluence limit in the PTLR. In accordance with Davis-Besse TS 5.6.4 requirements, the PTLR shall be periodically updated to include new P-T limits based on revised neutron fluence values corresponding to later operating periods or new credible RV surveillance data, and the revised PTLR shall be submitted to the NRC. The applicant stated that a revised PTLR will be submitted to the NRC in accordance with TS 5.6.4 requirements, before the current P-T limits expires. According to the applicant, the P-T limit curves, as established in the PTLR, will be updated as necessary through the period of extended operation as part of the RV Surveillance Program.

Based on the information above, the applicant concluded that the effects of neutron embrittlement on the RCS P-T limits will be adequately managed during the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii).

4.2.4.2 Staff Evaluation

The staff reviewed LRA Section 4.2.4 on P-T limits to verify, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of RV neutron embrittlement on the P-T limits will be adequately managed for the period of extended operation.

The staff reviewed the applicant's TLAA consistent with the review procedures in SRP-LR, Section 4.2.3.1.3.3, which state that updated P-T limits for the period of extended operation must be available prior to entering the period of extended operation. SRP-LR, Revision 2, Section 4.2.3.1.3.3 also states that either the 10 CFR 50.90 process for P-T limits located in the TS limiting conditions for operation (LCOs) or the TS administrative controls process for P-T limits contained in PTLRs can be considered adequate aging management programs (AMPs) within the scope of 10 CFR 54.21(c)(1)(iii), such that P-T limits will be appropriately maintained through the period of extended operation.

10 CFR Part 50, Appendix G, provides fracture toughness requirements for ferritic materials in the RCPB, including requirements for calculating the RCS P-T limits. Section IV.A.2 of 10 CFR Part 50, Appendix G requires that P-T limits be at least as conservative as those determined in accordance with the ASME Code, Section XI, Appendix G. The P-T limits shall also incorporate a 40 °F temperature shift above the ASME Code, Section XI, Appendix G, limits for core criticality and incorporate the minimum temperature requirements, as specified in Table 1 of the rule. Additionally, the rule requires that the P-T limit calculations account for the effects of neutron radiation on the properties of the RV beltline materials and that these calculations incorporate relevant RV surveillance capsule data. The NRC's guidelines for calculating the effects of neutron radiation on the RV beltline material properties, specifically the ART values, are provided in RG 1.99, Revision 2. P-T limits must be established for HU/CD operations with the core critical and not critical, for hydrostatic pressure tests, and for leak testing conditions. P-T limits specify the maximum RCS HU/CD rates as well as the EFPY operating period corresponding to the fluence level for which the curves are valid. Since the

ART for RV beltline materials increase as a function of neutron fluence, which changes with time, the P-T limits must be updated to ensure that they bound the plant's operating conditions.

LRA Section 4.2.4 provided a discussion of the P-T limits. The current Davis-Besse P-T limits, valid through 32 EFPY or April 22, 2017, whichever is earlier, are established in the Davis-Besse PTLR. The content of the Davis-Besse PTLR is administratively controlled in accordance with Davis-Besse TS Section 5.6.4. TS 5.6.4a specifies that P-T limits shall be established in the PTLR for all required operating and leak testing conditions for operation of the RCS in accordance with TS LCO 3.4.3, which requires that the applicant operate the RCS within the limits specified in the PTLR. TS 5.6.4b specifies the analytical methods used to calculate the P-T limits contained in the PTLR, including the NRC-approved alternative methodology described in BAW-2308, Revision 1-A and Revision 2-A, for determining the initial RT_{NDT} and M values for the Linde 80 weld materials. TS 5.6.4c requires that the PTLR be updated for each RV fluence period or for any revision or supplement thereto, thereby meeting the acceptance criterion from SRP-LR Section 4.2.2.1.3.3 that allows the P-T limits to be managed by the TS administrative controls process for P-T limits contained in PTLRs. The applicant will update the PTLR for new fluence limits prior to operating beyond the current period. The Davis-Besse TS requirements concerning RCS operation and PTLR content ensure that the structural integrity of the RCPB will be maintained in accordance with the requirements of 10 CFR Part 50, Appendix G. Additionally, the applicant specified that the RV Surveillance Program, described in LRA Section B.2.35, will maintain the Davis-Besse P-T limit curves to ensure compliance with 10 CFR Part 50, Appendix G, through the end of the period of extended operation. Since information from the RV Surveillance Program, such as neutron fluence and updated RT_{NDT} values, will be used to adjust the P-T limits as necessary, the staff finds the RV Surveillance Program is also appropriate for managing the P-T limits TLAA.

LRA Table 4.2-4 provided 52 EFPY ART values at the 1/4T and 3/4T locations for all RV beltline materials. The staff confirmed that the 52 EFPY ART values were calculated in accordance with RG 1.99, Revision 2, Position 1.1. The Cu and Ni content values used for the ART calculations are consistent with those listed in the PTLR and the RVID. The staff confirmed that initial RT_{NDT} values and M values listed in LRA Table 4.2-4 for welds WF-233 and WF-182-1 were determined using the alternative methods for Linde 80 welds described in BAW-2308, Revision 1-A. All other initial RT_{NDT} values and M values are consistent with those listed in the RVID. The staff noted that the current P-T limits in the PTLR are for 32 EFPY, but the ART values listed there are for 52 EFPY, and these ART values are consistent with those provided in LRA Table 4.2-4. As discussed in SER Section 4.2.3.2, the applicant is authorized, in License Amendment No. 282, to use the alternative methods of BAW-2308, Revision 1-A and Revision 2-A, for determining the initial RT_{NDT} values and M values for these Linde 80 weld materials. Therefore, future Davis-Besse PTLRs may use the BAW-2308, Revision 1-A and Revision 2-A, methods for determining RT_{NDT} values and M values for Linde 80 beltline welds.

The staff noted that Position 2.1 of RG 1.99, Revision 2, was not used for the 52 EFPY ART calculation because two sets of credible surveillance data are not available for determining the ART values for the RV beltline materials. The staff verified that the use of Position 1.1 of RG 1.99 for the ART calculation is consistent with the latest 32 EFPY PTLR, which was approved by the staff in an SE titled "Davis-Besse Nuclear Power Station, Unit No. 1 Safety Evaluation Regardng the Reactor Coolant System Pressure and Temperature Limits Report, Revision 1," issued by letter dated May 11, 2012. This is also consistent with the staff's RVID. Section 3.3 of the PTLR states that ART values were not calculated using surveillance data since the data was determined to be non-credible. The 32 EFPY PTLR provides a reference to the vendor report that documents the credibility determination, based on the procedures of RG

1.99, Revision 2. Therefore, the licensee's use of the methods of Position 1.1 of RG 1.99, Revision 2 for the 52 EFPY ART calculation is acceptable.

The staff noted that LRA Section 4.2.4 states that "[f]luence values at the 1/4T and 3/4T locations for the RV inlet and outlet nozzles and associated welds that connect the nozzles to the nozzle belt forging were obtained by adding the attenuation from both the inside and outside surface." The staff confirmed the adequacy of this approach by independently calculating the 1/4T and 3/4T neutron fluence values using the attenuation equation (equation 3) in RG 1.99, Revision 2. At the 1/4T location, the staff determined that the applicant's fluence values are approximately thirteen percent higher than the values calculated by the staff using equation 3 from the RG. At the 3/4T location, the staff determined that the applicant's fluence values are approximately 3.8 times higher than the values calculated by the staff using equation 3 from RG 1.99. Therefore, the applicant's method for calculating 1/4T and 3/4T neutron fluence values for these components leads to higher and therefore more conservative ART values than those suggested in RG 1.99, Revision 2. Accordingly, the staff finds acceptable the applicant's methods for calculating neutron fluence values at the 1/4T and 3/4T locations for the RV inlet and outlet nozzles and associated welds that connect these nozzles to the nozzle belt forging, because the values used are more conservative than those calculated from the approved guidance of RG 1.99, Revision 2.

The current Davis-Besse PTLR contains P-T limit curves that are valid through 32 EFPY, calculated using adjusted RT_{NDT} values for the limiting RV beltline shell material. The current Davis-Besse PTLR addresses P-T limit curve calculations for only the limiting RV beltline shell material and the staff could find no indication that the P-T limit calculations considered any RV materials or other RCPB components outside the RV beltline shell region.

The staff notes that 10 CFR Part 50, Appendix G, Paragraph IV.A states the following:

> The pressure-retaining components of the [RCPB] that are made of ferritic materials must meet the requirements of the ASME Code, [Section III], supplemented by the additional requirements set forth in [Paragraph IV.A.2, 'Pressure-Temperature Limits and Minimum Temperature Requirements'], for fracture toughness….

Therefore, 10 CFR Part 50, Appendix G requires that P-T limits be developed for the ferritic materials in the RV beltline (neutron fluence greater than or equal to 1 x 10^{17} n/cm^2, E greater than 1 MeV), as well as ferritic materials not in the RV beltline (neutron fluence less than 1 x 10^{17} n/cm^2, E greater than 1 MeV). Further, 10 CFR Part 50, Appendix G requires that all RCPB components must meet the ASME Code, Section III requirements. The relevant ASME Code, Section III requirement that will affect the P-T limits is the lowest service temperature requirement for all RCPB components specified in Section III, NB-2332(b).

The staff was concerned that consideration of non-RV beltline shell materials and other ferritic RCPB components may define P-T curves that are more limiting than those calculated for the RV beltline shell materials. This may be due to the following factors:

- RV nozzles, penetrations, and other discontinuities, have complex geometries that may exhibit significantly higher stresses than those for the RV beltline shell region. These higher stresses can potentially result in more restrictive P-T limits, even if the reference temperature (RT_{NDT}) for these components is not as high as that of RV beltline shell materials that have simpler geometries.

- Ferritic RCPB components that are not part of the RV may have initial RT_{NDT} values, which may define a more restrictive lowest operating temperature in the P-T limits than those for the RV beltline shell materials.

Therefore, by letter dated May 31, 2012, and as supplemented by letter dated July 26, 2012, the staff issued RAI 4.2.2-4, requesting that the applicant describe how the P-T limit curves to be developed for use in the period of extended operation, and the methodology used to develop these curves, will consider all RV materials (beltline and non-beltline) and the lowest service temperature of all ferritic RCPB materials, consistent with the requirements of 10 CFR Part 50, Appendix G.

In its response dated August 24, 2012, the applicant stated that it used the methods described in B&W topical report BAW-10046-A, Revision 2, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR Part 50, Appendix G," dated June 1986, to develop the P-T limits for Davis-Besse. The applicant also stated that this report addresses all beltline and non-beltline RCPB components. The applicant noted that the staff reviewed and approved the methods described in the topical report for implementation by all B&W plants in its April 30, 1986, SE.

The applicant stated that the current Davis-Besse PTLR includes P-T limits that are valid until 32 EFPY or April 22, 2017, whichever occurs first. These P-T limits were generated using the methods of BAW-10046-A, Revision 2, and the ASME Code, Section XI, Appendix G. The applicant also stated that non-beltline RCPB components have always been considered in the development of P-T limits, based on the analyses performed in BAW-10046-A, Revision 2. BAW-10046-A, Revision 2, determined that the three most controlling regions of the RCPB relative to the P-T limits are the RV closure head region, the RV outlet nozzles, and the RV beltline region. According to the applicant, BAW-10046-A, Revision 2, determined that the inside corner region of the RV outlet nozzles are subject to the highest local stresses due the nozzles' large diameter, and the RV outlet nozzles are more limiting relative to stress than any other RCPB nozzle or piping component. The applicant stated that, considering the above stress concentration effects, the current P-T limit curves are defined by the limiting RV beltline shell material, weld WF-182-1, based on this weld's high degree of embrittlement relative to the outlet nozzles.

With regard to replacement ferritic RCPB components, such as the RV closure head and any future replacement of RCPB components, which would not necessarily be bounded by the RT_{NDT} values assumed in BAW-10046-A for these components, the applicant stated that the ASME Code, Section III, NB-3211(d) requires that protection against nonductile fracture be provided by satisfying one of the following provisions:

1. performing an evaluation of service and test conditions by methods similar to those contained in Appendix G [of the ASME Code, Section III]; or
2. for piping, pump, and valve material with thickness greater than 2.5 in. (64 mm) establishing a lowest service temperature that is not lower than RT_{NDT} (NB-2331) + 100°F (56°C);
3. for piping, pump, and valve material with thickness equal to or less than 2.5 in (64 mm), the requirements of NB-2332(a) shall be met at or below the lowest service temperature as established in the design specification.

Therefore, for replacement components, the applicant noted that an ASME Code, Section III analysis is required to ensure that the new component is bounded by the ASME Code, Section XI, Appendix G, analysis of the RV used to derive the P-T limits.

The staff agreed with the applicant's statement that applying the requirements of the ASME Code, Section III, NB-3211(d) (which includes the lowest service temperature requirement of NB-2332(b)) will ensure that the fracture toughness of replacement ferritic RCPB components at Davis-Besse will comply with the requirements of 10 CFR Part 50, Appendix G. The original RCPB components at Davis-Besse were designed and fabricated to meet the requirements of the ASME Code, Section III, 1968 edition through summer 1968 addenda. This Code edition and addenda predate the lowest service temperature and fracture toughness requirements of NB-3211(d) and NB-2332(b). However, for original ferritic RCPB components not part of the RV, BAW-10046-A, Revision 2, provides assurance that adequate protection against nonductile fracture is provided, based on the establishment of bounding P-T limits for the RV.

The staff notes that, although the BAW-10046-A, Revision 2, methodology, which is used for the development of the current P-T limits for Davis-Besse, has been determined to be acceptable by the staff for demonstrating compliance with 10 CFR Part 50, Appendix G, this methodology assumes that all RV components outside the beltline shell region are not subject to significant neutron embrittlement. Specifically, the BAW-10046-A, Revision-2, analysis of the RV outlet nozzles assumes the nozzles are in the unirradiated state. The staff agrees that the effects of neutron irradiation on the nozzles should be insignificant during the initial 40-year operating period. However, for the period of extended operation, the outlet nozzles are beltline components, and at 52 EFPY, the projected ART value for the nozzles is 82 °F at the 1/4T location, which is significantly greater than the unirradiated RT_{NDT} value of 3 °F. The staff determined that future P-T limit curves to be developed for the period of extended operation should take into consideration both neutron embrittlement effects and high localized stresses at the nozzles' inside corner region to ensure compliance with 10 CFR Part 50, Appendix G, which the initial RAI response did not address.

On October 23, 2012, the staff held a telephone conference call with the applicant to discuss its concerns with the applicant's response to RAI 4.2.4-1. During the telephone conference call, the staff stated that the applicant's response did not provide information that addressed how future P-T limit curves would be developed for the period of extended operation taking into account the neutron embrittlement effects on the extended beltline region and the localized stresses of the inlet and outlet nozzles. During the telephone conference call, the applicant stated that the P-T limit curves are currently limited to 32 EFPY and that additional analysis is required to develop new curves for operation during the period of extended operation. Based on the telephone conference call discussion, the applicant agreed to submit a supplemental response to RAI 4.2.4-1, along with the appropriate LRA revisions, to address how P-T limits for the period of extended operation will take into consideration nozzle embrittlement.

In its supplemental response dated November 2, 2012, the applicant stated that the RV beltline region for 40 years includes only RV shell forgings and welds, whereas the beltline region for the period of extended operation also includes the inlet and outlet nozzles, the Dutchman forging, welds connecting the inlet and outlet nozzles to the nozzle belt forging, and the weld connecting the Dutchman forging to the lower shell forging (collectively referred to as the extended beltline components). Of all extended beltline components, the outlet nozzles are projected to have the highest ART, 82 °F at 52 EFPY, as shown in LRA Table 4.2-4. The applicant stated that P-T limits for the period of extended operation will take into consideration the evaluation of the effects of neutron embrittlement for the extended beltline materials as well as the high localized stresses in the closure head region of the RV and the inside corner of the RV outlet nozzles, which are the largest diameter nozzles in the RCPB.

The applicant's supplemental response included LRA Amendment 35, which revised LRA Section 4.2.4 and the corresponding USAR supplement section provided in LRA Section A.2.2.4 to state that future P-T limit curves for the period of extended operation will be developed based on an evaluation of the effects of neutron embrittlement for the 60-year beltline materials, the stresses in the RV closure head region, and the stresses in the RV outlet nozzles. The revision to LRA Sections 4.2.4 and A.2.2.4 included a comprehensive list of all 60-year beltline components, which includes the 40-year beltline components and the extended beltline components discussed above.

On May 21, 2013, the staff held a telephone conference call with the applicant to request that additional information be added to LRA Section 4.2.4 and and the USAR supplement (LRA Section A.2.2.4) to ensure that the effects of embrittlement on future P-T limit calculations will be addressed for any RV materials that could experience inside surface fluence greater than 1.0E17 n/cm^2 (E > 1.0 MeV), consistent with Revision 2 of the GALL Report. Therefore, by letter dated June 3, 2013, the applicant provided LRA Amendment 43, which further supplemented LRA Sections 4.2.4 and A.2.2.4 to state that the revised P-T limits for the period of extended operation will be generated by taking into consideration the effects of embrittlement on the 60-year RV beltline and extended beltline materials (as listed in these LRA sections), as well as any other components that could experience 52 EFPY inside surface fluence greater than 1.0E17 n/cm^2 (E > 1.0 MeV) during the period of extended operation.

Based on its review, the staff finds the applicant's response, as supplemented, acceptable because the applicant appropriately described how the P-T limit curves to be developed for use in the period of extended operation will consider all ferritic RCPB components, consistent with the requirements of 10 CFR Part 50, Appendix G. Additionally, the applicant adequately described how future P-T limits will consider the effects of neutron embrittlement on all 60-year RV beltline components, as well as the impact of the high localized stresses at the inside corner of the outlet nozzles and the RV closure head region. For ferritic RCPB components that are not part of the RV, which may be replaced in the future, the staff finds that the applicant adequately described how the fracture toughness of these components will be evaluated in accordance with ASME Code, Section III requirements, and any effects on the P-T limits will be accounted for. BAW-10046-A, Revision 2, provides the basis that the fracture toughness of existing ferritic RCPB components outside the RV complies with 10 CFR Part 50, Appendix G, requirements.

The staff finds that the LRA Amendment 35 revisions to LRA Sections 4.2.4 and A.2.2.4 ensure that the effects of both neutron embrittlement and stress concentrations in the extended beltline components will be addressed in future analyses for developing P-T limits for the period of extended operation. Therefore, these LRA revisions provide adequate assurance that future P-T limit curves will be developed such that they are bounding for all ferritic RCPB materials during the period of extended operation, consistent with the requirements of 10 CFR Part 50, Appendix G. Based on the information above, the staff's concern described in RAI 4.2.4-1 is resolved, and OI 4.2.4-1 is closed.

The staff reviewed the applicant's P-T limits TLAA in LRA Section 4.2.4, the applicant's response to RAI 4.2.4-1, the Davis-Besse PTLR, and the TS requirements for RCS P-T limits and PTLR contents, as documented above. Based on this review, the staff found that the applicant adequately demonstrated that the P-T limits at Davis-Besse will be managed under the TS administrative controls process and the RV Surveillance Program to ensure compliance with the requirements of 10 CFR Part 50, Appendix G, through the end of the period of extended operation.

Based on the above evaluation, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the Davis-Besse P-T limits will be adequately managed by the TS administrative controls process and the RV Surveillance Program for the period of extended operation. Additionally, the staff finds that the applicant's TLAA meets the acceptance criterion in SRP-LR Section 4.2.2.1.3.3 because the P-T limits will be adequately managed by the TS administrative controls process for the period of extended operation, as required by 10 CFR 54.21(c)(1)(iii).

4.2.4.3 USAR Supplement

LRA Section A.2.2.3 provides the USAR supplement summary description for the P-T limits TLAA evaluation. Based on its review of the USAR supplement, as revised by LRA Amendment 35, the staff concludes that the information in the USAR supplement is an adequate summary description of the evaluation, as required by 10 CFR 54.21(d), and is consistent with SRP-LR Section 4.2.3.2.

4.2.4.4 Conclusion

On the basis of 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 RCS P-T limits will be adequately managed for the period of extended operation. The staff also concludes that the USAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d), and, therefore, is acceptable.

4.2.5 Low-Temperature Overpressure Protection Limits

4.2.5.1 Summary of Technical Information in the Application

LRA Section 4.2.5 describes the applicant's TLAA for the low-temperature overpressure protection (LTOP) limits. The applicant stated that LTOP is provided in one of the following ways at Davis-Besse:

- Administrative controls are used to assure protection within the existing P-T limits when the pressurizer PORV and the safety valves are not providing over-pressure protection.
- A relief valve in the decay heat removal system suction piping is placed into service when the RCS temperature is below 280 °F.

The applicant stated that the TS LTOP limits were developed based on the NRC-approved methodology in topical report BAW-10046-A, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR Part 50, Appendix G," Revision 2, June 1986. According to the applicant, maintaining the LTOP limits in accordance with ASME Code, Section XI, Appendix G limits, as required by Appendix G of 10 CFR Part 50, assures that the effects of aging on the RCS will be adequately managed for the period of extended operation. The applicant stated that the LTOP limits will be managed during the period of extended operation under the RV Surveillance Program.

Based on the information above, the applicant concluded that the effects of neutron embrittlement on the LTOP limits will be appropriately managed during the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii).

4.2.5.2 Staff Evaluation

The staff reviewed LRA Section 4.2.5 on the LTOP limits to verify, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of RV neutron embrittlement on the LTOP limits will be adequately managed for the period of extended operation.

The staff reviewed the applicant's TLAA consistent with the review procedures in SRP-LR Section 4.7.3.1.3, which state that the applicant shall propose to manage the aging effects associated with the TLAA using an AMP in the same manner as described in the integrated plant assessment (IPA) in 10 CFR 54.21(a)(3). SRP-LR Section 4.7.3.1.3 also states that the applicable AMP is reviewed to verify that the effects of aging on the intended functions are adequately managed consistent with the CLB for the period of extended operation.

In LRA Section 4.2.5, the applicant proposed to manage the aging effects associated with the LTOP limits using the RV Surveillance Program, which is described in LRA Section B.2.35. As stated in LRA Section 4.2.5, LTOP is provided through (1) TS administrative controls, which are used to assure protection within the existing P-T limits when the pressurizer PORV and the two pressurizer safety valves are not in service, and (2) a relief valve in the decay heat removal system suction piping which is placed into service when the RCS temperature is below 280 °F. TS LCOs for the RCS specify that the pressurizer, pressurizer PORV, and pressurizer safety valves shall be operable when the reactor core is critical (Modes 1 and 2—reactor power operation and startup, respectively, as defined in TS Table 1.1-1) or the reactor is in hot standby (Mode 3).

The Davis-Besse LTOP system limits are established directly in the Davis-Besse TSs. Davis-Besse TS 3.4.12 provides LTOP requirements, wherein LCO 3.4.12 specifies that the decay heat removal system relief valve shall be operable with a lift setting less than or equal to 330 pounds per square inch gauge (psig) when the reactor is in hot shutdown (Mode 4), cold shutdown (Mode 5), or Mode 6 (refueling) with the RV head in place. Consistent with information provided in LRA Section 4.2.5, TS Table 1.1 specifies that, when the reactor is in Modes 4, 5, or 6, RCS average temperature shall be less than 280 °F. The staff agreed that these TS LTOP limits are appropriate for ensuring protection of the RV from low-temperature overpressurization events because the applicability of the LTOP TS requirements to Modes 4, 5, and 6 ensures that all ferritic RV materials will be protected from brittle fracture due to overpressurization events at temperatures less than 280 °F, which exceeds the 52 EFPY ART values for the RV beltline materials by a substantial margin. The staff also determined that the decay heat removal system relief valve lift setpoint of less than or equal to 330 psig for Modes 4, 5, and 6, as specified in TS LCO 3.4.12, also ensures that that RCS operation is bounded by the CLB P-T limits, as established in the PTLR, at temperatures less than 280 °F. Therefore, the staff found that the applicant's LTOP limits are appropriately maintained in the TSs to ensure that the RCS is operated in accordance with 10 CFR Part 50, Appendix G requirements.

The staff confirmed that the current TS requirements for LTOP are valid through 32 EFPY, and the Davis-Besse LTOP limits will be managed through continued implementation of the RV Surveillance Program during the period of extended operation, as stated by the applicant in LRA Section 4.2.5. Since the LTOP limits are related to the P-T limits, and the temperature at which the LTOP system must be operable is related to the limiting material RT_{NDT}, both of which are adjusted based on information from the RV Surveillance Program, the staff finds the RV Surveillance Program is appropriate for management of the LTOP limits. All future revisions to the TS LTOP requirements will be implemented through the license amendment process, in accordance with 10 CFR 50.90.

Based on the above considerations and the staff's determination that the P-T limits will be adequately managed for the period of extended operation, as discussed in SER Section 4.2.4, the staff determined that the Davis-Besse LTOP limits will be appropriately managed by a combination of the TS administrative controls process and the RV Surveillance Program for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii).

Based on the above evaluation, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the Davis-Besse LTOP limits will be adequately managed by the TS administrative controls process and the RV Surveillance Program for the period of extended operation. Additionally, the staff finds that the applicant's TLAA meets the acceptance criteria in SRP-LR Section 4.7.2.1 because the effects of aging on the intended function will be adequately managed for the period of extended operation.

4.2.5.3 USAR Supplement

LRA Section A.2.2.5 provides the USAR supplement summary description for the LTOP limits TLAA evaluation. Based on its review of the USAR supplement, the staff concludes that the information in the USAR supplement is an adequate summary description of the evaluation, as required by 10 CFR 54.21(d), and is consistent with SRP-LR Section 4.7.3.2.

4.2.5.4 Conclusion

On the basis of 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 RCS LTOP limits will be adequately managed for the period of extended operation. The staff also concludes that the USAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d), and, therefore, is acceptable.

4.2.6 Intergranular Separation (Underclad Cracking)

4.2.6.1 Summary of Technical Information in the Application

LRA Section 4.2.6 describes the applicant's TLAA for underclad cracking in the Davis-Besse SA-508, Class 2 RV forgings. According to the applicant, BAW-10013-A, "Study of Intergranular Separations in Low Alloy Steel Heat Affected Zones under Austenitic Steel Weld Cladding," February 1972, documents a fracture mechanics analysis for demonstrating that the critical crack size required to initiate fast fracture of underclad cracks is several orders of magnitude greater than the assumed maximum flaw size plus predicted flaw growth due to design fatigue cycles. The applicant stated that this analysis is based on 40 year cyclic loading, and a 32 EFPY assessment of radiation embrittlement for determining end-of-life fracture toughness properties. The LRA states that the report concluded that the intergranular separations found in B&W vessels would not lead to vessel failure during a 40-year/32 EFPY operating life.

The applicant stated that the evaluation of underclad cracking in the Davis-Besse SA-508 Class 2 forgings for the period of extended operation is consistent with the methodology described in Appendix C of BAW-2251-A, "Demonstration of the Management of Aging Effects for the Reactor Vessel," August 1999. The applicant also stated that the plant-specific analysis was performed for 60-years using the 52 EFPY fracture toughness information, applied stress intensity factor solutions, and fatigue crack growth correlations for SA-508, Class 2 material.

The applicant stated that the underclad crack analysis was applied to the beltline and the nozzle belt regions of the RV and that both axially- and circumferentially-oriented flaws were considered in the evaluation. The applicant stated that for an axially-oriented flaw, the limiting location for satisfying the requirements of IWB-3612 is at the lower end of the nozzle belt forging, where the thickness transitions from 8.438 to 12.0 in. The applicant stated that the maximum crack growth, considering normal/upset condition transients with associated 60-year projected cycles for the period of extended operation was determined to be 0.043 in., which results in a final flaw depth of 0.396 in. The applicant also stated that the maximum applied stress intensity factor for normal and upset conditions results in a fracture toughness margin of 3.67, which is greater than the acceptance criterion of 3.16. The applicant further stated that the maximum applied stress intensity factor for emergency and faulted conditions results in a fracture toughness margin of 1.43, which is greater than the acceptance criterion of 1.41. The applicant concluded that the postulated underclad cracks in the Davis-Besse RV forgings are acceptable for continued safe operation through the period of extended operation.

Based on the information above, the applicant concluded that its analysis of RV forging underclad cracking has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

LRA Section 4.2.6, as amended by letter dated March 9, 2012 (LRA Amendment 24), includes the applicant's disposition of intergranular separation (underclad cracking) flaw growth for its new RV head installed in the fall of 2011. The applicant stated that this TLAA was applicable to the RV closure head that was in place at the time of development of the LRA, but it does not apply to the new replacement RV closure head installed during the October 2011 outage in accordance with Confirmatory Action Letter No. 3-10-001. The applicant stated that the replacement RV closure head installed during the October 2011 outage was fabricated using SA-508, Class 3 material that is not susceptible to intergranular separations. Therefore, amended LRA Table 4.1-1 notes that the underclad cracking flaw growth is not a TLAA for the RV head.

4.2.6.2 Staff Evaluation

The staff reviewed LRA Section 4.2.6 on RV forging underclad cracking to verify, pursuant to 10 CFR 54.21(c)(1)(ii), that the applicant provided an acceptable analysis of projected underclad cracking in its SA-508, Class 2 RV forgings for the period of extended operation. The staff also reviewed this section to verify that the TLAA on intergranular separation (underclad cracking) flaw growth does not apply to its new RV closure head replaced in fall 2011.

In Confirmatory Action Letter Number 3-10-001, the applicant voluntarily committed to shutdown the Davis-Besse plant no later than October 1, 2011, and replace the RV closure head. The applicant stated that the replacement RV closure head/head flange was fabricated using SA-508 Class 3 material, which is not susceptible to intergranular separations. The staff reviewed the applicant's Confirmatory Action Letter No. 03-10-001 and confirmed that the vendor is to supply the replacement RV head with SA-508, Class 3 forging, and that the SA-508 specification for Class 3 materials reduced the maximum allowable chromium alloying content from those specified in the SA-508 specification for corresponding Class 2 forging materials. The staff noted that industry literature indicates that this change in the alloying content makes the SA-508, Class 3 forging materials more resistant to the phenomenon of underclad cracking than are SA-508, Class 2 forging materials. By letter dated November 7, 2011 (L-11-301), the applicant notified the NRC that it had completed the actions required by Confirmatory Action Letter 3-10-001, including replacing the RV head. The attachment to this letter specified that, as of October 30, 2011, the new RV head was placed into containment and the prior head had been removed from containment.

On September 30, 2011, the NRC completed an integrated inspection at Davis-Besse. The results of this inspection are documented in Davis-Besse Integrated Inspection Report 05000346/2011004, dated October 26, 2011. As documented in the inspection report, the NRC inspector reviewed the following documents related to the replacement RV closure head/head flange: (1) Certified Material Test Report: JQA-02-173, “Closure Head Forging—Japan Steel Works, LTD,” dated August 27, 2002; and (2) Drawing: AREVA-02-5053158E-00, “Replacement Reactor Vessel Closure Head,” Revision 6. Based on his review of the certified material test report and drawing for the replacement RV closure head/head flange, the inspector confirmed that the RV closure head/head flange was fabricated using SA-508, Class 3 forging material, which is more resistant to underclad cracking.

Based on this review, the staff finds that the intergranular separation (underclad cracking) flaw TLAA in SRP-LR Table 4.1-3 is not applicable to the flange in the new upper RV closure head and does not need to be identified as a TLAA for the following reasons:

- The analyses are only applicable to forgings designed to SA-508, Class 2 specifications.
- The flange for the new upper RV closure head is designed and fabricated from specification requirements (i.e., SA-508, Class 3 specification requirements), which make the material more resistant to the effect of RV underclad cracking.

For the remainder of the RV shell, the staff reviewed the applicant’s TLAA, consistent with the review procedures in SRP-LR Section 4.7.3.1.2, which state that the “documented results of the revised analyses are reviewed to verify that their period of evaluation is extended, such that they are valid for the period of extended operation (e.g., 60 years).” The staff also assessed the applicant’s criteria against the staff’s recommended position and criteria on RV underclad cracking in RG 1.43, “Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components,” May 1973.

Intergranular cracking in the HAZ of low-alloy steel RV forgings underneath stainless steel welded cladding (i.e., underclad cracking) has been observed for specific materials and cladding process conditions. In accordance with RG 1.43, underclad cracking has been reported only in SA-508, Class 2 RV forgings manufactured to a coarse-grain practice when clad using “high-heat-input” submerged arc welding processes. Cracking has not been observed in SA-508, Class 2 materials clad using “low-heat-input” processes, which are controlled to minimize heating of the base metal.

All SA-508, Class 2 RV beltline forgings at Davis-Besse are deemed potentially susceptible to underclad cracking because these forgings are manufactured to a coarse grain practice and clad using “high-heat-input” submerged arc welding processes. The applicant stated that a plant-specific fracture mechanics analysis of the susceptible RV forgings was performed for 60 years of facility operation. This analysis was based on 52 EFPY fracture toughness parameters; applied stress intensity factor solutions for normal, upset, emergency, and faulted conditions; and SA-508, Class 2 fatigue crack growth correlations based on 60-year projected cycles for the period of extended operation. The applicant performed the analysis for the limiting RV beltline region and the nozzle belt region.

The staff-approved BAW-2274, "Fracture Mechanics Analysis of Postulated Underclad Cracks in B&W Designed Reactor Vessels for the Period of Extended Operation," December 1996, in Appendix C of its final SE for BAW-2251. BAW-2274 updated the BAW-10013 underclad cracking analysis for license renewal considerations. The staff approved referencing of both BAW-2251-A and BAW-2274 in LRAs by letter dated April 26, 1999. In Appendix C of its final SE for BAW-2251, the staff concluded that the B&WOG methodology for flaw evaluations of postulated underclad cracks for the period of extended operation, as described in BAW-2274, is consistent with the current well-established flaw evaluation procedure and acceptance criteria in the ASME Code, Section XI, IWB-3612; therefore, it is adequate. The additional conservatism associated with the B&WOG methodology includes the following:

- using the maximum crack depth of 0.165 in. reported by the industry as the initial crack depth instead of the depth of 0.10 in. reported on the B&W RVs
- assuming all underclad cracks are surface cracks
- using the fatigue crack growth rate for surface flaws in a water reactor environment
- producing results equivalent to a safety factor 17 percent more than that specified by the ASME Code, Section XI, IWB-3612 acceptance criteria for Service Level A and B loading and 72 percent more than that specified by IWB-3612 for Service Level C and D loading

The staff noted that, as stated in LRA Section 4.2.6, the plant-specific underclad cracking analysis was consistent with the NRC-approved methods for B&W plants, with the exception of the applicant's assumed initial axial flaw depth of 0.353 in. and flaw length of 2.12 in. These initial flaw dimensions are more than twice as large as the largest detected underclad flaws used as the basis for the BAW-2274 report. Therefore, the applicant's assumed initial flaw dimensions ensure that the plant-specific analysis is based on a conservative assumption.

The applicant reported that the results of the 60-year plant-specific analysis indicated that the postulated underclad axial flaws remain in compliance with the acceptance criteria specified in IWB-3612 through the period of extended operation for Service Levels A, B, C and D, accounting for fatigue crack growth through 60 years. LRA Section 4.2.6 states that axially-oriented underclad flaws located in the thickness transition at the lower end of the nozzle belt region were determined to be the most bounding due to the stress intensity factor solutions for these flaws at this location in the RV.

The applicant applied the maximum stress intensity factors for normal/upset and emergency/faulted conditions to determine the minimum acceptable fracture toughness values, in accordance with the acceptance criteria of IWB-3612. For both sets of conditions, the applicant determined that the actual material fracture toughness exceeded the minimum fracture toughness requirements at the end of the period of extended operation. Accordingly, the applicant concluded, in LRA Section 4.2.6, that the postulated underclad cracks in the Davis-Besse RV are acceptable for continued operation through the period of extended operation.

Based on its review of the applicant's evaluation of its RV forging underclad cracking TLAA, as documented above, the staff determined that the applicant described its underclad cracking analysis in sufficient detail and, therefore, adequately demonstrated that the underclad cracking satisfied the flaw analytical acceptance criteria in IWB-3612 for the period of extended operation. Furthermore, the staff found that the applicant's description of the plant-specific 60-year analysis of the postulated underclad cracks is consistent with the methodologies used in BAW-2274, with the exception that a more conservative initial flaw size was used in the plant-specific analysis. However, since the document for the 60-year plant-specific analysis of underclad cracking is not in the list of references provided in LRA Section 4.8, the staff requested this reference by letter dated March 17, 2011, in RAI 4.2.6-1.

In its response dated April 15, 2011, the applicant provided a non-proprietary version of the plant-specific report, AREVA Document 86-910440-000, "Fracture Mechanics Analysis of Postulated Underclad Cracks in the DB-1 Reactor Vessel for 60 Years," July 2010. The staff reviewed the report and found the fracture mechanics analysis of the postulated underclad cracks documented in the AREVA report to be acceptable, and consistent with LRA Section 4.2.6. The staff's concern described in RAI 4.2.6-1 is resolved.

Based on its evaluation of the applicant's plant-specific 60-year analysis of postulated underclad cracking in the SA-508, Class 2 RV forgings at Davis-Besse, including its response to RAI 4.2.6-1, the staff found that the underclad cracking TLAA has been projected to meet the ASME Code, Section XI, IWB-3612, flaw evaluation analytical acceptance criteria for Levels A, B, C, and D service loadings through the period of extended operation.

Based on its evaluation, as discussed above, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis of RV forging underclad cracking has been projected to the end of the period of extended operation. Additionally, the staff finds that the applicant's TLAA meets the acceptance criteria in SRP-LR Section 4.7.2.1 because the analysis of RV forging underclad cracking has been projected to the end of the period of extended operation.

4.2.6.3 USAR Supplement

LRA Section A.2.2.6 provides the USAR supplement summary description for the RV forging underclad cracking TLAA evaluation. By letter dated March 9, 2012, the applicant provided LRA Amendment 24. This amendment, in part, revised LRA Section A.2.2.6, to indicate that (1) the RV closure head/head flange was replaced in the fall of 2011, and (2) the replacement RV closure head/head flange was fabricated using SA-508, Class 3 material, which is not susceptible to intergranular separations and underclad cracking. Based on its review of the USAR supplement, the staff concludes that the information in the USAR supplement is an adequate summary description of the evaluation, as required by 10 CFR 54.21(d), and is consistent with SRP-LR Section 4.7.3.2.

4.2.6.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis of RV forging underclad cracking has been projected to the end of the period of extended operation. The staff also concludes that the USAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d), and, therefore, is acceptable.

4.2.7 Reduction in Fracture Toughness of Reactor Vessel Internals

4.2.7.1 Summary of Technical Information in the Application

LRA Section 4.2.7 describes the applicant's TLAA for the reduction in fracture toughness of the RVI. The applicant cited USAR Appendix 4A, which describes the detailed stress analysis of the internals under accident conditions for the current term of operation. According to the applicant, the analysis shows that the internals will not fail because the stresses are within established limits. The applicant stated that the effect of irradiation on the mechanical properties and deformation limits for the RVI was also evaluated for the current term of operation. The applicant also stated that the aforementioned analysis concluded that the RVI will have adequate ductility to absorb local strain at the regions of maximum stress intensity and that irradiation will not adversely affect deformation limits.

The applicant stated that the impact of the MUR power uprate on the structural integrity of the RVI components was evaluated. The applicant concluded that the temperature changes due to the MUR power uprate are bounded by those used in the existing analyses. As part of the MUR power uprate, the applicant stated that, "[a]s appropriate, FENOC commits to incorporate recommendations from MRP [Materials Reliability Program] inspection guidelines into the RVI program at Davis-Besse Nuclear Power Station, Unit, No. 1."

The applicant stated that this TLAA will be managed during the period of extended operation through the implementation of the PWR RVI Program.

Based on the information above, the applicant concluded that the effects of neutron embrittlement on the reduction in fracture toughness for the RVI will be appropriately managed during the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii).

4.2.7.2 Staff Evaluation

The staff reviewed LRA Section 4.2.7 on the reduction in fracture toughness for the RVI to verify, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of neutron embrittlement on the RVI will be adequately managed for the period of extended operation.

The staff reviewed the applicant's TLAA consistent with the review procedures in SRP-LR Section 4.7.3.1.3, which state that the applicant shall propose to manage the aging effects associated with the TLAA using an AMP in the same manner as described in the IPA in 10 CFR 54.21(a)(3). SRP-LR Section 4.7.3.1.3 also states that the applicable AMP is reviewed to verify that the effects of aging on the intended functions are adequately managed, consistent with the CLB for the period of extended operation.

Exposure of stainless steel RVI components to high-energy neutron radiation during the period of extended operation could result in a significant reduction in fracture toughness, depending on the material, irradiation temperature, and neutron fluence.

The staff determined that the reduction in fracture toughness of the stainless steel RVI is a TLAA that should be managed during the period of extended operation. The staff reviewed LRA Section 4.2.7 to determine if the applicant's TLAA of the reduction in fracture toughness for the RVI demonstrates that the effects of embrittlement on these components will be adequately managed during the period of extended operation, pursuant to 10 CFR 54.21(c)(1)(iii). The applicant appropriately referenced the Davis-Besse PWR RVI Program for managing the loss of fracture toughness for the stainless steel RVI components. The staff reviewed the Davis-Besse

PWR RVI Program, as described in LRA Section B.2.32, and amended through LRA Amendment 15, and confirmed that it manages loss of fracture toughness due to neutron embrittlement of RVI components. The staff's review of the Davis-Besse PWR RVI Program is provided in SER Section 3.0.3.3.6.

The staff noted that cast austenitic stainless steel (CASS) components are susceptible to thermal embrittlement in addition to neutron embrittlement. As such, the reduction in fracture toughness for CASS RVI components should account for the effects of both neutron embrittlement and thermal embrittlement. The staff noted that LRA Aging Management Review (AMR) Results for the RVI (LRA Table 3.1.2-2) list many CASS RVI components. By letter dated March 17, 2011, the staff issued RAI 4.2.7-1 requesting that the applicant discuss how the effects of thermal embrittlement will be addressed for managing the reduction in fracture toughness for the CASS RVI components, with respect to thermal embrittlement susceptibility screening, supplemental examinations of CASS components, and component-specific evaluations of reduction in fracture toughness for the susceptible CASS RVI components under the Davis-Besse PWR RVI Program.

In its April 15, 2011, response to RAI 4.2.7-1, the applicant stated that the screening of the CASS RVI components and component-specific evaluations of the reduction in fracture toughness due to neutron embrittlement and thermal aging was performed as part of the development of the EPRI MRP PWR RVI inspection and evaluation guidelines.

These guidelines are documented in EPRI MRP Topical Report No. 1016596 (MRP-227), Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," January 2009. The staff noted that the NRC-approved version of the MRP-227 report, MRP-227-A, includes the staff's December 2011 final SE on MRP-227 as an attachment. The MRP-227-A report provides inspection and evaluation guidelines acceptable to the staff for implementation in plant-specific PWR internals programs. The Davis-Besse PWR RVI Program, as amended, is based on the MRP-227 inspection and evaluation guidelines. As an input document to MRP-227, the MRP-189 report (EPRI Report No. 1018292, "Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items (MRP-189, Revision 1)," March 2009 (Agencywide Document Access and Management (ADAMS) Accession No. ML091671777)) performed screening of the CASS RVI components and included ferrite and molybdenum content in the parameters screened. LRA Table 3.1.2-2 lists four RVI component types made from CASS, for which reduction in fracture toughness is managed under the PWR RVI Program.

The staff found the applicant's response to RAI 4.2.7-1 acceptable because the screening of CASS RVI components to determine susceptibility to thermal aging is performed in accordance with MRP-227, and component-specific evaluations of reduction in fracture toughness due to both neutron embrittlement and thermal embrittlement were performed as part of the development of MRP-227 for the CASS RVI components listed in LRA Table 3.1.2-2. In addition, the staff's concern with appropriate management of fracture toughness reduction for CASS RVI components will be addressed by the applicant through implementation of its PWR RVI Program. Therefore, the staff's concern described in RAI 4.2.7-1 is resolved.

The staff noted that applicant's proposal to implement MRP-227 as the basis for its plant-specific PWR RVI Program must address all of the plant-specific and vendor-specific action items associated with plant-specific implementation of MRP-227, as specified in Section 4.2 of the staff's SE on MRP-227. The staff's evaluation of the PWR RVI Program is provided in SER Section 3.0.3.3.6.

Based on the above evaluation, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the fracture toughness of the RVI will be adequately managed by the PWR RVI Program for the period of extended operation. Additionally, the staff finds that the applicant's TLAA meets the acceptance criteria in SRP-LR Section 4.7.2.1 because the effects of aging on the intended function will be adequately managed for the period of extended operation.

4.2.7.3 USAR Supplement

LRA Section A.2.2.7 provides the USAR supplement summary description for the reduction in fracture toughness of the RVI TLAA evaluation. Based on its review of the USAR supplement, the staff concludes that the information in the USAR supplement is an adequate summary description of the evaluation, as required by 10 CFR 54.21(d), and is consistent with SRP-LR Section 4.7.3.2.

4.2.7.4 Conclusion

On the basis of 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 RVI will be adequately managed for the period of extended operation. The staff also concludes that the USAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d), and, therefore, is acceptable.

4.3 <u>Metal Fatigue</u>

LRA Section 4.3 provides the assessment of metal fatigue as a TLAA for Davis-Besse license renewal. The applicant's assessment is documented in the following major subsections of LRA Section 4.3:

- LRA Section 4.3.1—describes significant characteristics of fatigue cycles and the monitoring activities performed by the applicant's Fatigue Monitoring Program.
- LRA Section 4.3.2—Class 1 vessels, pumps, and major components (in 4.3.2.2) and Class 1 piping and valves (in 4.3.2.3)
- LRA Section 4.3.3—non-Class 1 piping and in-line components (in 4.3.3.1) and non-Class 1 major components (in 4.3.3.2)
- LRA Section 4.3.4—effects of reactor coolant environment on fatigue

The staff's evaluation of LRA Section 4.3.1 is documented in SER Section 4.3.1.2 below. The description and staff's evaluation of above listed Sections 4.3.2, 4.3.3, and 4.3.4 are documented in SER Sections 4.3.2, 4.3.3, and 4.3.4, respectively.

4.3.1 Fatigue Cycles

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 fatigue TLAAs. The applicant stated that its ASME Code Class 1 components are designed for cyclic loads due to temperature and pressure changes in the RCS, expected from normal unit load transients, reactor trips, startup and shutdown operations, and earthquakes. USAR Table 5.1-8 lists the 14 original design transients for the RCS; however, over the life of the plant, additional transients were identified, including analyzed transients for new components and non-RCS components. As an example, in evaluating its response to NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," the applicant redefined the HU/CD transients. These redefined transients, and other transients modified to include thermal stratification and striping, were provided in LRA Table 4.3-1 along with the 14 original design transients.

The number of design cycles and 60-year projections for these transients are provided in LRA Table 4.3-1. These projections were obtained by first compiling the number of cycles accrued from plant startup until February 2008 and, then, linearly extrapolating to 60 years of operation. When the projected cycles were compared with the number of design cycles in LRA Table 4.3-1, transients 9C, 9D, and 32 were the only transients affecting Class 1 components for which the 60-year projected cycles exceeded the design cycles. The applicant provided further discussion of these transients and stated that, since the components affected by these transients may be reanalyzed for other reasons, it will manage fatigue of these components for the period of extended operation rather than reanalyze for the possible additional cycles. The applicant also stated that its Fatigue Monitoring Program manages metal fatigue by monitoring the cycles incurred and assures that corrective action will be taken prior to any analyzed numbers of events being exceeded.

4.3.1.2 Staff Evaluation

The staff reviewed LRA Section 4.3.1 to verify that the design transients (original and modified), which are significant fatigue contributors, are monitored to ensure that the applicant's design basis fatigue evaluations remain valid. The staff also reviewed the methodology used by the applicant to obtain the 60-year projections. TS Section 5.5.5, "Allowable Operating Transient Cycles Program," (AOTC) requires controls to track the cyclic and transient occurrences provided in USAR Section 5 and USAR Table 5.1-8 to ensure that components are maintained within the design limits. The staff's review of the Fatigue Monitoring Program, which includes the AOTC monitoring activities, is documented in SER Section 3.0.3.2.6.

During its review of the Fatigue Monitoring Program, the staff was not able to verify which transients are monitored and are considered fatigue-significant because there were several differences in various transient descriptions and cycle counts between LRA Table 4.3-1, the AOTC procedure, and USAR Table 5.1-8. By letter dated April 20, 2011, the staff issued RAI B.2.16-1 requesting that the applicant clarify and justify the discrepancies between the program implementation procedure, USAR Table 5.1-8, and LRA Table 4.3-1. The staff's evaluation of the applicant's response to RAI B.2.16-1 is documented in SER Section 3.0.3.2.6.

The staff noted that USAR Table 5.1-8 includes the classification of transients by the plant condition (normal, upset, etc.); however, LRA Table 4.3-1 includes several transients that are not listed in the USAR, along with the classification of the transient. LRA Section 4.3.2.2 states that cumulative usage factors (CUFs) for the Class 1 components are calculated based on normal and upset design transient definitions contained in the component design specifications. However, the staff noted that transient 9, "Rapid Depressurization," is classified as "Emergency" in USAR Table 5.1-8. Therefore, it is not clear if LRA Table 4.3-1 includes all emergency transients that were used in the fatigue analyses. By letter dated May 2, 2011, the staff issued RAI 4.3-1, Request 1, asking the applicant to clarify whether all fatigue significant transients in the fatigue TLAAs have been listed in LRA Table 4.3-1 and to identify the plant condition for each transient. The staff also requested in Request 2 that the applicant confirm whether the design basis fatigue evaluations included emergency and test conditions in addition to the normal and upset conditions.

In its response to Request 1, dated June 17, 2011, the applicant stated that LRA Table 4.3-1 includes all fatigue significant transients that are included in the metal fatigue TLAAs. Furthermore, these transients are consistent with the applicant's AOTC procedure and RCS functional specification, which is the primary source of design transients for the B&W-supplied RCS components. The staff noted that LRA Table 4.3-1 was previously amended in response to RAI B.2.16-1 by letter dated June 3, 2011. The staff's review of the applicant's response to RAI B.2.16-1 is documented in SER Section 3.0.3.2.6.

The staff finds the applicant's response to Request 1 of RAI 4.3-1 acceptable because the applicant confirmed that all fatigue significant transients included in its metal fatigue TLAAs are captured in amended LRA Table 4.3-1. These transients are incorporated into the applicant's Fatigue Monitoring Program, which ensures that the component fatigue usage does not exceed the design limit during the period of extended operation.

In its response to RAI 4.3-1, Request 2, dated June 17, 2011, the applicant stated that, from its review of the design report summaries for RCS components, the fatigue analyses include test transients, normal and upset transients, and operational basis earthquakes. The applicant also stated that the only CUF that included an emergency event was for the RV studs where the design CUF of 0.70 was conservatively increased by 0.026 to include 20 natural circulation cooldown events. The applicant stated that the incremental fatigue due to the emergency event is not required by ASME Code, Section III, Subsection NB-3224.5, and it amended LRA Section 4.3.2.2 to more accurately reflect the ASME Code Section III requirements.

The staff finds the applicant's response to Request 2 of RAI 4.3-1 acceptable because the applicant confirmed that test transients, normal and upset transients, and operational basis earthquakes are included in the metal fatigue TLAAs, which are tracked by the applicant's Fatigue Monitoring Program, and the analysis for the RV studs were the only components that included an emergency event. The staff's concerns described in RAI 4.3-1, Requests 1 and 2, are resolved.

LRA Table 4.3-1 states that transients 19, 20A, 20B, 20C, 23A, 23B, 23C, and 23D are not fatigue significant events, and transients 25A and 25B are not fatigue events; therefore, the applicant determined that the monitoring of these transients is not needed. However, the staff did not find a basis in the LRA to explain why these transients are not fatigue contributors and do not need to be monitored. By letter dated May 2, 2011, the staff issued RAI 4.3-10 requesting that the applicant justify why these transients are not considered fatigue significant events and why these transients do not need to be monitored by the Fatigue Monitoring Program during the period of extended operation.

In its response dated June 17, 2011, the applicant stated that the transients from LRA Table 4.3-1 do not need to be counted under the cycle-counting activities of its Fatigue Monitoring Program, as described below.

The applicant clarified that transient 19, "Feed and Bleed operation," occurs when RCS boron concentration is changed by introducing borated or deborated water through the makeup system. The stress analysis for the makeup nozzle was reviewed by the applicant, which indicated that transient 19 does not have any fatigue contribution. The applicant stated that the expected cycles for transient 20, "Makeup and Pressurizer Spray transients," are low compared to the large number of design cycles; therefore, transient 20 has very little impact on fatigue.

Transient 23, “Steam Generator Filling, Draining, Flushing, and Cleaning,” occurs at temperatures less than 225 °F; therefore, the applicant determined that little or no contribution to fatigue of the steam generators (SGs) is expected. Transient 25, “Pressurizer Heaters,” is only applicable to the electrical heaters; therefore, the applicant determined that there is no contribution to fatigue of the pressurizer or pressurizer heater elements.

The staff finds the applicant’s basis for not monitoring transient 19 acceptable because the stress analysis confirmed that the contribution to fatigue is insignificant to the CUF value. The staff also noted that in order for the applicant to reach the cycle limit for the transient 20C it must occur once a day for 60 years without considering RFOs. Similarly, the staff noted that transients 20A and 20B must occur approximately 183 and 1.4 cycles, respectively, per day for 60 years, without considering RFOs, in order to reach the cycle limit. The staff reviewed the applicant’s USAR and noted that the pressurizer spray system and makeup system operate in conjunction to accommodate changes in the reactor coolant volume due to small temperature changes. Therefore, the staff finds it reasonable that the applicant does not monitor transient 20 because the design cycles are large compared to the number of expected cycles and small temperature changes will not result in significant accumulation of fatigue usage. The staff finds the applicant’s basis for not monitoring the transient 23 from LRA Table 4.3-1 acceptable because the components affected by this transient remain below the temperature threshold of 225 °F; therefore, the cyclic fluctuation in temperatures is not substantial enough to cause a significant impact to the CUF of SG components. The staff finds the applicant’s basis for not monitoring transient 25 acceptable because this transient is only applicable to the electrical heaters and is not applicable to those components in the pressurizer with a CUF value.

The staff finds the applicant’s response acceptable because the applicant is monitoring all transients that cause cyclic strain that are significant contributors to the fatigue usage factor that have been included in the applicant’s design basis fatigue evaluations, consistent with the recommendations in the GALL Report AMP X.M1. Further, the applicant justified not monitoring select transients, as described above, and the staff found these justifications acceptable. The staff’s concern described in RAI 4.3-10 is resolved.

LRA Table 4.3-1 indicated that transient 22A, “Test-High Pressure Injection System,” corresponds to transient 12 in USAR Table 5.1-8 and transient 3, “Power change 8-100%,” and transient 4, “Power change 100-8%,” correspond to transient 3 in USAR Table 5.1-8. The applicant provided technical justifications for not monitoring these transients in its Fatigue Monitoring Program in LRA Table 4.3-1. However, the staff noted that TS 5.5.5 requires cycle counting of the applicant’s design basis transients in USAR Table 5.1-8, unless the USAR specifically explains why monitoring is not required. The staff noted that the Revision 26 of USAR Table 5.1-8 indicates that these transients are applicable to TS 5.5.5 and are not excluded from monitoring. By letter dated May 2, 2011, the staff issued RAI 4.3-11 requesting that the applicant confirm that the “Test-High Pressure Injection System,” “Power change 8-100%,” and “Power change 100-8%” transients are the only transients in LRA Table 4.3-1 and USAR Table 5.1-8 that require counting per TS 5.5.5 but are not counted by the Fatigue Monitoring Program. If not, the staff asked the applicant to identify any additional transients that require counting per TS 5.5.5 but are not counted by the Fatigue Monitoring Program. The staff also asked the applicant to clarify whether USAR Table 5.1-8 and TS 5.5.5 currently do not require these transients to be monitored and to justify why these transients can be omitted from monitoring without justification in the USAR and the applicant’s cycle-counting procedure.

In its response dated June 17, 2011, the applicant stated that a response was previously provided in letter dated June 3, 2011, in response to RAI B.2.16-1, to address whether USAR

Table 5.1-8 currently required the three aforementioned transients to be monitored. The staff noted that the applicant will update USAR Table 5.1-8, which is part of the applicant's CLB and provide the technical justification as to why the monitoring of these transients can be omitted. LRA Table 4.3-1, as amended by letter dated June 3, 2011, states the "Power change 8-100%" and "Power change 100-8%" transients could not credibly approach the large number of design cycles during the period of extended operation because the plant is not a load following plant. The staff noted that power changes at a base-loaded plant are normally the result of RFOs, maintenance, post-reactor trip startups, or TS action statements. Since there is a large number of design cycles for these two transients and the power changes at the applicant's plant are not the result of load following, the staff finds it reasonable that these two transients are not monitored by the Fatigue Monitoring Program. The staff's evaluation of RAI 4.3-2 discusses the basis that the "Test-High Pressure Injection System" transient can be omitted from monitoring by the Fatigue Monitoring Program. The staff's review of RAI B.2.16-1 is documented in SER Section 3.0.3.2.6.

The staff finds the applicant's response acceptable because, with the revision of USAR Table 5.1-8 to provide the technical justifications for not monitoring these transients, the applicant ensured that its CLB accurately reflects the transients that are monitored by the Fatigue Monitoring Program. The staff's concern described in RAI 4.3-11 is resolved.

In its review of LRA Section 4.3.1.2, "Projected Cycles," the staff noted that the elbowlets in HPI nozzles 1-1 and 1-2 were limited to 13 cycles of transients 9A, "Rapid RCS Depressurization 1-1," and 9B, "Rapid RCS Depressurization 1-2." The current cycles are 9 and 8 for HPI nozzles 1-1 and 1-2, respectively. During its audit, the staff noted discrepancies in the cycle count for transient 8, "Rapid Depressurization," of USAR Table 5.1-8, as described in the applicant's Fatigue Monitoring Program (AOTC procedure) logs. In the AOTC log, dated February 1990, a total of 11 cycles were recorded for this transient, out of the design limit of 13. In addition, the AOTC log, dated March 2003, stated that a total of 9 cycles were recorded for this transient, out of the design limit of 13. By letter dated May 2, 2011, the staff issued RAI 4.3-2, Request 1, asking the applicant to describe and justify the discrepancy between cycle counts for transients 9A–9D in LRA Table 4.3-1 and the AOTC logs dated February 1990 and March 2003. The staff also requested in, Request 2, that the applicant clarify whether corrective actions were taken, based on the cycle count exceeding the applicant's 75 percent action limit. Finally, the applicant was asked, in Request 3, to identify the design transients and associated cycle limits that were used in the design basis fatigue evaluations of the HPI nozzles and elbowlets.

In its response to Request 1 of RAI 4.3-2, dated June 17, 2011, the applicant stated that, during the review of the AOTC program as part of the Cycle 13 RFO (ended March 27, 2004) restart effort, the AOTC status log was updated. This updated status log included the latest transient 9 (now transient 22 A2) cycle counts and limits (13-cycle limit for train 1 and 40-cycle limit for train 2) based on that review. In addition, the number of cycles for the individual nozzles were separated starting with this updated status log and, as a result of the review and update, the event counts as of May 22, 2003, were as follows:

- 9 cycles for HPI nozzle 1-1
- 8 cycles for HPI nozzle 1-2
- 20 cycles for HPI nozzle 2-1
- 15 cycles for HPI nozzle 2-2

The staff finds the applicant's response to Request 1 acceptable because the applicant clarified the discrepancy by explaining that, prior to May 22, 2003, the event cycles for each HPI nozzle were not documented individually in the AOTC status log. The applicant confirmed the cycle counts, as listed above, for each HPI nozzle from its review of the event logs.

In its response to Request 2 of RAI 4.3-2, dated June 17, 2011, the applicant stated that the February 19, 1990, AOTC status log showed a total of 11 events for transient 9 (now transient 22 A2). The applicant reviewed the AOTC event logs up to that date and confirmed that 11 cycles were logged for nozzle 2-1, 2 cycles were logged for nozzle 2-2, 3 cycles were logged for nozzle 1-1, and 2 cycles were logged for nozzle 1-2. The staff noted that the cycle counts for the train 2 nozzles (2-1 and 2-2) cycle counts were below the 40-cycle limit, and the train 1 nozzles (1-1 and 1-2) were also below the 13-cycle limit. The applicant stated that these transient cycle counts did not exceed the 75 percent action limit.

The staff finds the applicant's response to Request 2 of RAI 4.3-2 acceptable because the applicant's review of the event logs confirmed that the transient cycle counts for the train 1 and train 2 HPI nozzles never exceeded the 75 percent action limit; therefore, no corrective actions were needed.

In its response to Request 3 of RAI 4.3-2, dated June 17, 2011, the applicant stated the objective of transient 9 (now transient 9A) in LRA Table 4.3-1 is to isolate a SG tube leak and results in the actuation of HPI. This is the only upset event in the RCS functional specification that results in HPI actuation. The applicant stated that the design cycle limit for this transient is 40. The applicant also stated that the 40-cycle limit was reduced to 13 (for HPI lines 1-1 and 1-2 with the elbowlets as the limiting location) in 1983 by a Bechtel evaluation of the HPI lines in response to IEB 79-14, and HPI lines 2-1 and 2-2 were qualified for 40 cycles.

The applicant clarified that transient 22 (now transient 22 A1), "HPI System Test," is the only other transient in the RCS functional specification that results in HPI actuation. The applicant stated that this test, as defined in the RCS functional specification, includes HPI flow through all four HPI nozzles for 10 seconds with RCS pressure of 2,200 psig and RCS temperature of 550 °F. However, the staff noted that since the HPI pump shutoff head at the applicant's site is approximately 1,600 psig, which is less than the RCS pressure of 2,000 psig, the flow from this test never comes into contact with the four HPI nozzles, and the HPI pump recirculates back to the BWST. The applicant stated that no inventory is added to the RCS for its plant configuration, and transient 22 (now transient 22 A1) is not applicable but is conservatively included in the design basis fatigue evaluations of the HPI nozzles and HPI elbowlets. The staff finds is reasonable that this test transient is not monitored by the applicant's Fatigue Monitoring Program, based on its plant-specific configuration, because the HPI flow from transient 22 A1 does not come into contact with the four HPI nozzles to create a temperature differential. Therefore, there is no contribution to fatigue usage on the four HPI nozzles from this test transient.

The applicant clarified that, in 1987, it initiated a test transient entitled, "HPI System Pressure Isolation Integrity Test-Back-to-Back Check Valves," which isolates makeup flow to the HPI nozzle used for reactor makeup with RCS pressure and temperature at 2,155 psig and 532 °F, respectively, for approximately 15 minutes and then resumed. The applicant clarified that the purpose of the test is to ensure that the HPI/makeup check valves work properly and isolate the HPI/makeup system from the RCS. This test did not fit the RCS functional specification definitions for transient 9 (now transient 9A) or transient 22 (now transient 22 A1) and was considered a new transient with the number of test cycles defined as 40. The staff noted that these new transients were included as transients 9A–9D (now transient 22 A2 for each of the HPI nozzles) in the applicant's Fatigue Monitoring Program.

The staff finds the applicant's response to RAI 4.3-2, Request 3, acceptable because the applicant monitors all transients that are applicable to the design basis fatigue evaluations of the HPI nozzles and elbowlets with its Fatigue Monitoring Program, with the exception of transient 22 A1 as described above, to ensure that the CUF, including environmental effects, as applicable, does not exceed the 1.0 design limit. Additionally, the applicant conservatively assumed the occurrence of transient 22 A1 in its design basis fatigue evaluation even though the test transient is not applicable to its plant. The staff's concerns described in RAI 4.3-2 are resolved.

LRA Section 4.3.1.2 states that "transients 9C, 9D, and 32 are the only transients affecting Class 1 components where the 60-year projected cycles exceed the design cycles." It was not clear to the staff if there are components, other than HPI nozzles 2-1 and 2-2, which are limited to 40 cycles of transients 9C and 9D, respectively, in the design basis fatigue evaluations and whether these components will be affected if the 60-year projected cycles are exceeded. By letter dated May 2, 2011, the staff issued RAI 4.3-15 requesting that the applicant clarify if there are other components that consider transients 9C or 9D in the design basis fatigue evaluations and to identify the number of design cycles in those evaluations, along with the associated justification for the disposition of the fatigue TLAA for these components.

In its response dated June 17, 2011, the applicant stated that transients 9C (now transient 22 A2, HPI nozzle 2-1) and 9D (now transient 22 A2, HPI nozzle 2-2) are only applicable to HPI nozzles 2-1 and 2-2, respectively. The applicant clarified that by letter dated June 3, 2011, in the response to RAI B.2.16-1, it amended LRA Table 4.3-1 such that the previously listed transients of 9A–9D are renamed as the HPI system pressure isolation integrity tests and are now grouped under transient 22 A2 (HPI nozzles 1-1, 1-2, 2-1, and 2-2). The staff's review of RAI B.2.16-1 is documented in SER Section 3.0.3.2.6.

The staff finds the applicant's response acceptable because the applicant confirmed that the only components affected by transients 9C (now transient 22 A2, HPI nozzle 2-1) and 9D (now transient 22 A2, HPI nozzle 2-2) are HPI nozzles 2-1 and 2-2, respectively. The applicant's Fatigue Monitoring Program counts these transient cycles to ensure the allowable cycle limits used in the design basis fatigue evaluations are not exceeded. The staff's concern described in RAI 4.3-15 is resolved.

LRA Section 4.3.1.2 indicates that the number of cycles accrued as of February 2008 were compiled and linearly extrapolated to the 60 years of operation to determine whether the incurred cycles would remain below the number of design cycles. However, the applicant did not justify the use of a linear extrapolation to determine the number of cycles for 60 years and whether it is conservative, based on its plant-specific operating history. By letter dated May 2, 2011, the staff issued RAI 4.3-9 requesting that the applicant explain the methodology used for the linear extrapolation of design transient cycle counts and to justify that its use is valid and conservative, based on the plant-specific operating history.

In its response dated June 17, 2011, the applicant stated that, based on its plant-specific operating history, transient occurrences were frequent early in plant life and, as issues were resolved, the transient frequency at the plant decreased. The applicant added that its fuel cycle has been increased to 2 years in duration, which results in further decreases in transient cycles. Therefore, the applicant determined that linear extrapolation of cycles, based on the entire operating history of the plant, to project 60-year cycles is conservative. The applicant provided an example of its projection methodology for the plant heat-up transient in its response to the RAI. The staff noted that this example demonstrated that the rate of occurrence based on recent plant-specific operating history is less than the projected rate of occurrence assumed by the applicant. The staff finds the use of this linear extrapolation conservative because the applicant considered the time period when it experienced frequent transient occurrences into its extrapolation and did not only consider its recent improved operating history.

The staff finds the applicant's response acceptable because the applicant considered operating history early in plant life into its projection methodology and demonstrated that its projection methodology is conservative. The applicant's Fatigue Monitoring Program counts transient cycles to ensure allowable cycle limits used in the fatigue analyses are not exceeded. The staff's concern described in RAI 4.3-9 is resolved.

Based on its review, the staff finds the applicant demonstrated that its projection methodology for projecting design transients to the end of the period of extended operation is conservative. The applicant will monitor all transients, that cause cyclic strains which are significant contributors to the fatigue usage factor, with its Fatigue Monitoring Program, such that corrective actions are taken prior to the design limit exceeding 1.0, including environmental effects when applicable.

4.3.1.3 USAR Supplement

LRA Sections A.2.3.1, A.2.3.1.1, A.2.3.1.2, and A.2.3.1.3 provide the USAR supplement summarizing the applicant's Class 1 Code fatigue requirements and 60-year projections of the transients that will be monitored by the Fatigue Monitoring Program. The staff reviewed LRA Section A.2.3.1 consistent with the review procedures in SRP-LR Section 4.3.3.3, which state that the reviewer should verify that the applicant has provided information to be included in the USAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA.

Based on its review of the USAR supplement, the staff finds it meets the acceptance criteria in SRP-LR Section 4.3.2.3. Additionally, the staff determines that the applicant provided an adequate summary description for the 60-year projections of the transients that will be monitored under its Fatigue Monitoring Program, as required by 10 CFR 54.21(d).

4.3.1.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an adequate description and acceptable basis for monitoring design transients and cycles with its Fatigue Monitoring Program, which are also consistent with the CLB and the design basis fatigue evaluations. Also, the staff concludes that the applicant provided an acceptable cycle projection basis for the design transients in LRA Table 4.3-1, "60-Year Projected Cycles," and provided action limits that ensure corrective actions are taken prior to exceeding the design limit during the period of extended operation. The staff also concludes that the USAR supplement contains an appropriate summary description of the cycle projection bases of transients and design cycles, as required by 10 CFR 54.21(d).

4.3.2 Class 1 Fatigue

LRA Section 4.3.2 provides the TLAAs for metal fatigue of Class 1 components within the scope of license renewal, in the following subsections:

- LRA Section 4.3.2.1—Class 1 background
- LRA Section 4.3.2.2—Class 1 vessels, pumps, and major components
- LRA Section 4.3.2.3—Class 1 piping and valves

4.3.2.1 Class 1 Background

4.3.2.1.1 Summary of Technical Information in the Application

The applicant stated that the primary code governing design and construction of Class 1 systems and components, as given in USAR Table 3.2-2, was the ASME Boiler and Pressure Vessel Code Section III, which required fatigue usage calculations based on applicable thermal and mechanical transient load cycles.

4.3.2.2 Class 1 Vessels, Pumps, and Major Components

4.3.2.2.1 Summary of Technical Information in the Application

LRA Section 4.3.2.2 describes the metal fatigue TLAAs for ASME Code, Section III Class 1 vessels, pumps, and major components that include the RV, the control rod drives (CRDs), the RCPs, the pressurizer, and the SGs. The applicant stated that CUFs of Class 1 components are based on the service and test loading definitions contained in the component design specifications. The design transients used to generate the CUF are discussed in LRA Section 4.3.1.

Reactor Vessel. LRA Section 4.3.2.2.1 describes the fatigue analyses conducted for the RV, which was designed to Class A requirements in accordance with the 1968 edition of the ASME Code, Section III, inclusive of the 1968 summer addenda. The entire vessel assembly was analyzed for the primary and secondary stresses under both steady-state and transient operations, and the resulting fatigue analysis was performed by the original equipment manufacturer (OEM). The design CUFs for RV assembly locations were less than 1.0. The applicant's Fatigue Monitoring Program tracks the number of occurrences of design transients to ensure that action is taken before the analyzed numbers of transients are reached. The applicant dispositioned the fatigue TLAA of the RPV in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the RV will be managed for the period of extended operation by the Fatigue Monitoring Program.

Reactor Vessel Internals. LRA Section 4.3.2.2.2 describes the RVI components that include the plenum assembly and the core support assembly consisting of the core support shield, core barrel, lower grid, flow distributor, incore instrument guide tubes, thermal shield, and surveillance specimen holder tubes. The applicant's metal fatigue TLAAs of RVIs are summarized below:

Low-Cycle Fatigue. LRA Section 4.3.2.2.2.1 states that the design of the RVIs meets the stress requirements of ASME Code, Section III, but the design code did not require a fatigue analysis to be performed. The applicant stated that it performed fatigue analyses for the Alloy X-750 HTH bolts, which were designed to ASME Code, Section III, to replace the majority of the vessel internals Alloy A-286 bolts. The applicant also stated that the CUFs for the Alloy X-750 HTH replacement bolts were based on the system design transients in LRA Table 4.3-1 and were found to be less than 1.0. The upper thermal shield bolts, flow distributor bolts, and guide block bolts have not been replaced. The applicant dispositioned the low-cycle fatigue TLAA of the

RVIs in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of aging due to fatigue will be managed for the period of extended operation by the Fatigue Monitoring Program.

Reactor Vessel Internals and Incore Instrument Nozzles Flow-Induced Vibration. LRA Section 4.3.2.2.2.2 discusses metal fatigue of RVIs and incore instrument nozzles subjected to flow-induced vibrations (FIV). The applicant stated that the FIV fatigue TLAA is based on the endurance limit approach that established the maximum allowable stress limit for an infinite life. The applicant stated that to implement this approach to high-cycle fatigue due to FIV, it extended the ASME Code fatigue curve to 1.0E+12 cycles as the upper-bound for a 40-year design life in its original analysis. This resulted in an allowable stress limit of 20,400 pounds per square inch (psi), and the applicant further reduced this to a conservative design limit of 18,000 psi. The applicant stated that for 60-years of operation, the extrapolated fatigue curve at 1.5E+12 cycles is approximately 20,200 psi, still above the 18,000 psi that was used as the endurance limit. The applicant dispositioned the fatigue TLAA for the FIV of RVIs in accordance with 10 CFR 54.21(c)(1)(i), that the existing analysis remains valid for the period of extended operation.

Surveillance Capsule Holder Tubes Flow-Induced Vibration. LRA Section 4.3.2.2.2.3, as amended by letter dated June 17, 2011, discusses the metal fatigue analysis of the surveillance capsule holder tubes subject to FIV. The applicant stated that the original analysis for a 40-year design life resulted in a CUF of 0.00042 for the holder tubes and an additional 20 years of operations would result in a CUF of 0.00063, which remains below the Code design limit of 1.0. The applicant stated that it dispositioned the FIV fatigue TLAA for the surveillance capsule holder tubes in accordance with 10 CFR 54.21(c)(1)(ii), that the analysis has been projected to the end of the period of extended operation.

Control Rod Drive Housings Fatigue. LRA Section 4.3.2.2.3 describes the fatigue analysis for the CRD housings that act as the pressure retaining enclosures for the drive mechanisms. These housings were designed to the 1968 edition of ASME Code, Section III, inclusive of the 1970 summer addenda and the CUFs for various CRD locations, which are less than 1.0, were based on the system design transients given in LRA Table 4.3-1. The applicant stated that its Fatigue Monitoring Program tracks the incurred cycles of these design transients to ensure action is taken before reaching their design number of cycles. The applicant dispositioned the TLAA of CRD housings in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of aging due to fatigue will be managed for the period of extended operation by the Fatigue Monitoring Program.

Reactor Coolant Pump Casings Fatigue. LRA Section 4.3.2.2.4 describes the fatigue analysis for the RCP casings, which are welded into the piping system. The casings were designed to the 1968 edition of ASME Code, Section III, inclusive of the 1968 winter addenda, and the CUFs for the RCP casings, which are less than 1.0, were based on the system design transients given in LRA Table 4.3-1. The applicant stated that its Fatigue Monitoring Program tracks the incurred cycles of these design transients to ensure action is taken before reaching their design number of cycles. The applicant dispositioned the TLAA of RCP casings in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of aging due to fatigue will be managed for the period of extended operation by the Fatigue Monitoring Program.

Pressurizer Fatigue. LRA Section 4.3.2.2.5 describes the fatigue analysis for the pressurizer, which consists of a vertical cylindrical vessel connected by the surge line to the reactor outlet piping, with nozzles and heater bundle (closures) attached to the vessel. The pressurizer was designed to the 1968 edition of ASME Code, Section III, inclusive of 1968 summer addenda, and the CUFs for pressurizer locations, which are less than 1.0, were based on the system design transients given in LRA Table 4.3-1. The applicant stated that its Fatigue Monitoring Program tracks the incurred cycles of these design transients to ensure action is taken before their design number of cycles is reached. The applicant dispositioned the TLAA of pressurizer in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of aging due to fatigue will be managed for the period of extended operation by the Fatigue Monitoring Program.

<u>Once-Through Steam Generators</u>. LRA Section 4.3.2.2.6 states that the once-through steam generators (OTSGs) components exposed to RCS pressure are the hemispherical heads, the tubesheet, and the straight inconel tubes between the tubesheets. The applicant's metal fatigue TLAAs related to the OSTGs is separated into four parts, as summarized below.

<u>*OTSGs Fatigue*</u>. LRA Section 4.3.2.2.6.1 states that the primary (tube) and secondary (shell) sides of the OTSGs were designed to the 1968 edition of ASME Code, Section III, inclusive of 1968 summer addenda, and were analyzed for fatigue by the OEM. The CUFs for OTSGs locations, which are less than 1.0, were based on the system design transients given in LRA Table 4.3-1. The applicant stated that the SG remote weld plugs have a limited design life of 33 HU/CD cycles to maintain a fatigue usage of less than 1.0. The applicant's Fatigue Monitoring Program tracks the incurred cycles of these design transients to ensure action is taken before reaching their design number of cycles. The applicant dispositioned the TLAA for the OTSGs in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of aging due to fatigue will be managed for the period of extended operation by the Fatigue Monitoring Program.

<u>*OTSGs Tube Sleeves Fatigue*</u>. LRA Section 4.3.2.2.6.2 describes the fatigue analysis for the tube sleeves that were used to repair leaking tubes of the OTSGs. In accordance with USAR Section 5.5.2.3, the applicant stated that the SG tubes may be plugged or repaired by mechanical (rolled) sleeving; however, Section III of the ASME Code does not provide design rules for mechanically roll-expanded attachments, and theoretical stress analyses are inadequate. The applicant stated that, in accordance with provisions of Appendix II, Section 1500, of ASME Code, Section III, fatigue tests were performed to demonstrate the structural adequacy of the sleeves to withstand cyclic loadings based on the design transients. The pressure cycling tests used 360 startup cycles to bound all B&W 177 fuel assembly plants. The applicant stated that, per USAR Table 5.1-8, its design basis is 240 startups and it projected only 128 startups for 60 years of operation, as described in LRA Table 4.3-1. The applicant dispositioned the TLAA associated with fatigue testing of the OTSG tube sleeves in accordance with 10 CFR 54.21(c)(1)(i) disposition, that the analysis will remain valid for the period of extended operation.

<u>*OTSGs Auxiliary Feedwater Modification*</u>. LRA Section 4.3.2.2.6.3 describes the fatigue analysis for the repair to the OTSGs auxiliary feedwater (AFW) system. The modification was installed (in 1982) with an external header on each SG. The applicant stated that the AFW thermal sleeve stresses were also analyzed by B&W, and the analysis, performed in accordance with the requirements of the ASME Code for Class 1 components, provided a basis for demonstrating that the AFW thermal sleeve is capable of withstanding 40,000 cycles of AFW injection transients. The riser flange attachment to the SG shell was also analyzed per ASME Code requirements and was acceptable for a design life of 875 cycles of HU/CD, bolt-up and unbolt, and AFW initiations. Transients 30A and 30B in LRA Table 4.3-1, which have 60-year projections of 387 and 442 cycles, respectively, are each less than the 875 design cycles for the riser flange attachment. The applicant stated that design transients are tracked for the number of occurrences under its Fatigue Monitoring Program to ensure that action is taken before the design cycles are reached. The applicant dispositioned the TLAAs of AFW

repair in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the AFW modification will be managed for the period of extended operation by the Fatigue Monitoring Program.

OTSGs Tubes and Tube Stabilizers Flow-Induced Vibration. LRA Section 4.3.2.2.6.4 describes the fatigue analysis performed for FIV of the OTSGs tubes and the tube stabilizers. The applicant stated that its latest analysis report showed the highest CUF for any existing tube configuration was 0.443 for an un-repaired tube next to the open lane, and the 60-year projected CUF value of 0.665 is acceptable. The applicant stated that the 60-year projected CUFs for the 3/8-in. tube-stabilizers, calculated using both high-cycle (FIV) and low-cycle (transients) fatigue, remains below the design limit of 1.0. The applicant dispositioned the fatigue TLAA associated with FIV of SG tubes and tube stabilizers in accordance with 10 CFR 54.21(c)(1)(ii), that the TLAAs have been projected through the period of extended operation.

4.3.2.2.2 Staff Evaluation

The staff reviewed LRA Section 4.3.2.2, which consists of metal fatigue TLAAs for ASME Code Section III Class 1 vessels, pumps, and major components, to confirm, pursuant to the following and dependent on the applicant's evaluation:

- 10 CFR 54.21(c)(1)(i), that the analyses remain valid during the period of extended operation
- 10 CFR 54.21(c)(1)(ii), that the analyses have been projected to the end of the period of extended operation
- 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

Reactor Vessel. The staff reviewed LRA Section 4.3.2.2.1 on fatigue of the RV to verify, 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 4.3.3.1.1.3, which state that the reviewer should verify 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 verify that the applicant identified the appropriate program, as described and evaluated in the GALL Report. Furthermore, the reviewer should also 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 noted that the LRA did not provide the CUF values for most of the ASME Code Section III Class 1 components described in LRA Section 4.3.2. Without these values, the staff could not ascertain whether the CUF values for these components exceed the allowable limit or evaluate the applicant's dispositions of these metal fatigue TLAAs, in accordance with 10 CFR 54.21(c). By letter dated May 2, 2011, the staff issued RAI 4.3-12 asking the applicant to provide the 40-year design-basis CUF values for all components or critical locations, or both, that are dispositioned in LRA Section 4.3.2.

In its response dated June 17, 2011, the applicant stated that the design (40-year) CUFs for all its Class 1 components are provided in Tables 3-1 through 3-9 of AREVA Document 51-9157140-001, "DB-1 Design CUFs and NUREG/CR-6260 Screening for License Renewal," dated June 10, 2011. The applicant provided a copy of the report as an enclosure to its June 17, 2011, letter, and the CUF values provided by the applicant in response to RAI 4.3-12 allow the staff to determine if the applicant's TLAA are appropriately dispositioned in accordance with 10 CFR 54.21(c)(1). The staff's review of the applicant's metal fatigue TLAAs and its dispositions of specific ASME Code Class 1 components are documented in SER Section 4.3.2. The staff's concern described in RAI 4.3-12 is resolved.

LRA Section 4.3.2.2.1 states that the bottom head of the RV assembly is penetrated by the instrumentation nozzles, and the design CUFs for all RV locations were calculated to be less than 1.0. During its audit, staff noted the applicant's basis documents for metal fatigue indicated the CUF values for the instrument nozzle weld locations vary from 0.0–0.323. Furthermore, LRA Section 4.3.2.2.2.3 states that the incore instrumentation nozzles were analyzed for fatigue due to FIV with the resulting CUF of 0.59 for a 40-year life and was projected to have a CUF of 0.885 for a 60-year life. LRA Section 4.3.4.2 states that the maximum design CUF for nickel-based alloy incore instrument nozzle is 0.77. It was not clear to the staff if the generic reference of "Instrument Nozzles" in the applicant's basis documents and LRA sections refer to the same location. By letter dated May 2, 2011, the staff issued RAI 4.3-4 requesting the applicant to clarify the location(s) that are being referenced by the "Instrument Nozzle" CUFs in LRA Sections 4.3.2.2.1, 4.3.2.2.2.3, and 4.3.4.2, as well as the applicant's basis documents for metal fatigue TLAAs. The staff also asked the applicant to clarify which of these locations for the instrumentation nozzle of the RV assembly support the aforementioned statement in LRA Section 4.3.2.2.1 and is considered the limiting location.

In its response dated June 17, 2011, the applicant provided a summary table that describes the documents in which instrument nozzles were discussed. The applicant stated that the discussion of instrument nozzles in LRA Section 4.3.2.2.1 is consistent with its bases documents, which only state that all vessel CUFs are less than 1.0. The CUF value of 0.59 in LRA Section 4.3.2.2.2.3 was reported in error, and the LRA was amended to provide clarification. The applicant's review of its source document revealed that the value of 0.59 was a typical value of B&W-designed plants and that the evaluation of FIV for the instrument nozzles is documented in LRA Section 4.3.2.2.2.2. The staff's evaluation of FIV for the instrument nozzles is documented elsewhere in this SER section. The applicant stated that the CUF values of 0.000 to 0.323 were reported in the RV stress report summary for the two different styles of nozzle bodies and were not discussed in the LRA since they are not the limiting locations. The staff noted that the weld between the incore instrument nozzle and RV lower head is the limiting location, which is discussed in LRA Section 4.3.4.2, and has a 40-year design CUF value of 0.770. The applicant also described how it obtained an environmentally-assisted fatigue (EAF) CUF value of 0.857 for the nozzle to vessel weld. The staff's review of the applicant's EAF evaluations is described in SER Section 4.3.4.2.

The staff finds the applicant's response acceptable because the applicant clarified the specific locations for the instrument nozzles and the associated CUF values that were referenced in the applicant's basis documents and LRA. In addition, the applicant identified the limiting location as the weld between the incore instrument nozzle and RV lower head, which has been evaluated for the effects of reactor coolant environment as a NUREG/CR-6260 location, consistent with the recommendations in SRP-LR Section 4.3.2.1.3. The staff's concern described in RAI 4.3-4 is resolved.

The staff reviewed the CUF values provided by the applicant, in Table 3-1 of AREVA Document 51-9157140-001, in response to RAI 4.3-12 (letter dated June 17, 2011) and confirmed that the design CUF values for the Class 1 components associated with the RV are less than the design limit of 1.0. The staff noted that the applicant credited the cycle-counting activities of its Fatigue Monitoring Program as the basis for managing cumulative fatigue damage that may occur in the RV during the period of extended operation and initiate corrective actions to ensure the design cycles and design limit of 1.0 will not be exceeded. Consistent with the recommendation of GALL Report AMP X.M1, the staff noted that the cycle-counting activities in the applicant's Fatigue Monitoring Program are an acceptable approach to manage CUF values for RCPB components, consistent with 10 CFR 54.21(c)(1)(iii). The staff's evaluation of the applicant's Fatigue Monitoring Program is documented in SER Section 3.0.3.2.6.

The staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging related to fatigue analyses of the RV 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 for the following reasons:

- The applicant's Fatigue Monitoring Program monitors and tracks the number of design basis transients that will occur through the period of extended operation.
- The applicant's Fatigue Monitoring Program includes action limits and corrective actions that will ensure that the CUF design limit of 1.0 will not be exceeded during the period of extended operation.
- The use of the applicant's Fatigue Monitoring Program is consistent with the recommendations of the GALL Report AMP X.M1 for managing cumulative fatigue damage.

Reactor Vessel Internals

Low-Cycle Fatigue. The staff reviewed LRA Section 4.3.2.2.2.1 on low-cycle fatigue of the RVI to verify, 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 4.3.3.1.1.3, which state that the reviewer should verify 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 verify that the applicant identified the appropriate program, as described and evaluated in the GALL Report. Furthermore, the reviewer should also 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 noted that, as discussed in LRA Section 4.3.2.2.2.1, the applicant has not replaced the upper thermal shield bolts, flow distributor bolts, or guide block bolts, and no fatigue analysis was performed for these bolts because it was not required during the original design. However, the staff noted that LRA Table 3.1.2-2, Row Nos. 42 and 110, for upper thermal shield bolts and flow distribution bolts, respectively, credit a TLAA to manage cumulative fatigue damage. It was not clear to the staff what TLAA was being referenced, since LRA Section 4.3.2.2.2.1 states that fatigue analyses were not performed for the RVIs. By letter dated May 2, 2011, the staff issued RAI 4.3-3 requesting that the applicant identify the fatigue TLAA that is being credited to manage cumulative fatigue damage of the components identified by the AMR line items in LRA Table 3.1.2-2, Row Nos. 42 and 110.

In its response dated June 17, 2011, the applicant stated that it has not replaced the upper thermal shield bolts, flow distributor bolts, or guide block bolt; therefore, a correction is required to Row Nos. 42 and 110 of LRA Table 3.1.2-2. The staff noted that the applicant amended LRA Table 3.1.2-2 to remove the AMR line items associated with stainless steel upper thermal shield bolts and flow distributor bolts exposed to borated reactor coolant that are being managed for cracking due to fatigue by a TLAA. Although these components do not have a fatigue TLAA associated with them, the staff noted that they will be managed by the applicant's PWR RVI Program for cracking during the period of extended operation. The staff finds the removal of these AMR line items acceptable because a fatigue analysis was not performed for these components; therefore, they do not have a TLAA associated with them. The staff's concern described in RAI 4.3-3 is resolved.

The staff reviewed the CUF values provided by the applicant, in Table 3-1 of AREVA Document 51-9157140-001, in response to RAI 4.3-12 (letter dated June 17, 2011) and confirmed that the design CUF values for the RVIs are less than the design limit of 1.0. These components are the upper and lower core barrel bolts, the lower thermal shield bolts, and the bolts associated with the surveillance specimen holder tube. The staff noted that the applicant credited the cycle-counting activities of its Fatigue Monitoring Program as the basis for managing cumulative fatigue damage that may occur in the RV during the period of extended operation and will initiate corrective actions to ensure the design cycles and design limit of 1.0 will not be exceeded. Consistent with the recommendation of GALL Report AMP X.M1, the staff noted that the cycle-counting activities in the applicant's Fatigue Monitoring Program is an acceptable approach to manage CUF values for RCPB components, consistent with 10 CFR 54.21(c)(1)(iii). The staff's evaluation of the applicant's Fatigue Monitoring Program is documented in SER Section 3.0.3.2.6.

The staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging related to low-cycle fatigue analyses of the RVIs 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 for the following reasons:

- The applicant's Fatigue Monitoring Program monitors and tracks the number of design basis transients that will occur through the period of extended operation.
- The applicant's Fatigue Monitoring Program includes action limits and corrective actions that will ensure that the CUF design limit of 1.0 will not be exceeded during the period of extended operation.
- The use of the applicant's Fatigue Monitoring Program is consistent with the recommendations of the GALL Report AMP X.M1 for managing cumulative fatigue damage.

<u>*Reactor Vessel Internals and Incore Instrumentation Nozzles Flow-Induced Vibration*</u>. The staff reviewed LRA Subsection 4.3.2.2.2.2 on FIV of the RVIs and incore instrument nozzles to verify, pursuant to 10 CFR 54.21(c)(1)(i), that the analysis remains valid during 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.

The staff noted that, as discussed in LRA Section 4.3.2.2.2.2, the fatigue of RVIs and incore instrument nozzles subject to FIV is based on the endurance limit approach, which establishes the allowable stress limit for infinite fatigue life. While this approach does not produce a CUF value or use the design transients, the staff noted that the effective CUF is implicitly demonstrated to be zero, based on maintaining the stress amplitude below the endurance limit. Mandatory Appendix I of the ASME Code Section III provides design fatigue curves. However, LRA Section 4.3.2.2.2.2 states that the ASME Code fatigue curve was extended because the 60-year projection used in the vessel internals fatigue TLAA exceeded the Code design curves. It was not clear to the staff which Appendix I design curve was used by the applicant and the method of extrapolation that was used to establish the endurance limit for the 40-year analysis and the 60-year projection. By letter dated May 2, 2011, the staff issued RAI 4.3-5 requesting the applicant to clarify and justify the ASME Code Section III (Mandatory Appendix I) design curves used in the extrapolation described in LRA Section 4.3.2.2.2.2 for all the vessel internal materials subject to the FIV.

In its response dated June 17, 2011, the applicant stated that the specific curve extrapolated in the original FIV analysis was Figure I-9.2 of the 1971 edition of the ASME Code Section III. The applicant explained that for the 40-year design, the allowable stress of 20,400 psi for 1.0E+12 cycles was identified on the fatigue curve for austenitic stainless steel. The applicant stated that it conservatively assumed the endurance limit as 18,000 psi in the FIV analysis and that the maximum calculated peak stress intensity provided in the FIV analysis is 8,260 psi for the upper thermal shield support blocks.

The applicant stated that the ASME Code fatigue curve had previously been extended from 1.0 x 10E6 cycles to 1.0 x 10E12 cycles based on the curve fit for the data found in the ASME Code transactions; for license renewal, this extrapolated curve was extended from 1.0E+12 cycles (the upper bound on the number of cycles for a 40-year design life) to 1.5E+12 cycles (the upper bound on the number of cycles for a 60-year design life). The stress value corresponding to 1.5E+12 cycles on the extrapolated fatigue curve is approximately 20,200 psi. The staff confirmed that Figure I-9.2 in the Mandatory Appendix I of the 1971 ASME Code Section III is for austenitic stainless steel. The staff noted that the endurance limit of 18,000 psi is still below, and remains valid, compared to the 20,200 psi identified on the 60-year extrapolated design curve. Furthermore, the staff noted that the maximum calculated peak stress intensity of 8,260 psi remains below the endurance limit.

The staff finds the applicant’s response acceptable because the applicant’s clarification justified that the endurance limit assumed in the original FIV analysis remains valid, as demonstrated above, which implicitly demonstrates that the CUF remains zero. The staff’s concern described in RAI 4.3-5 is resolved.

Based on its review, the staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the FIV analysis of the RVIs and incore instrument nozzles remains valid during the period of extended operation. Additionally, it meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.1 because the endurance limit assumed in the original analysis would not be exceeded and the implicit CUF value of zero remains valid during the period of extended operation.

Surveillance Capsule Holder Tubes Flow-Induced Vibration. The staff reviewed LRA Subsection 4.3.2.2.2.3 on FIV of the surveillance capsule holder tubes to verify, pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis has been projected the period 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 4.3.3.1.1.2, which 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 noted that the 40-year design CUF due to flow induced vibrations of the surveillance capsule holder tubes is 0.00042, and the applicant calculated the projected 60-year CUF by multiplying 0.00042 by a factor 1.5. The staff noted that the resulting CUF of 0.00063 remains far below the design limit of 1.0. The staff finds the use of a 1.5 factor projection basis reasonable for design basis CUF values that are based on a 40-year design life for cases in which there are no planned changes to plant operations or configuration that would call into question this projection. The resultant estimated 60-year CUF value(s) provide a gauge of how much margin is available before the design limit of 1.0 is reached. The staff noted that the 40-year design CUF due to low-cycle fatigue is 0.02 for this component, which was reviewed in SER Section 4.3.2.2.2.

The staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the surveillance capsule holder tubes FIV analysis have been projected to the end of the period of extended operation. Additionally, it meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.2 because the applicant demonstrated that the 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.

Control Rod Drive Housings Fatigue. The staff reviewed LRA Section 4.3.2.2.3 on fatigue of the CRD housings to verify, 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 4.3.3.1.1.3, which state that the reviewer should verify 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 verify that the applicant identified the appropriate program, as described and evaluated in the GALL Report. Furthermore, the reviewer should also 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 USAR Table 5.2-1 confirmed that the design code for the CRD housings is the 1968 edition of ASME Code, Section III, inclusive of the 1970 summer addenda. The staff reviewed the CUF values provided by the applicant, in Table 3-2 of AREVA Document 51-9157140-001, in response to RAI 4.3-12 (letter dated June 17, 2011) and confirmed that the design CUF values for the Class 1 components associated with the CRD housings are less than the design limit of 1.0. The staff noted that the applicant credited the cycle-counting activities of its Fatigue Monitoring Program as the basis for managing cumulative fatigue damage that may occur in the RV during the period of extended operation, and it will initiate corrective actions to ensure the design cycles and design limit of 1.0 will not be exceeded. Consistent with the recommendation of GALL Report AMP X.M1, the staff noted that the cycle-counting activities in the applicant's Fatigue Monitoring Program are an acceptable approach to manage CUF values for RCPB components and are consistent with

10 CFR 54.21(c)(1)(iii). The staff's evaluation of the applicant's Fatigue Monitoring Program is documented in SER Section 3.0.3.2.6.

The staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging related to fatigue analyses of the CRD housings 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 for the following reason:

- The applicant's Fatigue Monitoring Program monitors and tracks the number of design basis transients that will occur through the period of extended operation.
- The applicant's Fatigue Monitoring Program includes action limits and corrective actions that will ensure that the CUF design limit of 1.0 will not be exceeded during the period of extended operation.
- The use of the applicant's Fatigue Monitoring Program is consistent with the recommendations of the GALL Report AMP X.M1 for managing cumulative fatigue damage.

<u>Reactor Coolant Pump Casings Fatigue</u>. The staff reviewed LRA Section 4.3.2.2.4 on fatigue of the RCP casings to verify, 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 4.3.3.1.1.3, which state that the reviewer should verify 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 verify that the applicant identified the appropriate program, as described and evaluated in the GALL Report. Furthermore, the reviewer should also 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.

During its audit, the staff noted the applicant's basis documents for metal fatigue TLAAs indicated that the cooling hole ligament of the RCP cover has a CUF value of 0.56, which was calculated with an exception to the ASME Code rules. It was not clear to the staff what the exception was and whether this exception affects the applicant's disposition for the TLAA. By letter dated May 2, 2011, the staff issued RAI 4.3-6 requesting that the applicant clarify the exception used for the fatigue analysis of the cooling hole ligament of the RCP cover and justify that the exception does not affect the TLAA disposition of the RCP casing fatigue evaluation.

In its response dated June 17, 2011, the applicant stated that, during the course of evaluating thermal cracking of RCP covers in 1980s, the reanalysis of the ligament region at the cooling holes revealed stress values that were higher than those calculated in the original design report. The applicant also stated that the revised CUF for the cooling hole ligament exceeded 1.0 and, to demonstrate that the fatigue life of the cover cooling hole ligament is acceptable for the current term of operation, B&W used the vendor stress analysis and developed an alternate simplified elastic-plastic methodology for fatigue calculation. Specifically, in this methodology, a stress and fatigue analysis of the pump cover cooling hole ligament was performed in accordance with paragraph N-415 of the 1968 edition of ASME Code, Section III, with the exception that the limit on the range of primary-plus-secondary stress intensity may be waived if certain conditions were satisfied. The applicant stated that it satisfied the conditions of the exception, and the total usage factor of 0.56 was calculated in accordance with ASME Code, Section III, paragraph N-415. The applicant explained that the CUF of 0.56 included contributions of 0.41 for HU/CD transients (205 cycles), 0.09 for the combined hydrostatic tests and heatup transient (35 cycles), and approximately 0.06 for the remaining events. The staff noted that the applicant credited the cycle-counting activities of its Fatigue Monitoring Program as the basis for managing cumulative fatigue damage that may occur in RCP casings during the period of extended operation.

The staff finds the applicant's response acceptable because all fatigue-significant transients included in the analysis are captured in LRA Table 4.3-1, and these transients are incorporated into the applicant's Fatigue Monitoring Program, which ensures that pump cover cooling hole ligament fatigue usage does not exceed the design limit during the period of extended operation. The applicant described and confirmed that it satisfied the conditions for the "exception" in paragraph N-415 of the 1968 edition of ASME Code, Section III. The staff's concern described in RAI 4.3-6 is resolved.

LRA Table 3.1.1, item 3.1.1-55, states that the aging of these pump casings will be managed by the applicant's ISI Program and invokes the use of ASME Code Case N-481 as an alternative for the AMP. The staff's review of the Code Case N-481 and the required justification in support of the alternative inspection requirements for the pump casings identified that the applicant had submitted its evaluation (ADAMS Accession No. ML011200090) that included a time-dependent fatigue flaw growth analysis. However, the LRA did not discuss the TLAA disposition of ASME Code Case N-481. By letter dated May 2, 2011, the staff issued RAI 4.1-2 asking the applicant to justify the absence of TLAA identification in the LRA for the RCP casing regarding the application of Code Case N-481. The staff's evaluation of RAI 4.1-2 is documented in SER Section 4.7.6.

The staff reviewed the CUF values provided by the applicant, in Table 3-3 of AREVA Document 51-9157140-001, in response to RAI 4.3-12 (letter dated June 17, 2011) and confirmed that the design CUF values for the Class 1 components associated with the RCP are less than the design limit of 1.0. The staff noted that the applicant credited the cycle-counting activities of its Fatigue Monitoring Program as the basis for managing cumulative fatigue damage that may occur in the RV during the period of extended operation, and it will initiate corrective actions to ensure the design cycles and design limit of 1.0 will not be exceeded. Consistent with the recommendation of GALL Report AMP X.M1, the staff noted that the cycle-counting activities in the applicant's Fatigue Monitoring Program are an acceptable approach to manage CUF values for RCPB components and are consistent with 10 CFR 54.21(c)(1)(iii). The staff's evaluation of the applicant's Fatigue Monitoring Program is documented in SER Section 3.0.3.2.6.

The staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging related to fatigue analyses of the RCP casings 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 for the following reasons:

- The applicant's Fatigue Monitoring Program monitors and tracks the number of design basis transients that will occur through the period of extended operation.
- The applicant's Fatigue Monitoring Program includes action limits and corrective actions that will ensure that the CUF design limit of 1.0 will not be exceeded during the period of extended operation.

- The use of the applicant's Fatigue Monitoring Program is consistent with the recommendations of the GALL Report AMP X.M1 for managing cumulative fatigue damage.

Pressurizer Fatigue. The staff reviewed LRA Section 4.3.2.2.5 on fatigue of the pressurizer to verify, 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 4.3.3.1.1.3, which state that the reviewer should verify 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 verify that the applicant identified the appropriate program, as described and evaluated in the GALL Report. Furthermore, the reviewer should also 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 USAR Table 5.2-1 confirmed that the design code for the pressurizer was the 1968 edition of ASME Code, Section III, inclusive of the 1968 summer addenda. The staff reviewed the CUF values provided by the applicant, in Table 3-4 of AREVA Document 51-9157140-001, in response to RAI 4.3-12 (letter dated June 17, 2011), and confirmed that the design CUF values for the Class 1 components associated with the pressurizer are less than the design limit of 1.0. The staff noted that the applicant credited the cycle-counting activities of its Fatigue Monitoring Program as the basis for managing cumulative fatigue damage that may occur in the RV during the period of extended operation, and it will initiate corrective actions to ensure the design cycles and design limit of 1.0 will not be exceeded. Consistent with the recommendation of GALL Report AMP X.M1, the staff noted that the cycle-counting activities in the applicant's Fatigue Monitoring Program are an acceptable approach to manage CUF values for RCPB and are consistent with 10 CFR 54.21(c)(1)(iii). The staff's evaluation of the applicant's Fatigue Monitoring Program is documented in SER Section 3.0.3.2.6.

The staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging related to fatigue analyses of the pressurizer 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 for the following reasons:

- The applicant's Fatigue Monitoring Program monitors and tracks the number of design basis transients that will occur through the period of extended operation.
- The applicant's Fatigue Monitoring Program includes action limits and corrective actions that will ensure that the CUF design limit of 1.0 will not be exceeded during the period of extended operation.
- The use of the applicant's Fatigue Monitoring Program is consistent with the recommendations of the GALL Report AMP X.M1 for managing cumulative fatigue damage.

Once-Through Steam Generators

Once-Through Steam Generators Fatigue. The staff reviewed LRA Section 4.3.2.2.6.1 on fatigue of the OTSGs to verify, 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 4.3.3.1.1.3, which state that the reviewer should verify 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 verify that the applicant identified the appropriate program, as described and evaluated in the GALL Report. Furthermore, the reviewer should also 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 noted that LRA Section 4.3 states that the new design cycle limit for the remotely welded plugs was reduced to 33 cycles (transient 32 in LRA Table 4.3-1). During its audit, the staff noted in the applicant's basis documents for metal fatigue TLAAs that manually welded plugs may also be limited to 33 cycles. The staff noted that the applicant's documentation also describes other OTSG tube plug types. Furthermore, the staff noted that by letter dated November 3, 2003, the applicant responded to the staff's RAI regarding the 2002 SG tube inspection (ADAMS Accession No. ML033100370) and stated that there are 36 construction-era welded plugs and 2 of them were repaired in 2003 with remote welded plugs. It is not clear to the staff if other types of weld plugs, such as the 36 construction-era welded plugs and the 2 repaired welded plugs that were not discussed in the LRA Section 4.3.2.2.6.1, have applicable design fatigue evaluations. It is also not clear to the staff whether these other types of plugs are bounded by the remotely welded plugs, which have a limit of 33 cycles for transient 32. By letter dated May 2, 2011, the staff issued RAI 4.3-7 requesting that the applicant clarify whether there are other types of plugs in addition to remote welded plugs and whether these additional types of plugs have applicable fatigue design analyses. In addition, the applicant was asked to provide the applicable design transients and associated cycle limits for these plugs.

In its response dated June 17, 2011, the applicant stated that the OTSG remote weld plugs have a limited design life of 33 HU/CD cycles to maintain a fatigue usage of less than 1.0. The applicant also stated that the OTSG tube repairs include explosive tube plugs, welded U-cup plugs, rolled tube plugs, sleeve plugs, mechanical plugs, and welded tube plugs. The applicant clarified that only the remotely or manually welded tube plugs, which includes construction era and repaired welded plug have fatigue analyses. The applicant stated that the remote welded plugs are the most limiting and, therefore, bound the other welded tube plugs.

The staff finds the applicant's response acceptable because the applicant confirmed that only the construction era and repair welded tube plugs have fatigue analyses associated with their design. Additionally, the fatigue analysis for the repair welded tube plugs bound the construction era welded tube plugs, which are managed by the applicant's Fatigue Monitoring Program to ensure that the fatigue usage limit for the repaired welded tube plugs are not exceeded. The staff's concern described in RAI 4.3-7 is resolved.

The staff noted that, in LRA Section 4.3.2.2.6.1, the applicant stated that the SGs were analyzed for fatigue by the OEM and that the CUFs for limiting locations were calculated to be less than 1.0 based on the design transients. The staff reviewed the CUF values provided by the applicant, in Tables 3-5 and 3-6 of AREVA Document 51-9157140-001, in response to RAI 4.3-12 (letter dated June 17, 2011), and confirmed that the design CUF values for the Class 1 components associated with the OTSGs are less than the design limit of 1.0. The staff noted that the applicant credited the cycle-counting activities of its Fatigue Monitoring Program as the basis for managing cumulative fatigue damage that may occur in the RV during the period of extended operation, and it will initiate corrective actions to ensure the design cycles and design limit of 1.0 will not be exceeded. Consistent with the recommendation of GALL Report AMP X.M1, the staff noted that the cycle-counting activities in the applicant's Fatigue Monitoring Program are an acceptable approach to manage CUF values for RCPB components and are consistent with 10 CFR 54.21(c)(1)(iii). The staff's evaluation of the applicant's Fatigue Monitoring Program is documented in SER Section 3.0.3.2.6.

The staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging related to fatigue analyses of the OTSGs 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 for the following reasons:

- The applicant's Fatigue Monitoring Program monitors and tracks the number of design basis transients that will occur through the period of extended operation.
- The applicant's Fatigue Monitoring Program includes action limits and corrective actions that will ensure that the CUF design limit of 1.0 will not be exceeded during the period of extended operation.
- The use of the applicant's Fatigue Monitoring Program is consistent with the recommendations of the GALL Report AMP X.M1 for managing cumulative fatigue damage.

Once-Through Steam Generators Tube Sleeves Fatigue. The staff reviewed LRA Subsection 4.3.2.2.6.2 on fatigue of OTSG tube sleeves (rolls) to verify, pursuant to 10 CFR 54.21(c)(1)(i), that the analysis remains valid during 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.

The staff noted that LRA Subsection 4.3.2.2.6.2 states that the sleeves are used as a repair option for which ASME Code, Section III, does not provide design rules but allows the demonstration of structural adequacy to withstand cyclic loadings via fatigue testing per ASME Code, Section III, Appendix II-1500. The staff reviewed Appendix II, paragraph II-1500, of the ASME Code, Section III, and confirmed that the Code allows the use of fatigue testing as a means to demonstrate the adequacy of a component to withstand cyclic loading. During its audit, the staff also reviewed the applicant's basis document for metal fatigue TLAAs and noted that the fatigue tests were based on the startup design transients for OTSG with a bounding number of 360 cycles. The staff compared the limit with the USAR Table 5.1-8 transient allowable cycle limit of 240 startup cycles and concluded that the bounding number of 360 cycles will not be exceeded during the period of extended operation.

Based on its review, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue analysis of the OTSG tube sleeves remains valid during the period of extended operation. Additionally, it meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.1 because the number of cycles for the startup transient will not exceed the limits established in the OTSG tube sleeve fatigue tests.

Once-Through Steam Generators Auxiliary Feedwater Modification. The staff reviewed LRA Section 4.3.2.2.6.3 on fatigue of the OTSG AFW modification to verify, 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 4.3.3.1.1.3, which state that the reviewer should verify 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 verify that the applicant identified the appropriate program, as described and evaluated in the GALL Report. Furthermore, the reviewer should also 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.

LRA Section 4.3.2.2.6.3 states that AFW initiations (transients 30A and 30B in LRA Table 4.3-1) are currently at 196.5 and 224.5 cycles, respectively. The staff noted that transients 30A and 30B are projected to a maximum of 387 and 442 cycles, respectively, through the period of extended operation, and these 60-year projections exceed the 300 design cycles for the AFW thermal sleeve. The staff noted that transients 30A and 30B in LRA Table 4.3-1 are identified as "Auxiliary Feedwater Bolted Nozzle" (1-1 and 1-2). It is not clear to the staff whether these AFW injection transients refer to those transients identified in LRA Table 4.3-1. During its audit, the staff noted that the applicant's basis documents for the metal fatigue TLAA indicated that the 3-in. diameter AFW nozzles are limited to 1,447 cycles of AFW initiation based on the CUF of 1.0 for the studs. It is not clear to the staff whether the design cycle limit of 1,447 cycles for "AFW initiation" is tracked in the applicant's Fatigue Monitoring Program.

By letter dated May 2, 2011, the staff issued RAI 4.3-8 requesting that the applicant clarify how the "auxiliary feedwater injection transient" for the modified AFW thermal sleeve design is related to transient 30A, "Auxiliary Feedwater Bolted Nozzle 1-1," and transient 30B, "Auxiliary Feedwater Bolted Nozzle 1-2," in LRA Table 4.3-1. The staff also asked the applicant to clarify the cycle limit of 1,447 for the "AFW initiations" transient discussed in the basis document and to explain whether the "AFW initiation" transient will be monitored by the Fatigue Monitoring Program during the period of extended operation.

In its response dated June 17, 2011, the applicant stated that the AFW injection transient was used to evaluate the AFW nozzle thermal sleeves, AFW nozzle studs, and AFW nozzle flange. The applicant explained that the AFW nozzle thermal sleeves were initially qualified for 300 AFW cycles using conservative analytical techniques, and the thermal sleeves were reanalyzed in December 1982 using numerical methods and were re-qualified for 40,000 AFW cycles. The applicant revised LRA Sections 4.3.2.2.6.3 and A.2.3.2.7 to reflect that the AFW nozzle thermal sleeve is qualified for 40,000 cycles. The applicant stated that the AFW nozzle stud fatigue analysis included the bounding transients of HU/CD, boltup and unbolt, and AFW initiation, and the allowable cycles were reduced from 7,000 to 1,447 to obtain a CUF, for the studs, of less than 1.0.

The applicant also stated that the AFW nozzle flange fatigue analysis included the bounding transients of HU/CD, boltup and unbolt, and AFW initiation, and the cycles were reduced from 7,000 to 875 in the analysis, which resulted in a CUF value of 0.55 for the flange. The applicant stated that transients 30A and 30B in LRA Table 4.3-1, identified as "Auxiliary Feedwater Bolted Nozzle" (1-1 and 1-2), are applicable to the AFW nozzle flanges. The flange is the limiting component; therefore, the transient design cycle limit is set to 875. The applicant revised LRA

Sections 4.3.2.2.6.3 and A.2.3.2.7 to reflect that the AFW nozzle flange is the location with a limit on the number of design cycles of 875. The staff noted that LRA Table 4.3-1 indicates the correct limiting number of design cycle of 875 for transients 30A and 30B.

The staff finds the applicant's response acceptable because the applicant's clarification and revision clearly identifies the proper limiting number of design cycles for monitoring by the Fatigue Monitoring Program to ensure that the fatigue usage limits for the OTSG AFW nozzles will not be exceeded. The staff's concerns described in RAI 4.3-8 are resolved.

The staff reviewed the CUF values provided by the applicant, in Tables 3-5 and 3-6 of AREVA Document 51-9157140-001, in response to RAI 4.3-12 (letter dated June 17, 2011), and confirmed that the design CUF values for the Class 1 components associated with the OTSGs are less than the design limit of 1.0. The staff noted that the applicant credited the cycle-counting activities of its Fatigue Monitoring Program as the basis for managing cumulative fatigue damage that may occur in the RV during the period of extended operation, and it will initiate corrective actions to ensure the design cycles and design limit of 1.0 will not be exceeded. Consistent with the recommendation of GALL Report AMP X.M1, the staff noted that the cycle-counting activities in the applicant's Fatigue Monitoring Program are an acceptable approach to manage CUF values for RCPB components and are consistent with 10 CFR 54.21(c)(1)(iii). The staff's evaluation of the applicant's Fatigue Monitoring Program is documented in SER Section 3.0.3.2.6.

The staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging related to fatigue analyses of the OTSGs AFW header modification 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 for the following reasons:

- The applicant's Fatigue Monitoring Program monitors and tracks the number of design basis transients that will occur through the period of extended operation.
- The applicant's Fatigue Monitoring Program includes action limits and corrective actions that will ensure that the CUF design limit of 1.0 will not be exceeded during the period of extended operation.
- The use of the applicant's Fatigue Monitoring Program is consistent with the recommendations of the GALL Report AMP X.M1 for managing cumulative fatigue damage.

Once-Through Steam Generators Tubes and Tube Stabilizers Flow-Induced Vibration. The staff reviewed LRA Section 4.3.2.2.6.4 on FIV of the OTSG tubes and tube stabilizers to verify, pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis has been projected the period 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 4.3.3.1.1.2, which 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 noted that the 40-year design CUF of an un-repaired tube next to the open lane is 0.443, and the applicant calculated the projected 60-year CUF by multiplying 0.443 by a factor of 1.5. The staff noted that the resulting CUF of 0.665 remains less than the design limit of 1.0. The staff finds the use of a 1.5 factor projection basis reasonable for design basis CUF values that are based on a 40-year design life because 1.5 provides a reasonable scale to project to 60 years from the 40-year design CUF, and the resulting estimated 60-year CUF provides a gauge of how much margin is available before the design limit of 1.0 is reached.

The staff noted that, in LRA Section 4.3.2.2.6.4, the applicant stated the CUF for the 3/8-in. tube stabilizers was calculated using both high-cycle (FIV) and low-cycle (transients) fatigue, with 40-year design basis CUF values of 0.12 for the tube-to-stabilizer weld and 0.07 for the nail. However, based on the description provided in the LRA, it was not clear to the staff whether the CUF values of 0.12 and 0.07 for the tube-to-stabilizer weld and the nail, respectively, included both high-cycle and low-cycle fatigue. It was also not clear why only the FIV portion of these CUF values are increased by 1.5 to demonstrate that the TLAA is valid for the period of extended operation. By letter dated May 2, 2011, the staff issued RAI 4.3-20 requesting that the applicant clarify whether the CUF values are calculated considering both high-cycle and low-cycle fatigue. The staff also asked the applicant to provide the disposition, with the associated basis, in accordance with 10 CFR 54.21(c)(1) for the low-cycle (transient) portion of the fatigue TLAA for the tube-to-stabilizer weld and nail.

In its response dated June 17, 2011, the applicant confirmed that the CUFs for the 3/8-in. tube stabilizers are calculated using both high-cycle (FIV) and low-cycle (thermal transients) fatigue. The applicant clarified that LRA Section 4.3.2.2.6.4 applies to the disposition of the high-cycle fatigue TLAA for the SG tubes and stabilizers. The low-cycle fatigue TLAA of the OTSG locations, including the stabilizers, is addressed in LRA Section 4.3.2.2.6.1. The staff noted that the 40-year design CUF values of 0.12 and 0.07 for the tube-to-stabilizer weld and the nail, respectively, and the applicant calculated the projected 60-year CUF values by multiplying them by a factor 1.5. The staff finds the use of a 1.5 projection reasonable, as described above.

The staff finds the applicant’s response acceptable because the applicant clarified that the high-cycle fatigue TLAA for the SG stabilizer is addressed in LRA Section 4.3.2.2.6.4, and the applicant projected the CUF to remain valid for the period of extended operation. The staff also finds the applicant’s response acceptable because the applicant clarified that the low-cycle fatigue TLAA for the SG stabilizer is addressed in LRA Section 4.3.2.2.6.1, and the applicant is managing cumulative fatigue damage with its Fatigue Monitoring Program, which ensures that the component fatigue usage limit is not exceeded. The staff’s concern described in RAI 4.3-20 is resolved.

The staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis for OTSG tubes and tube stabilizers FIV analysis have been projected to the end of the period of extended operation. Additionally, it meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.2 because the applicant demonstrated that the projected CUF values will be less than the ASME Code, Section III, design limit of 1.0 through the period of extended operation.

4.3.2.2.3 USAR Supplement

LRA Section A.2.3 provides the USAR supplement summarizing the metal fatigue TLAAs. The staff reviewed LRA Sections A.2.3.1.1, A.2.3.1.5, and A.2.3.2 consistent with the review procedures in SRP-LR Section 4.3.3.3, which state that the reviewer verifies that the applicant provided information to be included in the USAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA.

The staff's review of USAR supplement Sections A.2.3.1.1, A.2.3.1.5, and A.2.3.2 of the LRA found that the applicant did not include summary description subsections for the fatigue TLAAs of Class 1 components and piping discussed in LRA Sections 4.3.2.2 and 4.3.2.3:

- RV assembly components, including shells, heads, flanges, nozzles, and bolts of LRA Section 4.3.2.2.1
- OTSGs primary and secondary shell components of LRA Section 4.3.2.2.6.1
- Class 1 piping of LRA Section 4.3.2.3.1

The staff noted that 10 CFR 54.21(d) requires that the USAR supplement contain an appropriate summary description of all TLAA evaluations. By letter dated May 2, 2011, the staff issued RAI 4.3-22 asking the applicant to justify why LRA Section A.2.3 does not include summary descriptions for the aforementioned fatigue TLAAs of Class 1 components and piping.

In its response dated June 17, 2011, the applicant stated that LRA Section A.2.3.2 is revised to include summary descriptions of the TLAA evaluations for the RV, the Class 1 piping, and the OTSGs in the USAR supplement. The staff confirmed that the applicant provided an acceptable summary description of the TLAA evaluations, which included the dispositions of the TLAAs, in accordance with 10 CFR 54.21(c)(1).

The staff finds the applicant's response acceptable because the applicant amended LRA Section A.2.3.2 to include an USAR supplement that contains the summary description of the TLAAs for the RV, the Class 1 piping, and the OTSGs, in accordance with 10 CFR 54.21(d). The staff's concern described in RAI 4.3-22 is resolved.

Based on its review of the USAR supplement, as amended by letter dated June 17, 2011, the staff finds it meets the acceptance criteria in SRP-LR Section 4.3.2.3. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address fatigue TLAAs of Class 1 vessels, pumps, and major components, as required by 10 CFR 54.21(d).

4.3.2.2.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the low cycle fatigue and FIV analyses for the RVIs and the SG tube sleeves remain valid during the period of extended operation. The staff also concludes that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses for FIV of the surveillance capsule holder tubes of RVIs, the SG tubes and SG tube stabilizers, 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 effects of aging related to fatigue of the RV, RVIs, CRD housings, RCP casings, pressurizer, OTSGs primary and secondary shell components, OTSGs SG tube sleeves and welded plugs, and

OTSGs AFW header modification will be adequately managed for the period of extended operation. The staff also concludes that the USAR supplement contains appropriate summary descriptions of the TLAAs, as required by 10 CFR 54.21(d).

4.3.2.3 Class 1 Piping and Valves

4.3.2.3.1 Summary of Technical Information in the Application

LRA Section 4.3.2.3 describes the applicant's TLAAs for metal fatigue of ASME Code, Section III, Class 1 piping and valves. The applicant stated that the RCS piping and the RCPB piping in other systems were designed to the requirements of American National Standards Institute (ANSI) B31.7 draft, (February 1968 with Errata, June 1968) and also meet the design requirements of ANSI B31.7, 1969 edition. The B31.7 Piping Code requires the evaluation of transient thermal and mechanical load cycles and the determination of fatigue usage for Class 1 piping. In addition, the reactor head vent and other piping, designated as quality Group A, B, or C, is designed to ASME Code, Section III, 1971 edition, Class 1, 2, or 3, respectively. The applicant stated that it has no Class 1 piping designed to ANSI B31.1. The applicant's evaluation is documented in the following subsections:

- LRA Section 4.3.2.3.1—Class 1 piping fatigue
- LRA Section 4.3.2.3.2—Class 1 valves fatigue
- LRA Section 4.3.2.3.3—High-energy line break (HELB) postulation

<u>Class 1 Piping Fatigue</u>. The applicant described the following fatigue analyses for Class 1 piping, in LRA Section 4.3.2.3.1:

- reactor coolant piping
- pressurizer surge line
- reactor coolant drains and letdown lines
- HPI lines
- decay heat removal lines
- core flooding lines
- pressurizer safety and relief valve lines

For the reactor coolant piping and the pressurizer safety and relief valve lines, the applicant stated that the CUF values of record were all less than 1.0 based on the design transients identified in LRA Table 4.3-1. LRA Section 4.3.4 contains the details of pressurizer surge line fatigue analyses, where the effects of reactor coolant environment on fatigue are also addressed. For the reactor coolant drains and letdown lines, the HPI lines, the decay heat removal lines, and the core flooding lines, the applicant stated that the original fatigue analyses were updated based on NRC Bulletin 79-14, and the resulting CUFs were all less than 1.0 for the design transients listed in LRA Table 4.3-1.

For the HPI line, the applicant stated that the fatigue usage for the normal makeup nozzle was mainly due to HPI flow tests. The estimated CUF value of 0.558 consists of 0.513 from 40 flow tests, which is the design number of cycles of flow tests, and all other transients contribute 0.045 of the usage factor. The applicant stated that it will monitor transient cycles, and fatigue of the nozzle will be managed through the period of extended operation.

The applicant stated that all CUFs calculated for Class 1 piping are less than 1.0 based on the design transients identified in LRA Table 4.3-1 and that the Fatigue Monitoring Program will monitor these transients for the period of extended operation to ensure that action is taken before the design cycles are reached. The applicant dispositioned all of these TLAAs in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of aging due to fatigue of the Class 1 piping will be managed for the period of extended operation by the Fatigue Monitoring Program.

Class 1 Valves Fatigue. LRA Section 4.3.2.3.2 provides the applicant's assessment of the potential metal fatigue TLAA for Class 1 valves. The applicant stated that it reviewed its licensing basis documents to determine if they contained fatigue analyses for Class 1 valves and that 12 valves of 4 in. or greater diameter were identified. The applicant's review of its QA records located the stress reports of record for each of the 12 valves, but no associated fatigue reports were identified. On the basis of this review, the applicant concluded that fatigue analyses for Class 1 valves were not performed, and there is no metal fatigue TLAA for Class 1 valves. The applicant stated that its conclusion is consistent with industry practice at the time the plant was designed and that valve bodies and pump casings were considered robust compared to the piping systems in which they were located; therefore, fatigue of the attached piping was understood to bound the fatigue of the valve bodies. The applicant stated that this is not a TLAA because no fatigue analyses were identified for Class 1 valves.

High-Energy Line Break Postulations. LRA Section 4.3.2.3.3 describes the disposition of the TLAA associated with the use of high-energy line break (HELB) postulations. The applicant stated that, in accordance with USAR Section 3.6.2.1, the criteria given in SRP Sections 3.6.1 and 3.6.2, including Branch Technical Position (BTP) MEB 3-1, were used to determine the pipe break locations, and allowed the elimination of potential break locations where the CUFs were less than 0.1. The identification of HELB locations for the RCS hot leg and cold leg piping was replaced by LBB criteria (in 1990), as discussed in LRA Section 4.7.1. The applicant indicated that its identification of HELB piping locations used the CUFs based on the design transients that are counted by the Fatigue Monitoring Program. The applicant further stated that this program will require action if any of the design cycles are approached, including a review of the HELB location selections, and that the effects of fatigue on the HELB location selection will be managed by the Fatigue Monitoring Program for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii).

4.3.2.3.2 Staff Evaluation

The staff reviewed LRA Section 4.3.2.3 concerning fatigue TLAAs of the Class 1 piping and valves, and its evaluation of the applicant's disposition of these TLAAs is documented in three subsections.

Class 1 Piping Fatigue. The staff reviewed LRA Section 4.3.2.3.1 on fatigue of Class 1 piping to verify, 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 4.3.3.1.1.3, which state that the reviewer should verify 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 verify that the applicant identified the appropriate program, as described and evaluated in the GALL Report. Furthermore, the reviewer should also 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 noted that the applicant's fatigue TLAAs included the following Class 1 piping of the RCS and RCPB:

- reactor coolant piping
- pressurizer surge line

- reactor coolant drains and letdown lines
- HPI lines
- decay heat removal lines
- core flooding lines
- pressurizer safety and relief valve lines

Based on its review of USAR Table 5.2-1, the staff noted that the design of all of this piping is based on ANSI B31.7 draft (February 1968 with Errata of June 1968) or the 1971 ASME Code, Section III. The staff determined that all of the applicable design codes for the Class 1 piping require CUF-based fatigue analyses.

LRA Section 4.3.2.3.1 states that all CUF values are less than 1.0 for these piping based on the design transients listed in LRA Table 4.3-1. The staff reviewed the CUF values provided by the applicant, in Table 3-7 of AREVA Document 51-9157140-001, in response to RAI 4.3-12 (letter dated June 17, 2011), and confirmed that the design CUF values for the Class 1 piping are less than the design limit of 1.0.

The staff noted that the applicant credits its Fatigue Monitoring Program as the basis for managing cumulative fatigue damage that may occur in Class 1 piping during the period of extended operation. The applicant's program includes monitoring and tracking the number of critical thermal and pressure transients that are significant contributors to the fatigue usage factor, which involves the systematic counting of transient cycles and the evaluation of operating data to ensure that the allowable cycle limits are not exceeded. The staff also noted that the applicant's program incorporates action limits and acceptance criteria to ensure that corrective actions are taken to prevent the fatigue TLAAs from exceeding their acceptance criteria, and to assure that the fatigue usage resulting from actual operational transients does not exceed the Code design limit of 1.0. Consistent with the recommendation of GALL Report AMP X.M1, the staff noted that the cycle-counting activities in the applicant's Fatigue Monitoring Program are an acceptable approach to manage CUF values for RCPB components and are consistent with 10 CFR 54.21(c)(1)(iii). The staff's evaluation of the applicant's Fatigue Monitoring Program is documented in SER Section 3.0.3.2.6.

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 of Class 1 piping 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 for the following reasons:

- The Fatigue Monitoring Program monitors and tracks the number of design basis transients that will occur through the period of extended operation.
- The Fatigue Monitoring Program includes corrective actions that will ensure that the Code design limit of 1.0 will not be exceeded during the period of extended operation.
- The use of the Fatigue Monitoring Program is consistent with the recommendations of the GALL Report AMP X.M1 for managing cumulative fatigue damage.

Class 1 Valves Fatigue. The staff reviewed LRA Section 4.3.2.3.2 and the applicant's evaluation of the absence of TLAAs for Class 1 valves fatigue to verify the applicant's basis. The staff reviewed the applicant's evaluation and conclusion consistent with the review procedures in SRP-LR Section 4.1.3, which state that the reviewer verifies that the selected analyses do not meet at least one of the six criteria of a TLAA, as defined in 10 CFR 54.3(a).

The staff reviewed the applicant’s CLB relevant to the RCPB Class 1 valves but was not able to ascertain the applicable design code(s) for the valves. Specifically, the staff noted that USAR Table 5.2-1 states the following:

- Relief valves and pressurizer safety valves were designed to ASME Code draft Pump and Valve Code, November 1968 edition.
- Loop isolation valves and other valves were designed to ASME Code, Section III, 1971 edition or later.
- Pressurizer pilot-operated relief isolation valves were designed to ASME Code, Section III, 1974 edition with addenda through summer 1976.
- Pressurizer spray line isolation valves were designed to ASME Code, Section III, 1986 edition.

For valves larger than 4-in. nominal pipe size (NPS) that are designed to these Codes, the staff noted that these valves must meet the requirements of NB-3530 through NB-3550 (or Article 4 of 1968 edition of draft Pump and Valve Code). The adequacy of these valves for cyclic conditions is confirmed in accordance with Subsection NB-3553 (or Subarticle 454 of the 1968 edition of the draft Pump and Valve Code), which requires the I_t fatigue usage factors for the valves to be less than a design limit of 1.0. It was unclear to the staff why the fatigue analyses of Class 1 valves were not performed as required by the design code and, as such, why these Class 1 valves are identified as not being a TLAA.

To address this issue, the staff issued RAI 4.1-1, Request 1, Part A, by letter dated May 2, 2011. This RAI requested the applicant to clarify which edition of the ASME Code, Section III, was used for the design of the following valves:

- seal injection flow isolation valve
- pump seal return isolation valve
- letdown cooler inlet valve
- HPI valve
- seal return isolation valve
- makeup isolation valve
- letdown cooler isolation valve
- pressure spray control valve
- pressurizer low-pressure injection valve
- each of the decay heat removal outlet valves

The staff also asked the applicant to justify why an I_t fatigue analysis would not have been required by the applicable ASME Code, Section III edition of record for each valve. If an I_t fatigue analysis was required for the valve, the staff asked the applicant to justify why the analysis would not need to be identified as a TLAA in accordance with the requirements of 10 CFR 54.21(c)(1).

The applicant’s July 22, 2011, response to RAI 4.1-1, Request 1, Part A, stated that, for ASME Code, Section III editions equal to the 1971 edition or more recent editions, the NB 3500 requirements would have required a CUF or I_t fatigue analysis if the valves were greater than 4-in. NPS (i.e., if the valves were designated as large bore valves). The applicant clarified that the 1968 draft Pump and Valve Code imparts equivalent I_t fatigue analysis requirements for those valves greater than 4-in. NPS.

The applicant clarified that it reviewed the Class 1 piping and instrument diagrams (P&IDs) to identify those Class 1 valves that are greater than 4-in. NPS and, as a result of this search, the following 12 Class 1 valves were identified as large bore Class 1 valves:

- low-pressure injection system outboard containment isolation valves (Valve Nos. DH1A and DH1B)—10-in. NPS, designed to the 1968 draft ASME Pump and Valve Code
- decay heat removal outlet system containment isolation valves (Valve Nos. DH11 and DH12)—12-in. NPS, designed to the 1968 draft ASME Pump and Valve Code
- low-pressure injection system stop check inside containment isolation valves (Valve Nos. DH76 and DH77)—10-in. NPS, designed to the 1971 edition of the ASME Code, Section III, inclusive of the summer 1971 addenda
- decay heat removal outlet system bypass valves (Valve Nos. DH21 and DH23 in the bypass lines around isolation valves DH11 and DH12)—8-in. NPS, designed to the 1971 edition of the ASME Code, Section III
- core flood system stop check isolation valves (Valve Nos. CF28, CF29, CF30, and CF31)—14-in. NPS, designed to the 1971 ASME Code, Section III, inclusive of the winter 1972 addenda

However, the applicant clarified that its search of the Davis-Besse plant records did not locate any CUF or I_t fatigue analyses for these large bore Class 1 valves. The applicant stated that, to address this issue, it is amending LRA Appendix A and Table A-1 (the LRA Commitment Table) to include Commitment No. 46, as follows, to perform CUF or I_t fatigue analyses for these large bore Class 1 valves:

> FENOC commits to perform a fatigue evaluation in accordance with the requirements of the ASME Code of record for the Davis Besse Class 1 valves that are greater than 4 inches nominal pipe size. The applicable valve identification numbers are CF28, CF29, CF30, CF31, DH76, DH77, DH11, DH12, DH1A, DH1B, DH21, and DH23.

The applicant also provided a new USAR supplement summary description in LRA Section A.2.3.2.13, "Class 1 Valves Fatigue," for large-bore Class 1 Valve TLAAs, as required by 10 CFR 54.21(d).

The staff confirmed that the draft 1968 ASME Pump and Valve Code and the cited editions of the ASME Code, Section III, would not require the performance of CUF or I_t metal fatigue analyses for Class 1 valves that were less than or equal to 4-in. NPS. However, it would have required either a CUF or I_t metal fatigue analysis if the valve was greater than 4-in. NPS. Thus, based on the applicant's response, the staff concluded that the 12 large bore Class 1 valves identified by the applicant were the appropriate Class 1 valve components that should have metal fatigue analyses, unless the applicant could demonstrate that it was appropriate to have waived a particular large bore Class 1 valve from a fatigue analysis under applicable Code fatigue analysis exemption or waiver provisions. The staff found that the applicant provided an acceptable basis for resolving the question in RAI 4.1-1, Request 1, Part A. The staff found also that the applicant appropriately identified which Class 1 valves would need to be analyzed in accordance with either a CUF or I_t fatigue analysis for the following reasons:

- The applicant's basis is in compliance with the applicable design code provisions.

- The applicant appropriately amended the LRA to identify that the 12 referenced large bore Class 1 valves were required to receive appropriate metal fatigue analyses.
- The applicant treated the future CUF analyses for these valve components (Commitment No. 46) as a TLAA for the application.

The staff review and evaluation of the new TLAA and associated USAR supplement section for these large bore Class 1 valves is provided below in this SER Section, including an evaluation on whether the commitment in LRA Commitment No. 46 provides an acceptable basis for dispositioning the TLAA for these valves in accordance with 10 CFR 54.21(c)(1)(iii). With the identification of these analyses as TLAAs, the staff's concern described in RAI 4.1-1, Request 1, Part A, is resolved.

By letter dated May 2, 2011, RAI 4.1-1, Request 1, Part B, requested that the applicant resolve apparent inconsistencies between relevant information in USAR Table 5.1-1b and USAR Table 5.2-1. Specifically, the staff asked the applicant to confirm that the "pressurizer relief isolation valve" in USAR Table 5.1-1b correlates to the "relief valve" in USAR Table 5.2-1. The staff also asked the applicant to confirm that the "pressurizer pilot-operated relief valve (PORV)" in USAR Table 5.1-1b correlates to the "pressurizer pilot-operated relief isolation valve" in USAR Table 5.2-1. If these items cannot be confirmed, the applicant was asked to identify which design codes of record are applicable for the design of the "pressurizer relief isolation valve" and the "pressurizer pilot-operated relief valve," as listed in USAR Table 5.1-1b.

The applicant's July 22, 2011, response to RAI 4.1-1, Request 1, Part B, confirmed the component identification correlations that were presumed by the staff during its review. Specifically the applicant confirmed that the "pressurizer relief isolation valve" in USAR Table 5.1-1b does correlate to the "relief valve" in USAR Table 5.2-1 and that the "pressurizer pilot-operated relief valve (PORV)" in USAR Table 5.1-1 b does correlate to the "pressurizer pilot-operated relief isolation valve" in USAR Table 5.2-1. The applicant also administratively confirmed that the design code for the pressurizer relief isolation valve is the 1974 edition of the ASME Code, Section III, inclusive of the summer 1976 addenda, and the design code for the pressurizer PORV is the 1968 edition of the draft ASME Pump and Valve Code.

The staff noted that the applicant's response confirmed that nomenclature inconsistencies between USAR Tables 5.1-1b and 5.2-1 for the pressurizer relief isolation valve and the pressure PORV were really referring to the same component and to identify the specific design codes that were used for design, fabrication, and installation of these valves at the facility. Thus, the staff found that the applicant's responses to RAI 4.1-1, Request 1, Part B, was acceptable because it provided the administrative clarifications necessary to confirm the pressurizer valve component nomenclature correlations that were presumed by the staff during its review. The applicant's response also identified the appropriate design codes used for the design of the pressurizer relief isolation valve and the pressurizer PORV. The staff's concern described in RAI 4.1-1, Request 1, Part B, is resolved.

By letter dated May 2, 2011, RAI 4.1-1, Request 2, requested that the applicant justify why an I_t fatigue analysis was not required for the pressurizer safety valves and PORV under the provisions of the 1968 draft ASME Pump and Valve Code, as part of the design basis listed under USAR Table 5.2-1, since Sections 452 and 454 of this Code include applicable time-dependent cyclic or fatigue assessment criteria. If an I_t analysis was performed as part of the design basis for these valves, the staff asked the applicant to justify why these analyses do not need to be identified as a TLAA in accordance with 10 CFR 54.21(c)(1).

The applicant's July 22, 2011, response to RAI 4.1-1, Request 2, clarified that the pressurizer safety valves (Valve Nos. RC13A and RC13B at the facility) and the pressurizer PORV (Valve No. RC2A) were all designed to the draft 1968 ASME Pump and Valve Code. The applicant confirmed that the 1968 draft ASME Pump and Valve Code required an I_t fatigue analysis to be performed if the valves were greater than 4-in. in NPS, or for valves less than or equal to 4-in. NPS if specified in the owner's design specification. The applicant clarified that these valves were all less than or equal to 4-in. NPS, and confirmed that the owner's design specifications for the valves did not require the applicant to perform an I_t fatigue analysis for the valves.

The staff noted that the applicant's response was consistent with the staff's understanding of the requirements in the draft 1968 draft ASME Pump and Valve Code. Specifically, the protocols used in the 1968 draft ASME Pump and Valve Code establish when an I_t fatigue analysis would have been required for a Class 1 valve procured, designed, analyzed, and installed in accordance with the 1968 draft ASME Pump and Valve Code specification's design rules. Based on this review, the staff found that the applicant's response to RAI 4.1-1, Request 2, provided an acceptable basis for concluding that I_t fatigue analyses would not be needed for the pressurizer safety valves and the pressurizer PORV for the following reasons:

- The staff has confirmed that the 1968 draft ASME Pump and Valve Code would not have required the applicant to perform CUF or I_t fatigue analyses for the valves based on their NPS.
- The owner's procurement specifications for these valves did not require the performance of either a CUF or I_t fatigue analysis as a condition for valve procurement.

The staff's concern described in RAI 4.1-1, Request 2, is resolved.

By letter dated May 2, 2011, RAI 4.1-1, Request 3, requested that the applicant justify why an I_t fatigue analysis was not required for the pressurizer pilot-operated relief isolation valve (as referenced in USAR Table 5.2-1) in accordance with paragraphs NB-3545.3 and NB-3550 of the 1974 edition of the ASME Code, Section III, and the provisions for performing I_t fatigue analyses in paragraph NB-3553. If an I_t analysis was performed as part of the design basis for the pressurizer pilot-operated relief isolation valve, the staff asked the applicant to justify its basis for concluding that the I_t fatigue analysis for the valve would not need to be identified as a TLAA in accordance with the requirement in 10 CFR 54.21(c)(1).

By letter dated May 2, 2011, RAI 4.1-1, Request 4, requested that the applicant justify why an I_t fatigue analysis was not required for the pressurizer spray line isolation valve in accordance with paragraphs NB-3545.3 and NB-3550 of the 1986 edition of the ASME Code, Section III, and the provisions for performing I_t fatigue analyses in paragraph NB-3553. If an I_t analysis was performed as part of the design basis for the pressurizer spray line isolation valve, the staff asked the applicant to justify its conclusion that the I_t fatigue analysis for the valve would not need to be identified as a TLAA for the LRA in accordance with 10 CFR 54.21(c)(1).

The applicant's July 22, 2011, response to RAI 4.1-1, Requests 3 and 4, clarified that the NB-3513 and NB-3563 paragraph provisions in the 1974 and 1986 editions of the ASME Code, Section III, did not require the applicant to perform an I_t fatigue analysis for a Class 1 valve that was designed and analyzed to the Code's design criteria if the valve was less than or equal to 4-in. NPS. The applicant also clarified that the pressurizer relief isolation valve (Valve No. RC11) and pressurizer spray isolation valve (Valve No. RC10) are only 2½-in. NPS and, therefore, were not required to be analyzed in accordance with a ASME Code, Section III, paragraph NB-3553 I_t fatigue analysis. The applicant clarified that it confirmed that the owner's purchasing specifications for these valves did not specifically request the performance of CUF or I_t fatigue analyses as a condition for valve procurement.

The staff reviewed the 1974 and 1986 editions of the ASME Code, Section III, Subarticle NB-3000 requirements to verify when the Code rules would require the performance of an I_t or CUF fatigue analysis for Class 1 valve. The staff confirmed that paragraph NB-3513 for small-bore Class 1 valves (i.e., valves less than or equal to 4-in. NPS) only require applicant's to perform P-T rating, hydrostatic test rating, and minimum wall thickness and neck thickness assessments for the valves, and did not require the small-bore valves to be analyzed to the cyclical metal fatigue analysis requirements in NB-3553. Based on this review, the staff found that the applicant's response to RAI 4.1-1, Requests 3 and 4, provided an acceptable basis for concluding that CUF and I_t fatigue analyses were not needed for the pressurizer relief isolation valve and the pressurizer spray isolation valve for the following reasons:

- The staff confirmed that the draft 1974 and 1986 editions of the ASME Code, Section III, would not have required the applicant to perform CUF or I_t fatigue analyses for the valves based on their NPS.
- The owner's procurement specifications for these valves did not require the performance of either a CUF or I_t fatigue analysis as a condition for valve procurement.

The staff's concerns described in RAI 4.1-1, Requests 3 and 4, are resolved.

Based on this review, the staff determined that the applicant's metal fatigue analysis basis for the Class 1 valves, as amended by the applicant's responses to the requests of RAI 4.1-1, is acceptable for the following reasons:

- The applicant appropriately amended the application to identify the metal fatigue analyses for the 12 identified large bore Class 1 valves as a TLAA for the LRA.
- The applicant demonstrated, and the staff confirmed, that the design codes for the small Class 1 valves would not have required the applicant to perform metal fatigue analyses as part of the design requirements for the valves.

Therefore, RAI 4.1-1 is resolved.

In its response to RAI 4.1-1, Request 1, Part A, by letter dated July 22, 2011, the applicant amended LRA Sections 4.3.2.3.2 and A.2.3.2.13. The staff reviewed the amended LRA Section 4.3.2.3.2 and the fatigue TLAA for Class 1 valves to verify, 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. This review was consistent with the review procedures in SRP-LR Section 4.3.3.1.1.3, which state that the reviewer should verify 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 verify that the applicant identified the appropriate program, as described and evaluated in the GALL Report. Furthermore, the reviewer should also 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.

However, the staff noted that, in the applicant's letter dated July 22, 2011, it did not provide clarifying information regarding whether there were any ASME Code, Section III, NB-3222.4(d) fatigue waiver assessments (or equivalent waiver assessments permitted by the 1968 draft

ASME Pump and Valve Code) for the 12 large-bore Class 1 valves referenced in Commitment No. 46. Therefore, the staff requested additional information regarding whether fatigue calculations were required for these valves as part of the applicant's CLB.

By letter dated August 11, 2011, the staff issued RAI 4.3.2.3.2-1 requesting that the applicant clarify whether the CUF or I_t analyses for each of the 12 large bore Class 1 valves are required as part of the applicant's CLB. In its response dated October 7, 2011, the applicant stated that, in order to provide the fatigue evaluation requested by the staff, it withdrew Commitment No. 46 and provided a new regulatory commitment in its place to read as follows:

> FENOC will perform a fatigue evaluation in accordance with the requirements of the ASME Code of record for the Davis-Besse Class 1 valves that are greater than 4 inches diameter nominal pipe size. The applicable valve identification numbers are CF28, CF29, CF30, CF31, DH76, DH77, DH11, DH12, DH1A, DH1B, DH21 and DH23. LRA Sections 4.3.2.3.2 and A.2.3.2.13, both titled "Class 1 Valves Fatigue," will be revised to include the results of the fatigue evaluations, and these changes will be submitted as an amendment to the Davis Besse LRA no later than May 31, 2012.

The staff confirmed that Commitment No. 46 has been removed from LRA Table A-1 and is defunct. The staff found the removal of Commitment No. 46 reasonable because the staff will complete its review of this issue once the applicant submits the fatigue evaluations for these Class 1 valves and supplements LRA Sections 4.3.2.3.2 and A.2.3.2.13 to include those results.

In its supplemental response to RAI 4.3.2.3.2-1, dated May 25, 2012, the applicant stated that fatigue analyses were prepared for each of the 12 large-bore Class 1 valves in accordance with paragraph NB-3550 of ASME Code, Section III, 1974 edition with addenda through the summer of 1976. The applicant also stated that the CUFs calculated for the Class 1 valves, which are based on nuclear steam supply system design transients, are less than the design limit of 1.0 and the number of occurrences of design transients is tracked by the Fatigue Monitoring Program. The applicant also revised LRA Sections 4.3.2.3.2 and A.2.3.2.13 to include the results of the fatigue evaluations. The staff noted that the code edition and design transients information addressed the staff's request whether fatigue analyses were required as part of the applicant's CLB.

Based on its review, the staff finds the applicant's response to RAI 4.3.2.3.2-1 acceptable because the applicant revised the LRA to provide the metal fatigue TLAA disposition and associated supporting information for its Class 1 valves. Therefore, the staff's concern described in RAI 4.3.2.3.2-1 is resolved. The staff's review of the revised LRA Section 4.3.2.3.2 is documented below.

The staff noted that the applicant credits its Fatigue Monitoring Program as the basis for managing cumulative fatigue damage that may occur in ASME Code, Class 1 valves during the period of extended operation. The applicant's program includes monitoring and tracking the number of critical thermal and pressure transients that are significant contributors to the fatigue usage factor, which involves the systematic counting of transient cycles and the evaluation of operating data to ensure that the allowable cycle limits are not exceeded. The staff also noted that the applicant's program incorporates acceptance criteria to ensure that corrective actions are taken to prevent these fatigue TLAAs from exceeding their acceptance criteria, and to assure that the fatigue usage resulting from actual operational transients does not exceed the Code design limit of 1.0. Consistent with the recommendation of GALL Report AMP X.M1, the staff noted that the cycle-counting activities in the applicant's Fatigue Monitoring Program are an acceptable approach to manage CUF values for its Class 1 valves and are consistent with 10 CFR 54.21(c)(1)(iii). The staff's evaluation of the applicant's Fatigue Monitoring Program is documented in SER Section 3.0.3.2.6.

The staff finds the applicant has demonstrated pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue of the ASME Code, Class 1 valves will be adequately managed for the period of extended operation. Additionally, it meets the acceptance criteria in SRP–LR Section 4.3.2.1.2.3 because the applicant's Fatigue Monitoring Program tracks the number design basis transients that will occur through the period of extended operation and includes corrective actions that will ensure that the assumption made in these ASME Code, Class 1 valves fatigue analyses will not be exceeded during the period of extended operation.

High-Energy Line Break Postulations. The staff reviewed LRA Section 4.3.2.3.3 and the TLAA disposition of HELB postulations to verify, 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 4.3.3.1.1.3, which state that the reviewer should verify the appropriateness of the applicant's program for monitoring and tracking the number of critical thermal and pressure transients.

The staff reviewed USAR Section 3.6, which includes the criteria used by the applicant for HELB postulation, which is based on SRP Sections 3.6.1 and 3.6.2, including BTP MEB 3-1. The staff noted that one of these criteria, which allows HELB locations to be eliminated, is whether the CUF value will be 0.1 or less. The staff noted that the cycle limits in LRA Table 4.3-1 correspond to the CUF design limit of 1.0 under the ASME Code, Section III, criteria. Therefore, the staff needs additional information related to the cycle limits that are applicable to HELB piping locations because they may be less than those provided in LRA Table 4.3-1 and monitored by the applicant's Fatigue Monitoring Program. The staff also noted that the Fatigue Monitoring Program does not address the acceptance criteria or the cycle-based action limits for these HELB locations. By letter dated May 2, 2011, the staff issued RAI 4.3-13 requesting that the applicant provide the design-basis transients and associated cycle limits applicable to each of the HELB piping locations that are within the scope of LRA Section 4.3.2.3.3. The staff also asked the applicant to justify that the Fatigue Monitoring Program can adequately ensure the CUF for HELB locations remain below 0.1 by using systematic counting of plant transient cycles associated with HELB analysis, and to provide any appropriate revisions to the Fatigue Monitoring Program.

In its response dated June 17, 2011, the applicant stated that the HELB postulation based on fatigue usage is applicable to the following ASME Code Class 1 piping locations: low-pressure injection lines, core flooding lines, letdown lines, and decay heat removal lines. The response also provided a table that lists the design transients that were considered in the fatigue analyses for these ASME Code Class 1 piping locations, with the associated analyzed cycles and 60-year projected cycles. The staff noted that the number of cycles projected in 60-years is less than the number of cycles assumed in the fatigue analyses for these HELB locations, with the exception of transient 3 "Power change 8-100%," transient 4 "Power change 100-8%," and transient 11, "Rod Withdrawal Accident." The applicant stated that transients 3 and 4 are not monitored, and its site is a based loaded plant. The staff's evaluation of the applicant's basis for not monitoring transients 3 and 4 is documented in SER Section 4.3.1.2, where the staff agreed with the applicant that it need not monitor for these transients. The staff also noted that transient 11 has not occurred as of February 19, 2008, and finds that the applicant conservatively assumed the 60-year projected cycles is equal to the number of design cycles (40 cycles) to allow for future occurrence. Other than the three transients described above, the staff noted that there is a significant margin between the number of cycles projected to occur after 60-years of operation and the number of cycles assumed in the applicant's fatigue analyses for HELB postulation. The applicant also amended LRA Sections 4.3.2.2.3 and A.2.3.2.1.2 to indicate that the TLAAs for HELB postulations are dispositioned in accordance with 10 CFR 54.21(c)(1)(i), such that the Class 1 HELB postulations remain valid for the period of extended operation.

The staff finds the applicant's response acceptable because the applicant demonstrated that, for these specific Class 1 piping locations associated with the TLAAs for HELB postulations, the number of cycles projected to occur after 60-years of operation are less than the number of cycles assumed in the fatigue analyses for HELB postulations. Additionally, the applicant revised the TLAA disposition indicating that it will be dispositioned in accordance with 10 CFR 54.21(c)(1)(i). The staff's evaluation of the applicant's projection methodology for design transients is documented in SER Section 4.3.1.2. The staff's concern described in RAI 4.3-13 is resolved.

Based on its review, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the HELB postulations remain valid during the period of extended operation. Additionally, it meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.1 because there is margin between the number of cycles projected to occur after 60 years of operation and the number of assumed cycles in the fatigue analyses of the HELB postulations, and the assumptions in the fatigue TLAAs will not be exceeded during the period of extended operation.

4.3.2.3.3 USAR Supplement

LRA Section A.2.3, as amended by letter dated June 17, 2011, provides the USAR supplement summarizing the metal fatigue TLAAs. The staff reviewed LRA Sections A.2.3.1.2, A.2.3.1.4, and A.2.3.2.11, as amended by letter dated June 17, 2011, consistent with the review procedures in SRP-LR Section 4.3.3.3, which state that the reviewer should verify that the applicant provided information to be included in the USAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA. The SRP-LR also states that the reviewer should verify that the applicant identified and committed in the LRA to any future aging management activities, including enhancements and commitments to be completed before the period of extended operation.

The staff's review of USAR supplement Section A.2.3 of the LRA determined that the applicant did not include a summary description for several metal fatigue TLAAs, including the TLAA of Class 1 piping discussed in LRA Section 4.3.2.3.1.

By letter dated May 2, 2011, the staff issued RAI 4.3-22 requesting that the applicant provide a basis for not including a summary description for the TLAAs of Class 1 piping as a subsection to LRA Section A.2.3.2. In its response dated June 17, 2011, the applicant included LRA Section A.2.3.2.11 "Class 1 Piping," that provided a summary description for its Class 1 Piping. The staff's evaluation of RAI 4.3-22 is documented in SER Section 4.3.2.2.3.

LRA Section A.2.3.2.13, as amended by letters dated July 22, 2011, and May 25, 2012, provides the USAR supplement summarizing the fatigue TLAA for Class 1 valves. The staff reviewed LRA Section A.2.3.2.13, consistent with the review procedures in SRP-LR

Section 4.3.3.3, which states that the reviewer verifies that the applicant provided information, to be included in the USAR supplement, that includes a summary description of the evaluation for the fatigue TLAA of Class 1 valves. The applicant provided Commitment No. 46 in its letter dated July 22, 2011. Based on the staff's concern in RAI 4.3.2.3.2-1, the applicant deleted Commitment No. 46 in its letters dated October 7, 2011. The staff found this deletion acceptable, as described in its evaluation of RAI 4.3.2.3.2-1 which is documented in SER Section 4.3.2.3.2.

Based on its review, the staff finds that the information in the USAR supplement, as amended by letters dated June 17, 2011, July 22, 2011, and May 25, 2012, meets the acceptance criteria in SRP-LR Section 4.3.2.3. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address fatigue TLAAs of Class 1 piping, Class 1 valves and HELB postulations, as required by 10 CFR 54.21(d).

4.3.2.3.4 Conclusion

On the basis of its review, the staff concludes that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging related to fatigue analyses of Class 1 piping and Class 1 valves will be adequately managed for the period of extended operation. The staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the HELB postulations remain valid during the period of extended operation. The staff also concludes that the USAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.3 Non-Class 1 Fatigue Analyses

LRA Section 4.3.3 provides the TLAA evaluation for metal fatigue of non-Class 1 mechanical components. The applicant stated that non-Class 1 components that are quality Group B or C are largely designed and constructed to the ASME Boiler and Pressure Vessel Code, but certain components are built to other codes including B31.1, American Water Works Association (AWWA), and the draft Pump and Valve Code. The applicant's evaluation is documented in the following subsections:

- LRA Section 4.3.3.1—non-Class 1 piping and in-line components
- LRA Section 4.3.3.2—non-Class 1 major components

The staff's review and assessment of these fatigue TLAAs is documented in the two subsections below, which correspond to the applicant's LRA Subsections 4.3.3.1 and 4.3.3.2.

4.3.3.1 Non-Class 1 Piping and In-Line Components

4.3.3.1.1 Summary of Technical Information in the Application

LRA Section 4.3.3.1 describes the applicant's metal fatigue TLAAs for non-Class 1 piping and in-line components. The applicant stated that the fatigue analyses of these non-Class 1 components were based on the design codes that include ASME Code, Section III, Class 2 and Class 3 with respect to thermal (expansion) stresses and ANSI B31.1 for quality Group D components with respect to the number of thermal cycles. In both of these codes, a stress range reduction factor of less than 1.0 is applied to the allowable stress range if the number of stress cycles exceeds 7,000. The applicant compared this cycle limit against its 60-year projections for its thermal transients, listed in LRA Table 4.3-1, as applicable to these non-Class 1 components and determined that the 7,000-cycles limit will not be exceeded. The applicant dispositioned these metal fatigue TLAAs in accordance with 10 CFR 54.21(c)(1)(i), that the fatigue analyses (stress range reduction factor) for non-Class 1 piping and in-line components remain valid through the period of extended operation.

4.3.3.1.2 Staff Evaluation

The staff reviewed LRA Section 4.3.3.1 on fatigue of non-Class 1 Piping and in-line components to verify, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses will remain valid during 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, which state that the staff should review relevant information in the TLAA, operating plant transient history, design basis, and CLB (including TS cycle-counting requirements). The SRP-LR also states that the staff should verify that the maximum allowable stress range values for the existing fatigue analysis remain valid for the period of extended operation and that the allowable limit for full thermal range transients will not be exceeded during the period of extended operation.

The staff noted that the applicable design code requirements, to which systems and components important to safety were designed, are listed in USAR Table 3.2-2. The staff noted that these metal fatigue TLAAs are based on the criteria for performing implicit fatigue analyses, as given in the ANSI B31.1 design code (for Group D components) and in ASME Code, Section III (NC-3000 for Class 2 and ND-3000 for Class 3 components), which require an allowable stress range reduction only if the number of full thermal cycles exceeds the limit of 7,000. The staff also reviewed the projected number of occurrences for various plant transients for 60-years of operation, as given in LRA Table 4.3-1, as well as the applicant's estimates for other thermal cycles that do not require monitoring or counting.

The 60-year projections in LRA Table 4.3-1 are based on the number of cycles for each transient that have been accrued as of February 2008 and then linearly extrapolated to the 60 years of operation. As described in SER Section 4.3.1.2, the staff determined that the applicant conservatively considered operating history early in plant life, when transient occurrences were more frequent compared to current operating history, into the 60-year projection methodology. The staff noted that the applicant's Fatigue Monitoring Program counts transient cycles to ensure allowable cycle limits used in the design basis fatigue evaluations analysis are not exceeded during the period of extended operation.

The staff noted that the piping connected to the RCS, main steam system, auxiliary steam system, and main feedwater system experience the same transients that were used in the design of the RCS, as listed in LRA Table 4.3-1. The staff confirmed that there is significant margin between the total number of design transients projected to occur after 60 years of operation, as shown in LRA Table 4.3-1, and the full thermal range transient cycle limit of 7,000.

The staff noted that the transients that occur in the following piping and piping components are routine and based on predictable surveillance testing or periodic cycling. These piping and piping components include those associated with

- the emergency diesels
- the fire pump diesel engine
- the station blackout (SBO) diesel

- the containment air system
- the gaseous radwaste system
- the sampling systems
- the auxiliary steam system
- the station heating system
- piping that connects the fire water storage tank heat exchanger to the fire water storage tank

The staff also noted that the applicant conservatively assumed unanticipated operation and cycling that may occur for the systems described above. The staff noted that, in all instances, the number of estimated cycles in these systems for 60-years of operation were significantly less than the full thermal range transient cycle limit of 7,000. The staff finds it reasonable that the full thermal range transient cycle design limit of 7,000 will not be exceeded during the period of extended operation because of the significant margin between the estimated number of transient cycles and the design limit of 7,000 cycles. These transient occurrences for the piping and piping components in these systems are predictable and routine, and the applicant conservatively incorporated unanticipated events that would lead to additional transients.

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 non-Class 1 piping and in-line components will not be exceeded during the extended period of operation; therefore, the maximum allowable stress range values for the existing analyses remains valid. The staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the TLAAs of non-Class 1 piping and in-line components fatigue analyses remain valid during the period of extended operation. Additionally, the analyses meet 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 non-Class 1 piping and in-line components does not exceed the 7,000-cycle limit.

4.3.3.1.3 USAR Supplement

LRA Section A.2.3.3.1 provides the USAR supplement summarizing the TLAA for non-Class 1 piping and in-line components fatigue analyses. The staff reviewed LRA Section A.2.3.3.1 consistent with the review procedures in SRP-LR Section 4.3.3.3, which state that the reviewer verifies that the applicant provided information to be included in the USAR supplement that includes a summary description of the evaluation of the TLAA.

Based on its review of the USAR supplement, the staff finds that the supplement meets the acceptance criteria in SRP-LR Section 4.3.2.3. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the TLAA of non-Class 1 piping and in-line components fatigue analysis, as required by 10 CFR 54.21(d).

4.3.3.1.4 Conclusion

On the basis of 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 non-Class 1 piping and in-line components remain valid during the period of extended operation. The staff also concludes that the USAR supplement contains an adequate summary description of the evaluated TLAAs, as required by 10 CFR 54.21(d).

4.3.3.2 Non-Class 1 Major Components

4.3.3.2.1 Summary of Technical Information in the Application

LRA Section 4.3.3.2 describes the applicant's assessment of non-Class 1 major components subject to fatigue. The applicant stated that the need for fatigue evaluation of non-piping components, which includes heat exchangers, storage tanks, and pumps, is limited based on the design code used. A review conducted by the applicant of the component design codes determined that the applicable design codes include: ASME Code, Section III (Class C or Class 3); ASME Code, Section VIII; the draft ASME Code for Pumps and Valves 1968 (Class 2); Section VIII Division 1; AWWA; Manufacturers Standardization Society; and National Electrical Manufacturers Association. The applicant stated that none of these design codes require fatigue analyses. The applicant stated there are no fatigue analyses and, hence, no TLAAs associated with the non-Class 1 major (non-piping) components.

4.3.3.2.2 Staff Evaluation

The staff reviewed LRA Section 4.3.3.2 and the applicant's evaluation for these non-Class 1 major (non-piping) components to verify the applicant's basis for determining there are no fatigue analyses and, hence, no TLAAs. The staff reviewed the applicant's evaluation and conclusion consistent with the review procedures in SRP-LR Section 4.1.3, which state that the review verifies that the selected analyses do not meet at least one of the criteria of a TLAA, as defined in 10 CFR 54.3(a).

The staff reviewed USAR Table 3.2-2 to confirm the applicable design codes of record for the components that are discussed in LRA Section 4.3.3.2 and then reviewed the design code requirements for fatigue evaluation for these non-Class 1 major (non-piping) components. The staff confirmed that ASME Code, Section III, does not require a fatigue analysis for Class 2 and 3 tanks (less than 15 psig). The staff also confirmed that only pressure vessel and heat exchangers designed under ASME Code, Section VIII, Division 2, Alternative Rules and ASME Code, Section III, NC-3200, explicitly require fatigue analyses.

For the fire water storage tank heat exchanger, the borated water tank heater, and the 10-psig condensate tank designed under ASME Code, Section VIII, Division 1, the staff confirmed that this design code does not require a fatigue analysis. For decay heat removal pumps designed under 1968 draft ASME Code for Pumps and Valves Class 2, the staff confirmed that this design code does not require a fatigue analysis for these components. For the waste gas surge tank and the decay heat removal coolers designed under ASME Code, Section III, Class C, the staff confirmed that this design code does not require a fatigue analysis. For the pressurizer quench tank designed under ASME Code, Section III, Class 3, the staff confirmed that this design code does not require a fatigue analysis. For the AFW pump turbine casings, the intake structure unit heater heat exchangers, the evaporator package condensate drain pumps, the degasifier package drain pumps, and the condensate pumps, the staff noted that, based on each system's function and the nonsafety-related classification, these components would not experience transients that can cause substantial cyclic strains that are significant contributors to the fatigue usage. Therefore, the staff finds it reasonable that fatigue analyses were not required during the design of these components.

Based on its review, the staff finds the applicant's conclusion, that there are no specific TLAAs associated with non-Class 1 major (non-piping) components, acceptable because the applicant demonstrated that its CLB does not contain analyses that consider cumulative fatigue damage for non-Class 1 major (non-piping) components. Therefore, metal fatigue of these non-Class 1 major (non-piping) components is not a TLAA, in accordance with Criterion 6 of 10 CFR 54.3(a).

4.3.3.2.3 USAR Supplement

LRA Section A.2.3.3.2 provides the USAR supplement summarizing the absence of TLAAs for non-Class 1 major (non-piping) components. The staff reviewed LRA Section A.2.3.3.2 consistent with the review procedures in SRP-LR Section 4.3.3.3, which state that the reviewer should verify that the applicant provided information to be included in the USAR supplement that includes a summary description of the evaluation of absence of TLAAs for non-Class 1 major (non-piping) components.

Based on its review of the USAR supplement, the staff finds it meets the acceptance criteria in SRP-LR Section 4.3.2.3. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the absence of TLAAs for non-Class 1 major (non-piping) components, as required by 10 CFR 54.21(d).

4.3.3.2.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration that the non-Class 1 major (non-piping) components do not have fatigue analyses that would be identified as a TLAA, in accordance with the requirements 10 CFR 54.3(a). The staff also concludes that the USAR supplement contains an appropriate summary description of the evaluation of the absence of TLAA, as required by 10 CFR 54.21(d).

4.3.4 Effects of Reactor Coolant Environment on Fatigue

4.3.4.1 Summary of Technical Information in the Application

LRA Section 4.3.4 describes the applicant's evaluation of the effect of reactor coolant environment on component fatigue life for the period of extended operation. The applicant stated that industry data indicates that certain environmental effects such as temperature and dissolved oxygen content in the reactor coolant could result in greater susceptibility to metal fatigue than those predicted by fatigue analyses based on the ASME Code, Section III, fatigue design curves. The LRA states that the Code design curves were based on laboratory tests in air and at low temperatures, which may not be sufficient to account for actual plant operating environments. As described in the LRA, EAF is evaluated for license renewal in accordance with the guidelines of NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," and EPRI report MRP-47, "Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application." The applicant stated that NUREG/CR-6260 identifies locations of interest for consideration of environmental effects for B&W PWRs.

The applicant stated that plant-specific locations corresponding to the NUREG/CR-6260 locations were identified and the ASME Code design fatigue CUFs were adjusted by the environmental life correction factors (F_{en}) to obtain the EAF results for these locations. F_{en} values were calculated using material-specific guidance contained in the following documents:

- NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels," for carbon and low alloy steels

- NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," for austenitic stainless steels
- NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," for Ni-based alloys

The design CUFs, adjusted CUFs, and environmentally-adjusted CUFs (U_{en}) for 15 plant-specific locations are summarized in LRA Table 4.3-2. The applicant stated that the U_{en} for all locations are less than 1.0 except for the HPI/makeup nozzle safe end and the associated welds. The applicant also stated that it will replace all four HPI/makeup nozzle safe ends and the associated welds prior to entering the period of extended operation.

The LRA states that the effects of EAF will be managed for the period of extended operation by the Fatigue Monitoring Program. The Fatigue Monitoring Program will be used to manage the effects of reactor coolant environment for each of the NUREG/CR-6260 locations by counting the design transients that were based upon in the EAF analyses. The applicant dispositioned the EAF evaluations for all the 15 NUREG/CR-6260 locations in accordance with 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.

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 Order No. CLI-10-17, dated July 8, 2010, the evaluations associated with the effects of the reactor coolant environment on component fatigue life do not fall within the definition of a TLAA in 10 CFR 54.3(a) because these evaluations are not in the applicant's CLB. Nevertheless, the applicant credited its Fatigue Monitoring Program to manage the effects of EAF. Therefore, the staff reviewed LRA Section 4.3.4 and the evaluations for EAF to verify, 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 EAF evaluations and the corresponding disposition consistent with the review procedures in SRP-LR, Revision 2, Section 4.3.3.1.3, which state that the reviewer should verify that the applicant has addressed the effects of the coolant environment on component fatigue life as AMPs are formulated in support of license renewal. If the applicant has chosen to assess the impact of the reactor coolant environment on a sample of critical components, the review verifies the following:

- The critical components include a sample of high-fatigue usage locations. This sample is to include the locations identified in NUREG/CR-6260, as a minimum, and propose alternatives based on plant configuration.
- The sample of critical components has been evaluated by applying environmental correction factors to the existing ASME Code fatigue analyses or using the methodology provided in NUREG/CR-6909. If the Class 1 component was designed to a Code not requiring CUF, a new environmental CUF calculation has been performed or addressed in an appropriate license renewal commitment.
- Formulae for calculating the F_{en} are those contained in several NUREG/CR reports as specified in SRP-LR, Revision 2, Section 4.3.3.1.3, or an approved technical equivalent.

LRA Section 4.3.4 discusses the methodology to determine the locations that require an EAF evaluation consistent with NUREG/CR-6260. The staff noted that LRA Table 4.3-2 contains 15 plant-specific locations, which are based on the six generic components identified in NUREG/CR-6260.

SRP-LR, Revision 2, Section 4.3.3.1.3 states that the impact of the reactor coolant environment on a sample of critical components should include the locations identified in NUREG/CR-6260, as a minimum, and that additional locations may be needed. It was not clear to the staff whether the applicant confirmed that the plant-specific locations listed in LRA Table 4.3-2 were bounding for the generic NUREG/CR-6260 components. Furthermore, the staff noted that the applicant's plant-specific configuration may contain locations that should be analyzed for the effects of the reactor coolant environment other than those identified in NUREG/CR-6260. By letter dated May 2, 2011, the staff issued RAI 4.3-14, asking the applicant to confirm and justify that the plant-specific locations listed in LRA Table 4.3-2 are bounding for the generic NUREG/CR-6260 components. Furthermore, the staff requested that the applicant confirm and justify that the locations listed in LRA Table 4.3-2 that were selected for EAF analyses consists of the most limiting locations for the plant (beyond the generic locations identified in the NUREG/CR-6260). If these locations in LRA Table 4.3-2 are not bounding for the plant, the staff asked the applicant to clarify the locations that require an EAF analysis and explain the actions that will be taken for these additional locations.

In its response dated June 17, 2011, the applicant stated that a response was previously provided in letter dated June 3, 2011, in response to RAI B.2.16-2, to address the issue of EAF for locations beyond the generic locations identified in the NUREG/CR-6260. The applicant also stated that it provided the required changes to LRA Sections A.1.16 and B.2.16 and LRA Table A-1. In its June 3, 2011, letter, the applicant compiled a listing of all its design CUFs, which were then multiplied by a maximum F_{en} value to determine the bounding EAF CUFs (CUF_{en}). The applicant provided the following bounding F_{en} values for a PWR reactor coolant environment and the associated NUREG/CR report that was used for each value:

- low alloy steel—F_{en} max of 2.45 (NUREG/CR-6583) [Note that the June 3, 2011, letter states a value of 2.54, which the applicant identified as a typographical error and corrected to 2.45 by letter dated Mary 25, 2011)
- carbon steel—F_{en} max of 1.74 (NUREG/CR-6583)
- stainless steel—F_{en} max of 15.35 (NUREG/CR-5704)
- Ni-based alloy—F_{en} max of 4.52 (NUREG/CR-6909)

As a result, the applicant provided a list of additional locations not evaluated in the LRA for EAF for which the bounding estimates of CUF_{en} exceeded the design limit of 1.0. The staff noted that the applicant provided an enhancement to the "scope of program" program element for its Fatigue Monitoring Program and committed (Commitment No. 42) to enhance this program to evaluate additional plant-specific component locations in the RCPB that may be more limiting than those considered in NUREG/CR-6260 for environmental effects. The staff's evaluation of the Fatigue Monitoring Program, RAI B.2.16-2, and the bounding F_{en} values listed above, is documented in SER Section 3.0.3.2.6.

The staff finds the applicant's response acceptable because the applicant committed to evaluate additional plant-specific component locations in the RCPB that may be more limiting than those considered in NUREG/CR-6260 for environmental effects as part of its Fatigue Monitoring Program, which is consistent with SRP-LR, Revision 2, Section 4.3.2.1.3 and GALL Report AMP X.M1. The staff's concern described in RAI 4.3-14 is resolved.

In LRA Table 4.3-2 for the RV inlet and outlet nozzles and the pressurizer surge nozzle safe-end, adjusted values of CUF were determined by identifying the incremental fatigue contributions attributed to the full NSSS design transient cycles for design CUF and reducing those incremental contribution based on the 60-year projected cycles. Specific to the HPI/makeup nozzle and stainless steel safe-end, the applicant stated that it still maintained the full-set of 40-year nuclear steam supply system (NSSS) design transients while conservatism in the design analyses was removed. It is not clear to the staff which incremental contributions were reduced based on the 60-year projected cycles and which transients and the associated numbers of cycles were used in the analysis for the RV inlet and outlet nozzles and the pressurizer surge nozzle safe-end. For the HPI/makeup nozzle and stainless steel safe-end, it is not clear to the staff which elements in the original design basis fatigue evaluations were adjusted to remove conservatism in the original design CUF, and the basis for these adjustments. By letter dated May 2, 2011, the staff issued RAI 4.3-16 asking the applicant to identify the changes that were made to reduce the conservatism and justify the reduction of conservatism in the original CUFs of record for each location.

In its response dated June 17, 2011, the applicant stated that for the RV inlet and outlet nozzles, the CUF was reduced by using the current design cycles for transients 3 and 4 and 60-year projections for transients 5 and 6 in LRA Table 4.3-1. The applicant stated that large contributions to fatigue were due to transients 3–6. Specifically, the original analyses used 48,000 cycles for transients 3 and 4, and 8,000 cycles for transients 5 and 6. The staff noted that the use of 48,000 cycles was conservative since the current design cycles for transients 3 and 4 are 1,800; the staff finds it reasonable that the evaluation uses the number of current design cycles. The applicant stated that the CUF reduction was obtained by using the current design of 1,800 cycles for transients 3 and 4 and the projected cycles for 60-year for transients 5 and 6, which are 67 and 140, respectively. The staff's review of the applicant's 60-year projection methodology is documented in SER Section 4.3.1.2. Therefore, the design CUF for the RV inlet nozzle was reduced from 0.829 to 0.146, and the design CUF for the RV outlet nozzle was reduced from 0.768 to 0.335 for the EAF evaluations reported in LRA Table 4.3-2. The staff noted that the CUF contribution for the remaining NSSS design transients listed in LRA Table 4.3-1 was unchanged.

Based on its review, the staff finds the conservatism removed by the applicant for the RV inlet and outlet nozzles EAF evaluations acceptable for the following reasons:

- The applicant used, for transients 3 and 4, the number of cycles from its current design.
- The applicant used, for transients 5 and 6, the 60-year projected cycles that were based on actual operating history of the plant.
- The applicant's Fatigue Monitoring Program will ensure that this evaluation remains valid and the Design Code limit of 1.0, including environmental effects, will not be exceeded during the period of extended operation.

The applicant stated that the original design CUF for the pressurizer surge nozzle safe end consists of 0.108 from HU/CD transients (the current design is 240 cycles for transients 1A and 1B) and all other NSSS design transients contribute a negligible amount. The applicant used the 60-year projections of 128 HU/CD transients to reduce the design CUF from 0.108 to 0.0581 for the EAF evaluation reported in LRA Table 4.3-2.

Based on its review, the staff finds the conservatism removed by the applicant for the pressurizer surge nozzle safe end EAF evaluation acceptable because, for transients 1A and

1B, the applicant used 60-year projected cycles for these transients in the calculations of CUF_{en}, which is based on the actual operating history of the plant. Use of data from the applicant's actual operating history provides a more realistic accumulated fatigue usage through the period of extended operation. Additionally, the applicant's Fatigue Monitoring Program will ensure that the evaluation remains valid and the Design Code limit of 1.0 is not exceeded during the period of extended operation.

The applicant stated that the original design CUF for the carbon steel HPI nozzle is 0.589, and the major contributions to this CUF is from transient 12 (hydro-test) and transient 23 (SG filing, draining, flushing and cleaning). The applicant also stated that the stress for transient 23, was conservatively calculated in the original analysis based on the same pressure as the hydro-test (i.e., 3125 psig) plus other stresses from thermal moments and mechanical loads. However, in accordance with its RCS functional specification, the pressure range permitted during transient 23 is only 485 psig. The staff finds it reasonable that the applicant reduced the stress due to pressure from transient 23 in its EAF evaluation because this reduction is consistent with the actual pressure permitted by the applicant's RCS functional specification during the transient. The applicant also stated that the stresses due to thermal moments and mechanical loads for transient 23 and the usage factor contributions from the other NSSS transients were not changed. The design usage factor for the carbon steel nozzle was reduced from 0.589 to 0.348 for the EAF evaluation reported in LRA Table 4.3-2.

Based on its review, the staff finds the conservatism removed by the applicant for the carbon steel HPI nozzle EAF evaluation acceptable because the evaluation considered the stresses from the actual pressure on the component that occurs during the transient, as defined by the applicant's RCS functional specifications. Additionally, the applicant's Fatigue Monitoring Program will ensure that this evaluation remains valid and the design Code limit of 1.0 will not be exceeded during the period of extended operation.

The applicant stated that the design CUF for the stainless steel HPI/makeup nozzle safe end was reduced from 0.664 to 0.550 for the EAF evaluation reported in LRA Table 4.3-2. The applicant clarified that transient 22 (now transient 22 A1), as defined in the RCS functional specification, cannot occur at the applicant's site because the HPI pump shutoff head is approximately 1,600 psig. The staff's evaluation of RAI 4.3-2, as documented in SER Section 4.3.1.2, discusses why this transient is not applicable to the applicant, why it does not need to be monitored by the Fatigue Monitoring Program, and why it does not contribute to fatigue usage of the HPI nozzles. The applicant stated that fatigue usage due to transient 22 was eliminated, and the usage factor contributions from the other NSSS transients were not changed for the EAF calculations.

Based on its review and the evaluation of RAI 4.3-2 in SER Section 4.3.1.2, the staff finds the conservatism removed by the applicant for the stainless steel HPI nozzle safe end EAF evaluation acceptable because the fatigue usage contribution from a test transient, which cannot occur at the applicant's site due to its plant-specific operating parameters, was removed from the evaluation. Additionally, the applicant's Fatigue Monitoring Program will ensure that this evaluation remains valid and the design Code limit of 1.0 is not exceeded during the period of extended operation.

The applicant stated that the RV inlet nozzle, RV outlet nozzle, pressurizer surge nozzle safe end, and HPI/makeup nozzle safe end are the only locations where selected 60-year transient projections were used to reduce the CUFs.

The staff finds the applicant's response acceptable because, for each EAF evaluation in which incremental fatigue contribution was reduced, the applicant provided an adequate justification for the conservatism that was removed, as described above. Additionally, the applicant confirmed that no CUF value of other components was reduced based on selected 60-year transient projections, and the Fatigue Monitoring Program tracks allowable cycles to ensure that these EAF evaluations remain valid during the period of extended operation. The staff's concern described in RAI 4.3-16 is resolved.

LRA Section 4.3.4.2 states that the surge line piping and HPI/makeup nozzle and safe end were evaluated using an integrated F_{en} approach consistent with EPRI Technical Report MRP-47; however, EPRI Technical Report MRP-47 has not been reviewed and approved by the NRC. The staff noted that, in Technical Report MRP-47, Section 4.2, the CUF and U_{en} are computed for each load pair, and an effective F_{en} is calculated by dividing the U_{en} by the CUF. LRA Section 4.3.4 states that the maximum U_{en} is calculated with a global F_{en} and the adjusted CUF is then obtained by dividing the U_{en} by the global F_{en}. Furthermore, Footnote 2 of LRA Table 4.3-2 states that the global F_{en} is calculated using the method from Section 4.2 of Technical Report MRP-47. However, the staff noted that the term "global F_{en}" was not discussed in Technical Report MRP-47, and the process of calculating the global F_{en} was not addressed in the LRA. Therefore, it is not clear how the applicant determined the U_{en} for the surge line piping and the HPI/makeup nozzle and safe end. By letter dated May 2, 2011, the staff issued RAI 4.3-17 asking the applicant to justify that the use of the integrated F_{en} approach in Technical Report MRP-47 is applicable and adequately conservative to calculate U_{en} for the period of extended operation. The staff also asked the applicant to clarify how the "global F_{en}" is calculated for each component and provide its relationship with the U_{en} calculation methodology discussed in TR MRP-47.

In its response to RAI 4.3-17 dated June 17, 2011, the applicant stated the EAF evaluation of the stainless steel surge line involved calculation of a separate F_{en} multiplier for each transient pair in the analysis. A value for the F_{en} value was computed using the worst-case strain rate of less than 0.0004 percent per second, dissolved oxygen content of less than 0.05 parts per million (ppm), and the appropriate temperature associated with each transient. The applicant stated that the 60-year projected numbers of NSSS design cycles in LRA Table 4.3-1 were used (except for the best estimate 60-year project cycles of 114 for HU/CD events), and the transient pairings were performed in accordance with ASME Code, Section III, rules. The applicant explained that the U_{en} for each transient pair is determined by multiplying the in-air CUF for that pair by the F_{en} calculated for that pair. The cumulative U_{en} for that specific location is obtained by adding up the U_{en} contribution for all transient pairs and the "global F_{en}" is then calculated by dividing the cumulative U_{en} by the total in-air CUF.

The staff determined that the dissolved oxygen concentration assumption is conservative for the F_{en} formulation of stainless steel materials because assuming a higher dissolved oxygen content results in a lower and less conservative F_{en} value. The staff determined the applicant's strain-rate assumption is conservative for the F_{en} formulation of stainless steel materials because assuming a higher strain rate results in a lower and less conservative F_{en} value.

The staff held a teleconference with the applicant on July 12, 2011, to discuss LRA Section 4.3.4.2 and Footnote 2 of LRA Table 4.3-2, which states that the adjusted CUF is obtained by dividing the U_{en} by the global F_{en}. By letter dated August 17, 2011, the applicant revised LRA Table 4.3-2 and LRA Section 4.3.4.2. Specifically, Footnote 2 of LRA Table 4.3-2 and LRA Section 4.3.4.2 was revised to state that, for the pressurizer surge line, the adjusted CUF was calculated using 60-year projected cycles (except for best-estimate 60-year project cycles of 114 used for HU/CD events). In addition, Footnote 9 was added to LRA Table 4.3-2 to describe the methods used by the applicant to determine the F_{en}, the U_{en}, and global F_{en}. The staff's review of the applicant's response to RAI 4.3-19 discusses the use of the best-estimate 60-year projected cycles of 114 used for HU/CD events for the EAF evaluation of the pressurizer surge line, which is documented below in SER Section 4.3.4.2. The staff's review of the applicant's assumptions in determining the F_{en} value is documented previously in SER Section 4.3.4.2.

In its response to RAI 4.3-17 dated June 17, 2011, the applicant also stated that the EAF evaluation of the stainless steel HPI/makeup nozzle safe end involved calculating the F_{en} value with an integrated approach at different temperatures and an overall F_{en} was obtained over the entire temperature range considered. The integrated F_{en} value was determined for each transient event that is applicable to the HPI nozzle safe end and was then applied to the incremental CUF associated with each transient. Since the U_{en} for this component was 4.417, the applicant committed in the LRA (Commitment No. 23) to replace all four HPI/makeup nozzle safe ends prior to the period of extended operation.

The staff noted that the integrated F_{en} approach, which computes F_{en} value over the entire range of temperature, gives a more refined F_{en} value to account for the environmental effects of reactor coolant on component fatigue life. The staff noted that the surge line piping and HPI/makeup nozzle EAF evaluations were dispositioned in accordance with 10 CFR 54.21(c)(1)(iii), where the effects of EAF will be managed by the Fatigue Monitoring Program. The Fatigue Monitoring Program, as amended by letter dated June 3, 2011, states that it prevents the EAF evaluations from becoming invalid by assuring that the fatigue usage resulting from actual operational transients does not exceed the Code design limit of 1.0, including environmental effects where applicable. The staff's evaluation of the Fatigue Monitoring Program is documented in SER Section 3.0.3.2.6.

The staff finds the applicant's response, as amended by letter dated August 17, 2011, acceptable for the following reasons:

- The applicant conservatively calculated the F_{en} values, as described above, in the surge line piping EAF evaluation.
- The applicant performed a rigorous integrated F_{en} calculation to obtain more refined F_{en} values for the transients used in the HPI/makeup nozzle EAF evaluation.
- The applicant's EAF evaluations for the surge line piping and HPI/makeup nozzle have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii) and managed with the Fatigue Monitoring Program, which counts transient cycles to ensure that allowable cycle limits used in these EAF evaluations and the Code design limit of 1.0 are not exceeded.

The staff's concern described in RAI 4.3-17 is resolved.

The applicant committed (Commitment No. 23) to evaluate the environmental effects on the replacement HPI nozzle safe ends and associated welds in accordance with NUREG/CR-6260 and the guidance of EPRI TR MRP-47, Revision 1. The staff noted that EPRI Technical Report MRP-47 has not been reviewed and approved by the NRC, and the applicant does not specify which portions of MRP-47 will be used in this evaluation of the replacement HPI nozzle safe ends and associated welds. The staff noted that the applicant claims that its Fatigue Monitoring Program, with enhancements, is consistent with GALL Report AMPX.M1 and that it addresses the effects of the reactor coolant environment on component fatigue life. By letter dated May 2, 2011, the staff issued RAI 4.3-18 asking the applicant to justify the use of EPRI Technical Report MRP-47 to evaluate the environmental effects on the replacement HPI nozzle safe ends and associated welds in lieu of managing cumulative fatigue damage as part of the Fatigue Monitoring Program, which is consistent with the recommendations of the GALL Report AMP X.M1.

In its response dated June 17, 2011, the applicant stated that it used Section 4.2 of Technical Report MRP-47 in the EAF calculations because there is no specific NRC guidance provided for the application of F_{en} factors, reported in NUREG/CR-5704, to an ASME Code fatigue evaluation. The applicant committed (Commitment No. 23), as amended by letter dated October 31, 2011, to replace the HPI nozzle safe end including the associated Alloy 82/182 weld for all four HPI nozzles prior to the period of extended operation. The applicant also amended Commitment No. 23 to credit the Fatigue Monitoring Program to evaluate the environmental effects and manage cumulative fatigue damage for the replacement HPI nozzle safe ends and associated welds. The Fatigue Monitoring Program, as amended by letter dated June 3, 2011, states that it prevents the fatigue TLAAs (i.e., EAF evaluations) from becoming invalid by assuring that the fatigue usage resulting from actual operational transients does not exceed the Code design limit of 1.0, including environmental effects where applicable. By letter dated October 31, 2011, the applicant stated that the Inservice Inspection Program has been augmented to include examination of the HPI/makeup nozzle thermal sleeves. In addition, the thermal sleeve for this nozzle has since been replaced during the Cycle 13 RFO that ended in March 2004 and the Inservice Inspection Program was revised to require an augmented VT-1 visual examination of the makeup nozzle thermal sleeve once every other RFO commencing with the Cycle 15 RFO. The staff's evaluation of the Fatigue Monitoring Program and Inservice Inspection Program are documented in SER Sections 3.0.3.2.6 and 3.0.3.1.12, respectively.

The staff finds the applicant's response acceptable because the applicant committed to replace the HPI nozzle safe ends prior to the period of extended operation. Additionally, the applicant credited its Fatigue Monitoring Program to manage cumulative fatigue damage on the replacement HPI nozzle safe ends and associated welds, including environmental effects, by ensuring the evaluation remains valid and Code design limit of 1.0 is not exceeded during the period of extended operation, which is consistent with the recommendations of GALL Report AMP X.M1. In addition, the staff finds acceptable the applicant plans to inspect the HPI/makeup nozzle thermal sleeves as part of its Inservice Inspection Program. The staff's concern described in RAI 4.3-18 is resolved.

LRA Section 4.3.4.2 states that the EAF evaluation for the stainless steel surge line piping used 60-year projected cycle with the exception of the 60-year projection of HU/CD in which a best estimate number of 114 cycles were used. The staff noted that LRA Table 4.3-1 states that the 60-year projected cycles for HUs/CDs are 128 cycles, which is based on the linear extrapolation method described in the LRA Section 4.3.1.2. The applicant committed (Commitment No. 9) to monitor any transient where the 60-year projected cycles were used in an EAF evaluation and establish an administrative limit that is equal to or less than the 60-year projected cycles. In this particular evaluation, for the stainless steel surge line piping, the analyzed number of cycles for the HU/CD transients is less than the 60-year projected cycle. By letter dated May 2, 2011, the staff issued RAI 4.3-19 requesting the applicant provide the basis for using 114 HU/CD transient cycles in the EAF evaluation and justify that the Fatigue Monitoring Program and Commitment No. 9 will ensure that corrective actions will be taken prior to the HU/CD transients exceeding the analyzed number of cycles of 114. The staff also asked the applicant to clarify whether there are any additional locations in which the analyzed transient cycles are less than the 60-year projected cycles listed in LRA Table 4.3-1.

In its response dated June 17, 2011, the applicant stated that it was not able to demonstrate in its EAF evaluation that the surge line piping was acceptable for 60 years of operation when it used the 60-year projected HU/CD cycles (128 cycles for each transient), which are provided in LRA Table 4.3-1. The applicant stated that, alternatively, it used the best-estimate 60-year projected cycles for the HU/CD cycles, which is based on more recent operating experience compared to the entire operation history of the plant. This resulted in a best-estimate 60-year projected cycles of 114 cycles for the HU/CD transients. The applicant clarified that all of its EAF evaluations used the 60-year projected cycles reported in LRA Table 4.3-1 with the exception of the surge line piping EAF evaluations that used the best estimate 60-year projected HU/CD cycles. The staff noted that the surge line piping EAF evaluation was dispositioned in accordance with 10 CFR 54.21(c)(1)(iii), where the effects of reactor coolant environment on component fatigue life will be managed by the Fatigue Monitoring Program. The Fatigue Monitoring Program, as amended by letter dated June 3, 2011, states that it prevents the fatigue TLAAs from becoming invalid by assuring that the fatigue usage resulting from actual operational transients does not exceed the Code design limit of 1.0, including environmental effects where applicable. The staff's evaluation of the Fatigue Monitoring Program is documented in SER Section 3.0.3.2.6.

The staff finds the applicant's response acceptable because the applicant's EAF evaluations, including the surge line piping, have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii), such that the effects of cumulative fatigue damage when considering reactor water environment will be managed by the Fatigue Monitoring Program. This program counts transient cycles to ensure that the allowable cycle limits (e.g., 128 cycles or 114 cycles) used in the EAF evaluations and the Code design limit of 1.0 are not exceeded during the period of extended operation. The staff's concern described in RAI 4.3-19 is resolved.

In its review of LRA Section 4.3.4.2, the staff noted that F_{en} values were determined using guidance contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-6909 for Ni-based alloys. The applicant stated that the calculated bounding values for F_{en} are 1.74 for carbon steel, 2.45 for low-alloy steel, and 4.16 for the Ni alloy incore instrument nozzles. Based on the guidance in NUREG/CR-6583 and NUREG/CR-6909, the F_{en} value can vary based on sulfur content, temperature, dissolved oxygen content, and strain rate. The staff noted that for Ni-based alloy components, when using the guidance in NUREG/CR-6909, the F_{en} value can be as high as 4.52. For carbon and low-alloy steel components, when using the guidance in NUREG/CR-6583, the F_{en} value can vary significantly depending on the plant's history for dissolved oxygen content in the reactor coolant. It is not clear to the staff what assumptions were used by the applicant when determining the bounding F_{en} values for the carbon and low-alloy steel and Ni-based alloy components described in LRA Section 4.3.4.2 and LRA Table 4.3-2.

By letter dated May 2, 2011, the staff issued RAI 4.3-21 asking the applicant to clarify how the bounding F_{en} values for carbon steel, low-alloy steel, and Ni-based alloy components were determined and to justify any assumptions used. Furthermore, specifically for carbon and low-alloy steel components, the applicant was requested to confirm that dissolved oxygen content remained less than 0.05 ppm since initial plant operation and justify that the dissolved oxygen content will remain less than 0.05 ppm during the period of extended operation.

In its response dated June 17, 2011, the applicant stated that the bounding F_{en} value for carbon steel and low-alloy steel were calculated from NUREG/CR-6583, equations 6.5a and 6.5b, respectively. In addition, the applicant stated that, in a PWR environment, the dissolved oxygen level is less than 0.05 ppm at RCS temperatures greater than 302 °F (150 °C); therefore, at this RCS temperature, the transformed dissolved oxygen is 0.0, and the bounding F_{en} values for carbon steel and low alloy steel are 1.74 and 2.45, respectively. The staff reviewed NUREG/CR-6583 and confirmed that, at RCS temperatures below 150 °C, the transformed metal service temperature in equations 6.5a and 6.5b is 0.0, and the bounding F_{en} values for carbon steel and low alloy steel are 1.74 and 2.45, respectively.

The applicant stated that NUREG/CR-6909 was used to determine the F_{en} value for the Ni-based alloy incore instrument nozzles. This evaluation assumed that the temperature of the incore instrument nozzles is 582 °F (305.6 °C), which corresponds to the average RCS temperature at 15 percent power for steady-state operations, as defined in the applicant's RCS functional specification. The applicant stated that, at 15 percent power, the RV inlet and outlet temperature are approximately 577 °F and 586 °F, respectively, with an average temperature of 582 °F. As power increases, the reactor inlet temperature decreases to 556.5 °F at full power. The staff noted that, during normal operation at full power, the average temperature between the RV inlet and outlet will be less than the assumed average temperature of 582 °F for the incore instrument nozzles in the EAF evaluation. The staff finds this assumption of 582 °F for the average temperature acceptable because it is conservative to assume a higher temperature, as an input to determine the transformed temperature value, when calculating the F_{en} value for Ni-based alloy components. Consistent with NUREG/CR-6909, the applicant stated that it used a conservative transformed temperature value, as described above, the bounding strain rate of 0.0004 percent per second and transformed dissolved oxygen content of 0.16 for PWR water.

The applicant confirmed that dissolved oxygen in the RCS has been historically less than 0.05 ppm with RCS temperatures greater than 302 °F (150 °C), and the only exceptions are short periods of time during selected heat-ups when the pressurizer temperature was elevated to approximately 425 °F, with the RCS temperature at approximately 100 °F. The applicant further explained the circumstances in which the dissolved oxygen content exceeds 0.05 ppm and determined that, in order to meet the dissolved oxygen requirements, a method of adding hydrazine directly to the pressurizer was needed. The applicant described the method that it would take to meet the meet dissolved oxygen content requirements and stated that the method was successfully employed during pressurizer heatup following February 2010 cycle 16 RFO.

The applicant stated that through the use of hydrazine addition, the sampling frequency and dissolved oxygen limits specified in its PWR Water Chemistry Program, it provides assurance that reactor coolant dissolved oxygen levels will continue to be maintained below 0.05 ppm at temperatures above 250 °F for the period of extended operation. The staff finds that the short periods of time when the dissolved oxygen levels exceeded 0.05 ppm do not have a significant impact to the overall F_{en} value because the duration of time that the plant operated in this manner is negligible when compared to the total operating time after 60 years with dissolved oxygen less than 0.05 ppm, and the resultant increase in F_{en} value is also negligible. The staff noted that dissolved oxygen content at temperatures less than 302 °F (150 °C) does not have an impact on the F_{en} value, based on the guidance provided in NUREG/CR-6583. This is only applicable for carbon and low-alloy steel components because the assumption of dissolved oxygen less than 0.05 ppm is conservative and bounding for calculating the F_{en} value for stainless steel components.

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 carbon steel, low-alloy steel, and Ni-based alloy components, which the staff confirmed were 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 less than 0.05 ppm during the period of extended operation.

The staff's concern described in RAI 4.3-21 is resolved.

Based on its review, the staff finds the applicant demonstrated, 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. Additionally, the applicant has met the acceptance criteria in SRP-LR, Revision 2, Section 4.3.2.1.3 because it has demonstrated that the impact of the reactor coolant environment on critical components has been adequately addressed and will be managed by the Fatigue Monitoring Program, such that the applicant's EAF evaluations will remain valid. Additionally, the Design 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 USAR Supplement

LRA Section A.2.3.4.2 provides the USAR supplement summarizing the effects of the reactor coolant environment on fatigue life of piping and components. The staff reviewed LRA Section A.2.3.4.2 consistent with the review procedures in SRP-LR Section 4.3.3.3, which state that the review verifies that the applicant has provided information, to be included in the USAR supplement that includes a summary description of the evaluation of the effects of reactor coolant environment on fatigue life. The SRP-LR also states that the reviewer should verify that the applicant identified and committed in the LRA to any future aging management activities, including enhancements and commitments to be completed before the period of extended operation.

The staff noted that, based on the discussions regarding the staff's concern in RAI 4.3-18, the applicant revised Commitment No. 23 in its letter dated June 17, 2011, as amended by letter dated October 31, 2011, to state the following:

> In association with the TLAA for effects of environmentally assisted fatigue of the high pressure injection (HPI) nozzle safe end including the associated Alloy 82/182 weld (weld that connects the safe end to the nozzle), replace the HPI nozzle safe end including the associated Alloy 82/182 weld for all four HPI nozzles prior to the period of extended operation. The Fatigue Monitoring Program will evaluate the environmental effects and manage cumulative fatigue damage for the replacement high pressure injection (HPI) nozzle safe ends and associated welds.

The staff's evaluation of RAI 4.3-18 is documented in SER Section 4.3.4.2.

Based on its review of the USAR supplement, the staff finds that the USAR supplement meets the acceptance criteria in SRP-LR Section 4.3.2.3. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the effects of reactor coolant environment on component fatigue life, as required by 10 CFR 54.21(d).

4.3.4.4 Conclusion

On the basis of its review, the staff concludes that 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), and consistent with Commission Order No. CLI-10-17. The staff also concludes that the applicant has 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 USAR supplement contains an appropriate summary description of the evaluation, as required by 10 CFR 54.21(d).

4.4 <u>Environmental Qualification of Electrical Equipment</u>

4.4.1 Summary of Technical Information in the Application

LRA Section 4.4 describes the applicant's TLAA for the evaluation of environmentally qualified electrical equipment for the period of extended operation. The applicant stated that environmentally qualified components not qualified for the current license term are to be refurbished, replaced, or have their qualification extended prior to reaching the limits established in the evaluation. The applicant also stated that equipment qualification evaluations for environmentally qualified components that specify a qualification of at least 40 years are considered TLAAs for license renewal.

The applicant dispositioned the EQ of Electrical Components Program TLAA in accordance with 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.

4.4.2 Staff Evaluation

The staff reviewed LRA Section 4.4 on the TLAA associated with the EQ of Electrical Components Program to verify, 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's review of the EQ of Electrical Components Program is documented in SER Section 3.0.3.1.7.

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. The TLAA of the EQ of electrical components 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 environments of the plant are those areas subject to environmental effects by a LOCA, HELB, or post-LOCA environment. EQ equipment comprises safety-related and nonsafety-related equipment, the failure of which could prevent satisfactory accomplishment of any safety-related function, and necessary post-accident monitoring equipment.

As required by 10 CFR 54.21(c)(1), the applicant must provide a list of EQ electrical equipment. The applicant shall demonstrate one of the following for each type of EQ equipment:

- The analyses remain valid for the period of extended operation.

- The analyses have been projected to the end of the period of extended operation.
- The effects of aging on the intended function(s) 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, pursuant to 10 CFR 54.21(c)(1)(iii), an applicant must demonstrate that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

The staff reviewed LRA Sections 4.4 and B.2.14, plant basis documents, additional information provided to the staff, and interviewed plant personnel to verify whether the applicant provided adequate information to meet the requirement of 10 CFR 54.21(c)(1). For 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. Per the GALL Report, plant EQ Programs that implement the requirements of 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 of 10 CFR 54.21(c)(1)(iii). 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.

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.2.14 and the program basis documents. LRA Section B.2.14 discusses the component reanalysis attributes, including analytical models, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions. On the basis of its audit, the staff finds that the EQ Program, which the applicant claimed to be consistent with GALL Report AMP X.E1, "Environment Qualification (EQ) of Electric Components," is consistent with the GALL Report, as described in SER Section 3.0.3.1.7. Therefore, the staff concludes that the applicant's EQ of Electric Equipment TLAA is implemented per the requirements of 10 CFR 54.21(c)(1)(iii).

The staff finds that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions of the EQ of electric equipment TLAA will be adequately managed for the period of extended operation. Additionally, the applicant's disposition of this TLAA meets the acceptance criteria in SRP-LR Section 4.4.2.1.3 because the applicant's EQ Program is 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 USAR Supplement

LRA Section A.2.4 provides the USAR supplement summarizing the EQ of Electrical Equipment TLAA. The staff reviewed LRA Section A.2.4 consistent with the review procedures in SRP-LR Section 4.4.3.3, which state that the detailed information on the evaluation of TLAAs is contained in the renewal application. A summary description of the evaluation of TLAAs for the period of extended operation is contained in the applicant's USAR supplement.

Based on its review of the USAR supplement, the staff finds it meets the acceptance criteria in SRP-LR Section 4.4.2.3. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the TLAA for the period of extended operation, as required by 10 CFR 54.21(d).

4.4.4 Conclusion

On the basis of 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 EQ of electrical equipment will be adequately managed for the period of extended operation. The staff also concludes that the USAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.5 Concrete Containment Tendon Prestress

4.5.1 Summary of Technical Information in the Application

LRA Section 4.5 and LRA Tables 4.1-1 and 4.1-2 state that the concrete containment tendon pre-stress analysis is not a TLAA. The LRA states that the containment is a cylindrical steel pressure vessel with hemispherical dome and ellipsoidal bottom, which is completely enclosed by a reinforced concrete shield building, and an annular space is provided between the wall of the containment vessel and the wall of the shield building. The applicant stated that the TLAA for tendon prestress is not applicable because the plant has a free-standing metal containment.

4.5.2 Staff Evaluation

The staff noted that SRP-LR Table 4.1-2 states that the concrete containment tendon analyses may be a generic TLAA applicable under an applicant’s CLB. The staff reviewed relevant containment design information in the USAR to evaluate the validity of the applicant’s basis. The staff noted that USAR Section 3.8.2 identifies the containment as a cylindrical steel pressure vessel that is enclosed by a reinforced concrete shield building, with annular space between the two buildings. The staff also noted that USAR Section 1.2.10 describes the containment vessel as a free-standing steel structure housed within a concrete shield building that has no structural ties with the containment vessel, other than the concrete foundation slab upon which the containment vessel is built. Thus, the staff confirmed that the containment does not use a pre-stressed concrete design, and does not use prestressed or preloaded tendons as the basis for structural reinforcement.

Based on its review, the staff confirmed that the design of the containment structure is not reinforced with pre-stressed tendons; therefore, the staff finds that this TLAA is not required. The staff also references its evaluation in SER Section 3.5.2.2.1.5, “Loss of Prestress due to Relaxation, Shrinkage, Creep and Elevated Temperature.”

4.5.3 USAR Supplement

The staff concludes that no USAR supplement is required because the applicant has no pre-stressed tendons in its containment.

4.5.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that this TLAA is not required.

4.6 Containment Fatigue Analyses

LRA Section 4.6 provides the assessment of containment fatigue as a TLAA for Davis-Besse license renewal. The applicant's assessment is documented in the following major subsections of LRA Section 4.6:

- LRA Section 4.6.1—Containment Vessel
- LRA Section 4.6.2—Containment Penetrations
- LRA Section 4.6.3—Permanent Canal Seal Plate

4.6.1 Containment Vessel

4.6.1.1 Summary of Technical Information in the Application

LRA Section 4.6.1 describes the applicant's evaluation of the TLAA for the containment vessel. The applicant stated that the containment vessel is a cylindrical steel pressure vessel with hemispherical dome and ellipsoidal bottom, which houses the RV, reactor coolant piping, pressurizer, pressurizer quench tank and coolers, RCPs, SGs, core flooding tanks, letdown coolers, and normal ventilating system. The LRA states that the containment vessel is a Class B vessel, as defined in the ASME Code, Section III, paragraph N-132, 1968 edition through summer 1969 addenda.

The LRA states that the containment vessel meets the requirements of ASME Code, Section III, paragraph N-415.1, thereby justifying the exclusion of cyclic or fatigue analyses in the design of the containment vessel. The LRA also stated that analysis of 400 pressure cycles (from -25 to 120 psi) and 400 temperature cycles (from 30 °F to 120 °F) were performed against the requirements of ASME Code, Section III, paragraph N-415.1. The LRA further states that, to date, the containment vessel has not seen any pressure cycles from -25 to 120 psi. The LRA further states that the values of 400 pressure and temperature cycles used to exclude fatigue analyses will not be exceeded for 60 years of operation.

The applicant dispositioned the containment vessel TLAA in accordance with 10 CFR 54.21(c)(1)(i), that the TLAA excluding the containment vessel from fatigue analysis remains valid during the period of extended operation.

4.6.1.2 Staff Evaluation

The staff reviewed LRA Section 4.6.1 and the containment vessel TLAA to verify, pursuant to 10 CFR 54.21(c)(1)(i), that the analysis remains valid during 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. This comparison confirms that the number of transients in the existing analyses will not be exceeded during the period of extended operation.

The staff reviewed the applicant's USAR Section 3.8.2.1.5 which states that the containment vessel design meets the requirements for paragraph N-415.1, Section III of the 1968 ASME Code, Section III, to justify the exclusion of cyclic or fatigue analyses.

The LRA states that the containment vessel was analyzed for 400 temperature cycles (from 30 °F to 120 °F) and 400 pressure cycles (from -25 to 120 psi) against the requirements for ASME Code, Section III, paragraph N-415.1 fatigue waiver. For the temperature cycles, the staff noted in USAR Section 3.8.2.1.4(e) that the containment is designed to a lowest metal service temperature of 30 °F and a maximum operating temperature of 120 °F. However, the staff could not determine the basis for a pressure range of -25 to 120 psi in the applicant's USAR. In addition, the staff noted that the USAR does not specify how and when the analysis used to satisfy the fatigue waiver was performed, nor does it address the 400 pressure and 400 temperature cycles used in a fatigue analysis. Therefore, by letter dated July 21, 2011, the staff issued RAI 4.6-1 requesting that the applicant describe the original design basis used to determine that the requirements of a fatigue waiver, per ASME Code, subparagraph N-415.1, were met for the containment vessel.

In its response dated August 17, 2011, the applicant stated that the fatigue waiver calculation for the containment vessel was performed in accordance with paragraphs N-415.1(a) through (f) of ASME Code, Section III. The applicant further stated that the calculation confirmed the requirements of N-415.1 for 400 pressure cycles (from -25 to 20 psi) and 400 temperature cycles (from 30 °F to 120 °F) and provided calculations to justify the basis for their determination. The applicant stated that although the pressure cycle range used in the fatigue waiver evaluation is from -25 to 20 psi, for a full range pressure fluctuation of 45 psi, the possible full range pressure is from -0.67 to 45 psig based on the containment vessel design allowable values. The applicant stated that this adjusted full range pressure fluctuation of 45.67 psig still meets the criteria of ASME Code, subparagraph N 415.1(b).

The staff reviewed the applicant's August 17, 2011, response and determined that additional information was needed to complete its review. The staff determined that the applicant's response to RAI 4.6-1 had not provided the basis for using 400 pressure and temperature cycles. In addition, the staff noted that, in its response to RAI 4.6-1, the applicant stated that it had used a pressure range of -25 to 20 psi in the fatigue analysis, which differed from the -25 to 120 psi stated in the LRA. In a teleconference held on September 13, 2011, the staff requested the applicant provide a supplement to the response for RAI 4.6-1 to include the basis for the 400 pressure and temperature cycles and resolve the discrepancy in pressure ranges between the RAI response and the LRA.

In a letter dated October 7, 2011, the applicant submitted a supplemental RAI response that revised the LRA to include details of its fatigue waiver analysis, provided the basis for using 400 cycles, and revised the error in the LRA. The applicant stated that the 400 cycles in the original fatigue analysis were based on a conservative estimate of anticipated cycles for 40 years of operation. The staff reviewed USAR Table 5.1-8 and noted that the original plant design was for 240 HU/CD cycles. The staff also reviewed LRA Table 4.3-1, which provided an updated analysis of transient cycles. LRA Table 4.3-1 shows that the plant is projected to reach 128 cycles through 60 years of operation. The staff noted that the analysis of 400 temperature cycles performed against the requirements of ASME Code, Section III, paragraph N-415.1, for the fatigue waiver is greater than the plant design number of transients for 40 years of operation and is also greater than the projected number of cycles anticipated through the period of extended operation. The staff finds this acceptable because the number of transients in the existing analyses will not be exceeded during the period of extended operation.

In its October 7, 2011, response letter, the applicant stated that the pressure range of -25 to 120 psi in the original LRA submittal was a typographical error, and the LRA should have read -25 to 20 psi. However, the applicant stated that the pressure range considered in the original fatigue analysis has since been replaced with the adjusted pressure range of -0.67 to 45 psig, which is based on the containment vessel design allowable negative pressure of -0.67 psig and the containment vessel pneumatic test pressure of 45 psig (design pressure of 36 psig x 1.25). The applicant re-performed their fatigue waiver calculation with the corrected values and determined that the adjusted pressure range did not exceed ASME Code requirements. The staff found that the pressure range of -0.67 psig to 45 psig provided in the supplemental RAI response is consistent with the design pressures stated in USAR Section 3.8.2.1.4(e) for the containment vessel. However, the staff could not determine the applicant's basis for having used a pressure range of -25 to 20 psi in the original fatigue waiver calculation, when that analysis was revised, and the basis for the change.

In a telephone conference held on October 26, 2011, the staff requested that the applicant explain the original basis for the pressure range of -25 to 20 psi used in the original fatigue waiver. By letter dated November 9, 2011, the applicant stated that based on a review of their CLB, the basis for the pressure range of -25 psi to 20 psi could not be determined. Therefore, the applicant re-performed the fatigue waiver with the values for the maximum possible full range pressure fluctuation in the containment vessel design. The staff found this acceptable because the analysis uses the design pressure range found in the USAR and is within the limits of the ASME Code requirements for the fatigue waiver. The staff's concern described in RAI 4.6-1 is resolved.

The staff reviewed USAR Section 3.7.3 to determine if the fatigue analysis for the containment vessel considered earthquake-induced cycles. The staff noted in USAR Section 3.7.3.1 that the applicant's earthquake analysis conservatively assumes 200 cycles of significant loading for Class 1 structures. The USAR states that the design criteria for both structures and equipment required that the calculated stresses from seismic, combined with other loads, remained below yield of the material. The USAR further states that since the calculated stresses were below yield, no cyclic loading was considered in the design, and for the small number of cycles that occur during an earthquake, fatigue modes of failure are not applicable for structures and equipment. The staff finds it acceptable that earthquake-induced cycles are not a part of the applicant's fatigue analysis for the containment vessel because the applicant's CLB does not include consideration of seismic stresses in the design of the containment vessel.

The staff finds that the TLAA for the containment vessel meets the acceptance criteria in SRP-LR Section 4.6.2.1.1.1 because the number of projected cycles remains below the original design assumptions and the number of cycles used in the fatigue waiver analysis. As such, the existing exemption from fatigue analysis for the containment vessel will remain valid for the period of extended operation. The staff finds the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue analysis exemption for the containment vessel remains valid during the period of extended operation.

4.6.1.3 USAR Supplement

LRA Section A.2.5.1 provides the USAR supplement summarizing the containment vessel TLAA. The staff reviewed LRA Section A.2.5.1 consistent with the review procedures in SRP-LR Section 4.6.3.2, which state that the applicant should provide a summary description of the fatigue evaluation of the containment vessel including the basis for determining that the applicant has dispositioned the evaluation in accordance with 10 CFR 54.21(c)(1).

Based on its review of the USAR supplement, as amended by letter dated August 17, 2011, the staff finds it meets the acceptance criteria in SRP-LR Section 4.6.2.2. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the containment vessel fatigue analysis, as required by 10 CFR 54.21(d).

4.6.1.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the analysis for the containment vessel TLAA remains valid during the period of extended operation. The staff also concludes that the USAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.6.2 Containment Penetrations

4.6.2.1 Summary of Technical Information in the Application

LRA Section 4.6.2 describes the applicant's evaluation of the existence of a TLAA for the containment penetrations. The applicant stated that the piping penetrations (of the containment vessel) are either large diameter, high energy, hot piping (main steam and feedwater lines), or small diameter lower energy piping (general piping). The applicant also stated that each main steam and main feedwater containment penetration consists of a process pipe, a guard pipe, a flued head, and a penetration bellows assembly.

The applicant stated that, consistent with the exclusion of cyclic fatigue analyses in containment vessel design (reviewed by the staff in SER Section 4.6.1), a search of the Davis-Besse CLB did not identify any pressurization cycles or fatigue analyses for containment penetration assemblies. Therefore, the applicant stated that there are no TLAAs associated with the containment vessel penetrations.

4.6.2.2 Staff Evaluation

The staff reviewed LRA Section 4.6.2 and the applicant's CLB to verify that there is no TLAA for containment penetrations, in accordance with the definition of a TLAA in 10 CFR 54.3. Section 54.3(a) of 10 CFR states, in part, that a TLAA must involve time-limited assumptions defined by the current operating term. The staff reviewed the applicant's USAR Section 3.8.2.1.3 and noted that the containment vessel, including the penetrations, was designed using the ASME Boiler and Pressure Vessel Code, Section III, 1968 through summer addenda 1969. ASME Code, Section III, Paragraph N-451, "General Requirements for Openings," part (a) states, in part, that "for vessels or parts thereof which meet the requirements of N-415.1, analysis showing satisfaction of the (fatigue analysis) requirements of N-414.1, N-414.2, N-414.3, and N-414.4 in the immediate vicinity of the openings is not required. The staff reviewed and confirmed the applicability of the provisions of ASME Code, Section N-415.1, for the containment vessel in SER Section 4.6.1. In addition, the staff noted that the applicant's review and an independent staff review did not identify any cycle-dependent analysis for containment penetrations. The staff finds the applicant's determination that there is no TLAA associated with the containment vessel penetrations acceptable because the analysis does not satisfy criterion (3) of the definition of a TLAA given by 10 CFR 54.3, that there must be time-limited assumptions defined by the current operating term.

4.6.2.3 USAR Supplement

LRA Section A.2.5.2 provides the USAR supplement, which states that there is no TLAA associated with the containment vessel penetrations. The staff reviewed LRA Section A.2.5.2 consistent with the review procedures in SRP-LR Section 4.1.3, which state that the staff should verify that the analysis does not meet at least one of the criteria defining a TLAA in 10 CFR 54.3.

Based on its review of the USAR supplement, the staff finds it meets the acceptance criteria in SRP-LR Section 4.1.2. Additionally, the staff determines that the applicant provided an adequate summary description of the lack of a TLAA for containment penetrations in accordance with the 10 CFR 54.3 definition of TLAA, as required by 10 CFR 54.21(d).

4.6.2.4 Conclusion

On the basis of its review, the staff concludes that the applicant has provided an acceptable demonstration that there is no TLAA for containment penetrations. The staff also concludes that the USAR supplement contains an appropriate summary of the evaluation for a TLAA, as required by 10 CFR 54.21(d).

4.6.3 Permanent Canal Seal Plate

4.6.3.1 Summary of Technical Information in the Application

LRA Section 4.6.3 describes the applicant's evaluation of the TLAA for the permanent canal seal plate (also known as the permanent reactor cavity seal plate). The applicant also stated that the permanent canal seal plate spans the gap between the RV and the fuel transfer canal floor and retains water in the canal when the canal is flooded. The applicant further stated that the fatigue analysis of the permanent canal seal plate seal membrane, which was installed in 2004, shows that the maximum fatigue usage factor, at the inner leg to the RV seal ledge weld, is based on 50 full HU/CD cycles. In LRA Table 4.3-1, transient 31A, the permanent canal seal plate is projected to experience 51 HU/CD cycles through the end of the period of extended operation.

The applicant stated that the number of occurrences of permanent canal seal plate HU/CD is tracked by the Fatigue Monitoring Program to ensure that action is taken before the analyzed number of transients is reached. The applicant dispositioned the permanent canal seal plate TLAA in accordance with 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.

4.6.3.2 Staff Evaluation

The staff reviewed LRA Section 4.6.3 and the permanent canal seal plate TLAA to verify, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended function of the permanent canal seal plate 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, if the proposed AMP relies on mitigation or inspection, it shall be evaluated against the 10 elements described in BTP RLSB-1 of Appendix A of the SRP-LR. If the applicant proposes a component replacement before its CUF exceeds 1.0, the CUF for the replacement will remain less than or equal to 1.0 during the period of extended operation.

The staff reviewed the applicant's statement that the permanent canal seal plate membrane was designed to 50 HU/CD cycles and that the design for 50 cycles corresponds to the maximum CUF based on the fatigue properties of the materials and expected fatigue service of the membrane. The staff searched the applicant's USAR and did not find a statement showing that 50 cycles was used in the original fatigue analysis. The staff needed more information to complete its review; therefore, by letter dated July 21, 2011, as Part 2 of RAI 4.6-1, the staff requested that the applicant explain the basis for the 50 cycles stated in the LRA.

By letter dated August 17, 2011, the applicant responded that the actual maximum fatigue usage factor for the permanent canal seal plate, which is at the inner leg to the RV seal ledge weld, has a value of 1.2, based on 60 cycles. The applicant stated that to satisfy the ASME Code requirement that the maximum usage factor not exceed 1.0, the number of HU/CD cycles is established at 50 cycles. The staff could not determine whether the applicant performed the ASME Code Division III analysis for 50 cycles and calculated a CUF of less than 1.0 or if the 50 cycles was a linear extrapolation from 60 cycles with CUF of 1.2. In a September 13, 2011, telephone conference, the staff asked the applicant to confirm that they performed the ASME Code Division III fatigue analysis for 50 cycles and calculated a CUF which did not exceed 1.0. The applicant stated that the analysis was performed for 50 cycles, with a CUF of 1.0. The staff finds this acceptable because the applicant confirmed that 50 cycles was used for the fatigue analysis of the permanent canal seal plate. The staff's concern in Part 2 of RAI 4.6-1 is resolved.

The staff noted that, in accordance with LRA Table 4.3-1, transient 31A, the permanent canal seal plate was installed on January 25, 2003, and it is expected to exceed the anticipated number of HU/CD cycles during the period of extended operation. As of February 19, 2008, the permanent canal seal plate has experienced 7.5 HU/CD cycles and a 60-year projection anticipates the component will experience a total of 51 cycles, which is greater than the allowed 50 cycles. The applicant is using the Fatigue Monitoring Program to track the number of HU/CD cycles experienced by the permanent canal seal plate. The staff noted that the applicant's Fatigue Monitoring Program is based on tracking transient cycle counts and comparing them with design limits on fatigue transients to manage cumulative fatigue damage of select components. The staff's review of the applicant's Fatigue Monitoring Program is located in Section 3.0.3.2.6 of this SER. The Fatigue Monitoring Program has measures to ensure that fatigue usage calculations are updated, as needed, prior to the accrued cycles exceeding the allowable cycle limit, which meets the acceptance criteria in SRP-LR Section 4.6.2.1.1.3.

The staff finds the use of the Fatigue Monitoring Program acceptable because the program will track the transients analyzed in the design of the permanent canal seal membrane, and it will implement appropriate actions to maintain the CUF of the permanent canal seal plate with allowable limits through the period of extended operation.

The staff finds the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions of the permanent canal seal plate will be adequately managed for the period of extended operation.

4.6.3.3 USAR Supplement

LRA Section A.2.5.3 provides the USAR supplement summarizing the permanent canal seal plate TLAA. Based on its review of the USAR supplement, the staff finds it meets the acceptance criteria in SRP-LR Section 4.6.2.2. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the effects of aging on the intended functions will be adequately managed for the period of extended operation for the permanent canal seal plate TLAA, as required by 10 CFR 54.21(d).

4.6.3.4 Conclusion

On the basis of 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 permanent canal seal plate will be adequately managed for the period of extended operation. The staff also concludes that the USAR 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 Leak-Before-Break

LRA Section 4.7.1 provides the background for the applicant's use of leak-before-break (LBB). The use of LBB is based on the plant's ability to detect leakage from a through-wall flaw in piping and take appropriate action before the flaw could grow to the point of failure. Topical report BAW-1847, Revision 1, "Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSS," September 1985 (ADAMS Accession Nos. 8511180489 and 8511180499), presents the LBB evaluations of primary coolant system piping in several B&W plants, including the 36-in. hot leg and 28-in. cold leg piping at Davis-Besse. The inputs to the LBB analyses include RCS piping structural loads, leakage flaw size determination, and RCS piping material properties.

In 2010, in accordance with NRC approval, the applicant installed at Davis-Besse, nickel-based Alloy 52 weld overlays on the Alloy 82/182 dissimilar metal welds (DMWs) at the RCP suction and discharge nozzles to mitigate primary water stress corrosion cracking (PWSCC). The RCP nozzles are part of piping approved for LBB. As part of the weld overlay installation, the applicant updated its original LBB evaluation to reflect the new weld configuration with the weld overlays in place.

The applicant evaluated the TLAA for use of LBB in terms of the fatigue flaw growth analysis in LRA Section 4.7.1.1, the thermal aging analyses for CASS in LRA Section 4.7.1.2, and the PWSCC analyses in LRA Section 4.7.1.3.

4.7.1.1 Fatigue Flaw Growth

4.7.1.1.1 Summary of Technical Information in the Application

LRA Section 4.7.1.1 describes the applicant's TLAA for fatigue flaw growth in the LBB evaluation. As part of the LBB analysis, the applicant postulated surface flaws at the piping system locations having the highest stress coincident with the lower bound of the material properties for base metal and welds. The applicant calculated growth due to fatigue for a postulated surface flaw to demonstrate that should the surface flaw propagate in the through-wall direction, an identifiable leak would develop and the operator would take corrective actions before the flaw would propagate circumferentially around the pipe and cause a double-ended pipe rupture under faulted conditions.

The applicant used plant design transients in the fatigue flaw growth analysis. For the updated analysis, the applicant used 1.5 times the design cycles for the RCP suction and discharge weld overlays. The applicant stated that the transient cycles are being monitored by the Fatigue Monitoring Program. The applicant also stated that if a transient cycle count approaches the allowable design limit, corrective actions will be taken. The applicant further stated that the effects of fatigue flaw growth on piping approved for LBB will be managed by the Fatigue Monitoring Program for the period of extended operation.

The applicant dispositioned the TLAAs for the fatigue flaw growth analysis in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of aging will be adequately managed for the period of extended operation.

4.7.1.1.2 Staff Evaluation

The staff reviewed LRA Section 4.7.1.1 to verify, 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 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 effects are properly addressed through the period of extended operation.

By letter dated March 21, 2011, the staff issued RAI 4.7.1.1-3, requesting the applicant to explain the differences among the updated analysis, the fatigue flaw growth analysis, and the LBB analysis, as discussed in LRA Section 4.7.1.1. In its response, dated April 20, 2011, the applicant stated that the original LBB evaluation for the Davis-Besse RCS primary piping is contained in Topical Report BAW-1847, Revision 1. The applicant stated that the original LBB evaluation included fatigue flaw growth analyses, flaw stability analyses, and limit load analyses. The NRC approved the original LBB evaluation in BAW-1847, Revision 1, by letter dated December 12, 1985, from Dennis M. Crutchfield, NRC to L.C. Oakes, B&WOG, Subject "Safety Evaluation of B&W Group Reports Dealing with Elimination of Postulated Pipe Breaks in PWER Primary Main Loop."

By letter dated September 28, 2009, the applicant submitted the updated LBB evaluation "License Amendment Request to Update the Leak-Before-Break Evaluation for the Reactor Coolant Pump Suction and Discharge Nozzle Dissimilar Metal Welds," (ADAMS Accession No. ML092790438). The updated LBB evaluation was submitted to address application of weld overlays on the Alloy 82/182 DMWs of the RCP suction and discharge nozzles. As part of the updated LBB evaluation, the applicant analyzed fatigue crack growth for the Alloy 82/182 DMWs to demonstrate that the post-weld overlay crack growth is minimal for balance of plant life.

By letter dated March 24, 2010, the NRC approved the updated LBB evaluation in Amendment No. 281, from Michael Mahoney of NRC to Barry S. Allen of FENOC, "Davis-Besse Nuclear Power Station, Unit 1—Issuance of Amendment RE: Application to update the Leak-Before-Break Evaluation for the Reactor Coolant Pump Suction and Discharge Nozzle Dissimilar Metal Welds" (ADAMS Accession No. ML100640506).

The staff finds that both the original LBB evaluation and updated LBB evaluation include fatigue crack growth calculations. The staff finds the applicant's response to RAI 4.7.1.1-3 acceptable because the applicant has clarified the differences among the updated analysis, the fatigue flaw growth analysis, and the LBB analysis. The staff's concern described in RAI 4.7.1.1-3 is resolved.

The staff evaluated the applicant's original and updated fatigue crack growth calculations in terms of TLAA parameters (e.g., the number of transient cycles) as follows. By letter dated

March 21, 2011, the staff issued RAI 4.7.1.1-1, requesting the applicant to provide details of the original fatigue flaw growth analysis, such as the number of postulated surface flaws, initial and final flaw sizes, the plant design transients, and the number of the transient cycles.

In letter dated April 20, 2011, in response to RAI 4.7.1.1-1, the applicant stated that BAW-1847, Revision 1, included fatigue flaw growth analyses for the smallest and largest pipe straight sections to demonstrate that postulated surface flaws are likely to propagate in the through-wall direction and develop leakage before they will propagate circumferentially around the pipe and result in pipe failure. A fatigue flaw in the longitudinal and circumferential direction was postulated for each diameter of the pipe. The minimum postulated flaws are all at least 30 percent through-wall.

The applicant's postulated fatigue flaw sizes that grow through-wall are many times larger than those permitted by the ASME Code. The applicant noted that whether or not the flaws grow through-wall does not affect the conclusion of the LBB evaluation because the LBB methodology allows the pipe to leak. The important factor is that postulated flaws would propagate radially, go through-wall, and produce leakage before they could propagate circumferentially and potentially produce pipe failure.

The applicant stated that six categories of NSSS design transients were used in the fatigue flaw growth evaluation, as shown in Tables 4-3, 4-4, and 4-5 of BAW-1847, Revision 1. Category 1 includes deadweight, thermal expansion, and operating pressure associated with 240 HU/CD cycles. Categories 2 to 5 include thermal stresses due to four groupings of NSSS design transients. Category 6 includes 22 safe shutdown earthquake events. The applicant selected the generic NSSS design transients used in the BAW-1847, Revision 1, to bound the participating B&W plants (including Davis-Besse) for a 28-in. cold leg straight section and a 38-in. hot leg section.

The applicant provided a comparison of the NSSS design cycles used in the fatigue crack growth evaluation in BAW-1847, Revision 1, to the Davis-Besse plant-specific NSSS design cycles. The staff finds that the analyzed cycles used in the original LBB fatigue flaw growth analysis bound the projected transient cycles at the end of 60 years.

In BAW-1847, Revision 1, the fatigue flaw growth analysis was based on linear elastic fracture mechanics and used the equations in Appendix A to Section XI of the ASME Code. This method is applicable to surface flaws that have not fully penetrated the wall. Results of the fatigue flaw growth analysis show the minimum surface flaw depths (semi-elliptical shape) that will grow through the pipe wall. In letter dated April 20, 2011, in response to RAI 4.7.1.1-1, the applicant stated that the analyzed cycles used in the LBB fatigue flaw growth analysis bound the 60 year projected cycles. The staff has confirmed that the applicant has shown that the fatigue flaw growth calculation of LBB piping in the original LBB evaluation used transient cycle numbers that exceed the projected design cycles at the end of 60 years. Therefore, the fatigue flaw growth calculation in the original LBB evaluation is valid for the period of extended operation. The staff's concern described in RAI 4.7.1.1-1 is resolved.

By letter dated March 21, 2011, the staff issued RAI 4.7.1.1-2, which asked the applicant to clarify the fatigue crack growth calculation in the updated LBB evaluation. Specifically, the staff asked about the design cycles and associated cycle numbers, whether the multiple of 1.5 times the design cycles is adequate for the period of extended operation, and how many years the design cycles will cover. In its response, dated April 20, 2011, the applicant stated that, as part of the updated LBB evaluation, it analyzed the fatigue crack growth for the overlaid Alloy 82/182 DMWs using transient cycles that were 1.5 times higher than the 40-year design cycles to bound the 60-year projected cycles. The applicant stated that the analyzed cycles used in the fatigue crack growth analyses would remain valid for at least 90 years of operation, based on current cycle projections.

The staff finds the applicant's response acceptable because the applicant demonstrated that the transient cycles that were used in the fatigue crack growth calculation in the updated LBB evaluation exceed the projected transient cycles at the end of 60 years. Therefore, the fatigue flaw growth analysis in the updated LBB evaluation is valid for the period of extended operation. The staff's concern described in RAI 4.7.1.1-2 is resolved.

LRA Section 4.7.1.1 states that, per 10 CFR 54.21(c)(1)(iii), the effects of fatigue flaw growth on piping approved for LBB will be managed by the Fatigue Monitoring Program for the period of extended operation. By letter dated March 21, 2011, the staff issued RAI 4.7.1.1-4, which asked the applicant to describe the processes and procedures to explain how the actual transient cycles are monitored and how corrective actions will be implemented by the Fatigue Monitoring Program. In its response, dated April 20, 2011, the applicant stated that it has elected to disposition the fatigue flaw growth analysis in accordance with 10 CFR 54.21(c)(1)(i), instead of 10 CFR 54.21(c)(1)(iii), as stated in LRA Section 4.7.1.1. The applicant further stated that it will not credit the Fatigue Monitoring Program for managing the effects of fatigue flaw growth on piping approved for LBB. The applicant explained that it compared the design cycles that were used in the original fatigue flaw growth evaluation for LBB piping provided in BAW-1847, Revision 1, to the Davis-Besse 60-year projected cycles and determined that the analyzed cycles bound the 60-year projected cycles. The applicant further stated that the fatigue flaw growth calculation in the original LBB evaluation remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i). The staff finds the applicant's response acceptable for the following reasons:

- The staff confirmed that the transient cycles used in the original LBB evaluation bound the projected 60-year transient cycles.
- The staff compared the design cycles that were used in the fatigue crack growth analyses for the Alloy 82/182 DMWs in the updated LBB evaluation to the 60-year projected cycles and confirmed that the analyzed cycles bound the 60-year projected cycles.
- The staff finds that the fatigue crack growth analysis in the updated LBB evaluation remains 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 4.7.1.1-4 is resolved.

As shown in the enclosure to the April 20, 2011, letter, LRA Amendment 4 revised LRA Section 4.7.1.1 to disposition the fatigue crack growth calculations of LBB piping pursuant to 10 CFR 54.21(c)(1)(i).

The staff reviewed the applicant's revised disposition of this TLAA consistent with the review procedures in SRP-LR Section 4.7.3.1.1, which state that justification provided by the applicant is reviewed to verify that the existing analyses are valid for the period of extended operation and the existing analyses should be shown to be bounding even during the period of extended operation. The staff finds the revision to LRA Section 4.7.1.1 acceptable because the staff has confirmed that the existing fatigue flaw growth analyses in the original and updated LBB evaluations are valid for the period of extended operation and the transient cycles used in the LBB evaluations bound the projected 60-year transient cycles.

4.7.1.1.3 USAR Supplement

LRA Section A.2.7.1 provides the USAR supplement summarizing the TLAA evaluation of the fatigue flaw growth calculation for LBB piping. The applicant amended LRA Section A.2.7.1 to include its response to RAIs 4.7.1.1-1 and 4.7.1.1-2, as documented in LRA Amendment 4 (April 20, 2011). The staff reviewed the amended LRA Section A.2.7.1 consistent with the review procedures in SRP-LR Section 4.7.3.2, which state that the staff verifies that the USAR supplement includes a summary description of the evaluation of each TLAA.

Based on its review of the revised USAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.7.2.2. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the TLAA of the fatigue flaw growth calculation for LBB piping, as required by 10 CFR 54.21(d).

4.7.1.1.4 Conclusion

On the basis of its review, the staff concludes that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue crack growth analyses for the LBB piping will remain valid for the period of extended operation. The staff also concludes that the amended USAR supplement contains an appropriate summary description of the fatigue crack growth analyses for the LBB piping, as required by 10 CFR 54.21(d).

4.7.1.2 Thermal Aging

4.7.1.2.1 Summary of Technical Information in the Application

LRA Section 4.7.1.2 states that the only stainless steel components in the LBB analysis are the safe ends welded to the RCP casings and the pump casings themselves, with the pump casings being the only cast stainless steel. The RCP casings at Davis-Besse, including the suction and discharge nozzles, are made of annealed SA 351 CF-8M and were statically cast.

The applicant stated that the updated LBB analysis was based on saturated embrittlement of the CASS casings, such that there is no embrittlement TLAA. The applicant also stated that reduction of fracture toughness due to thermal embrittlement of CASS components is an aging effect requiring management for the RCP casings and is managed by the ISI Program. The applicant further stated that the acceptability of a 10-year inspection interval for these weld overlays was demonstrated in the updated LBB analysis. The applicant also stated that this analysis does not justify operation of the weld overlays for the life of the plant but for the 10 years between inspections. Therefore, the effects of thermal aging on CASS components in the approved LBB piping will be managed by the ISI Program for the period of extended operation.

The applicant stated that thermal aging of CASS is not a TLAA. The effects of thermal aging on CASS components in the approved LBB piping will be managed by the ISI Program for the period of extended operation.

4.7.1.2.2 Staff Evaluation

The staff confirmed, pursuant to 10 CFR 54.3, that the analysis of thermal aging of CASS related to LBB is not a TLAA for the period of extended operation.

The staff reviewed the applicant's evaluation and conclusion consistent with the review procedures in SRP-LR Section 4.1.3, which state that the reviewer verifies that the selected analyses do not meet at least one of the six criteria of a TLAA, as defined in 10 CFR 54.3(a).

By letter dated March 21, 2011, the staff issued RAI 4.7.1.2-1, requesting the applicant to explain why thermal aging of CASS component is not a TLAA if it is monitored by the ISI Program, as stated in LRA Section 4.7.1.2. In its response dated April 20, 2011, the applicant stated that thermal aging of CASS components (i.e., RCP casings including the pump suction and discharge nozzles) in the LBB piping is not a TLAA because saturated embrittlement (the lowest and worst-case fracture toughness) was used in the updated LBB analysis, and thus 10 CFR 54.21(c)(1)(iii) is not applicable.

By letter dated March 21, 2011, the staff issued RAI 4.7.1.2-4, requesting the applicant to demonstrate that the value of fracture toughness used in the updated LBB analysis represents the lowest and worst-case fracture toughness value for the RCP casing. In its response dated April 20, 2011, the applicant stated that the updated LBB analysis used actual material properties to develop lower bound J-R curves and J_{Ic}/K_{Ic} for the RCP pump CASS material heats with consideration of thermal embrittlement.

The applicant used the lowest (saturated) fracture toughness property of the CASS material, which bounds the fracture toughness value at the end of 60 years, and, thereby, has demonstrated that the structural integrity of the CASS components will be acceptable at the end of 60 years. SER Section 4.7.6.2 discusses in detail the staff's evaluation of the fracture toughness value used in the applicant's pump casing analysis. The staff notes that the applicant has addressed the staff's concern regarding thermal embrittlement of CASS satisfactorily because the LBB evaluation has considered the thermal embrittlement of CASS using the worst case fracture toughness of the LBB piping, and the LBB piping has satisfied the safety margins in SRP-LR Section 3.6.3. The staff's concerns described in RAI 4.7.1.2-1 and RAI 4.7.1.2-4 are resolved.

As part of its response to RAI 4.7.1.2-1, the applicant proposed in LRA amendment No. 4 to delete the following two sentences from LRA Section 4.7.1.2 and the associated TLAA summary in Section A.2.7.1: "The acceptability of a 10 year inspection interval for these weld overlays was demonstrated in the updated LBB analysis. This analysis does not justify operation of the weld overlays for the life of the plant, but for the 10 years between inspections…" The proposed deletion to LRA Section 4.7.1.2 is documented in the enclosure to the April 20, 2011, letter. The staff finds the proposed revision to LRA Section 4.7.1.2 acceptable for two reasons: 1) the staff found the inspection and analysis of the subject weld overlays acceptable in its March 24, 2010, SE on the updated LBB in accordance with the requirements of 10 CFR Part 50, and 2) deletion of these two sentences does not affect the staff's evaluation of whether the weld overlay analysis is a TLAA.

The staff is not aware of any ultrasonic examination technique that has been qualified to detect flaws in CASS material in accordance with the ASME Code, Section XI, Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems." By letter dated March 20, 2011, the staff issued RAI 4.7.1.2-2, which asked the applicant to discuss the ASME Code-qualified ISI method(s) that will be used to monitor thermal aging effect in the

CASS components and the associated inspection frequency. In its response dated April 20, 2011, the applicant stated that the primary inspection of CASS components (i.e., valve bodies and pump casings) is external visual examination. Internal visual inspections and volumetric inspections are performed only when a valve or pump is disassembled for maintenance. The applicant stated that the GALL Report, Section XI.M12 (the CASS AMP), states that screening for susceptibility to thermal aging is not required for pump casings and valve bodies based on the assessment documented in the letter dated May 19, 2000, from Christopher Grimes, NRC, to Douglas Walters, NEI. GALL Report AMP XI.M12 further states that the existing ASME Code, Section XI, inspection requirements, including the alternative requirements of ASME Code Case N-481 for pump casings, are adequate for all pump casings and valve bodies.

For pump casings, the applicant stated that ASME Code, Section XI, ISI requirements for pressure retaining welds of pump casings (Examination Category B-L-I) are delineated in ASME Code, Section XI, Table IWB-2500-1. The applicant stated that Examination Category B-L-1 requires volumetric examination of pump casing welds. The applicant stated that Davis-Besse uses ASME Code Case N-481 in lieu of the examination requirements of this Code category. The applicant also stated that these alternate requirements consist of visual inspections and an analytical evaluation to demonstrate the safety and serviceability of the pump casings in the presence of an assumed flaw and that each of the four RCP casings is visually examined every 10-year ISI interval.

For valve bodies, the applicant stated that there are no ASME Code Category B-M-1 welds in valves installed at Davis-Besse and there are 10 Code Category B-M-2 valves in four groups. The applicant stated that one valve per group will be examined over each ISI interval when disassembled for routine maintenance, repair, or volumetric examination. The applicant also stated that, per ASME Code, Section XI, Table IWB-2500-1, item B.12.30, the inspection of Code Category B-M-2 valves less than 4-in. NPS is limited to surface examination. The applicant further stated that per ASME Code, Section XI, Table IWB-2500-1, items B.12.40 and B.12.50, the inspection of valves greater than 4-in. NPS includes volumetric and visual (VT-3) examination. However, the inspection is limited to an external visual (VT-3) inspection unless the valve is opened for maintenance and repair. Therefore, the staff finds that it is acceptable that Davis-Besse will inspect the CASS pump casings and valves per the ASME Code, Section XI, Table IWB-2500-1. The staff's concern described in RAI 4.7.1.2-2 is resolved. SER Section 4.7.6 provides additional information and the staff's review regarding Code Case N-481.

By letter dated March 20, 2011, the staff issued RAI 4.7.1.2-3 requesting the applicant to explain why the LRA does not include an AMP to monitor thermal aging embrittlement of CASS that is similar to GALL Report AMP XI.M12. In letter dated April 20, 2011, the applicant stated that the LRA does not include a Thermal Aging Embrittlement of CASS Program similar to GALL Report AMP XI.M12 because Davis-Besse has no CASS components other than pump casings and valve bodies subject to thermal embrittlement. The applicant further stated that a program similar to GALL Report AMP XI.M12 is not required as reduction of fracture toughness of these component types is managed by the ISI Program, as shown in LRA Section B.2.24.

The applicant noted that GALL Report AMP XI.M12 specifically exempts pump casings and valve bodies from this program and states the following:

> For pump casings and valve bodies, based on the assessment documented in the letter dated May 19, 2000 from Christopher Grimes, Nuclear Regulatory Commission (NRC), to Douglas Walters, Nuclear Energy Institute (NEI),

> screening for susceptibility to thermal aging embrittlement is not required. The existing ASME [Code] Section XI inspection requirements, including the alternative requirements of ASME Code Case N-481 for pump casings, are adequate for all pump casings and valve bodies.

The applicant explained that this position is re-iterated in the GALL Report, item IV.C2-6, which states the following:

> For pump casings and valve bodies, screening for susceptibility to thermal aging is not necessary. The ASME [Code] Section XI inspection requirements are sufficient for managing the effects of loss of fracture toughness due to thermal aging embrittlement of CASS pump casings and valve bodies. Alternatively, the requirements of ASME Code Case N-481 for pump casings are sufficient for managing the effects of loss of fracture toughness due to thermal aging embrittlement of CASS pump casings.

Also, in response to RAI 4.7.1.2-1, the applicant noted that LRA Table 3.1.1, item 3.1.1-55, shows that the ISI Program is credited with management of thermal aging for CASS components. The applicant stated that this is consistent with SRP-LR Table 3.1-1, which identifies the following under "Aging Management Program": "Inservice [I]nspection (IWB, IWC, and IWD). Thermal aging susceptibility screening is not necessary, inservice inspection requirements are sufficient for managing these aging effects. ASME Code Case N-481 also provides an alternative for pump casings."

The applicant noted, in response to RAI 4.7.1.2-3, that LRA Table 3.1.2-3 for the RCPB shows only three CASS component types managed for reduction of fracture toughness—RCP casings (Row 196), small bore valve bodies (Row 234), and large bore valve bodies (Row 255). The applicant further stated that, as there are no piping, piping components, or piping elements, other than pump casings and valve bodies, GALL Report AMP XI.M12 is not required.

Based on the review of the applicant's response, the LRA, and the USAR, the staff finds that Davis-Besse has no piping, piping components, or piping elements that are fabricated with CASS other than pump casings, valve bodies, and valve components. GALL Report AMP XI.M12 exempts pump casings and valve bodies from monitoring. Therefore, the staff finds that it is acceptable that Davis-Besse does not implement an AMP to monitor thermal aging of the CASS material. The staff finds it is acceptable that, in lieu of GALL Report AMP XI.M12, CASS pump casings and valve bodies will be examined under the ISI requirements of the ASME Code Section XI. The staff's concern described in RAI 4.7.1.2-3 is resolved.

Based on its review of the information provided by the applicant, the staff finds that the analysis of thermal aging of CASS within its LBB analyses is not a TLAA because it does not meet criterion 3 of the 10 CFR 54.3 definition of a TLAA, which states that the analysis involves "time-limited assumptions defined by the current operating term, for example 40 years."

4.7.1.2.3 USAR Supplement

LRA Section A.2.7.1 provides the USAR supplement summarizing the absence of TLAAs for thermal aging of CASS components in LBB piping. The applicant amended LRA Section A.2.7.1, as documented in LRA Amendment 4 (April 20, 2011). The staff reviewed amended LRA Section A.2.7.1 consistent with the review procedures in SRP-LR Section 4.7.3.2, which state that the staff verifies that the USAR supplement includes a summary description of the evaluation of each TLAA. Based on its review of the USAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.7.2.2. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the absence of TLAA for thermal aging of CASS components in LBB piping, as required by 10 CFR 54.21(d).

4.7.1.2.4 Conclusion

On the basis of its review, the staff concludes that, in accordance with 10 CFR 54.3(a)(3), thermal aging of RCP pump casing is not a TLAA because the fracture toughness value used in the RCP pump casing analysis does not involve time-limited assumptions defined by the current operating term. The staff also concludes that the amended USAR supplement, as shown in April 20, 2011, letter, contains an appropriate summary description of the basis for absence of a TLAA for thermal aging of CASS components, as required by 10 CFR 54.21(d).

4.7.1.3 Primary Water Stress Corrosion Cracking

4.7.1.3.1 Summary of Technical Information in the Application

LRA Section 4.7.1.3 states that the applicant submitted and the NRC approved a relief request to install weld overlays on certain Alloy 600 components and Alloy 82/182 DMWs for mitigation of PWSCC in LBB piping. The applicant updated the original LBB calculations, demonstrating that the weld overlays resolve the concerns for original DMWs susceptibility to PWSCC. The applicant stated that critical crack sizes and leakage rates with the weld overlay in place were evaluated to demonstrate that margins exist for detection of leakage (i.e., the conclusions of the existing LBB analysis remain valid).

The applicant stated that PWSCC is an aging effect requiring management for the period of extended operation and is managed by the ISI Program and the Nickel Alloy Management Program. The applicant further stated that PWSCC is not a TLAA and the effects of PWSCC on the RCS piping will be managed by the ISI Program and the Nickel Alloy Management Program for the period of extended operation.

4.7.1.3.2 Staff Evaluation

The staff reviewed the applicant’s evaluation and conclusion consistent with the review procedures in SRP-LR Section 4.1.3, which state that the reviewer verifies that the selected analyses do not meet at least one of the six criteria of a TLAA, as defined in 10 CFR 54.3(a).

The staff notes that the weld overlay is installed on piping to mitigate PWSCC in nickel-based Alloy 82/182 dissimilar metal welds. By letter dated March 21, 2011, the staff issued RAI 4.7.1.3-1, in which the staff noted that, as part of NRC approval, the design of weld overlays is required to include a fatigue flaw growth calculation based on a postulated or an actual detected flaw in the Alloy 82/182 dissimilar metal weld. The fatigue flaw growth calculation uses transient cycles, which are time dependent. Therefore, as described in SER Section 4.7.1.2.2, the staff found that the fatigue flaw growth calculation in the weld overlay design is a TLAA. However, the staff notes that PWSCC is not a TLAA concern as part of the weld overlay design.

In its response to RAI 4.7.1.3-1 dated April 20, 2011, the applicant stated that it addressed the fatigue flaw growth TLAA for the Alloy 82/182 DMWs at RCP nozzles (in response to RAI 4.7.1.3-1) and provided a revision to LRA Section 4.7.1.1 to include the subject TLAA (LRA

Amendment 4). As part of its response to RAI 4.7.1.3-1, the applicant proposed to delete LRA Section 4.7.1.3, as shown in LRA Amendment 4 (April 20, 2011).

The staff finds acceptable the applicant's proposal to delete LRA Section 4.7.1.3 because PWSCC is not a TLAA, and the revised LRA Section 4.7.1 addresses the issues of PWSCC and weld overlay installation on RCP nozzles. The staff's concern described in RAI 4.7.1.3-1 is resolved.

4.7.1.3.3 USAR Supplement

LRA Amendment 4 deleted the USAR supplement summary of PWSCC in LBB piping in LRA Section A.2.7.1. The staff finds the deletion acceptable because PWSCC is not a TLAA and resolution of aging effects due to PWSCC is discussed in amended LRA Section 4.7.1.

4.7.1.3.4 Conclusion

LRA Amendment 4 (April 20, 2011) deleted LRA Section 4.7.1.3 and the PWSCC portion of LRA Section A.2.7.1. The staff concludes that the deletion is acceptable because PWSCC is not a TLAA and resolution of aging effects due to PWSCC is discussed in the amended LRA Section 4.7.1.1.

4.7.2 Metal Corrosion Allowance for Pressurizer Instrument Nozzles

4.7.2.1 Summary of Technical Information in the Application

LRA Section 4.7.2 states that USAR Section 5.2.3.2 indicates that pressurizer nozzle repairs and replacements have resulted in a portion of the carbon steel pressurizer nozzle bore being exposed to reactor coolant. The applicant stated that this resulted in an increase of the general corrosion rate (GCR) of the pressurizer shell base metal in the nozzle bores from 0 to 1.42 thousandths of an inch (mils) per year. The applicant also stated that, over the 9 years from the installation of this modification to the end of the original licensed period, general corrosion will result in a loss of 13 mils of the pressurizer carbon steel shell in the nozzle annular regions. The applicant further stated that the allowable radial corrosion limit, calculated per Section III of the ASME Code, is 293 mils for the level instrument nozzles, 493 mils for the sample nozzle, and 495 mils for the vent and thermowell nozzles. The applicant stated that the projected loss of material can be extrapolated to 60-years by multiplying the 1.42 mils per year corrosion rate times the 29 years from the date of installation to the end of the period of extended operation. The applicant further stated that the projected loss of 41.2 mils (29 years x 1.42 mils per year) remains below the allowable radial corrosion limits.

The applicant dispositions the TLAA associated with the metal corrosion allowance for the pressurizer instrument nozzles in accordance with 10 CFR 54.21(c)(1)(ii), that the analysis has been projected to the end of the period of extended operation.

4.7.2.2 Staff Evaluation

The staff reviewed LRA Section 4.7.2 on metal corrosion of the pressurizer instrument nozzles to verify, pursuant to 10 CFR 54.21(c)(1)(ii), that the metal corrosion of the pressurizer instrument nozzles has been projected to the end of the period of extended operation. The staff reviewed the applicant's TLAA consistent with the review procedures in SRP-LR Section 4.7.3.1.2, which state "the documented results of the revised analyses are reviewed to verify that their period of evaluation is extended, such that they are valid for the period of extended operation (e.g., 60 years)."

By letter dated March 21, 2011, the staff issued RAI 4.7.2-1 requesting the applicant to discuss in detail how the corrosion rate was obtained. The staff also asked the applicant to discuss verification of the corrosion rate and to explain whether the corrosion rate increases as the component ages.

In its response dated April 20, 2011, the applicant stated the following:

> General Corrosion rate of 1.42 mils per year was developed in Structural Integrity Associates, Inc., Report SIR-07-188-NPS, "Evaluation of the Corrosion of Carbon Steel and Low Alloy Steel in Portions of Pressurizer Vessels Exposed to Primary Water Following Repair of Small Bore Instrument Nozzles," dated November 2007.
>
> All of the repairs of small bore instrument nozzles or other penetrations that expose carbon steel to primary coolant by "uncovering" some carbon steel vessel material will expose that steel to borated water of nominal boron concentrations under immersion conditions rather than conditions of dripping and evaporation that occur for borated water that leaks from the pressure boundary. The corrosion rate methodology, described in Report SIR-07-188-NPS, used the corrosion rates reported from the literature for such worst case full immersion conditions compared to steam at high temperature, low temperature, and very low oxygen environments (e.g., normal operation) or higher oxygen environments that may occur during refueling. The overall metal loss is the sum of the products of the time at given conditions and the corrosion rates for each of those environments. Assuming that [Davis-Besse] operates 85 [percent] of the year at high temperatures, spends 10 [percent] of the year under shutdown conditions and 5 [percent] of the year at intermediate temperatures, the total general corrosion rate (GCR), actually an average annualized metal loss rate, would be the following: GCR = 0.85x0.6 (operated at 500 °F) + 0.1x8 (operated at 100 °F) + 0.05x2.2 (operated at 300 °F) = 1.42 mils per year
>
> The allowable radial corrosion limits, provided in LRA Section 4.7.2, were developed in Structural Integrity Associates, Inc., Calculation Package DB-09Q-303, "Determination of Allowable Corrosion of Pressurizer Vessel Shell," Revision 1, dated November 17, 2007. In this calculation, the maximum allowable corrosion of the pressurizer material in the penetration bore was quantified by determining the corroded radius such that [the] resulting stresses due to primary loads in the repair pad still meet ASME Code, Section III, design conditions allowable values. General primary membrane and primary membrane-plus-bending stress intensity values due to pressure and mechanical loads (where applicable) were determined and compared to ASME Code allowable values using the maximum corroded bore radius that is possible.
>
> For carbon steel and low alloy steel, oxidation of metallic iron to ferrous ion (Fe^{+2}, a soluble ionic species) or ferric ion (Fe^{+3}, an insoluble ion) on the low-alloy steel provides a level of protection against continuing corrosion. Therefore, the general corrosion rate of 1.42 mils per year is an average annualized metal loss rate and is assumed to be constant throughout the remaining life of the plant.

To verify the applicant's corrosion rate, the staff used a Westinghouse topical report as a reference. On January 12, 2005, the staff approved Westinghouse topic report, WCAP-15973-P, Revision 1, "Low-Alloy Steel Component Corrosion Analysis Supporting Small Diameter Alloy 600/690 Nozzle Repair/Replacement Program" (ADAMS Accession No. ML050180528). The corrosion rate of 1.42 mils per year used by Davis-Besse is slightly lower than the corrosion rate specified for Combustion Engineering plants in WCAP-15973-P. Considering the operating and system design differences between the Combustion Engineering plants and B&W plants such as Davis-Besse, the staff finds the slight difference in the GCR acceptable. The staff finds acceptable the applicant's use of 1.42 mils as the corrosion rate, since the corrosion rate equation used is similar to the method found acceptable to the staff in WCAP-15973-P. Based on this finding, the staff's concern described in RAI 4.7.2-1 is resolved.

The staff finds that the applicant's evaluation of this TLAA acceptable because, pursuant to 10 CFR 54.21(c)(1)(ii), the applicant used an adequate corrosion rate to project the corrosion of the low alloy steel of the pressurizer instrument nozzles to the end of period of extended operation. Further, the corrosion at the end of the period of extended operation was within the allowable limits.

4.7.2.3 USAR Supplement

LRA Section A.2.7.2 provides the USAR supplement summarizing the TLAA of the metal corrosion allowance for pressurizer instrument nozzles. The staff reviewed LRA Section A.2.7.2 consistent with the review procedures in SRP-LR Section 4.7.3.2, which state that the staff verifies that the USAR supplement includes a summary description of the evaluation of each TLAA. Based on its review of the USAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.7.2.2. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the TLAA of metal corrosion for pressurizer instrument nozzles, as required by 10 CFR 54.21(d).

4.7.2.4 Conclusion

On the basis of its review, the staff concludes that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the metal corrosion allowance for pressurizer instrument nozzles has been projected to the end of the period of the extended operation, such that the projected metal loss in the instrument nozzles will be within the allowable limits at the end of the period of extended operation. The staff also concludes that the USAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.3 Reactor Vessel Thermal Shock due to Borated Water Storage Tank Injection

4.7.3.1 Summary of Technical Information in the Application

LRA Section 4.7.3 describes the applicant's TLAA for RV thermal shock due to postulated BWST injection. The applicant stated that USAR Section 5.2 addresses integrity of the RCPB and the analysis to demonstrate that the RV can safely accommodate the PTS condition that is associated with a postulated small steam line break and subsequent low temperature injection from the BWST by the emergency core cooling system at the end of the RV design life. The applicant stated that this transient generates the greatest level of stress in the RV. The applicant also stated that this analysis was revised for the LRA to use RV embrittlement values that bound the period of extended operation.

The applicant further stated that the integrity of the RV was analyzed for this postulated thermal shock event taking into consideration RV embrittlement through 52 EFPY and the 35 °F minimum temperature for the water in the BWST. Several locations in the RV were analyzed for this postulated thermal shock event, and all locations have demonstrated service life greater than 52 EFPY.

Based on the information above, the applicant concluded that the RV integrity analysis under postulated thermal shock conditions associated with BWST injection has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

4.7.3.2 Staff Evaluation

The staff reviewed LRA Section 4.7.3 on postulated RV thermal shock due to BWST injection to verify, pursuant to 10 CFR 54.21(c)(1)(ii), that the applicant provided an acceptable analysis for demonstrating that RV integrity will be maintained during the subject thermal shock event considering projected neutron embrittlement for the period of extended operation.

The staff reviewed the applicant's TLAA consistent with the review procedures in SRP-LR Section 4.7.3.1.2, which state "the documented results of the revised analyses are reviewed to verify that their period of evaluation is extended, such that they are valid for the period of extended operation (e.g., 60 years)."

The staff reviewed USAR Section 5.2 and could not find information regarding the applicant's analysis for demonstrating that RV integrity will be maintained under the postulated thermal shock conditions associated with BWST injection following a small steam line break. The staff determined that the applicant must provide sufficient information or references for this analysis, taking into consideration embrittlement of the RV beltline materials at 52 EPFY. Therefore, by letter dated October 11, 2011, the staff issued RAI 4.7.3-1 requesting that the applicant state the USAR section and page number where the summary of the CLB analysis of the subject thermal shock event is located and provide the reports or calculations documenting the projected 52 EFPY analysis of RV integrity under the subject postulated thermal shock conditions.

In its response dated October 31, 2011, the applicant addressed the 52 EFPY analysis of RV integrity during the subject thermal shock event and the recent update to USAR Section 5.2 describing this 52 EFPY analysis. In its RAI response, the applicant stated that the analysis is addressed in USAR Section 5.2, pages 5.2-2 and 5.2-3. The applicant stated that the original analysis of the RV for this thermal shock event used a non-conservative water injection temperature for the postulated operation of the emergency core cooling system. The applicant also stated that pages 5.2-2 and 5.2-3 of USAR Section 5.2 were revised on May 26, 2011, under an approved USAR change notice, to address the recent reanalysis of RV integrity under the specific postulated thermal shock condition associated with BWST water injection at 35 °F and RV embrittlement through 52 EFPY, as described in LRA Section 4.7.3. The revisions to USAR Section 5.2 were provided by the applicant as part of its response to RAI 4.7.3-1. The staff reviewed the USAR Section 5.2 revisions and confirmed that USAR Section 5.2 does contain an adequate summary description of the analysis for demonstrating RV integrity under these thermal shock conditions. Furthermore, the staff confirmed that the revision to USAR Section 5.2 also states that this analysis was performed using RV fluence levels corresponding to 52 EFPY.

In its response to RAI 4.7.3-1, the applicant stated that the fracture mechanics analysis for evaluating RV integrity during the subject postulated thermal shock event is documented in AREVA Calculation 32-9124893-001, "DB-1 Pressurized Thermal Shock (PTS) Analysis for 32 and 52 EFPY," dated December 14, 2009. The applicant provided this proprietary report in Enclosure C of its October 31, 2011, response to RAI 4.7.3-1. This report documents a linear elastic fracture mechanics analysis for demonstrating that RV integrity will be maintained during the subject postulated thermal shock event, accounting for RV embrittlement at 32 EFPY and 52 EFPY. This linear elastic fracture mechanics analysis is based on the calculation of applied stress intensity factors for the subject thermal shock transient for postulated shallow inside-diameter flaws ranging in depth from one-fortieth of the wall thickness (1/40T) to 1/4T, in increments of 1/40T (i.e., ten postulated flaws: 1/40T, 2/40T, 3/40T….1/4T). The beltline material fracture toughness property (K_{Ic}) for the initiation of flaw growth is based on the formula in the ASME Code, Section XI, Appendix G, which specifies fracture toughness for the initiation of flaw growth as a function of flaw tip temperature and RT_{NDT}, which must be adjusted for neutron embrittlement. The staff confirmed that the analysis was performed for all beltine locations, and the fracture toughness for each location and postulated flaw depth was calculated using flaw tip temperatures based on the thermal shock transient conditions and adjusted RT_{NDT} (ART) values based on neutron embrittlement for 32 EFPY and 52 EFPY. The staff confirmed that 52 EFPY ART values for each of the postulated flaw depths were calculated using the staff's recommended procedures in RG 1.99, Revision 2. The ART values were also calculated using appropriate input values. Specifically, the initial RT_{NDT}, chemistry factor, and margin term inputs correspond to those found acceptable by the staff in SER Sections 4.2.3 and 4.2.4, and the neutron fluence values at the inner surface of the RV correspond to those found acceptable by the staff in SER Section 4.2.1.

Based on its review of the report, the staff determined that this analysis conclusively demonstrates that none of the postulated flaws would initiate growth under the postulated thermal shock condition associated with a small steam line break, followed by 35 °F water injection from the BWST, accounting for neutron embrittlement of the RV beltline materials at 32 EFPY and 52 EFPY. The staff also determined that the analytical methods and assumptions described in the report are consistent with those described in LRA Section 4.7.3 and USAR Section 5.2, as revised on May 26, 2011, to include a reference to this report.

The staff determined that the applicant's response to RAI 4.7.3-1 is acceptable because the applicant provided the detailed analysis of RV integrity under the subject postulated thermal shock conditions for 52 EFPY and identified the revised USAR Section 5.2 text describing this 52 EFPY analysis. The staff also confirmed that the 52 EFPY RV integrity analysis adequately demonstrates that RV integrity would be maintained during this postulated thermal shock event through the period of extended operation, consistent with the description in LRA Section 4.7.3. Therefore, the staff's concern described in RAI 4.7.3-1 is resolved.

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis of RV thermal shock due to postulated BWST water injection has been projected to the end of the period of extended operation. Additionally, the staff finds that the applicant's TLAA meets the acceptance criteria in SRP-LR Section 4.7.2.1 because the analysis of RV thermal shock due to postulated BWST water injection has been projected to the end of the period of extended operation.

4.7.3.3 USAR Supplement

LRA Section A.2.7.3 provides the USAR supplement summary description for the TLAA of RV thermal shock due to BWST injection. The staff reviewed the applicant's USAR supplement summary description for this TLAA and determined that it is consistent with the TLAA discussed in LRA Section 4.7.3. The staff also concludes that the information in the USAR is consistent with SRP-LR Section 4.7.3.2. Based on its review of the USAR supplement, the staff concludes that the applicant provided an adequate summary description of its actions to address the RV thermal shock due to BWST injection for the period of extended operation, as required by 10 CFR 54.21(d).

4.7.3.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis of RV thermal shock due to postulated BWST water injection has been projected to the end of the period of extended operation. The staff also concludes that the USAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.4 High-Pressure Injection/Makeup Nozzle Thermal Sleeves

4.7.4.1 Summary of Technical Information in the Application

LRA Section 4.7.4 describes the applicant's TLAA for the HPI/makeup nozzle thermal sleeves. The applicant stated that during the Cycle 5 RFO a failed thermal sleeve for HPI/makeup nozzle A-1 was discovered. The applicant described its corrective actions, which included assessment and preservation of the structural integrity of the nozzle, which had experienced thermal cycling due to the thermal sleeve failure. The applicant stated that the makeup flow path was re-routed from nozzle A-1 to nozzle A-2 during the Cycle 6 RFO (1990) as one of the corrective actions. The applicant also stated that it performed a fracture mechanics analysis of nozzle thermal sleeve life under various makeup flow cycling conditions, which predicted a lifetime exceeding twenty 18-month operating cycles under the current re-routed makeup flow control conditions.

The applicant stated that accounting for the extended (approximately 2 year) Cycle 13 RFO, the conversion to a 24-month fuel cycle, and the MUR power uprate conditions, the predicted end-of-life for the HPI/makeup nozzle thermal sleeve is approximately 2022, based on the predicted number of makeup thermal cycles. The applicant stated that the current operating license for Davis-Besse expires in April 2017. The applicant committed (Commitment No. 23) to replace all four makeup nozzle thermal sleeves prior to the period of extended operation.

The LRA concludes that cracking of the HPI/makeup thermal sleeves will be managed through the period of extended operation by the Fatigue Monitoring Program, in accordance with 10 CFR 54.21(c)(1)(iii).

4.7.4.2 Staff Evaluation

The staff reviewed LRA Section 4.7.4 on the HPI/makeup nozzle thermal sleeves to verify, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the thermal sleeves will be adequately managed for the period of extended operation.

The staff reviewed the applicant's TLAA consistent with the review procedures in SRP-LR Section 4.7.3.1.3, which state that the applicant shall propose to manage the aging effects associated with the TLAA using an AMP in the same manner as described in the IPA in 10 CFR 54.21(a)(3). SRP-LR Section 4.7.3.1.3 also states that the applicable AMP is reviewed to verify that the effects of aging on the intended functions are adequately managed consistent with the CLB for the period of extended operation.

LRA Section 4.7.4 describes the 1990 failure of HPI/makeup nozzle thermal sleeve A-1 and the applicant's proposal to manage the effects of fatigue on the other HPI/makeup nozzle thermal sleeves during the period of extended operation under the Fatigue Monitoring Program. The staff's review of the applicant's Fatigue Monitoring Program is discussed in SER Section 3.0.3.2.6.

The staff noted that LRA Section 4.7.4 states that the HPI/makeup flow path was re-routed from HPI/makeup nozzle A-1 to nozzle A-2 during the Cycle 6 RFO (1990) as one of the corrective actions for the subject failed thermal sleeve discovered during the Cycle 5 RFO. By letter dated March 17, 2011, the staff issued RAI 4.7.4-1, requesting that the applicant state which specific HPI/makeup nozzle thermal sleeves were analyzed, as discussed in LRA Section 4.7.4.

In its RAI response dated April 15, 2011, the applicant stated that the fracture mechanics analysis of thermal sleeve life under various makeup flow cycling conditions was performed to predict the life of the thermal sleeve for the HPI nozzle that is used for both HPI and makeup duty, nozzle A-2. The staff found the applicant's response to RAI 4.7.4-1 acceptable because the applicant confirmed that the analysis predicting thermal sleeve end-of-life in 2022 specifically applied to the thermal sleeve on HPI/makeup nozzle A-2. The staff's concern described in RAI 4.7.4-1 is resolved.

By letter dated March 17, 2011, the staff issued RAI 4.7.4-2 requesting that the applicant provide a reference for the subject thermal sleeve fracture mechanics analysis discussed in LRA Section 4.7.4. The applicant's April 15, 2011, response to RAI 4.7.4-2 stated that a description of the methodology and results of the HPI/makeup nozzle thermal sleeve analysis are provided in SIR-91-047, Revision 0 report, "Fracture Mechanics Evaluation of Davis-Besse HPI/Makeup Nozzle Thermal Sleeve," which had been submitted to the NRC by letter dated August 23, 1991 (ADAMS Accession No. 910903009, Microfiche Document). By SE issued on January 28, 1991 (ADAMS Accession No. 9102140250, Microfiche Document), the staff approved continued operation of Davis-Besse with the HPI/makeup nozzle thermal sleeve installed at that time. As documented in the January 1991 SE, the staff reviewed an earlier (1990) fracture mechanics evaluation for the HPI/makeup nozzle and approved continued operation of the Davis-Besse unit for Cycle 7 and beyond with the installed HPI/makeup nozzle thermal sleeve under the re-routed makeup flow control conditions. As documented in the January 1991 SE, the staff's approval of continued operation of Davis-Besse was based, in part, on the applicant's commitment to perform a subsequent fracture mechanics analysis of the HPI/makeup nozzle thermal sleeve. Accordingly, the SIR-91-047, Revision 0, report is the "subsequent analysis" provided for in the Davis-Besse commitment. The staff found the applicant's response to RAI 4.7.4-2 acceptable because the applicant identified the key reference for the staff, which helped the staff to locate all documents related to this issue. The staff's concern described in RAI 4.7.4-2 is resolved.

From the January 28, 1991, SE, the staff found that the fatigue crack growth part of the fracture mechanics analysis of the HPI/makeup nozzle was based on an assumed 240 startup and shutdown cycles and 80 HPI transient cycles; these cycle counts bound those expected for the period of extended operation. As discussed in the applicant's August 23, 1991, letter, the fracture mechanics analysis of the thermal sleeve predicts a thermal sleeve lifetime exceeding twenty 18-month operating cycles for the nozzle used for both HPI and makeup service under the re-routed makeup flow control conditions. The staff reviewed the description of the methods used for the 1991 fracture mechanics analysis of the HPI/makeup nozzle thermal sleeve used for both HPI and makeup service. The staff determined that the analysis is acceptable for ensuring thermal sleeve functionality through 2022 because the fracture mechanics model used to analyze the HPI/makeup nozzle thermal sleeve is representative of the fracture behavior of the failed thermal sleeve. Furthermore, as discussed in the August 23, 1991, letter, the stress intensity factor calculations were performed based on a postulated initial flaw size in the unflawed sleeve equal to the initial flaw size observed in the metallurgical analysis of the failed sleeve, and account for flaw growth due to fatigue over twenty 18-month operating cycles. Finally, the staff noted that USAR supplement Commitment No. 23 ensures that all HPI/makeup nozzle thermal sleeves will be replaced prior to entering the period of extended operation (April 2017), which is well before the predicted end-of-life for these sleeves.

LRA Section 4.7.4 stated that the effects of fatigue on the HPI/makeup nozzle thermal sleeve will be managed during the period of extended operation under the Fatigue Monitoring Program (LRA Section B.2.16), in accordance with 10 CFR 54.21(c)(1)(iii). The staff reviewed the applicant's Fatigue Monitoring Program (SER Section 3.0.3.2.6) and determined that this program is not acceptable for managing the effects of aging on the HPI/makeup nozzle thermal sleeve. This determination was made because the applicant's Fatigue Monitoring Program is structured to count transient cycles to ensure that the plant's design-basis transient cycles are not exceeded, thereby ensuring that the ASME Code, Section III cumulative fatigue usage limits are not exceeded. The CUF analyses are evaluated as separate TLAAs in LRA Section 4.3. LRA Section 4.3 CUF analyses do not address the growth of preexisting flaws or postulated flaws. The applicant's Fatigue Monitoring Program is not structured to count transient cycles against any analysis that is based upon the growth of preexisting or postulated flaws.

Based on the above concern, by letter dated April 20, 2011, the staff issued RAI B.2.16-7, requesting that the applicant justify the use of cycle counting, as described in the Fatigue Monitoring Program, for the analysis described in LRA Section 4.7.4, without an update to applicable TS requirements and cycle counting procedures, and without enhancements to the applicable Fatigue Monitoring Program Elements. Note that staff evaluation of this RAI response is described in SER Section 3.0.3.2.6.

By letter dated June 3, 2011, the applicant provided a response to RAI B.2.16-7, indicating that the 10 CFR 54.21(c)(1)(iii) disposition and Fatigue Monitoring Program are no longer used for the LRA Section 4.7.4 TLAA of the HPI/makeup nozzle thermal sleeve. In an enclosure to the RAI response, the applicant provided Amendment 8 to the Davis-Besse LRA. LRA Amendment 8 revised the disposition for the analysis of the HPI/makeup nozzle thermal sleeve in LRA Section 4.7.4 from "10 CFR 54.21(c)(1)(iii)" to "Not a TLAA." The Amendment 8 revision of LRA Section 4.7.4 states that "[b]ased on [the USAR supplement] commitment [to replace the subject thermal sleeve], the HPI/Makeup nozzle thermal sleeves are short-lived (not 40-year) parts and therefore this analysis is not a TLAA." The staff found this disposition not acceptable because aging of the subject thermal sleeve, as discussed in LRA Section 4.7.4, results from an aging mechanism that is time-dependent (i.e., it is dependent on the number of transient cycles incurred), consistent with the definition of a TLAA in 10 CFR 54.3.

By letter dated October 11, 2011, the staff issued RAI 4.7.4-1 requesting that the applicant amend LRA Sections 4.1, 4.7.4, and A.2.7.4 to identify HPI/makeup nozzle thermal sleeve aging as a TLAA. The staff also requested, in RAI 4.7.4-1, that the applicant select an appropriate disposition as required by 10 CFR 54.21(c)(1). The staff noted that if the applicant proposes a

10 CFR 54.21(c)(1)(iii) disposition for this analysis, then the applicant should amend LRA Sections 4.7.4 and A.2.7.4 to propose an appropriate AMP for managing the effects of aging on the HPI/makeup nozzle thermal sleeve. The AMP identified in LRA Sections 4.7.4 and A.2.7.4 for a 10 CFR 54.21(c)(1)(iii) disposition of this analysis should ensure that the effects of aging on the subject thermal sleeve will be appropriately managed for the period of extended operation.

In its response dated October 31, 2011, the applicant provided LRA Amendment 21, which revised several interrelated portions of the LRA. First, the applicant revised LRA Sections 4.1 (Table 4.1-1), 4.7.4, and A.2.7.4 to identify HPI/makeup nozzle thermal sleeve aging as a TLAA. The applicant stated that the effects of aging on the HPI/makeup nozzle thermal sleeve will be managed by the ISI Program during the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii). Additionally, the applicant revised LRA Section B.2.24, "Inservice Inspection Program," to include augmented examination of the HPI/makeup nozzle thermal sleeve. Finally, the applicant revised LRA Table A-1, Commitment No. 23 in the USAR supplement to delete thermal sleeve replacement as a required action under this commitment.

The staff reviewed the LRA revisions provided in LRA Amendment 21 and noted that LRA Sections 4.1 (Table 4.1-1), 4.7.4, and A.2.7.4 were appropriately revised to identify HPI/makeup nozzle thermal sleeve aging as a TLAA. The staff also noted that LRA Sections 4.7.4 and A.2.7.4, as revised, now identify 10 CFR 54.21(c)(1)(iii) as the disposition for this TLAA and state that the ISI Program will manage the effects of aging on the HPI/makeup nozzle thermal sleeve during the period of extended operation. Furthermore, LRA Section 4.7.4 now states the following:

> [a]fter re-routing the makeup flow path through HPI nozzle A-2, the thermal sleeve for nozzle A-2 has since been replaced during the Cycle 13 refueling outage that ended in March 2004. In addition, the ISI Program was revised to require an augmented VT-1 visual examination of the HPI/makeup nozzle thermal sleeve once every other refueling outage commencing with the cycle 15 RFO.

The staff confirmed that the ISI Program, as described in LRA Amendment 21 Section B.2.24, was augmented to require a VT-1 visual examination of the HPI/makeup nozzle thermal sleeve once every other RFO, corresponding to once every 4 calendar years, commencing with the cycle 15 RFO (2008). Finally, the staff noted that Commitment No. 23 in Table A-1 of the USAR supplement was revised to delete thermal sleeve replacement as a required action under this commitment.

Based on its review of the LRA, as amended in LRA Amendment 21, the staff determined that the applicant's proposal to manage the effects of aging for the HPI/makeup nozzle thermal sleeve using the augmented ISI program is acceptable because the affected HPI/makeup nozzle thermal sleeve (HPI nozzle A-2) was replaced in 2004, and the implementation of a VT-1 visual examination of the HPI/makeup nozzle thermal sleeve once every other RFO will provide for detection of cracking prior to thermal sleeve failure.

In a December 8, 2011, telephone conference call, the staff requested that the applicant clarify why Commitment No. 23 was revised to remove the action to replace the HPI/makeup nozzle thermal sleeve. During the teleconference discussion, the applicant stated that, since the HPI/makeup nozzle thermal sleeve TLAA was dispositioned as required by 10 CFR 54.21(c)(1)(iii) using the augmented ISI Program to manage the effects of aging, the previous commitment to replace the thermal sleeve was determined to be no longer applicable.

The applicant noted that the VT-1 examinations every other RFO started during the cycle 15 RFO and will continue during the period of extended operation. The applicant also noted that VT-1 examinations conducted during the cycle 15 RFO found no indications of thermal sleeve cracking. The staff determined that the applicant provided adequate clarification regarding the deletion of the thermal sleeve replacement action from Commitment No. 23. Based on the acceptable LRA revisions provided in LRA Amendment 21, the replacement of the HPI/makeup nozzle (Nozzle A-2) thermal sleeve in 2004, and the clarification provided during the December 8, 2011, teleconference, the staff determined that the applicant's proposal to manage the effects of aging for the HPI/makeup nozzle thermal sleeve using the augmented ISI Program is acceptable. Therefore, the staff's concern described in RAI 4.7.4-1 is resolved.

Based on the above evaluation, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the HPI/makeup nozzle thermal sleeve will be adequately managed by the ISI Program for the period of extended operation. Additionally, the staff finds that the applicant's TLAA meets the acceptance criteria in SRP-LR Section 4.7.2.1 because the effects of aging on the intended function will be adequately managed for the period of extended operation.

4.7.4.3 USAR Supplement

LRA Section A.2.7.4, as revised by LRA Amendment 21, provides the USAR supplement summary description for the TLAA of the HPI/makeup nozzle thermal sleeve. The staff reviewed the applicant's USAR supplement summary description for this TLAA and determined that it is consistent with the TLAA discussed in LRA Amendment 21 Section 4.7.4. The staff also concludes that the information in the USAR is consistent with SRP-LR Section 4.7.3.2. Based on its review of the USAR supplement, the staff concludes that the applicant provided an adequate summary description of its evaluation of the HPI/makeup nozzle thermal sleeve for the period of extended operation, as required by 10 CFR 54.21(d).

4.7.4.4 Conclusion

On the basis of 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 HPI/makeup nozzle thermal sleeve will be adequately managed for the period of extended operation. The staff also concludes that the USAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d) and therefore, is acceptable.

4.7.5 Inservice Inspection—Fracture Mechanics Analyses

The applicant stated that pursuant to 10 CFR 50.55a(g), an ISI Program is required to verify the integrity of ASME Code Class 1, 2, and 3 components. The ASME Code, Section XI, Table IWB-2500-1, requires the use of nondestructive examination techniques to detect and characterize flaws. Flaws detected in Class 1 components during examination are compared to acceptance standards established in the ASME Code, Section XI, IWB-3500. Unacceptable flaws require detailed analyses, repair, or component replacement.

The applicant stated that fracture mechanics analysis requires a prediction of flaw growth for a specific evaluation period. Flaw indications that are determined not to grow beyond acceptance limits during the evaluation period are justified for continued operation. Fracture mechanics analyses performed for the life of the plant are TLAAs that typically involve the same design transient cycle assumptions considered in the CLB.

The applicant performed a search of Davis-Besse ISI reports and docketed correspondence to identify analytical evaluations of ASME Code components. The applicant identified two analyses based, in part, on fracture or fatigue projections, or both, for the current license term. As such, these analyses are identified as TLAAs in LRA Section 4.7.5.

4.7.5.1 Reactor Coolant System Loop 1 Cold Leg Drain Line Weld Overlay Repair

4.7.5.1.1 Summary of Technical Information in the Application

LRA Section 4.7.5.1 describes the full structural weld overlay repair that was performed for an axial indication found on the RCS Loop 1 cold leg drain line nozzle during the Cycle 14 (calendar year 2006) RFO. The applicant stated that the structural weld overlay for the cold leg drain line nozzle was designed consistent with the requirements of ASME Code Case N-504-2, "Alternative Rules for Repair of Class 1, 2, and 3 Austenitic Stainless Steel Piping, Section XI, Division 1," March 12, 1997 and non-mandatory Appendix Q of the ASME Code, Section XI, "Weld Overlay Repair of Classes 1, 2, and 3 Austenitic Stainless Steel Piping Weldments." The applicant stated that the weld overlay design was supplemented by additional design considerations specific to the unique nature of the geometry and materials of the subject cold leg drain line nozzle-to-elbow weld.

The applicant stated that the weld overlay was designed as a full structural overlay that assumes the as-found flaw propagates to extend 100 percent through the pipe wall. Therefore, according to the applicant, an analytical evaluation of the as-found flaw was not required.

The applicant stated that a fatigue analysis was performed for the repaired configuration. According to the applicant, the fatigue analysis conservatively assumed cycles for 60 years at 1.5 times the number of original design cycles. The applicant also stated that this analysis is a TLAA because it is based on a specific number of cycles. The applicant further stated that the Fatigue Monitoring Program manages the effects of fatigue on the RCS drain line weld overlay repair by counting the thermal cycles incurred through the period of extended operation.

Based on the information above, the applicant concluded that the effects of fatigue on the subject weld overlay will be appropriately managed the during the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii).

4.7.5.1.2 Staff Evaluation

The staff reviewed LRA Section 4.7.5.1 on the analysis of the weld overlay repair for the axial flaw found in the RCS Loop 1 cold leg drain line nozzle-to-elbow weld to verify, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the weld overlay 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 4.3.3.1.1.3, which state that the reviewer should verify 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 verify that the applicant identified the appropriate program, as described and evaluated in the GALL Report. Furthermore, the reviewer should also 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 reviewed the information provided by the applicant in LRA Section 4.7.5.1 concerning the weld overlay repair for the axial flaw found in the RCS Loop 1 cold leg drain line nozzle-to-elbow weld during the Cycle 14 RFO. The repair of the drain line using a full structural weld overlay was approved by the staff, as documented in its SE dated October 19, 2006 (ADAMS Accession No. ML062440478). From its review of the 2006 SE, the staff determined that the applicant's description of its weld overlay repair is consistent with the established ASME Code criteria for acceptable weld overlay repairs of an RCS component because the repair was performed in accordance with the criteria specified in ASME Code Case N-504-2 and nonmandatory Appendix Q of the ASME Code, Section XI, and the subject repair was designed as a full structural weld overlay that conservatively assumes that the as-found flaw has propagated 100 percent through the pipe wall.

According to the applicant, the fatigue analysis of the repaired configuration assumed thermal cycles for 60 years of plant operation. The staff needed additional information in order to confirm that the subject weld overlay repair was properly designed and analyzed for 60 years of thermal cycles at 1.5 times the number of original design cycles. Therefore, by letter dated March 17, 2011, the staff issued RAI 4.7.5.1-1 requesting that the applicant provide a reference for the fatigue analysis of the repaired configuration discussed above.

In its response dated April 15, 2011, the applicant provided the requested reference for the design and fatigue analysis of the subject weld overlay repair. The applicant stated that, by letter dated May 22, 2006 (ADAMS Accession No. ML061440282), it submitted a summary of the calculation packages for the weld overlay to support its 10 CFR 50.55a request for alternative to implement the subject weld overlay repair. The fatigue analysis is addressed in Calculation Number DB-06Q-304, "RCS Cold Leg Letdown Line Nozzle Weld Overlay Repair ASME Code, Section III, Evaluation," Revision 1. This calculation summary demonstrates that the fatigue evaluation of the repaired configuration will meet the ASME Code, Section III, Class 1 design acceptance criteria, considering thermal cycles for 60 years of facility operation at 1.5 times the number of original design cycles. The calculation summary also lists several weld overlay design analyses required by ASME Code Case N-504-2, paragraphs F and G, including calculations of the required weld overlay dimensions, finite element models for calculating thermal, mechanical, and residual stresses, and a weld shrinkage analysis. The staff found that the fatigue analysis identified in Calculation Number DB-06Q-304 adequately demonstrated that the subject weld overlay would remain in compliance with the ASME Code, Section III, Class 1 design acceptance criteria for 60 years of operation because the CUF for the weld overlay is less than 1.0. The CUF is based on an assumed number of transient cycles equal to 1.5 times the number of original design cycles, which bounds the number of cycles projected for the period of extended operation.

The staff noted that the applicant credits its Fatigue Monitoring Program as the basis for managing cumulative fatigue damage during the period of extended operation. The applicant's program includes monitoring and tracking the number of critical thermal and pressure transients that are significant contributors to the fatigue usage factor, which involves the systematic counting of transient cycles and the evaluation of operating data to ensure that the allowable cycle limits are not exceeded. The staff also noted that the applicant's program incorporates action limits and acceptance criteria to ensure that corrective actions are taken to prevent the fatigue TLAAs from exceeding their acceptance criteria, and to assure that the fatigue usage resulting from actual operational transients does not exceed the Code design limit of 1.0. Consistent with the recommendation of GALL Report AMP X.M1, the staff noted that the cycle-counting activities in the applicant's Fatigue Monitoring Program are an acceptable approach to manage CUF values for RCPB components and are consistent with

10 CFR 54.21(c)(1)(iii). The staff's evaluation of the applicant's Fatigue Monitoring Program is documented in SER Section 3.0.3.2.6.

The staff noted the calculation summary for the design and analysis of the weld overlay also references an ASME Code, Section XI, crack growth analysis for the original flaw. This crack growth analysis is addressed in Calculation Number DB-06Q-307, "Predicting Fatigue Crack Growth for the DB Unit 1 RCP 1-1 Cold Leg Drain Nozzle With Design Weld Overlay," Revision 0. The Summary of Results for this calculation states, "[c]rack growth is not considered to be a significant factor affecting the weld overlay design based on the compressive stresses present in the nozzle weld due to the presence of the overlay." The staff questioned whether this analysis of fatigue crack growth in the original weld, even if it demonstrates negligible crack growth, should be identified as a TLAA and dispositioned for the period of extended operation, in accordance with 10 CFR 54.21(c)(1). Therefore, in a telephone conference call on June 21, 2012, the staff requested the applicant to provide additional information to address the analysis of crack growth for the period of extended operation. The staff also requested the applicant to appropriately disposition this analysis in accordance with 10 CFR 54.21(c)(1).

By letter dated July 5, 2012, the applicant provided a supplemental response to RAI 4.7.5.1-1, which stated that the crack growth analysis identified in Calculation Number DB-06Q-307 was performed to demonstrate that flaws of the maximum possible initial size "would not grow unacceptably in the nozzle, so as to undermine the basis for the weld overlay." The applicant emphasized that examinations of the nozzle weld did not precisely characterize the through-wall extent or orientation for the flaw, thereby necessitating a full structural weld overlay that assumed the as-found flaw had propagated 100 percent through-wall. However, the applicant stated that supplemental examinations confirmed that the flaw did not extend into the outer two-thirds of the original nozzle weld thickness, thereby indicating a maximum flaw size that is one-third through-wall from the inner surface of the weld. The applicant stated that its analysis considered six different types of flaw propagation paths, based on the maximum initial flaw size of one-third through the original nozzle weld. The applicant further stated that for each of the six types of flaw paths, the maximum and minimum applied stress intensity factors for the flaw, due to transient cycle loading, were determined to be negative, indicating that the flaw is always loaded in compression. Accordingly, the applicant's crack growth analysis projects zero crack growth for all six cases. The applicant stated that the fatigue crack growth analysis was performed based on an assumed number of transient cycles equal to 1.5 times the number of original design cycles, and therefore the analysis is a TLAA that requires disposition for the period of extended operation. The applicant stated that since the assumed number of cycles for the crack growth analysis bounds the 60-year projected number of cycles from LRA Table 4.3-1, the fatigue crack growth analysis remains valid for the period of extended operation. Accordingly, in LRA Amendment 27, the applicant revised LRA Section 4.7.5.1, to reflect this information, and identified the disposition for the fatigue crack growth analysis as 10 CFR 54.21(c)(1)(i), on the basis that the existing analysis of fatigue crack growth remains valid for the period of extended operation.

The staff reviewed the applicant's supplemental response to RAI 4.7.5.1-1 and found it acceptable for addressing the separate TLAA issue associated with crack growth in the original nozzle weld. The staff finds the applicant's determination that no crack growth would be expected for any number of transient cycles acceptable because the maximum and minimum stress intensity factors for a maximum initial flaw size equal to one-third of the original nozzle weld thickness are negative, due to the compressive loading conditions present for the various flaw orientations analyzed. The staff determined that, since this CLB analysis was performed based on an assumed number of cycles (1.5 times the number of original design cycles) that bounds the number of cycles projected for 60 years, the analysis remains valid for the period of extended operation. Therefore, the staff finds the applicant's selection of a 10 CFR 54.21(c)(1)(i) disposition for the fatigue crack growth analysis is appropriate. Based on the applicant's identification of the ASME Code, Section III, cumulative fatigue usage calculation for demonstrating weld overlay design acceptability, as well as its acceptable disposition of the separate TLAA issue associated with crack growth, as discussed above, the staff concern related to RAI 4.7.5.1-1 is resolved.

Based on its review of this TLAA and supporting documentation, the staff found that the applicant adequately demonstrated that fatigue of the subject weld overlay at Davis-Besse will be appropriately managed under the Fatigue Monitoring Program through the end of the period of extended operation. The staff also finds that management of the subject weld overlay under the Fatigue Monitoring Program will ensure compliance with the requirements of the ASME Code, Section III, 10 CFR 50.55a, and the Davis-Besse TS administrative controls through the end of the period of extended operation. The staff also finds that the applicant adequately demonstrated that the analysis of fatigue crack growth for the original flaw will remain valid for the period of extended operation, because no fatigue crack growth ensures that the original flaw will remain acceptable per the requirements of the ASME Code, Section XI, and ASME Code Case N-504-2.

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of cumulative fatigue damage on the structural integrity of the RCS Loop 1 cold leg drain line nozzle weld overlay will be adequately managed for the period of extended operation. The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that its analysis of the effects of fatigue crack growth on the as-found nozzle weld flaw will remain valid for the period of extended operation because no crack growth is projected through 60 years of facility operation. The staff finds that the applicant's TLAA related to the cumulative fatigue damage of the weld overlay meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.3 because the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The staff also finds that the applicant's TLAA related to the growth of the original flaw meets the acceptance criteria in SRP-LR Section 4.7.2.1 because the CLB analysis of flaw growth remains valid for the period of extended operation.

4.7.5.1.3 USAR Supplement

LRA Section A.2.6.1, as amended, provides the USAR supplement summary description for the TLAA of the RCS Loop 1 cold leg drain line weld overlay repair. The staff reviewed the applicant's USAR supplement summary description for this TLAA and determined that it is consistent with the TLAA discussed LRA Section 4.7.5.1, as amended. The staff also concludes that the information in the USAR supplement is consistent with SRP-LR Sections 4.3.3.3 and 4.7.3.2 for the TLAAs of weld overlay cumulative fatigue and fatigue crack growth, respectively. Based on its review of the USAR supplement, the staff concludes that the applicant provided an adequate summary description of its actions to address the RCS Loop 1 cold leg drain line weld overlay repair for the period of extended operation, as required by 10 CFR 54.21(d).

4.7.5.1.4 Conclusion

On the basis of 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 cumulative fatigue on the

RCS Loop 1 cold leg drain line nozzle weld overlay will be adequately managed 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)(i), that the existing fatigue crack analysis of the original flaw in the RCS Loop 1 cold leg drain line nozzle weld overlay will remain valid for the period of extended operation. The staff also concludes that the USAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.5.2 Once-Through Steam Generator 1-2 Flaw Evaluations

4.7.5.2.1 Summary of Technical Information in the Application

LRA Section 4.7.5.2 describes the applicant's TLAA for the OTSG 1-2 flaw evaluations. The applicant stated that many flaw indications were detected in SG 1-2, both in the shell near the steam outlet nozzle and in the shell welds near the lower tubesheet-to-shell juncture during the Cycle 5 RFO (May 1988). According to the applicant, two of the indications in the shell near the steam outlet nozzle were evaluated in accordance with ASME Code, Section XI, IWB-3612 requirements, with the remaining shell indications bounded by those evaluated. The applicant stated that five of the indications in the shell welds near the lower tubesheet-to-shell juncture were evaluated in accordance with ASME Code, Section XI, IWB-3612 requirements, with the remaining shell weld indications bounded by those evaluated.

The applicant stated that an evaluation of fatigue crack growth, based on 240 HU/CD cycles, concluded that there would be only slight crack growth, and the indications were found to be acceptable, in accordance with the ASME Code, Section XI, IWB-3612 analytical acceptance criteria. The applicant also stated that because these analyses are based on a specific number of cycles, they are TLAAs. The applicant further stated that the Fatigue Monitoring Program manages the effects of fatigue on the OTSG flaw evaluations by counting the thermal cycles incurred through the period of extended operation.

Based on the information above, the applicant concluded in the LRA that the effects of fatigue on the OTSG 1-2 flaws will be appropriately managed the during the period of extended operation by the Fatigue Monitoring Program (LRA Section B.2.16), in accordance with 10 CFR 54.21(c)(1)(iii).

4.7.5.2.2 Staff Evaluation

The staff reviewed LRA Section 4.7.5.2 on the OTSG 1-2 flaw evaluation to verify, pursuant to 10 CFR 54.21(c)(1)(iii), that the that the effects of fatigue on the OTSG 1-2 flaws will be adequately managed for the period of extended operation.

The staff reviewed the applicant's TLAA consistent with the review procedures in SRP-LR Section 4.7.3.1.3, which state that the applicant shall propose to manage the aging effects associated with the TLAA using an AMP in the same manner described in the IPA in 10 CFR 54.21(a)(3). SRP-LR Section 4.7.3.1.3 also states that the applicable AMP is reviewed to verify that the effects of aging on the intended functions are adequately managed consistent with the CLB for the period of extended operation.

The staff reviewed the information in LRA Section 4.7.5.2 concerning the ASME Code, Section XI evaluations of flaws discovered in the OTSG shell near the steam outlet nozzle and in the shell welds near the lower tubesheet-to-shell juncture. The staff agreed that the ASME Code, Section XI, evaluations of the detected flaws are TLAAs due to the dependence of the flaw crack growth on fatigue and thermal cycles incurred during the period of extended operation.

The staff issued RAI 4.7.5.2-1 on March 17, 2011. RAI 4.7.5.2-1 consisted of Parts a through g. The applicant responded to all parts of RAI 4.7.5.2-1 by letter dated April 15, 2011.

RAI 4.7.5.2-1, Part a, requested that the applicant state the number of flaw indications that were found that did not pass the initial ASME Code, Section XI, IWB-3500 screening criteria. In its response, the applicant stated that a total of 12 indications were found during the Cycle 5 RFO (1988) for OTSG 1-2 that did not meet the ASME Code, Section XI, IWB-3500 flaw screening criteria for disposition of flaws without further evaluation. The applicant noted that the flaws are in ASME Code Class 2 components; however, the ASME Code, Section XI, IWC-3000 acceptance standards for Class 2 components were in the course of preparation at that time. The applicant stated that, of the 12 indications, 10 are associated with the shell welds near the lower tubesheet-to-shell juncture, and two are associated with the shell-to-steam outlet nozzle welds. The applicant stated that the applicable edition of the ASME Code, Section XI, at that time was the 1977 edition with addenda through 1978 of the ASME Code, Section XI. The staff found the applicant's response acceptable because the applicant provided the necessary information concerning the number of flaws discovered that did not pass the IWB-3500 screening criteria.

RAI 4.7.5.2-1, Part b, requested that the applicant state whether the subject flaws were determined to be the result of service-induced degradation or fabrication defects. In its response, the applicant stated that the subject flaws were analyzed in accordance with IWB-3612, as required by the ASME Code, Section XI, acceptance standards, and all flaws were found to be acceptable for continued service. The staff did not find the applicant's response acceptable for addressing its concern as to whether the subject flaws were determined to be the result of service-induced aging or fabrication because the applicant did not address this issue. Accordingly, this issue is addressed in an October 11, 2011 supplemental RAI (RAI 4.7.5.2-2), which is discussed below.

RAI 4.7.5.2-1, Part c, requested that the applicant state whether the OTSG shell materials with flaws have received successive examinations, in accordance with ASME Code, Section XI, requirements, since May 1988. In its response, the applicant stated that initial re-examination of the subject OTSG shell materials was performed for all flawed regions during the Cycle 6 RFO (1990). The applicant stated that the subject materials were again re-examined during the Cycle 7 outage (1991), with the exception of the W axis longitudinal seam weld intersection with the shell-to-lower tubesheet weld. According to the applicant, these re-examinations met the ASME Code, Section XI, IWC-2420(b), requirements for successive inspections of components with flaws that were accepted for continued service in accordance with IWC-3122.3 requirements for acceptance by analytical evaluation.

The staff found the applicant's response to RAI 4.7.5.2-1, Part c, acceptable because the applicant provided the necessary information concerning successive examinations of the OTSG shell materials with flaws, as required by ASME Code, Section XI. The staff found that the applicant successively examined the subject components in accordance with ASME Code, Section XI, IWC-2420(b) requirements, which require the areas containing flaws or relevant conditions in components accepted for continued service, in accordance with IWC-3122.3 or IWC-3132.3, to be re-examined during the next inspection period listed in the schedule of the Inspection Program of IWC-2400. If the reexaminations reveal that the flaws or relevant conditions are essentially unchanged, IWC-2420(c) allows the component examination schedule to revert to the original schedule of subsequent inspections.

RAI 4.7.5.2-1, Part d, requested that the applicant state when the next inservice examination is scheduled for the OTSG components with flaws. In its response, the applicant stated that the only OTSG flaw location still scheduled for examination during the current (third) 10-year ISI interval is the OTSG 1-2 W/X axis outlet nozzle-to-shell weld, which was scheduled for examination (and in fact was examined) during the Cycle 17 mid-cycle outage (2011). The applicant stated that the OTSG 1-2 X/Y axis outlet nozzle-to-shell weld and lower tubesheet-to-shell weld were examined in the second and first periods of the third 10-year ISI interval, respectively. The staff found the applicant's response acceptable because the applicant provided the necessary information concerning the scheduled examination of the remaining flawed region for the third 10-year ISI interval, as well as information concerning the examinations of the flawed regions already completed for the third 10-year ISI interval. The staff finds that this information demonstrates that the flawed regions are being examined in accordance with requirements of the ASME Code, Section XI, IWC-2000, for the third 10-year ISI interval.

RAI 4.7.5.2-1, Part e, requested that the applicant state whether the dimensions of any of the flaws have increased since discovery in 1988. In its response, the applicant stated that no flaw growth was noted during either the Cycle 6 outage (1990) inspections, where all flawed regions were re-examined, or during the Cycle 7 outage (1991) inspections, where all flawed welds were re-examined except for the W/X axis longitudinal seam weld intersection with the shell-to-lower tubesheet weld. The staff reviewed the applicant's response and noted that the RAI response only stated that no flaw growth was noted during the ASME Code, Section IWC-2420(b)-required successive inspections performed in 1990 and the subsequent inspections performed in 1991. The applicant did not state whether any flaw growth was noted for the subject components from any examinations performed on the flawed regions after 1991. The staff identified the need for supplemental information regarding this response, as described below for RAI 4.7.5.2-2.

RAI 4.7.5.2-1, Part f, requested that the applicant state whether the existing flaw growth analyses for the subject flaws are bounding relative to the projected number of thermal cycles for the period of extended operation. In its response, the applicant stated that the projected flaw growth calculations are based on the design basis assumption of 240 HU/CD cycles, as stated in LRA Section 4.7.5.2. The applicant referred to LRA Table 4.3-1, where the 60-year projection of HU/CD cycles is 128, which is bounded by the assumed 240 cycles used in the subject flaw evaluations. The staff found the applicant's response to be acceptable because the existing flaw growth analyses are based on an assumed 240 thermal cycles, which bounds the number of thermal cycles (128 cycles) that is projected for the period of extended operation.

RAI 4.7.5.2-1, Part g, requested that the applicant provide references for all reports previously submitted to the NRC, which document ASME Code, Section XI, analytical evaluations of the subject flaws. In its response to RAI 4.7.5.2-1, the applicant provided the following references for the subject flaw evaluations:

- B&W Report 32-1172294-00, “Davis-Besse 1 SG Flaw Evaluation,” June 9, 1988
- B&W Report 32-1172294-01, “Davis-Besse 1 SG Flaw Evaluation,” July 18, 1988
- B&W Report 32-1172523-00, “DB-1 SG Flaw Evaluation,” July 18, 1988

The applicant provided a copy of each report in Enclosure D to its RAI response. Therefore, the staff reviewed these reports to determine whether they support operation of the OTSG during the period of extended operation.

The staff determined that the 1988 flaw evaluation reports state that the subject flaws were found to be acceptable, in accordance with the 1977 edition of the ASME Code, Section XI, IWB-3612 analytical acceptance standard. Based on its review, the staff determined that these flaw evaluation reports demonstrate that all of the subject OTSG shell flaws will meet the analytical acceptance criterion specified in the ASME Code, Section XI, IWB-3612, paragraph A for normal (including upset and test) operating conditions during the period of extended operation. Specifically, the flaw evaluation reports demonstrate that, for all of the flaws, the ratio of shell material fracture toughness to the maximum applied stress intensity factor for the limiting normal/upset condition transient is projected to be significantly greater than the square root of ten, as required by the ASME Code, Section XI, IWB-3612, paragraph A. The staff also confirmed that the applied stress intensity factors were calculated using the procedures in the ASME Code, Section XI, Appendix A, and consider projected flaw growth due to fatigue as a result of 240 HU/CD cycles (design cycles), which bounds the number of projected cycles for the period of extended operation (128 HU/CD cycles).

In reviewing the above flaw evaluation reports, the staff determined that the subject flaw evaluations were only performed for normal (including upset and test) operating conditions, as specified in the ASME Code, Section XI, IWB-3612, paragraph A. The staff determined that it needed clarification on the applicant’s evaluation of the subject flaws for emergency and faulted conditions, as required by the 1977 edition of the ASME Code, Section XI, IWB-3612, paragraph B. This request is described below for RAI 4.7.5.2-2.

LRA Section 4.7.5.2 states that the effects of fatigue on the OTSG 1-2 flaws will be managed during the period of extended operation under the Fatigue Monitoring Program (LRA Section B.2.16), in accordance with 10 CFR 54.21(c)(1)(iii). The staff reviewed the applicant’s Fatigue Monitoring Program (SER Section 3.0.3.2.6) and determined that this program is not acceptable for managing the effects of aging on OTSG 1-2 flaw growth. This determination was made because the applicant’s Fatigue Monitoring Program is structured to count transient cycles to ensure that the plant’s design-basis transient cycles are not exceeded, thereby ensuring that the ASME Code, Section III cumulative fatigue usage limits are not exceeded. The CUF analyses are evaluated as separate TLAAs in LRA Section 4.3. The LRA Section 4.3 CUF analyses do not address the growth of preexisting flaws. The applicant’s Fatigue Monitoring Program is not structured to count transient cycles against ASME Code, Section XI, IWB-3612 analyses, which address the growth of preexisting flaws.

Based on the above concern, by letter dated April 20, 2011, the staff issued RAI B.2.16-7, requesting that the applicant justify the use of cycle counting, as described in the Fatigue Monitoring Program, for the analysis described in LRA Section 4.7.5.2, without an update to applicable TS requirements and cycle counting procedures, and without enhancements to the applicable Fatigue Monitoring Program Elements.

By letter dated June 3, 2011, the applicant provided a response to RAI B.2.16-7, indicating that the 10 CFR 54.21(c)(1)(iii) disposition and Fatigue Monitoring Program are no longer used for the LRA Section 4.7.5.2 TLAA of the OTSG 1-2 flaws. In an Enclosure to the RAI response, the applicant provided LRA Amendment 8. LRA Amendment 8 revised the disposition for the analysis of the OTSG 1-2 flaws in LRA Section 4.7.5.2 from "10 CFR 54.21(c)(1)(iii)" to "10 CFR 54.21(c)(1)(i)." LRA Amendment 8 added the following to LRA Section 4.7.5.2:

> Simplified evaluation of fatigue crack growth, based on 240 [HU/CD] cycles, concluded that there would be only slight crack growth, and the indications were found to be acceptable by ASME [Code] Section XI, IWB-3612 standards. Because these analyses are based on a specific number of cycles, they are TLAAs. As shown in LRA Table 4.3-1, the 60-year projected cycles for [HU/CD] are 128 and are bounded by the analyzed number of 240. Therefore, the [SG] flaw growth analyses will remain valid through the period of extended operation.

The staff reviewed the applicant's justification for revising the disposition of the OTSG 1-2 flaw TLAA from 10 CFR 54.21(c)(1)(iii) to 10 CFR 54.21(c)(1)(i) and determined that it is appropriate because the existing flaw growth analyses are based on an assumed 240 thermal cycles, which bounds the number of thermal cycles (128 cycles) that is projected for the period of extended operation.

To address the concerns above regarding the applicant's response to RAI 4.7.5.2-1, Parts b, e, and g, by letter dated October 11, 2011, the staff issued RAI 4.7.5.2-2 requesting the following:

1) Taking into consideration the OTSG shell materials containing the flaws, the secondary side water and steam environment, and the secondary side thermal and pressure stresses to which these shell components are subjected, the staff requested that the applicant state whether any of the surface-breaking indications were believed to have been caused by stress corrosion cracking, or any other service-induced aging effect.

2) For any inservice examinations performed on the flawed regions of the OTSG shell after 1991, in particular the examinations performed for the OTSG X/Y axis outlet nozzle to shell weld and the lower tubesheet to shell weld during the first and second periods of the third 10-year ISI interval, the staff requested that the applicant state whether these examinations detected any increase in the flaw dimensions, relative to the 1988 flaw dimensions. (The staff notes that any measured increase in flaw dimensions could possibly invalidate the analyses performed in the 1988 flaw evaluation reports.)

3) The staff requested that the applicant state whether the subject flaws were analyzed for emergency and faulted conditions, as required by the ASME Code, Section XI, IWB-3612, paragraph B. If the subject flaws were analyzed for emergency and faulted conditions, as required by IWB-3612, paragraph B, the staff requested that the applicant provide the flaw analyses for these conditions, or explain how the IWB-3612, paragraph A analyses, as documented in the 1988 flaw evaluation reports, for normal, upset, and test conditions, would bound the flaw analyses for emergency and faulted conditions. If the subject flaws were not analyzed for emergency and faulted conditions, the staff requested that the applicant provide these analyses, as required by IWB-3612, paragraph B.

By letter dated November 23, 2011, the applicant provided its response to RAI 4.7.5.2-2, which addressed the three issues identified above.

The applicant's response to RAI 4.7.5.2-2, Part 1, stated that the OTSG shell material containing the flaws is carbon steel with an environment of treated water (liquid and steam phases). The applicant stated that the aging management review (AMR) of the OTSG components did not identify stress corrosion cracking as an aging management concern for the SG shell. The applicant also stated that stress corrosion cracking is an applicable aging effect for carbon steel exposed to treated water only if there is a potential for microbiologically-influenced corrosion (MIC) contamination, pH less than 10.5, temperature less than 210 °F, and use of nitrite corrosion inhibitor. The applicant referenced EPRI Report 1010639, “Non-Class 1 Mechanical Implementation Guidance and Mechanical Tools,” Revision 4, as the basis for its determination. Based on a review of plant-specific operating experience, the applicant identified instances of MIC only for open cycle cooling water systems. The applicant stated that MIC is not an age-related concern for the OTSG shell operating in a treated water environment at Davis-Besse. The applicant therefore concluded that the OTSG flaws were not caused by stress corrosion cracking.

The applicant also addressed whether the subject flaws could have been caused by fatigue. The applicant stated that the AMR addressed cracking due to fatigue as an aging effect requiring further evaluation and noted that carbon steel above 220 °F is susceptible to cracking due to fatigue. The applicant noted that the ASME Code, Section III, requires calculation of cumulative usage factors (CUFs) and states that CUFs shall be less than 1.0. The applicant stated that the OTSGs were analyzed for fatigue, and the CUFs for the limiting primary and secondary side OTSG locations, which were calculated based on design transients, are less than 1.0, based on the projected design cycles. The applicant pointed to LRA Table 4.3-1, “60-Year Projected Cycles,” and noted that the accrued cycles as of February 19, 2008 were less than the design cycles. The applicant therefore concluded that the OTSG flaws were not caused by cracking due to fatigue.

The applicant stated that the AMR also identified cracking due to the growth of preexisting flaws as an aging effect requiring management for the carbon steel components of the OTSGs that are exposed to the treated water environment. The applicant stated that components fabricated in accordance with the ASME Code are presumed to contain material and fabrication flaws whose sizes and character are below the detection threshold of the examination method employed, or less than the acceptance standards. According to the applicant, the presence of such flaws led to the recognition that these flaws might grow in size as a consequence of the loadings imposed on the component during the service lifetime.

The applicant stated that, based on its determination that the subject flaws were not caused by stress corrosion cracking or fatigue, it is believed that the subject flaws are pre-service flaws that were below the detection threshold of the examination method employed during fabrication of OTSG 1-2.

The staff found the applicant’s response to RAI 4.7.5.2-2, Part 1, acceptable because the applicant provided sufficient evidence for the staff to determine that the subject flaws were likely not caused by service-induced aging, although potential accelerating effects of reactor water environment on fatigue cracking cannot be ruled out. Specifically, based on its review of the applicant’s response to RAI 4.7.5.2-2, Part 1, the staff found that the applicant’s AMR results (LRA Table 3.1.2-4) for the OTSGs correctly identified that stress corrosion cracking is not an aging management concern for the carbon steel shell material because this aging mechanism would not be expected to occur for the carbon steel shell in a secondary side treated water environment, where MIC is not possible. The staff confirmed that the GALL Report, Revision 2, AMR results also do not identify stress corrosion cracking as an aging management concern for the OTSG shell material. With respect to cracking due to fatigue, the staff noted that LRA Table 3.1.2-4 AMR for the SGs did identify cracking due to fatigue as an aging effect applicable to the shell. However, consistent with the applicant’s response to RAI 4.7.5.2-2, Part 1, the staff confirmed that the CUFs for the limiting primary and secondary side OTSG locations are less than 1.0 (LRA Section 4.3.2.2.6.1), based on the design transients and projected design cycles. The staff noted that the total accumulated cycles as of February 19, 2008, are far less than the design cycles. Since the flaws were discovered in 1988, the staff determined that they are unlikely to have been caused by fatigue because the cycles incurred through that time were less than the cycles incurred through February 19, 2008, and therefore, were well bounded by the design cycles. The staff determined that no other known aging effects could have caused the subject flaws. Therefore, based on the above, the staff determined that the applicant provided sufficient information to conclude that service-induced aging likely did not cause the subject flaws.

The applicant's response to RAI 4.7.5.2-2, Part 2, addressed whether recent examinations detected any increase in the flaw dimensions. This response stated that examinations of the flawed OTSG shell regions were performed during the third 10-year ISI interval, and none of these examinations detected any flaw growth relative to the 1988 flaw dimensions. This includes the most recent examination of the OTSG 1-2 W/X axis outlet nozzle to shell weld, which was completed during the October 2011 midcycle outage. The staff found the applicant's response acceptable for resolving its concern because the applicant provided information confirming that the flaws have exhibited no measurable increase in size since initial detection.

The applicant's response to RAI 4.7.5.2-2, Part 3, stated that the subject OTSG shell flaws were not previously analyzed for emergency and faulted conditions, as required by the ASME Code, Section XI, IWB-3612, paragraph B. Accordingly, the applicant stated that the subject flaw evaluation reports have been revised to address the analysis of the subject flaws under emergency/faulted conditions, as required by the ASME Code, Section XI, IWB-3612, paragraph B. The applicant concluded that the flawed OTSG shell materials would remain acceptable for continued service during the period of extended operation, in accordance with the ASME Code, Section XI, IWB-3612, paragraph B.

Enclosures C and D of the applicant's response to RAI 4.7.5.2-2 included the revised flaw evaluation reports, which document the analytical evaluations of the subject flaws for both normal/upset and emergency/faulted conditions. The staff reviewed the revised flaw evaluation reports provided in Enclosures C and D. Based on its review, the staff determined that the revised flaw evaluation reports demonstrate that all of the subject flaws will meet the analytical acceptance criterion specified in the ASME Code, Section XI, IWB-3612, paragraph B for emergency/faulted conditions during the period of extended operation. Specifically, the revised flaw evaluation reports demonstrate that, for all of the subject flaws, the ratio of OTSG shell material fracture toughness to the maximum applied stress intensity factor due to applied loads during the limiting emergency transient is projected to be significantly greater than the square root of two, as required by the ASME Code, Section XI, IWB-3612, paragraph B. The staff also noted that the original flaw evaluations for normal/upset conditions were not changed in the revisions provided in Enclosures C and D. Therefore, the staff determined that the applicant's response to RAI 4.7.5.2-2, Part 3, is acceptable because the applicant demonstrated that all of the subject flaws will remain in compliance with analytical acceptance criteria specified in the ASME Code, Section XI, IWB-3612, paragraphs A and B for the period of extended operation.

Based on its evaluation, the staff determined that the applicant's responses to all parts of RAIs 4.7.5.2-1 and 4.7.5.2-2 are acceptable as described above, and the staff's concerns described in these RAIs are resolved.

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses of the OTSG 1-2 shell flaws remain valid for the period of extended operation. Additionally, the staff finds that the applicant's TLAA meets the acceptance criteria in SRP-LR Section 4.7.2.1 because the analyses of the OTSG 1-2 shell flaws remain valid for the period of extended operation

4.7.5.2.3 USAR Supplement

As revised in LRA Amendment 8 by letter dated June 3, 2011, LRA Section A.2.6.2 provides the USAR supplement summary description for the TLAA of the OTSG 1-2 flaw evaluations. LRA Amendment 8 revised the disposition for the analysis of the OTSG 1-2 flaws in LRA Section A.2.6.2 from 10 CFR 54.21(c)(1)(iii) to 10 CFR 54.21(c)(1)(i), consistent with the revised disposition identified in LRA Amendment 8, Section 4.7.5.2. The staff reviewed the applicant's amended USAR supplement summary description for this TLAA and determined that it is consistent with the TLAA discussed in LRA Section 4.7.5.2, as amended. The staff also concludes that the information in the USAR supplement is consistent with SRP-LR

Section 4.7.3.2. Based on its review of the USAR supplement, the staff concludes that the applicant provided an adequate summary description of its actions to address the OTSG 1-2 flaw evaluations for the period of extended operation, as required by 10 CFR 54.21(d).

4.7.5.2.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the existing analyses for the OTSG 1-2 flaws remain valid for the period of extended operation. The staff also concludes that the USAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.6 ASME Code Case N-481 Evaluation

4.7.6.1 Summary of Technical Information in the Application

By letter dated August 17, 2011, in response to RAI 4.1-2, the applicant provided LRA Amendment 13 to, in part, include LRA Section 4.7.6, which describes the TLAA related to the fatigue analysis associated with ASME Code Case N-481 of the RCP casings.

The applicant stated that it has invoked the use of ASME Code Case N-481 for its stainless steel RCP casings. The applicant stated that the staff has accepted Code Case N-481 for use in ISI inspection programs. The applicant also stated that this code case allows the replacement of volumetric examinations of primary loop pump casings with a fracture mechanics-based evaluation supplemented by specific visual examinations, and includes a fatigue crack growth analysis. The applicant further stated that the code case evaluation includes two areas that potentially involve time-dependency assumptions, the fracture toughness property assumed in the analysis for the RCP CASS material of fabrication and the fatigue flaw growth analysis for the casings. The applicant stated that the fracture toughness parameter is not time-dependent because the analysis used a lower-bound fracture toughness value of 139 ksi-in$^{1/2}$ that bounds the saturated fracture toughness of the CASS material.

The applicant further stated that the fatigue crack growth analysis, which is based on design cycles for a 40-year plant life, is a TLAA requiring disposition for license renewal. The applicant stated that the fatigue flaw growth analysis conservatively assumed 2,000 transient cycles, and the growth of postulated flaw remained below the critical crack size. The applicant stated that, since the 2,000 cycles assumed in the analysis bounds the number of 60-year projected cycles assumed for the transients in LRA Table 4.3-1, the ASME Code Case N-481 analysis of the RCP casings is acceptable, in accordance with 10 CFR 54.21(c)(1)(i), and remains valid for the period of extended operation.

4.7.6.2 Staff Evaluation

The staff reviewed new LRA Section 4.7.6 provided in LRA Amendment 13, on the fatigue analysis associated with the ASME Code Case N-481 evaluation of the RCP casings, to verify, 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 4.7.3.1.1, which state that the reviewer should verify that the existing analysis remains valid for the period of extended operation.

During its review of the LRA, the staff noted that LRA Section 4.3.2.2.4 describes the fatigue TLAA for the RCP casings and states that the casings were analyzed for fatigue by the OEM to meet the requirements of the ASME Code, Section III, 1968 edition through winter 1968 addenda. LRA Table 3.1.1, item 3.1.1-55, states that the aging of these pump casings will be managed by its ISI Program and invokes the use of ASME Code Case N-481. The staff noted that its endorsement of ASME Code Case N-481, as referenced in RG 1.147, Revision 14, and permitted in 10 CFR 50.55a(b), requires the performance of a crack-growth evaluation on the RCP casing in order to support justification of the alternative visual inspection requirements for the pump casing. The staff also noted that the LRA did not identify a TLAA disposition of ASME Code Case N-481. As described in SER Section 4.3.2.2, the staff issued RAI 4.1-2 by letter dated May 2, 2011, requesting the applicant to justify the absence of TLAA identification in the LRA for the RCP casing regarding the application of Code Case N-481.

The applicant's response dated June 17, 2011, described the analysis to justify use of ASME Code Case N-481. The applicant revised its response to RAI 4.1-2 by letter dated August 17, 2011, and provided LRA Amendment No. 13, which included new LRA Section 4.7.6.

The applicant's revised RAI response identified that Topical Report No. SIR-99-040, Revision 1, "ASME Code N-481 Evaluation of Davis Besse Reactor Coolant Pumps" (ADAMS Accession No. ML011200090), was used to support the alternative visual examination basis. FENOC submitted this analysis to the NRC by letter dated April 23, 2001. The applicant's revised response to RAI 4.1-2 identified two potential time-dependencies in SIR-99-040 Revision 1: (1) the fracture toughness property of the CASS material that was used to fabricate the RCP casing, and (2) the time-dependency on the fatigue flaw growth analysis that was performed.

For the fracture toughness property, the applicant's response stated that the fracture toughness of the CASS material is not time-dependent because the analysis assumed a lower-bound fracture toughness that bounded the fracture toughness of the CASS material under assumed saturated thermal aged conditions. The staff noted that the applicant's basis may be predicated on thermal aging data that are not up to date or conservative when compared to the most recent data for the industry.

By letter dated October 21, 2011, the applicant supplemented its response to RAI 4.1-2 and LRA Section 4.7.6 by comparing the thermal aging data used in the SIR-99-040 to the most-recent industry data in two NUREG reports. The applicant stated that the saturation fracture toughness value in SIR-99-040, Revision 1, was determined using the methodology outlined in NUREG/CP-0119, Volume 2, pages 151–178, "Proceedings of the USNRC, 19th Water Reactor Safety Information Meeting held at Bethesda, MD, October 28-30, 1991," considering all available certified material test report for the base material and welds of the Davis-Besse reactor coolant pump casings. The applicant also stated that the minimum saturation fracture toughness value has since been re-calculated using NUREG/CR-4513, Revision 1, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," which is based on the most recent published data on this subject matter. The applicant confirmed that using the methodology and correlation in this latest NUREG resulted in the same minimum saturation fracture toughness value for the pump casings as that used in SIR-99-040, Revision 1.

The applicant stated that the fracture toughness for thermal aging of welds has also been presented in NUREG/CR-6428, "Effects of Thermal Aging on Fracture Toughness and Charpy-Impact Strength of Stainless Steel Pipe Welds." Using a conservative J_{Ic} fracture toughness value of 40 kJ/m^2 based in this NUREG/CR report, the applicant determined that the corresponding K_{Ic} value is 80 ksi-in$^{1/2}$. The applicant concluded that the applied stress intensity factors, calculated in Table 4-5 of SIR-99-040, Revision 1, are bounded by the K_{Ic} value of 80 ksi-in$^{1/2}$, and thus the conclusion in SIR-99-040, Revision 1, remains valid. The staff reviewed NUREG/CR-6428 and confirmed that the J_{Ic} value of 40 kJ/m^2 represents an acceptable lower bound fracture toughness value based on the data in the NUREG report. The staff also confirmed that the stress intensity factors listed in SIR-99-040, Revision 1, for normal and upset conditions do not exceed the values of 80 ksi-in$^{1/2}$. Thus, the staff concluded that the applicant's calculation remain valid when comparing to the most-recent available data in the NUREG/CR-6428 and that there is not any time-dependency for the lower bound fracture toughness value that was assumed for the pump casing material. Thus the potential TLAA related to the fracture toughness property aspect of the ASME Code Case N-481 evaluation is not a TLAA in accordance with 10 CFR 54.3, because it does not involve time-limited assumptions defined by the current operating term.

Regarding the potential time dependency of the fatigue flaw growth analysis discussed in SIR-99-040 (Revision 1), the staff noted that the significant transients considered in the fatigue crack growth analysis are HU/CD, loss of secondary pressure, hydrotest, and leak test because these transients are associated with very high pressure and temperature changes. The staff confirmed that the fatigue flaw growth analysis in SIR-99-040, Revision 1, was analyzed to 2,000 cycles for a 40-year plant life. The staff also confirmed that LRA Table 4.3-1 indicates that the total number of cycles projected for these transients through 60 years of operations would be less than the value of 2,000 that was assumed in the fatigue crack growth analysis in the ASME Code Case N-481 evaluation. The staff's evaluation of the applicant's projection methodology for design transients is documented in SER Section 4.3.1.2.

The staff finds that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue flaw growth analysis associated with ASME Code Case N-481 for RCP casings is acceptable because the existing analysis in the CLB will remain valid for the period of extended operation. Additionally, the staff finds that the analysis meets the acceptance criteria in SRP-LR Section 4.7.2.1 because the 60-year projected number of cycles is less than the number assumed in the fatigue flaw growth analysis.

4.7.6.3 USAR Supplement

LRA Section A.2.7.5, as amended by letters dated August 17 and October 21, 2011, provides the USAR supplement summarizing the analysis of RCP casing associated with ASME Code Case N-481. The staff reviewed LRA Section A.2.7.5 consistent with the review procedures in SRP-LR Section 4.7.3.2, which state that the reviewer verifies that the applicant has provided information to be included in the USAR supplement that includes a summary description of the evaluation of the TLAA.

Based on its review of the amended USAR supplement, the staff finds the supplement meets the acceptance criteria in SRP-LR Section 4.7.2.2. Additionally, the staff determined that the applicant provided an adequate summary description associated with the TLAA for ASME Code Case N-481 of the RCP casings regarding the basis for determining that the applicant has made the demonstration required by 10 CFR 54.21(c)(1).

4.7.6.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue flaw growth analysis for the ASME Code Case N-481 evaluation of the RCP casings remains valid for the period of extended operation. The staff also concludes that the USAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.7 Crane Load Cycles

4.7.7.1 Summary of Technical Information in the Application

By letter dated October 7, 2011, the applicant provided LRA Amendment No. 19 to include new LRA Sections 4.7.7 and A.2.7.6, both titled “Crane Load Cycles,” to address the disposition of the TLAA associated with crane load cycles. This amendment was initiated in response to FENOC-generated OI Number OIN-378, which resulted from the NRC Region III implementation of Inspection Procedure IP-71002, “License Renewal Inspection,” during the week of August 22, 2011, to address an inspector request regarding crane cycles.

LRA Section 4.7.7 describes the applicant’s TLAA for crane load cycles. The applicant stated that the load cycle limits for cranes were identified as a potential TLAA, and the following cranes are in the scope of license renewal and have been identified as having a TLAA, which requires evaluation for 60 years:

- containment polar crane (including auxiliary hoist)
- reactor service crane
- spent fuel shipping cask crane (including auxiliary hoist)
- intake structure gantry crane

The applicant also stated that these cranes are designed in accordance with Bechtel design specifications, which require that the cranes be designed in accordance with the minimum requirements for Class A cranes as stated in Crane Manufacturers Association of America (CMAA) Specification 70 for Electric Overhead Traveling Cranes.

The applicant dispositioned the TLAA for crane load cycles in accordance with 10 CFR 54.21(c)(1)(i), that these analyses remain valid during the period of extended operation.

4.7.7.2 Staff Evaluation

During the NRC staff's inspection per Inspection Procedure IP-71002, "License Renewal Inspection," during the week of August 22, 2011, the staff raised a concern regarding the absence of discussion of fatigue TLAAs for steel cranes in LRA Section 4, as documented in Section 3.11 of Inspection Report 05000346/2011012, dated October 7, 2011. By letter dated October 7, 2011, the applicant revised the LRA to include new Sections 4.7.7 and A.2.7.6, both titled "Crane Load Cycles," to address the disposition of the TLAA associated with crane load cycles.

The staff reviewed LRA Section 4.7.7 on crane load cycles to verify, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses remains valid during the period of extended operation. This review was performed 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.

For the containment polar crane (including auxiliary hoist), the applicant stated that the rate of occurrence using this crane is based on refueling RFOs, mid-cycle outages with core off load, and the final core off load at the end of 60 years of operation, and a total of 22,000 cycles is expected through the period of extended operation. The staff noted that an additional 500 cycles was estimated for the pre-operational construction period, which is included in the estimate of 22,000 cycles.

For the reactor service crane, the applicant stated that the rate of occurrence is based on RFOs, mid-cycle outages with core off load, and the final core off load at the end of 60 years of operation, and a total of 8,000 cycles is expected through the period of extended operation. The staff noted that an additional 500 cycles was estimated for the pre-operational construction period, which is included in the estimate of 8,000 cycles.

For the spent fuel shipping cask crane (including auxiliary hoist), the applicant stated that the rate of occurrence is based on RFOs, mid-cycle outages with core off load, and the final core off load at the end of 60 years of operation, and a total of 18,000 cycles is expected through the period of extended operation. The staff noted that an additional 500 cycles and 3,600 cycles were estimated for crane usage during the pre-operational construction period and during non-outage periods, respectively, which are included in the estimate of 18,000 cycles.

For the intake structure gantry crane, the applicant stated the rate of occurrence is based on crane usage throughout the calendar year at 20 cycles per year and a total of 1,700 cycles is expected through the period of extended operation. The staff noted that an additional 500 cycles are estimated for the pre-operational construction period, which are included in the estimate of 1,700 cycles.

The staff reviewed CMAA No. 70 and confirmed that Service Class A cranes are designed for up to 100,000 load cycles. The staff finds that the applicant conservatively accounted for crane usage during the pre-operational construction period and during non-outage periods for the spent fuel shipping cask crane (including auxiliary hoist). The staff noted that the applicant's estimated use levels of these Service Class A cranes are based on operations that are routine and predictable, which occur during RFOs, mid-cycle outages, core off loads, and normal operation; therefore, the staff finds the applicant's estimates for its crane usage to be reasonable. For the containment polar crane (including auxiliary hoist), reactor service crane, spent fuel shipping cask crane (including auxiliary hoist) and intake structure gantry crane, the staff noted that the applicant's estimates for crane usage through the period of extended operation were no more than 22 percent of the 100,000 design load cycles specified in CMAA No.70. Additionally, the staff finds that there is a sufficient margin to account for unexpected crane use through the period of extended operation.

The staff finds the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses of load cycles for those cranes discussed above 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 estimated usage of the cranes described above is significantly less than the 100,000 design load cycles specified in CMAA No. 70 for Service Class A cranes, and these analyses bound the crane usage through the period of extended operation.

4.7.7.3 USAR Supplement

LRA Section A.2.7.6 provides the USAR supplement summarizing the TLAA for crane load cycles of the containment polar crane (including auxiliary hoist), reactor service crane, spent fuel shipping cask crane (including auxiliary hoist), and intake structure gantry crane. The staff reviewed LRA Section A.2.7.6 consistent with the review procedures in SRP-LR Section 4.7.3.2, which state that the reviewer verifies that the applicant has provided information to be included in the USAR supplement that includes a summary description of the evaluation of the TLAA.

Based on its review of the USAR supplement, the staff finds it meets the acceptance criteria in SRP-LR Section 4.7.2.2. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the TLAA for crane load cycles, as required by 10 CFR 54.21(d).

4.7.7.4 Conclusion

On the basis of its review, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses for the crane load cycles of the containment polar crane (including auxiliary hoist), reactor service crane, spent fuel shipping cask crane (including auxiliary hoist), and intake structure gantry crane remain valid for the period of extended operation. The staff also concludes that the USAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.8 Conclusion

The staff reviewed the information in LRA Section 4, "Time-Limited Aging Analyses." On the basis of its review, the staff concludes that the applicant provided a sufficient list of TLAAs, as defined in 10 CFR 54.3, and that the applicant has demonstrated the following:

- The TLAAs will remain valid for the period of extended operation, as required by 10 CFR 54.21(c)(1)(i).
- The TLAAs have been projected to the end of the period of extended operation, as required by 10 CFR 54.21(c)(1)(ii).
- The effects of aging on 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 USAR supplement for the TLAAs and finds that the supplement contains descriptions of the TLAAs sufficient to satisfy the requirements of 10 CFR 54.21(d). In addition, the staff concludes, as required by 10 CFR 54.21(c)(2), that no plant-specific, TLAA-based exemptions are in effect.

With regard to these matters, the staff concludes that there is reasonable assurance that the activities authorized by the renewed licenses will continue to be conducted in accordance with the CLB. Additionally, any changes made to the CLB, in order to comply with 10 CFR 54.29(a), are in accordance with the Atomic Energy Act of 1954, as amended, and NRC regulations.

SECTION 5

REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

In accordance with Title 10, Part 54, of the *Code of Federal Regulations*, the Advisory Committee on Reactor Safeguards (ACRS) will review the license renewal application (LRA) for Davis-Besse Nuclear power Station (Davis-Besse). The ACRS Subcommittee on Plant License Renewal will continue its detailed review of the LRA after this safety evaluation report (SER) is issued. FirstEnergy Nuclear Operating Company (FENOC or the applicant) and the staff of the U.S. Nuclear Regulatory Commission (NRC or the staff) will meet with the Subcommittee and the Full Committee to discuss issues associated with the review of the LRA.

After the ACRS completes its review of the LRA and SER, the Full Committee will issue a report discussing the results of the review. An update to this SER will include the ACRS report and the staff's response to any issues and concerns reported.

SECTION 6

CONCLUSION

The staff of the U.S. Nuclear Regulatory Commission (NRC or the staff) reviewed the license renewal application (LRA) for Davis-Besse Nuclear Power Station (Davis-Besse) in accordance with NRC regulations and NUREG-1800, Revision 2, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” (SRP-LR), dated December 2010. Title 10, Section 54.29, of the *Code of Federal Regulations* (10 CFR 54.29) sets the standards for issuance of a renewed license.

On the basis of its review of the LRA, the staff determines that the requirements of 10 CFR 54.29(a) have been met.

The staff noted that any requirements of 10 CFR Part 51, Subpart A, will be documented in a draft supplement to NUREG-1437, “Generic Environmental Impact Statement for License Renewal of Nuclear Plants Regarding Davis-Besse Nuclear Power Station,” to be issued at a later date.

APPENDIX A

DAVIS-BESSE NUCLEAR POWER STATION LICENSE RENEWAL COMMITMENTS

During the review of the Davis-Besse Nuclear Power Station (Davis-Besse) license renewal application (LRA) by the staff of the U.S. Nuclear Regulatory Commission (NRC or the staff), FirstEnergy Nuclear Operation Company (FENOC or the applicant) made commitments related to aging management programs (AMPs) to manage the aging effects of structures and components (SCs) prior to the period of extended operation. The following table lists these commitments, as well as the implementation schedules and the sources for each commitment.

Table A-1. Davis-Besse License Renewal Commitments

Item Number	Commitment	Updated Safety Analysis Report (USAR) Supplement Section No// Comments	Implementation Schedule	Source
1	Enhance the Aboveground Steel Tanks Inspection Program to do the following: • Include a volumetric examination of tank bottoms to detect evidence of loss of material due to crevice, general, or pitting corrosion, or to confirm a lack thereof. • Establish the examination technique, the inspection locations, and the acceptance criteria for the examination of the tank bottoms. • Require that unacceptable inspection results be entered into the FENOC Corrective Action Program. • Ensure that the volumetric examination of the tank bottoms will be performed within 5 years after entering the period of extended operation and that additional opportunistic tank bottom inspections will be performed whenever the tanks are drained.	A.1.2 B.2.2 Response to NRC RAI B.2.2-1 from NRC letter dated April 20, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letters L-11-153 and L-13-160
2	Implement the Boral® Monitoring Program as described in LRA Section B.2.5.	A.1.5 B.2.5 and Response to NRC RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letter L-13-160
3	Enhance the Buried Piping and Tanks Inspection Program to do the following: • Add bolting for buried Fire Protection System piping and the emergency diesel fuel oil storage tanks (DB-T153-1, DB-T153-2) to the scope of the program. • Conduct annual ground potential surveys of the cathodic protection system using the acceptance criteria listed in NACE RP0285 2002 and NACE SP0169-2007. • Monitor cathodic protection voltage and current monthly to determine the effectiveness of cathodic protection systems and, thereby, the effectiveness of corrosion mitigation. • Trend voltage, current, and ground potential readings and evaluate for adverse changes.	A.1.7 B.2.7 and Response to NRC RAI B.2.7-1 from NRC letter dated April 20, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letters L-11-153 and L-13-160

<table>
<tr><th>Item Number</th><th>Commitment</th><th>Updated Safety Analysis Report (USAR) Supplement Section No// Comments</th><th>Implementation Schedule</th><th>Source</th></tr>
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- Require that the activity of the jockey fire pump or equivalent parameter be monitored on at least a monthly interval.
- Conduct a flow test by the end of the next refueling outage when unexplained changes in jockey pump activity are observed.
- Require that the directed buried pipe inspection locations be selected based on risk.
- Require that the minimum number of buried in-scope piping inspections during the 30–40, 40–50, and 50–60 year operating period is one steel safety-related piping segment and one steel piping segment containing hazardous material.
- Perform the directed buried steel pipe and tank inspections each 10-year interval based upon the following table. Each inspection will have a minimum of 10 feet of piping inspected.

<table>
<tr><th>Preventive Actions</th><th># of inspections of safety related piping or tanks</th><th># of Hazmat inspections or % of pipe length</th></tr>
<tr><td>A</td><td>1 (Note 2)</td><td>1 (Note 2)</td></tr>
<tr><td>B</td><td>1</td><td>2%</td></tr>
<tr><td>C</td><td>4</td><td>5%</td></tr>
<tr><td>D</td><td>8</td><td>10%</td></tr>
</table>

Note 1: Preventive actions are categorized as follows:

A. Cathodic protection, in accordance with NACE SP0169-2007 or NACE RP0285-2002, was installed for at least 5 years prior to entering the period of extended operation and was operational for 90% of the time during that 5 years or cathodic protection was operational for 90% of the time since the last inspection conducted under this program.

B. Cathodic protection, in accordance with NACE SP0169-2007 or NACE RP0285-2002, was installed for less than 5 years prior to entering the period of extended operation or was operational for less than 90% of the time during that 5 years or cathodic inspection was operational for less than 90% of the time since the last inspection conducted under this program.

C. Protective coatings are in place and no mechanical coating damage due to the backfill, but cathodic protection is not provided or not in accordance with criteria A or B and the period of extended operation has not been entered.
</td><td></td><td></td><td></td></tr>
</table>

Item Number	Commitment	Updated Safety Analysis Report (USAR) Supplement Section No// Comments	Implementation Schedule	Source
	D. Criteria of A, B, and C are not met. Note 2: Only one inspection is required for piping which is both safety-related and contains hazardous material. • Require that the EDG Fuel Oil Storage Tanks (DB-T153-1 and DB-T153-2) be inspected prior to entering the period of extended operation. The inspection will be either a visual inspection of at least 25% of each tank and include at least some portion of the tank top and bottom or, an internal inspection consisting of UT measurements with at least one measurement per square foot of the surface of the tanks. These inspections are not required if it is demonstrated that the tanks are cathodically protected in accordance with NACE SP0169-2007 or NACE RP0285-2002. • Require that a visual and volumetric inspection of the underground piping within the borated water piping trench will be performed during each 10-year period beginning no sooner than 10 years prior to the entry into the period of extended operation. • Require that if adverse indications are detected, additional buried in-scope piping inspections be performed in order to provide reasonable assurance of the integrity of buried piping. • Base the selection of components to be examined on previous examination results, trending, risk ranking, and areas of cathodic protection failures or gaps, if applicable. • Continue additional sampling until reasonable assurance of the integrity of buried piping is provided. • Require that an inspection of buried Fire Protection System bolting will be performed when the bolting becomes accessible during opportunistic or focused inspections. • Require that the inspections of buried piping be conducted using visual (VT-3 or equivalent) inspection methods. Excavation shall be a minimum of 10 linear feet of piping, with all surfaces of the pipe exposed.			

Item Number	Commitment	Updated Safety Analysis Report (USAR) Supplement Section No// Comments	Implementation Schedule	Source
4	Implement the Collection, Drainage, and Treatment Components Inspection Program, as described in LRA Section B.2.9.	A.1.9 B.2.9 and Response to NRC RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letter L-13-160
5	Implement the Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Inspection as described in LRA Section B.2.11. Enhance the Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Inspection to include high-voltage connections to confirm the absence of aging effects for metallic electrical connections.	A.1.11 B.2.11 Response to NRC RAI 3.6-3 from NRC letter dated April 5, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letters L-11-134 and L-13-160
6	Implement the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program, as described in LRA Section B.2.12.	A.1.12 B.2.12 and Response to NRC RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letter L-13-160
7	Implement the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program, as described in LRA Section B.2.13.	A.1.13 B.2.13 and Response to NRC RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letter L-13-160
8	Enhance the External Surfaces Monitoring Program to do the following: • Add systems that credit the program for license renewal but do not have Maintenance Rule intended functions to the scope of the program. • Perform opportunistic inspections of surfaces that are inaccessible or not readily visible during normal plant operations or refueling outages, such as	A.1.15 B.2.15 and Responses to NRC RAIs 3.3.2.2.5-1 and B.2.2-2 from	Prior to October 22, 2016	LRA and FENOC letters L-11-153, L-11-166,

Item Number	Commitment	Updated Safety Analysis Report (USAR) Supplement Section No// Comments	Implementation Schedule	Source
	surfaces that are insulated. Surfaces that are accessible will be inspected at a frequency not to exceed one refueling cycle. • Perform, in conjunction with the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Program, inspection and surveillance of elastomers and polymers exposed to air-indoor uncontrolled or air-outdoor environments, but not replaced on a set frequency or interval (i.e., are long-lived), for evidence of cracking and change in material properties (hardening and loss of strength) and loss of material due to wear. Specify acceptance criteria of no unacceptable visual indications of cracks or discoloration that would lead to loss of function prior to the next inspection, and of no hardening as evidenced by a loss of suppleness during manipulation. • Perform inspection of the control room emergency ventilation system air-cooled condensing unit cooling coil tubes and fins and the station blackout diesel generator radiator tubes and fins for visible evidence of external surface conditions that could result in a reduction in heat transfer. Specify acceptance criteria of no unacceptable visual indications of fouling (build up of dirt or other foreign material) that would lead to loss of function prior to the next scheduled inspection. • Manage cracking of copper alloys with greater than 15% zinc and stainless steel components exposed to an outdoor air environment through plant system inspections and walkdowns for evidence of leakage. Specify acceptance criteria of no unacceptable visual indications of cracks that would lead to loss of function prior to the next scheduled inspection. • Include inspection parameters and acceptance criteria for polymers, elastomers and metallic components as applicable in system inspection and walkdown documentation. • Retain system inspection and walkdown documentation in plant records.	NRC letter dated April 20, 2011, NRC RAI 3.3.2-2 from NRC letter dated May 2, 2011, RAI 3.3.2.2.5-2 from NRC letter dated July 12, 2011, and Supplemental RAI OIN-352 from NRC Region III IP-71002 Inspection and RAI A.1-1 from NRC letter dated March 26, 2013		L-11-238, and L-13-160

Item Number	Commitment	Updated Safety Analysis Report (USAR) Supplement Section No// Comments	Implementation Schedule	Source
9	Enhance the Fatigue Monitoring Program to do the following: • Provide for updates of the fatigue usage calculations on an as needed basis if an allowable cycle limit is approached. When the number of accrued cycles is within 75% of the allowable cycle limit for any transient, a condition report will be generated. For any transient whose cycles are projected to exceed the allowable cycle limit by the end of the next plant operating cycle (Davis Besse operating cycles are normally two 2 years in duration), the program will require an update of the fatigue usage calculation for the affected component(s). • Establish an acceptance criterion for maintaining the cumulative fatigue usage below the Code design limit of 1.0 through the period of extended operation, including environmental effects where applicable.	A.1.16 B.2.16 Responses to NRC RAIs B.2.16-3, B.2.16-4 and B.2.16-5 from NRC letter dated April 20, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letter L-11-166 and L-13-160
10	Enhance the Fire Water Program to do the following: • Perform periodic ultrasonic testing for wall thickness of representative above-ground water suppression piping that is not periodically flow tested but contains, or has contained, stagnant water. The ultrasonic testing will be performed prior to the period of extended operation and at appropriate intervals thereafter, based on engineering evaluation of the initial results. • Perform at least one opportunistic or focused visual inspection of the internal surface of buried fire water piping and of similar above-ground fire water piping, within the 5-year period prior to the period of extended operation, to confirm whether conditions on the internal surface of above-ground fire water piping can be extrapolated to be indicative of conditions on the internal surface of buried fire water piping. • Perform representative sprinkler head sampling (laboratory field service testing) or replacement prior to 50 years in-service (installed), and at 10-year intervals thereafter, in accordance with NFPA 25, or until there are no untested sprinkler heads that will see 50 years of service through the end of the period of extended operation. • Perform opportunistic fire water supply and water-based suppression system internal inspections each time a fire water supply or water-based suppression system (including fire pumps) is breached for repair or maintenance. These internal visual inspections must be demonstrated to be representative of water supply and water-based suppression locations, performed on a reasonable basis (frequency), and capable of evaluating wall thickness and flow capability. If the internal inspections cannot be	A.1.18 B.2.18 and Response to NRC RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letter L-13-160

Item Number	Commitment	Updated Safety Analysis Report (USAR) Supplement Section No// Comments	Implementation Schedule	Source
	completed of a representative sample, then ultrasonic testing inspections will be used to complete the representative sample.			
11	Implement the Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program, as described in LRA Section B.2.21. Enhance the Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program to do the following: • Include inaccessible underground lower service voltage cables (400 VAC to 2 kV). • Not use 'significant voltage' (defined as being subjected to system voltage for more than 25% of the time) as a criterion for inclusion into the program. • Include inspection of electrical manholes which contain power cables within the scope of the program. • Inspect electrical manholes at least once per year. The frequency of inspections for accumulated water will be established and adjusted based on plant-specific inspection results. Also, manhole inspections will be performed in response to event-driven occurrences (e.g., heavy rain or flooding). • Include a requirement in preventive maintenance activities PM 4297, PM 4294, PM 8025, and PM 4296 to generate a condition report in cases where in scope inaccessible non-EQ power cable manhole inspection identifies submerged cables. Although the Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program, preventive maintenance activities exist for inspection of water accumulation in the manholes associated with the in scope inaccessible non-EQ power cables. • Perform cable testing on a frequency of at least every 6 years. Testing will be evaluated for more frequent performance based on test results and operating experience.	A.1.21 B.2.21 Responses to NRC RAIs B.2.21-1 and B.2.21-3 from NRC letter dated April 5, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letters L-11-134 and L-13-160

Item Number	Commitment	Updated Safety Analysis Report (USAR) Supplement Section No// Comments	Implementation Schedule	Source
12	Enhance the Masonry Wall Inspection to do the following: • Include and list the structures within the scope of license renewal that credit the program for aging management. • Add an action to follow the documentation requirement of 10 CFR 54.37, including submittal of records of structural evaluations to records management. • Specify that for each masonry wall, the extent of observed masonry cracking or degradation of steel edge supports or bracing is evaluated to ensure that the current evaluation basis is still valid. Corrective action is required if the extent of masonry cracking or steel degradation is sufficient to invalidate the evaluation basis. An option is to develop a new evaluation basis that accounts for the degraded condition of the wall (i.e., acceptance by further evaluation). • Specify that for the masonry walls within the scope of license renewal, inspections will be conducted at least once every 5 years, with provisions for more frequent inspections in areas where significant loss of material or cracking is observed to ensure there is no loss of intended function between inspections.	A.1.27 B.2.27 Response to NRC RAI B.2.39-5 from NRC letter dated April 5, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letter L-11-153 and L-13-160
13	Implement the One-Time Inspection, as described in LRA Section B.2.30. Enhance the One-Time Inspection to include enhanced visual (EVT-1 or equivalent) or surface examination (magnetic particle, liquid penetrant), or volumetric (RT or UT) inspections to detect and characterize cracking due to cyclic loading of the stainless steel makeup pump casings (DB-P37-1 and 2) of the makeup and purification system. The one-time inspections will provide verification of the absence of cracking due to cyclic loading.	A.1.30 B.2.30 Responses to NRC RAI 3.3.2.2.4.3-1 from NRC letter dated May 2, 2011, Supplemental Question—Makeup Pump Casing Inspections, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letters L-11-153, L-11-166, L-11-218, L-11-237, L-11-252, and L-13-160

Item Number	Commitment	Updated Safety Analysis Report (USAR) Supplement Section No// Comments	Implementation Schedule	Source
14	Implement the PWR Reactor Vessel Internals Program, as described in LRA Section B.2.32.	A.1.32 B.2.32 and Response to NRC RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letter L-13-160
15	In association with the PWR Reactor Vessel Internals Program, a plant-specific inspection plan for ensuring the implementation of MRP-227 program guidelines, as amended by the safety evaluation for MRP-227, and Davis-Besse's responses to the plant-specific action items, as identified in Section 4.2 of the safety evaluation for MRP-227, will be submitted for NRC review and approval. * NOTE: The inspection plan will be submitted no later than 2 years after issuance of the renewed operating license or 2 years prior to the beginning of the period of extended operation (April 22, 2015), whichever is earlier.	A.1.32 B.2.32 and Response to NRC RAI B.2.32-1 from NRC letter dated July 11, 2011	Prior to April 22, 2015 *	LRA and FENOC letter L-11-252
16	Enhance the Reactor Head Closure Studs Program as follows: • Select an alternate stable lubricant that is compatible with the fastener material and the environment. A specific precaution against the use of compounds containing sulfur (sulfide), including molybdenum disulfide (MoS_2), as a lubricant for the reactor head closure stud assemblies will be included in the program. • Preclude the future use of replacement closure stud bolting fabricated from material with actual measured yield strength greater than or equal to 150 ksi except for use of the existing spare reactor head closure stud bolting.	A.1.34 B.2.34 and Response to NRC RAI B.2.34-1 from NRC letter dated June 20, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letters L-11-218 and L-13-160
17	Enhance the Reactor Vessel Surveillance Program as follows: • The Capsule Insertion and Withdrawal Schedule for Davis-Besse will be revised to schedule testing of the TE1-C capsule.	A.1.35 B.2.35 and Response to NRC RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letter L-13-160

Item Number	Commitment	Updated Safety Analysis Report (USAR) Supplement Section No// Comments	Implementation Schedule	Source
18	Implement the Selective Leaching Inspection, as described in LRA Section B.2.36.	A.1.36 B.2.36 and Response to NRC RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letter L-13-160
19	Implement the Small Bore Class 1 Piping Inspection, as described in LRA Section B.2.37.	A.1.37 B.2.37 Response to NRC RAI B.2.37-2 from NRC letter dated April 20, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Completed within the 6-year period prior to October 22, 2016	LRA and FENOC letters L-11-153 and L--13-160
20	Enhance the Structures Monitoring Program to do the following: • Include and list the structures within the scope of license renewal that credit the program for aging management. • Include aging effect terminology (e.g., loss of material, cracking, change in material properties, and loss of form). • List ACI 349.3R and ANSI/ASCE 11-90 as references and indicate that they provide guidance for the selection of parameters monitored or inspected. • Clarify that a “structural component” for inspection includes each of the component types identified within the scope of license renewal as requiring aging management. • Require the responsible engineer to review site raw water pH, chlorides, and sulfates test results prior to the inspection to take into account the raw water chemistry for any unusual trends during the period of extended operation. Raw water chemistry data shall be collected at least once every 5 years. Data collection dates shall be staggered from year to year (summer-winter-summer) to account for seasonal variation. • Perform an inspection for loss of material for carbon steel structural components subject to aggressive groundwater. Require the use of the FENOC Corrective Action Program for identified concrete or steel degradation.	A.1.39 B.2.39 Responses to NRC RAIs B.2.39-4, B.2.39-5, B.2.39-6 and B.2.39-7 from NRC letter dated April 5, 2011, B.2.39-11 and 3.5.2.3.12-4 from NRC letter dated July 21, 2011, Supplemental RAI B.2.39-11 from telecon held with the NRC on September 13, 2011, Supplemental RAI OIN-380 from Region III IP-71002 Inspection. RAI B.2.4-1a from NRC letter dated November 14, 2012, RAI B.2.43-3a from NRC letter dated January 4, 2013, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA FENOC letters L-11-153, L-11-237, L-11-292, L-11-317, L-12-455, L-13-037 and L-13-160

Item Number	Commitment	Updated Safety Analysis Report (USAR) Supplement Section No// Comments	Implementation Schedule	Source
	• Specify that, upon notification that a below-grade structural wall or other in-scope concrete or metal structural component will become accessible through excavation, a followup action is initiated to the responsible engineer to inspect the exposed surfaces for age-related degradation. Such inspections will include concrete examination using acceptance criteria from NUREG-1801, XI.S6, program element 6. Degradation found that exceeds the acceptance criteria will be trended and processed through the FENOC Corrective Action Program. • List ACI 349.3R, ANSI/ASCE 11-90, and EPRI Report 1007933 as references and indicate that they provide guidance for detection of aging effects. • Add an action to follow the documentation requirement of 10 CFR 54.37, including submittal of records of structural evaluations to records management. • Revise to add sufficient acceptance criteria and critical parameters to trigger an increased level of inspection and initiation of corrective action. Indicate that ACI 349.3R provides acceptable guidelines which will be considered in developing acceptance criteria for concrete structural elements, steel liners, joints, and waterproofing membranes. The acceptance criteria for visual inspection of coatings on in-scope concrete structures will be in accordance with ACI 349.3R. Plant-specific quantitative degradation limits, similar to the three-tier hierarchy acceptance criteria from Chapter 5 of ACI 349.3R, will be developed and added to the inspection procedure. The Structures Monitoring Program procedure will also be enhanced to reflect the "Periodic Evaluation" criteria defined in Chapter 3.3 of ACI 349.3R. The Structures Monitoring Program procedure will include the "prioritization process" to develop a representative sample of areas to inspect in accordance with ACI 349.3R. • Require that personnel performing the structural inspections meet qualifications that are commensurate with ACI 349.3R, "Evaluation of Existing Nuclear Safety-Related Concrete Structures," Chapter 7, "Qualifications of Evaluation Team." • The program procedure will be enhanced by specifying that, for the structures within the scope of license renewal, inspections will be conducted at least once every 5 years.			

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	• Conduct a baseline inspection of the structures within the scope of license renewal prior to entering the period of extended operation. • Require optical aids, scaling technologies, mechanical lifts, ladders or scaffolding for tall structures or difficult to reach areas of structures to allow visual inspections that meet the guidelines of Chapter 5 of ACI 349.3R. Select the areas to be inspected in accordance with the guidelines of Chapter 5 of ACI 349.3R to reflect the "Periodic Evaluation" criteria defined in Chapter 3.3 of ACI 349.3R. Include the "prioritization process" in the selection methodology to develop a representative sample of areas to inspect in accordance with ACI 349.3R. • Monitor elastomeric vibration isolators and structural sealants for cracking, loss of material, and hardening. • Supplement visual inspection of elastomeric vibration isolation elements by feel to detect hardening if the vibration isolation function is suspect. • Identify that the following are true: – Loose bolts and nuts and cracked high strength bolts are not acceptable unless accepted by engineering evaluation. – Structural sealants are acceptable if the observed loss of material, cracking, and hardening will not result in loss of sealing. – Elastomeric vibration isolation elements are acceptable if there is no loss of material, cracking, or hardening that could lead to the reduction or loss of isolation function. • Require that high strength (i.e., ASTM A540 Grade B23) structural bolting materials with an actual measured yield strength greater than or equal to 150 ksi and greater than 1 inch in nominal diameter are monitored for stress corrosion cracking (SCC). Perform periodic visual inspections of susceptible ASTM A540 bolting to identify locations where A540 bolting may be exposed to a potentially corrosive environment for SCC. Complete the initial visual inspections prior to entering the period of extended operation, and perform recurring inspections at an interval not to exceed five years. Perform volumetric examination (i.e., ultrasonic testing) on a sampling basis of bolting exposed to a corrosive environment, as determined by engineering evaluation, to a depth of at least 12 inches.			

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	• Require that personnel performing ultrasonic testing (UT) examinations of structural bolting have a current ASME Code Section XI, Appendix VIII, Supplement 8 endorsement. • Revise the applicable structural bolting specifications to prevent future use of A540 bolting with measured yield strength equal to or exceeding 150 ksi.			
21	Enhance the Water Control Structures Inspection to do the following: • Include the service water discharge structure, which is within the scope of license renewal. • Include parameters monitored and inspected for water control structures, including the service water discharge structure, in accordance with applicable inspection elements listed in Section C.2 of Regulatory Guide 1.127, Revision 1. Descriptions of concrete conditions will conform with the appendix to ACI 201. The use of photographs for comparison of previous and present conditions will be included as a part of the inspection program. • Specify that water control structure periodic inspections are to be performed at least once every 5 years. • Add an action to follow the documentation requirement of 10 CFR 54.37, including submittal of records of structural evaluations to records management. • Add sufficient acceptance criteria and critical parameters to trigger an increased level of inspection and initiation of corrective action. Indicate that ACI 349.3R provides acceptable guidelines which will be considered in developing acceptance criteria for water control structures. Plant-specific quantitative degradation limits, similar to the three-tier hierarchy acceptance criteria from Chapter 5 of ACI 349.3R, will be developed and added to the inspection procedure. The Structures Monitoring Program procedure will also be enhanced to reflect the “Periodic Evaluation” criteria defined in Chapter 3.3 of ACI 349.3R. The Structures Monitoring Program procedure will include the “prioritization process” to develop a representative sample of areas to inspect in accordance with ACI 349.3R. • Conduct a baseline inspection of the structures within the scope of license renewal prior to entering the period of extended operation. • Require that loose bolts and nuts, cracked high strength bolts, and degradation of piles and sheeting (sheet pilings) are accepted by	A.1.40 B.2.40 Responses to NRC RAI B.2.39-6 from NRC letter dated April 5, 2011, Supplemental RAI OIN-379 from Region III IP-71002 Inspection, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letters L-11-153, L-11-292, and L-13-160

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	engineering evaluation or subject to corrective actions. Engineering evaluation will be documented and based on codes, specifications and standards such as American Institute of Steel Construction (AISC) specifications, Structural Engineering Institute / American Society of Civil Engineers (SEI/ASCE) 11, and codes, specifications or standards referenced in the Davis-Besse current licensing basis.			
22	Enclose or otherwise protect the safety-related station ventilation radiation monitors located in the turbine building such that leakage and spray from surrounding piping systems does not adversely impact the intended function of the radiation monitors.	Response to NRC RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letter L-13-160
23	In association with the TLAA for effects of environmentally assisted fatigue of the high-pressure injection (HPI) nozzle safe end including the associated Alloy 82/182 weld (weld that connects the safe end to the nozzle), replace the HPI nozzle safe end including the associated Alloy 82/182 weld for all four HPI nozzles prior to the period of extended operation. Apply the Fatigue Monitoring Program to evaluate the environmental effects and manage cumulative fatigue damage for the replacement HPI nozzle safe ends and associated welds.	A.2.3.4.2 A.2.7.4 Responses to NRC RAIs 4.7.4-1 from NRC letter dated April 15, 2011, 4.3-18 from NRC letter dated June 17, 2011, RAI 4.7.4-1 from NRC letter dated October 11, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letters L-11-107, L-11-203, L-11-334, and L-13-160
24	Apply the elements of corrective actions, confirmation process, and administrative controls in the Quality Assurance Program Manual to the credited AMPs and activities for safety-related and nonsafety-related structures and components determined to require aging management for the period of extended operation.	A.1 Response to NRC RAI 3.0 from NRC letter dated May 2, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letter L-11-166 and L-13-160
25	Enhance the Steam Generator Tube Integrity Program to do the following: • Include gross visual inspection of the steam generator tube-to-tubesheet welds coupled with eddy-current inspection (i.e., bobbin coil or rotating coil examinations) of the tubes to monitor for cracking and degradation of the tube-to-tubesheet welds (Alloy 600). Schedule the gross visual inspection of the tube-to-tubesheet welds concurrent with eddy-current inspection of the steam generator tubes that are scheduled in accordance with Davis-Besse Technical Specification 5.5.8 such that 100 % of the	A.1.38 B.2.38 Responses to NRC RAIs 3.1.2.2.16-2 from NRC letter dated November 8, 2011, RAI 3.1.2.2.16-3 from NRC letter dated	Prior to October 22, 2016	LRA and FENOC letters L-11-354,

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	tube-to-tubesheet welds (includes both the hot leg and cold leg welds) are inspected at sequential periods of 60 effective full power months. Perform the gross visual inspection of the tube-to-tubesheet welds through remote-visual examination using a manipulator camera to obtain a straight-on view of the weld with a visual acuity sufficient to detect evidence of degradation. Perform the gross visual inspections using personnel who are qualified for American Society of Mechanical Engineers (ASME) code visual examination (i.e., are certified VT-1 or VT-3 examiners) and are knowledgeable in the type of tube-to-tubesheet welds being examined (i.e., fillet welds). Define the acceptance criteria for the gross visual inspections and the eddy-current inspections as no indication of cracking or relevant conditions of degradation.	December 27, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013		L-12-001, and L-13-160
26	• Obtain and evaluate for degradation a concrete core bore from two representative inaccessible concrete components of an in-scope structure subjected to aggressive groundwater prior to entering the period of extended operation. Based on the results of the initial core bore sample, evaluate the need for collection and evaluation of representative concrete core bore samples at additional locations that may be identified during the period of extended operation as having aggressive groundwater infiltration. • Select additional core bore sample locations based on the duration of observed aggressive groundwater infiltration. • Document identified concrete or steel degradation in the FENOC Corrective Action Program.	Responses to NRC RAI B.2.39-3 from NRC letter dated April 5, 2011, RAI B.2.39-11 from NRC letter dated July 21, 2011, and Supplemental RAI B.2.39-11 from telecon held with the NRC on September 13, 2011	Prior to December 31, 2014	FENOC letters L-11-153, L-11-237, and L-11-292
27	Davis-Besse Surveillance Test Procedure DB-PF-03009, Revision 06, “Containment Vessel and Shielding Building Visual Inspection,” Subsection 2.1.2, shall be enhanced to state, Personnel who perform general visual examinations of the exterior surface of the Containment Vessel and the interior and exterior surfaces of the Shield Building shall meet the requirements for a general visual examiner in accordance with Nuclear Operating Procedure NOP-CC-5708, “Written Practice for the Qualification and Certification of Nondestructive Examination Personnel.” These individuals shall be knowledgeable of the types of conditions which may be expected to be identified during the examinations.	Response to NRC RAI B.2.1-1 from NRC letter dated April 5, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	FENOC letters L-11-134 and L-13-160

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28	Enhance the Fuel Oil Chemistry Program to do the following: • Require that internal surfaces of emergency diesel generator fuel oil storage tanks and day tanks, diesel oil storage tank, diesel fire pump day tank, and station blackout diesel generator day tank are periodically drained (at least once every 10 years) for cleaning and are visually inspected to detect potential degradation. If degradation is identified in a diesel fuel tank by visual inspections, a volumetric inspection is performed. • Require that biological activity be monitored and trended at least quarterly.	A.1.20 B.2.20 Responses to NRC RAIs B.2.20-1 and B.2.20-2 from NRC letter dated April 5, 2011, Supplemental RAI OIN-368 from NRC Region III IP-71002 Inspection, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letters L-11-134, L-11-238, and L-13-160
29	Enhance the Cranes and Hoists Inspection Program to include visual inspections for loose bolts and missing or loose nuts in crane, monorail, and hoist inspection procedures at the same frequency as inspections of rails and structural components.	A.1.10 B.2.10 Response to NRC RAI B.2.10-2 from NRC letter dated April 20, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letter L-11-153 and L-13-160

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30	Enhance the Leak Chase Monitoring Program to do the following: • Include acceptance criteria such that measurement of leakage from any monitoring line exceeding 15 ml/min will be documented in the Corrective Action Program for evaluation and potential corrective actions. Evaluation will include consideration of more frequent monitoring. • Analyze collected leak chase drainage for pH monthly and for iron every 6 months. The initial acceptance criteria will be 7.0–8.0 for pH. The results for iron will be monitored and trended to insure that there is no indication of corrosion of the reinforcing bars in the walls or floor of the pool and pits. An acceptance criterion for the iron analyses will be developed after 3 years of measurements. Analyses that exceed the limits will be documented in the Corrective Action Program. • Perform the leak chase inspection and cleaning recurring preventive maintenance (PM) activity every 18 months. • Inspect once per year for leakage migrating through the accessible outside walls and floor (from the ceiling side) of the pool and pits. Document the inspection results and retain in plant records. Indication of leakage through the walls will be documented in the Corrective Action Program.	A.1.25 B.2.25 Responses to NRC RAI B.2.25-5 from NRC letter dated April 5, 2011, and RAIs B.2.25-7, RAI B.2.39-10 from NRC letter dated July 21, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letters L-11-153, L-11-238, and L-13-160
31	Incorporate reference to and the preventative actions of the Research Council for Structural Connections, "Specification for Structural Joints Using ASTM A325 or A490 Bolts" into the Davis-Besse specifications and implementing procedures that address Davis-Besse structural bolting within the scope of license renewal.	Response to NRC RAI B.2.39-8 from NRC letter dated April 5, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	FENOC letters L-11-153 and L-13-160
32	Enhance the Closed Cooling Water Chemistry program to do the following: • Document the results of periodic inspections of opportunity, performed when components are opened for maintenance, repair, or surveillance. • Ensure that a representative sample of piping and components will be inspected on a 10-year interval, with the first inspection taking place prior to entering the period of extended operation. • Ensure that component cooling water radiochemistry is sampled on a weekly interval to verify the integrity of the letdown coolers and seal return coolers.	A.1.8 B.2.8 Response to NRC RAI B.2.8-1 from NRC letter dated April 20, 2011, Supplemental RAI 2.3.3.18-4 from telephone conference held on November 9, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letters L-11-153, L-11-354, and L-13-160

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33	Phase 1 Perform the following actions to reduce or mitigate the refueling canal leaks inside containment: 1. Select and implement a leak detection method to locate the leakage area. 2. Evaluate temporary and permanent repair methods to stop or significantly reduce the leakage, and implement a repair plan. Phase 2 Perform the following actions to evaluate the impact of refueling canal leaks on concrete and reinforcing steel structures. Discontinue core bores, testing and reinforcing steel inspections when indications of refueling canal leakage are no longer present: 1. Perform a core bore in the south wall of the east-west section of the core flood pipe tunnel. a. Assess borated water degradation of the concrete by testing the core bore sample for compressive strength and by petrographic examination and evaluate the results. b. Conduct a visual examination of the concrete and reinforcing steel to identify aging effects (e.g., concrete degradation or steel corrosion). Enter identified aging effects into the FENOC Corrective Action Program and evaluate in accordance with the requirements of the current licensing basis Maintenance Rule Program. 2. If leakage from the refueling canal has not been eliminated or resumes by the beginning of the period of extended operation, then evaluate the concrete structures in a manner similar to the way that they were evaluated under Phase 2, Action 1. However, use acceptance criteria from the ACI 349.3R for the evaluation. 3. If leakage from the refueling canal has not been eliminated or resumes during the period of extended operation, then evaluate the concrete structures again in a manner similar to the way that they were evaluated under Phase 2, Action 2. Perform evaluations every 10 years until the end of the period of extended operation.	Response to NRC RAI B.2.39-9 from NRC letter dated July 27, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Phase 1: Action 1 prior to December 31, 2014 Action 2 prior to October 22, 2016 Phase 2: Action 1 prior to December 31, 2014 Action 2 prior to December 31, 2023 Action 3—Ongoing	FENOC letters L-11-252 and L-13-160

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34	Enhance the Bolting Integrity Program to do the following: • Select an alternate stable lubricant that is compatible with the fastener material and the environment. A specific precaution against the use of compounds containing sulfur (sulfide), including MoS_2, as a lubricant will be included in the program.	A.1.4 B.2.4 Response to NRC RAI B.2.4-3 from NRC letter dated April 20, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letters L-11-153 and L-13-160
35	Perform the following actions for each of two examinations (Phase 1 and Phase 2) of the containment vessel in the sand pocket region: • Perform nondestructive examination (NDE) of the containment vessel from the outer surface at five areas of previously-identified groundwater in-leakage. – Examine the vessel at a minimum of three vertical grid locations at 12 in. nominal horizontal spacing at each area. Examine the containment vessel at a minimum of three elevations: 1. approximately 3 in. below the existing grout-to-vessel interface in the sand pocket region 2. at the existing grout-to-vessel interface level in the sand pocket region 3. approximately 3 in. above the existing grout-to-vessel interface in the sand pocket region • Compare the ultrasonic test (UT) thickness readings to minimum ASME Code vessel thickness requirements and to the results obtained during previous UT examinations of the containment vessel and determine the need for maintenance or repair of the containment vessel based on the results and evaluation of the examinations. • Document the results of each of the two examinations in the work order system. • Document and evaluate adverse conditions in accordance with the FENOC Corrective Action Program for an evaluation of potential degradation of the steel containment vessel thickness over the longer term.	Response to NRC RAI B.2.22-5 from NRC letter dated July 21, 2011	Phase 1 prior to December 31, 2014 and Phase 2 prior to December 31, 2025	FENOC letter L-11-252

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36	Perform the following actions related to the containment vessel sand pocket region each refueling outage: • Perform visual inspection of 100% of the accessible areas of the wetted outer surface of the containment vessel in the sand pocket region. • Perform visual inspection of accessible dry areas of the outer surface of the containment vessel in the sand pocket region and the areas above the grout-to-steel interface up to elevation 566 feet + 3 in., - 1 in. • Perform visual inspection for deterioration (e.g., missing or damaged grout) of accessible grout and the containment exterior moisture barrier in the sand pocket area. • Perform opportunistic visual inspections of inaccessible areas of the containment vessel in the sand pocket region when such areas are made accessible. • Perform opportunistic visual inspections for deterioration (e.g., missing or damaged grout) of inaccessible grout in the sand pocket region when such areas are made accessible. Inaccessible grout is the grout below the normally-exposed surface of the grout in the sand pocket area. • Address issues of pitting, microbiologically-influenced corrosion (MIC), degraded grout, moisture barrier or sealant identified during the inspections using the FENOC Corrective Action Program. • Sample the water in the sand pocket region when sufficient volumes are available. The number of sampled water volumes will be determined by the number of water volumes observed and the size of those water volumes. Analyze the sample(s) for pH, chlorides, iron and sulfates. Treat or wash (or a combination thereof) the sand pocket area to reduce measured chloride concentrations to less than 250 parts per million (ppm) if the concentration of chlorides in a sample exceeds 250 ppm. Note: Water samples may be taken at different times during each outage. Engineering judgment may be used to determine the priority of the chemical analyses to be performed if sufficient water is not available in a given sample for all analyses.	Response to NRC RAI B.2.22-5 from NRC letter dated July 21, 2011 and Supplemental RAI B.2.22-5 from telephone conferences held on October 5 and November 14, 2011	Ongoing	FENOC letter L-11-252 and L-11-354

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37	Perform and evaluate core bores of the ECCS Pump Room No. 1 wall and the Room 109 ceiling. • The core bores will be deep enough to expose reinforcing bar in the wall and ceiling. The core samples from the core bores will be examined for signs of corrosion or chemical effects of boric acid on the concrete or reinforcing bars. The examination will include a petrographic examination. The reinforcing steel that will be exposed for a visual inspection will have corrosion products collected for testing. Degradation identified from the samples will be entered into the FENOC Corrective Action Program. The core bores will be performed in areas where leakage has been observed in the past. • The first set of core bores will be performed prior to the end of 2014 (Phase 1). • The second set of core bores will be performed prior to the end of 2020 (Phase 2). • Further core bores will be conducted, if warranted, based on the evaluation of the results of the inspection and testing of the core bores or if spent fuel pool leakage through the wall or ceiling recurs after the second set of core bores is performed. If spent fuel pool leakage through another wall or ceiling is identified, then core bores will be performed in a manner similar to that stated for the ECCS Pump Room No. 1 wall and the Room 109 ceiling.	Responses to NRC RAI B.2.39-2 from NRC letter dated April 5, 2011, and RAI B.2.39-10 from NRC letter dated July 21, 2011	Phase 1 prior to December 31, 2014 and Phase 2 prior to December 31, 2020	FENOC letters L-11-153 and L-11-238
38	Evaluate the concrete cracking observed on the underside of the spent fuel pool for necessary repairs. Note: A core bore of the Room 109 ceiling will be performed by the end of 2014 (see license renewal commitment 37). Degradation identified from the samples will be entered into the FENOC Corrective Action Program. The condition of the concrete and the reinforcing steel will be evaluated at that time to assist in determining what repairs, if any, need to be made to the underside of the spent fuel pool concrete. The criterion for determining the need to repair the cracking will be the continued capability of the structures to perform their intended functions during the period of extended operation.	Responses to NRC RAI B.2.39-2 from NRC letter dated April 5, 2011, RAI B.2.39-10 from NRC letter dated July 21, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	FENOC letters L-11-153, L-11-238, and L-13-160

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39	Address the potential for borated water degradation of the steel containment vessel through the following actions: • Access the inside surface of the embedded steel containment at a vertical height no greater than 10 inches above bottom dead center. A core bore will be completed by the end of 2014 (Phase 1). • If necessary, a second core bore will be completed by the end of 2020 (Phase 2). • If there is evidence of the presence of borated water in contact with the steel containment vessel, conduct NDT to determine what effect, if any, the borated water has had on the steel containment vessel. • Based on the results of NDT, perform a study to determine the effect through the period of extended operation of any identified loss of thickness in the steel containment due to exposure to borated water.	Responses to NRC RAIs B.2.22-2 from NRC letter dated April 5, 2011, RAI B.2.22-6 from NRC letter dated July 27, 2011, and Supplemental RAI B.2.22-6 from NRC telephone conference call held on May 9, 2013	Phase 1 prior to December 31, 2014 and Phase 2 prior to December 31, 2020	FENOC letters L-11-153, L-11-237, and L-13-180
40	Implement the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Program, as described in LRA Section B.2.41.	A.1.41 B.2.41 Responses to NRC RAIs 3.3.2.2.5-1 and 3.3.2.71-2 from NRC letter dated April 20, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letters L-11-153 and L-13-160
41	Establish a preventive maintenance task to periodically replace the flexible connections exposed to fuel oil in the fuel oil system.	Response to NRC RAI 3.3.2.3.12-2 from NRC letter dated May 2, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	FENOC letters L-11-166 and L-13-160

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42	Enhance the Fatigue Monitoring Program to do the following: • Evaluate additional plant-specific component locations in the reactor coolant pressure boundary that may be more limiting than those considered in NUREG/CR-6260. This evaluation will include identification of the most limiting fatigue location exposed to reactor coolant for each material type (i.e., CS, LAS, SS, and NBA) and that each bounding material/location will be evaluated for the effects of the reactor coolant environment on fatigue usage. Nickel based alloy items will be evaluated using NUREG/CR-6909. • Submit the evaluation to the NRC 1 year prior to the period of extended operation.	A.1.16 B.2.16 Response to NRC RAI B.2.16-2 from NRC letter dated April 20, 2011	Prior to April 22, 2016	LRA and FENOC letter L-11-166
43	Ensure that the current station operating experience review process includes future reviews of plant-specific and industry operating experience to confirm the effectiveness of the license renewal AMPs to determine the need for programs to be enhanced, or indicate a need to develop new AMPs.	Response to NRC RAI B.1.4-1 from NRC letter dated May 19, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Complete	FENOC letters L-11-188 and L-13-160 L-13-257
44	Cathodically protect the EDG fuel oil storage tanks (DB-T153-1 and DB-T153-2) and the in-scope fuel oil and service water buried piping in accordance with NACE SP0169-2007 or NACE RP0285-2002.	Response to NRC RAI B.2.7-1 from NRC letter dated April 20, 2011, as modified per telecon with the NRC held on June 7, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	FENOC letter L-11-203 L-11-218, and L-13-160
45	Implement the Nuclear Safety-Related Coatings Program, as described in LRA Section B.2.42.	A.1.42 B.2.42 Response to NRC RAI XI.S8-1 from NRC letter dated April 5, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letters L-11-203, L-11-218, and L-13-160

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46	Implement the Shield Building Monitoring Program as described in LRA Section B.2.43.	A.1.43 B.2.43 Response to NRC RAI B.2.39-13 from NRC Letter dated December 27, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	FENOC letters L-12-028 and L-13-160
47	Enhance the Inservice Inspection (ISI) Program—IWE to include surface examinations to monitor for cracking of containment stainless steel penetration sleeves, dissimilar metal welds, bellows, and steel components that are subject to cyclic loading but have no current licensing basis fatigue analysis. The inspection sample size will include 10% of the containment penetration population that are subject to cyclic loading but have no current licensing basis fatigue analysis. Penetrations included in the inspection sample will be scheduled for examination in each 10-year ISI interval that occurs during the period of extended operation. Should fatigue analyses be performed in the future for the subject containment penetrations, the surface examinations will no longer be required.	A.1.22 B.2.22 Responses to NRC RAI B.2.22-7 from NRC letter dated July 21, 2011, Supplemental RAI B.2.22-7 from NRC Telecons on September 13 and 16, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letters L-11-238, L-11-292, and L-13-160
48	Complete an investigation and needed repairs or modification of the degraded portion of the safety-related intake canal embankment.	Response to NRC RAI B.2.40-2 from NRC letter dated July 21, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	FENOC letters L-11-238 and L-13-160
49	Enhance the Nickel-Alloy Management Program to provide for inspection of dissimilar metal butt welds in accordance with the requirements of 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," as modified by 10 CFR 50.55a(g)(6)(ii)(F).	A.1.28 B.2.28 Response to NRC RAI B.2.28-1 from NRC letter dated July 27, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC letters L-11-238 and L-13-160
50	Enhance the Inservice Inspection (ISI) Program - IWF to do the following: • Include monitoring of ASTM A490 high-strength bolting (i.e., actual measured yield strength greater than or equal to 150 ksi or 1,034 MPa) in sizes greater than 1-inch nominal diameter for cracking using volumetric examination. The volumetric examinations will be performed in accordance	A.1.23 B.2.23 Supplemental response to NRC RAI B.2.4-1b from NRC letter dated February 14, 2013, and	Prior to October 22, 2016	LRA and

Item Number	Commitment	Updated Safety Analysis Report (USAR) Supplement Section No// Comments	Implementation Schedule	Source
	with the requirements of ASME Boiler and Pressure Vessel Code, Section V, Article 5, Appendix IV, 2007 Edition through 2008 Addenda. The representative sample size will be equal to 20 percent (rounded up to the nearest whole number) of the entire IWF population of ASTM A490 high-strength bolts in sizes greater than 1-inch nominal diameter, with a maximum sample size of 25 bolts. The selection of the representative sample will consider susceptibility to stress corrosion cracking (SCC) (e.g., actual measured yield strength) and as low as reasonably achievable (ALARA) radiation dose reduction principles. The frequency of examination will be once each 10-year ISI interval beginning with the fourth interval that started on September 21, 2012. • Include monitoring of ASTM A540 high-strength bolting (i.e., actual measured yield strength greater than or equal to 150 ksi or 1,034 MPa) in sizes greater than 1-inch in nominal diameter for cracking. Periodic visual inspections of susceptible ASTM A540 bolting will be conducted prior to the period of extended operation and at an interval not to exceed 5 years to identify locations where the A540 bolting may be exposed to a potentially corrosive environment for SCC. If the visual inspections identify one or more bolts in a potentially corrosive environment, then an engineering evaluation will be performed to determine whether the bolting material had been subjected to a corrosive environment for SCC. The bolts determined to have been subiected to a corrosive environment for SCC comprise the population subject to sampling for volumetric examinations. The representative sample size is equal to 20 percent (rounded up to the nearest whole number) of the bolts in the sample population, with a maximum sample size of 25 bolts. The volumetric examinations are performed in accordance with the requirements of ASME Code Section V, Article 5, Appendix IV. Volumetric examinations will be performed no later than the subsequent refueling outage following visual identification of bolting subject to a corrosive environment. Deferral of volumetric examinations to the subsequent refueling outage is not permitted if the visual inspection indicates evidence of contaminant penetration through the coatings. The frequency of examination is once each 10-year ISI interval beginning with the 4th interval that started on September 21, 2012. For ASTM A540 hiqh-strength bolts that are not exposed to a corrosive environment, the volumetric examinations are waived based on plant-specific operating experience associated with the volumetric examination of the Davis-Besse reactor head closure studs (60 each) constructed of high-strength ASTM A540 material where the studs are examined once each ISI interval, and after three intervals, no unacceptable indications have been noted.	from telephone conference calls held on April 11, April 24, May 2, and May 28, 2013		FENOC letters L-13-181 and L-13-199

Item Number	Commitment	Updated Safety Analysis Report (USAR) Supplement Section No// Comments	Implementation Schedule	Source
	• As an alternative to the visual examinations and the subsequent volumetric examinations of ASTM A540 bolts subjected to a corrosive environment, the Inservice Inspection (ISI) Program - IWF provides an option to perform periodic volumetric examinations as follows. The program includes monitoring of ASTM A540 high strength bolting (i.e., actual measured yield strength greater than or equal to 150 ksi or 1,034 MPa) in sizes greater than 1-inch nominal diameter for cracking using volumetric examination. The volumetric examinations are performed in accordance with the requirements of ASME Code Section V, Article 5, Appendix IV. The representative sample size is equal to 20 percent (rounded up to the nearest whole number) of the entire IWF population of ASTM A540 high -strength bolts in sizes greater than 1-inch nominal diameter, with a maximum sample size of 25 bolts. The selection of the representative sample considers susceptibility to SCC (e.g., actual measured yield strength) and ALARA radiation dose reduction principles. The frequency of examination is once each 10-year ISI interval beginning with the 4th interval that started on September 21, 2012.			

APPENDIX B

CHRONOLOGY

This appendix contains a chronological listing of the routine correspondence between the staff of the U.S. Nuclear Regulatory Commission (NRC or the staff) and FirstEnergy Nuclear Operating Company (FENOC or the applicant) and other correspondence regarding the staff's reviews of the Davis-Besse Nuclear Power Station (Davis-Besse), Docket Number 50-346, license renewal application (LRA).

Table B-1. Chronology

Date	Subject
August 27, 2010	Davis-Besse Nuclear Power Station—License Renewal Application and Ohio Coastal Management Program Consistency Certification. (Accession No. ML102450565)
August 27, 2010	Davis-Besse Nuclear Power Station— License Renewal Application Boundary Drawings. (Accession No. ML102460429)
August 31, 2010	License Application for Facility Operating License (Amend/Renewal) DKT 50, FENOC, "Enclosures to Davis-Besse, Unit 1, Letter L-10-221, License Renewal Application, Section 3.5 through Appendix E, References 2.3-1." (Accession No. ML102450563)
August 31, 2010	License Application for Facility Operating License (Amend/Renewal) DKT 50, FENOC, "Enclosures to Davis-Besse, Unit 1, Letter L-10-221, License Renewal Application, Cover Page through Section 3, Page 3.4-112." (Accession No. ML102450567)
August 31, 2010	License Application for Facility Operating License (Amend/Renewal) DKT 50, FENOC, "Enclosures to Davis-Besse, Unit 1, Letter L-10-221, License Renewal Application, Appendix E, Section 2.4 through References E.11." (Accession No. ML102450568)
September 17, 2010	Federal Register Notice: Receipt and Availability of the License Renewal Application for the Davis-Besse Nuclear Power Station, Unit 1. (Accession No. ML102300325)
September 20, 2010	Press Release -10-164: NRC Announces Availability of License Renewal Application for Davis-Besse Nuclear Plant. (Accession No. ML102630380)
October 18, 2010	Letter to Allen B. S., FENOC, "Determination of Acceptability and Sufficiency for Docketing, Proposed Review Schedule, and Opportunity for a Hearing Regarding the Application from the FirstEnergy Nuclear Operating Company, for Renewal of the Operating License for the Davis-Besse Nuclear Power Station." (Accession No. ML102710584)
October 20, 2010	Federal Register Notice: Notice of Intent to Prepare an Environmental Impact Statement and Conduct Scoping Process for License Renewal for the Davis-Besse Nuclear Power Station, Unit 1. (Accession No. ML102700603)
October 26, 2010	Press Release-10-191: NRC Announces Opportunity for Hearing on Application to Renew Operating License for Davis-Besse Nuclear Power Plant. (Accession No. ML102990387)
October 28, 2010	Press Release-10-193: NRC to Conduct Environmental Scoping Meeting as Part of the License Renewal Application for Davis Besse; Meeting November 4, 2010. (Accession No. ML103010069)
November 4, 2010	Meeting Transcript: Davis Besse License Renewal Public Meeting - Afternoon Session. Pages 1-46. (Accession No. ML110140231)
November 4, 2010	Meeting Transcript: Davis Besse License Renewal Public Meeting - Evening Session. Pages 1-3. (Accession No. ML110140232)

Date	Subject
November 4, 2010	General FR Notice Comment Letter, from Kucinich D. J., U.S. House of Representatives, to Jaczko G. B., NRC/Chairman , “Comment (10) of Dennis J. Kucinich on Behalf of US House of Representatives, Opposing Davis-Besse for 20-year License Extension” (Accession No. ML110680518)
December 6, 2010	Letter to Kurkul P. A., U.S. Department of Commerce, National Oceanic & Atmospheric Admin (NOAA): Request for List of Protected Species Within the Area Under Evaluation for the Davis-Besse Nuclear Power Station License Renewal Application Review. (Accession No. ML102980692)
December 7, 2010	Letter to Epstein M., State of OH, Historic Preservation Office: Davis-Besse Nuclear Power Station, Unit 1, License Renewal Application Review. (Accession No. ML102980687)
December 21, 2010	Letter from Colligan M.A.,NOAA: Letter dated December 6, 2010, Requesting Information on the Presence of Listed Species in the Vicinity of the Davis-Besse Nuclear Power Station, Located 25 miles East of Toledo, Ohio. (Accession No. ML110140230)
February 3, 2011	Press Release-11-016: Licensing Board to Hear Oral Argument March 1 on Davis-Besse Reactor License Renewal Application. (Accession No. ML110340176)
February 17, 2011	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station—Fire Protection (TAC No. ME4640). (Accession No. ML110450046)
February 24, 2011	Letter to Allen B. S., FENOC: Davis-Besse Nuclear Power Station, Information Request for an NRC License Renewal Inspection. (Accession No. ML110550916)
February 28, 2011	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station—Section 2.4 (TAC No. ME4640). (Accession No. ML110420597)
February 28, 2011	Letter to Allen B. S., FENOC: Environmental Site Audit Regarding Davis-Besse Nuclear Power Station, Unit Number 1, License Renewal Application. (Accession No. ML110190113)
March 17, 2011	Letter to Allen B. S., FENOC: Request for Additional Information on the Reactor Vessel Surveillance Aging Management Program, Time-Limited Aging Analyses (TLAAs) for Neutron Embrittlement of the Reactor Vessel and Internals, and Other TLAAs for the Review of the Davis-Besse Nuclear Power Station (TAC No. ME4640). (Accession No. ML110680172)
March 18, 2011	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station—Section 2.2 & 2.3 (TAC No. ME4640). (Accession No. ML110700732)
March 18, 2011	Letter from Allen B. S., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640) and License Renewal Application Amendment No. 1. (Accession No. ML110830025)
March 21, 2011	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station—Section 4.7 (TAC No. ME4640). (Accession No. ML11068A000)
March 23, 2011	Letter from Allen B. S., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML110880058)
March 23, 2011	Letter from Allen B. S., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640), Section 2.4. (Accession No. ML110880059)

Date	Subject
March 30, 2011	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station—Section 2.1 (TAC No. ME4640). (Accession No. ML110820624)
April 5, 2011	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station—Batch 1 (TAC No. ME4640). (Accession No. ML110820490)
April 15, 2011	Letter from Allen B. S., FENOC: Reply to Request for Additional Information on the Reactor Vessel Surveillance Aging Management Program and Time-Limited Aging Analyses for Neutron Embrittlement for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640) and License Renewal Application Amendment No. 2. (Accession No. ML11109A083)
April 15, 2011	Letter from Allen B. S., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application, Sections 2.2 & 2.3 (TAC No. ME4640), License Renewal Application Amendment No. 3, and Revised License Renewal Application Boundary Drawings. (Accession No. ML11110A088)
April 19, 2011	Letter to Allen B. S., FENOC: Scoping and Screening Audit Report Regarding the Davis-Besse Nuclear Power Station License Renewal Application. (Accession No. ML111050091)
April 20, 2011	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis Bessie Nuclear Power Station—Batch 2 (TAC No. ME4640). (Accession No. ML110980718)
April 20, 2011	Letter from Allen B. S., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640), and License Renewal Application Amendment No. 4. (Accession No. ML11112A078)
April 29, 2011	Letter from Byrd K. W., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application, Section 2.1 (TAC No. ME4640), License Renewal Application Amendment No. 5, and Revised License Renewal Application Boundary Drawings. (Accession No. ML11126A016)
May 2, 2011	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station—Batch 3 (TAC No. ME4640). (Accession No. ML111170204)
May 5, 2011	Letter from Allen B. S., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application, Batch 1 (TAC No. ME4640) and License Renewal Application Amendment No. 6. (Accession No. ML11131A073)
May 19, 2011	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station—Operating Experience (TAC No. ME4640). (Accession No. ML11132A203)
May 24, 2011	Letter from Byrd K. W., FENOC: Reply to Requests for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1. License Renewal Application, Batch 2 and Batch 1 (TAC No. ME4640), and License Renewal Application Amendment No. 7. (Accession No. ML11151A090)
June 1, 2011	Letter to Allen B.S., FENOC: Audit Report Regarding the Davis-Besse Nuclear Power Station License Renewal Application. (Accession No. ML11122A014)

Date	Subject
June 3, 2011	Letter from Allen B. S., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application, Batch 3 (TAC No. ME4640), and License Renewal Application Amendment No. 8. (Accession No. ML11159A132)
June 17, 2011	Letter from Allen B. S., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640), and License Renewal Application Amendment No. 9. (Accession No. ML11172A389)
June 20, 2011	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station (TAC No. ME4640). (Accession No. ML11167A171)
June 24, 2011	Letter from Byrd K. W., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Batch 4 (TAC No. ME4640), and License Renewal Application Amendment No. 11. (Accession No. ML11180A060)
June 24, 2011	Letter from Byrd K. W., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4613) Environmental Report Severe Accident Mitigation Alternatives Analysis and License Renewal Application Amendment No. 10. (Accession No. ML11180A233)
June 27, 2011	Letter to Allen B., FENOC: Davis-Besse Nuclear Power Station NRC License Renewal Scoping, Screening, and Aging management Inspection Report 05000346/2011010 (Accession No. ML11179A134)
July 11, 2011	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station (TAC No. ME4640). (Accession No. ML11174A191)
July 12, 2011	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station (TAC No. ME4640). (Accession No. ML11189A043)
July 21, 2011	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station (TAC No. ME4640). (Accession No. ML11195A020)
July 21, 2011	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station (TAC No. ME4640). (Accession No. ML11196A127)
July 22, 2011	Letter from Allen B. S., FENOC: Reply to Requests for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1. License Renewal Application (TAC No. ME4640) and License Renewal Application Amendment No. 12. (Accession No. ML11208C274)
July 27, 2011	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station (TAC No. ME4640). (Accession No. ML11203A080)
August 11, 2011	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station (TAC No. ME4640). (Accession No. ML11216A236)
August 17, 2011	Letter from Byrd K. W., FENOC: Reply to Request for Additional Information for the Review of License Renewal Application and License Renewal Application Amendment No. 13. (Accession No. ML11231A966)
August 26, 2011	Letter from Byrd K. W., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640), and License Renewal Application Amendment No. 14. (Accession No. ML11242A166)
September 9, 2011	Summary of Telephone Conference Call Held on June 7, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests for Additional Information Pertaining to the Davis-Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML11242A007)

Date	Subject
September 12, 2011	Summary of Telephone Conference Call Held on August 4, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests for Additional Information Pertaining to the Davis-Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML11242A003)
September 16, 2011	Letter from Allen B. S., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640) and License Renewal Application Amendment No. 15. (Accession No. ML11264A059)
September 22, 2011	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station (TAC No. ME4640). (Accession No. ML11256A149)
September 30, 2011	Letter from Byrd K. W., FENOC: License Renewal Application Amendment No. 18 - Annual Update (TAC No. ME4640) (Accession No. ML11276A078)
October 7, 2011	Letter from Allen B. S., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). License Renewal Application Amendment No. 19, and Revised License Renewal Application Boundary Drawings. (Accession No. ML11285A064)
October 7, 2011	Letter to Allen B., FENOC: Davis-Besse Nuclear Power Station NRC License Renewal Aging Management Follow-up Inspection Report 05000346/2011012. (Accession No. ML11284A242)
October 11, 2011	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station (TAC No. ME4640). (Accession No. ML11271A147)
October 14, 2011	Congressional Correspondence—Letter from Markey E. J. to Jaczko G. B., NRC: Safe Operation of the Davis-Besse Nuclear Power Plant. (Accession No. ML11292A005)
October 21, 2011	Letter from Allen B. S., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640) and License Renewal Application Amendment No. 20. (Accession No. ML11298A097)
October 31, 2011	Letter from Allen B. S., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640) and License Renewal Application Amendment No. 21. (Accession No. ML11306A066)
October 31, 2011	Letter to Allen B. S., FENOC: Schedule Revision for the Environmental and Safety Review of the Davis-Besse Nuclear Power Station, Unit 1, License Renewal Application (TAC No. ME4613). (Accession No. ML11256A164)
November 8, 2011	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station License Renewal Application. (Accession No. ML11306A141)
November 21, 2011	Congressional Correspondence—Letter from Kucinich D. J. to Jaczko G. B., NRC: Cracks in the Concrete Wall of the Shield Building of the Davis-Besse Power Plant. (Accession No. ML11332A094)
November 22, 2011	Summary of Telephone Conference Call Held on October 26, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests for Additional Information Pertaining to the Davis-Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML11308A697)
November 23, 2011	Letter from Allen B. S., FENOC: Reply to Requests for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640), License Renewal Application Amendment No. 22, and Revised License Renewal Application Boundary Drawings. (Accession No. ML11335A223)

Date	Subject
December 2, 2011	Summary of Telephone Conference Call Held on September 29, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests for Additional Information Pertaining to the Davis-Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML11327A008)
December 2, 2011	Summary of Telephone Conference Call Held on July 27, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests for Additional Information Pertaining to the Davis-Besse Nuclear Power Station, License Renewal Application (TAC. No. ME4640). (Accession No. ML11327A079)
December 2, 2011	Summary of Telephone Conference Call Held on July 27, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests for Additional Information Pertaining to the Davis-Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML11327A087)
December 2, 2011	Letter to Allen B.S., FENOC: Davis-Besse Confirmatory Action Letter 3-11-001. (Accession No. ML11336A355)
December 2, 2011	Meeting Notice: December 15, 2011, Notice of Public Meeting with First Energy Nuclear Operating Company to Discuss Their Technical Analysis Regarding Cracking Identified in the Davis-Besse Shield Building. (Accession No. ML113360416)
December 7, 2011	Letter from Allen B. S., FENOC: Reply to Request for Supplemental Information for the Review of the License Renewal Application (TAC No. ME4640). (Accession No. ML11342A100)
December 12, 2011	Congressional Correspondence—Letter to Kucinich D.J., U.S. House of Representatives: December 12, 2011, Letter to Honorable Dennis Kucinich Providing Documents that First Energy Nuclear Operating Company Provided to the NRC. (Accession No. ML11347A341)
December 13, 2011	Summary of Telephone Conference Call Held on November 22, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests for Additional Information Pertaining to the Davis-Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML11339A086)
December 14, 2011	Summary of Telephone Conference Call Held on June 15, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests for Additional Information Pertaining to the Davis-Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML11341A118)
December 21, 2011	Summary of Telephone Conference Call Held on November 1, 2011 Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests for Additional Information Pertaining to the Davis Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML11348A021)
December 22, 2011	Meeting Notice: January 5, 2012, Notice of Public Meeting with First Energy Nuclear Operating Company to Discuss Present the Results of its Evaluation of Shield Building Cracking and for the NRC to Present the Results of its Independent Assessment of that Evaluation." (Accession No. ML113560164)
December 27, 2011	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML11333A396)
December 27, 2011	Letter to Allen B. S., FENOC: Schedule Revision for the Safety Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML11353A015)
December 29, 2011	Summary of Telephone Conference Call Held on December 8, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning RAIs Pertaining to the Davis-Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML11355A095)

Date	Subject
December 29, 2011	Press Release-III-11-037: NRC to Hold Meeting With FirstEnergy to Discuss Davis-Besse Shield Building Cracks. (Accession No. ML113630420)
January 13, 2012	Letter from Allen B. S., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640) and License Renewal Application Amendment No. 23. (Accession No. ML12018A338)
January 19, 2012	Summary of Telephone Conference Call Held on November 14, 2011,Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests for Additional Information Pertaining to the Davis-Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML11364A017)
January 23, 2012	Summary of Telephone Conference Call Held on June 29, 2011, Between the U.S. Nuclear regulatory Commission and First Energy Nuclear Operating Company, Concerning Request for Additional Information Pertaining to the Davis-Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML11363A167)
January 23, 2012	Summary of Telephone Conference Call held on December 12, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Request for Additional Information Pertaining to the Davis Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML11363A171)
January 23, 2012	Summary of Telephone Conference Call Held on November 9, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Request for Additional Information Pertaining to the Davis-Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML12018A165)
January 30, 2012	Summary of Telephone Conference Call Held on August 22, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Request for Additional Information Pertaining to the Davis-Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML11363A174)
January 31, 2012	Letter to Allen B. S., FENOC: Schedule Revision for the Environmental Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4613). (Accession No. ML12032A131)
February 1, 2012	Summary of Telephone Conference Call Held on June 28, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Request for Additional Information Pertaining to the Davis Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML12018A022)
February 3, 2012	Summary of Telephone Conference Call Held on June 30, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Request for Additional Information Pertaining to the Davis Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML12018A046)
February 9, 2012	Summary of Telephone Conference Call Held on June 16, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Request for Additional Information Pertaining to the Davis-Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML12018A146)
February 16, 2012	Summary Of Telephone Conference Call Held on January 24, 2012, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests for Additional Information Pertaining to the Davis-Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML12045A016)
February 21, 2012	Summary of Telephone Conference Call Held on October 31, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Request for Additional Information Pertaining to the Davis-Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML12024A276)

Date	Subject
February 22, 2012	Summary of Telephone Conference Call Held on September 7, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests for Additional Information Pertaining to the Davis-Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML12025A047)
February 22, 2012	Summary of Telephone Conference Call Held on October 5, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests for Additional Information Pertaining to the Davis-Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML12038A197)
February 24, 2012	Summary Of Telephone Conference Call Held On September 16, 2011, Between The U.S. Nuclear Regulatory Commission And Firstenergy Nuclear Operating Company Concerning Requests for Additional Information Pertaining to the Davis-Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML12039A013)
February 27, 2012	Summary of Telephone Conference Call Held on July 15, 2011, Between The U.S. Nuclear Regulatory Commission And FirstEnergy Nuclear Operating Company, Concerning Requests For Additional Information Pertaining To The Davis-Besse Nuclear Power Station, License Renewal Application (TAC No. ME4640). (Accession No. ML12052A171)
February 27, 2012	Summary of Telephone Conference Call Held on August 2, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests for Additional Information Pertaining to the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML12033A060)
February 28, 2012	Summary of Telephone Conference Call Held on August 29, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests for Additional Information Pertaining to the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML12052A285)
March 2, 2012	Summary of Telephone Conference Call Held on July 13, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests For Additional Information Pertaining to the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML12031A183)
March 9, 2012	Letter from Allen B. S., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML12094A383)
April 2, 2012	Summary of Telephone Conference Call Held on July 12, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests For Additional Information Pertaining to the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML12061A258)
April 2, 2012	Summary of Telephone Conference Call Held on February 9, 2012, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests For Additional Information Pertaining to the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML12059A078)
April 5, 2012	Letter from Imlay D. M., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML12097A520)
May 15, 2012	Summary of Telephone Conference Call Held on July 19, 2011, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests For Additional Information Pertaining to the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML12052A258)
May 25, 2012	Letter from Allen B. S., FENOC: Supplemental Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML12151A120)

Date	Subject
May 25, 2012	Letter from Allen B. S., FENOC: Correction of Typographical Error in Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML12151A119)
May 31, 2012	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station License Renewal (TAC No. ME4640). (Accession No. ML12144A038)
June 12, 2012	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station License Renewal (TAC No. ME4640). (Accession No. ML12160A016)
June 14, 2012	Letter from Allen B. S., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML12167A369)
June 22, 2012	Summary of Telephone Conference Call Held on April 26, 2012, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests For Additional Information Pertaining to the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No.ML12157A238)
June 22, 2012	Summary of Telephone Conference Call Held on April 17, 2012, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests For Additional Information Pertaining to the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML12157A231)
July 5, 2012	Letter from Allen B. S., FENOC: Supplemental Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML12191A037)
July 9, 2012	Summary of Telephone Conference Call Held on June 21, 2012, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests For Additional Information Pertaining to the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML12185A020)
July 11, 2012	Letter to Allen B. S., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station License Renewal (TAC No. ME4640). (Accession No. ML12191A192)
July 11, 2012	Letter from Allen B. S., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML12198A019)
July 26, 2012	Letter to Allen B. S., FENOC: Supplemental Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station License Renewal (TAC No. ME4640). (Accession No. ML12201B519)
August 9, 2012	Letter from Allen B. S., FENOC: License Renewal Application Amendment No. 30 - Annual Update (TAC No. ME4640). (Accession No. ML12229A139)
August 16, 2012	Letter from Imlay D. M., FENOC: Supplemental Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML12230A221)
August 16, 2012	Letter from Imlay D. M., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML12230A220)
August 24, 2012	Letter from Allen B. S., FENOC: Reply to Supplemental Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML12240A219)

Date	Subject
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October 26, 2012	Letter to Lieb R. A., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station (TAC No. ME4640). (Accession No. ML12292A627)
November 2, 2012	Letter from Lieb R. A., FENOC: Supplemental Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML12311A024)
November 8, 2012	Summary of Telephone Conference Call Held on August 1, 2012, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests For Additional Information Pertaining to the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML12299A396)
November 14, 2012	Letter to Lieb R. A., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station License Renewal Application Related to High Strength Bolting (TAC No. ME4640). (Accession No. ML12318A246)
November 20, 2012	Letter from Imlay D. M., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640) and License Renewal Application Amendment No. 36. (Accession No. ML12331A125)
December 11, 2012	Letter from FENOC: Davis-Besse, Unit 1, Enclosure A to L-12-444, Calculation 32-9195423-000, "DB-1 EMA of RPV Inlet & Outlet Nozzle-to-Shell Welds for 60 Years - Non-Proprietary." (Accession No. ML13009A374)
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December 20, 2012	Letter to Lieb R. A., FENOC: Supplemental Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1 (Davis-Besse), License Renewal Application, Section B2.2. (Accession No. ML12361A255)
January 4, 2013	Letter to Lieb R. A., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station License Renewal Application Related to the Shield Building Monitoring Program (TAC No. ME4640). (Accession No. ML12355A184)
January 7, 2013	Letter from FENOC: Submittal of Contractor Equivalent Margins Assessments for Reactor Vessel Welds (Nonproprietary Versions) (TAC No. ME4640). (Accession No. ML13009A373)
January 7, 2013	Letter from Lieb R. A., FENOC: Davis-Besse Nuclear Power Station, Unit 1, Notification of Closure of Commitments Related to the Review of the License Renewal Application (TAC No. ME4640). (Accession No. ML13008A330)
January 21, 2013	Letter from Imlay D. M., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML13023A113)
January 29, 2013	Letter to Lieb R. A., FENOC: Schedule Revision for the Safety and Environmental Review of the Davis-Besse Nuclear Power Station License Renewal Application (TAC NO. ME4640). (Accession No. ML13011A301)
February 12, 2013	Letter from Imlay D. M., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML13044A499)

Date	Subject
February 14, 2013	Summary of Telephone Conference Call Held on October 23, 2012, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests For Additional Information Pertaining to the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML13025a027)
February 14, 2013	Letter to Lieb R. A., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station License Renewal Application Related to the Bolting Integrity Program (TAC No. ME4640). (Accession No. ML13038A118)
February 28, 2013	Letter from Lieb R. A., FENOC: Revised Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML13063A033)
March 14, 2013	Letter from Lieb R. A., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML13077A391)
March 26, 2013	Letter to Lieb R. A., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML13051A056)
March 26, 2013	Summary of Telephone Conference Call Held on January 16, 2013, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests For Additional Information Pertaining to the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML13051A068)
March 26, 2013	Summary of Telephone Conference Call Held on January 23, 2013, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests For Additional Information Pertaining to the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML13051A660)
April 19, 2013	Letter from Lieb R. A., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML13114A254)
May 17, 2013	Letter from Lieb R. A., FENOC: Supplemental Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML13141A271)
May 21, 2013	Letter from Lieb R. A., FENOC: Supplemental Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML13144A079)
June 3, 2013	Letter from Lieb R. A., FENOC: Supplemental Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML13156A388)
June 4, 2013	Letter from Imlay D. M., FENOC: Supplemental Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML13158A302)
July 2, 2013	Summary of Telephone Conference Call Held on May 21, 2013, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests For Additional Information Pertaining to the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML13142A512)
July 9, 2013	Summary of Telephone Conference Call Held on May 9, 2013, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests For Additional Information Pertaining to the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML13149a288)

Date	Subject
July 12, 2013	Summary of Telephone Conference Call Held on May 2, 2013, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests For Additional Information Pertaining to the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML13149A286)
July 12, 2013	Summary of Telephone Conference Call Held on April 11, 2013, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests For Additional Information Pertaining to the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML13182A443)
July 17, 2013	Summary of Telephone Conference Call Held on October 16, 2012, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests For Additional Information Pertaining to the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML13182A408)
July 17, 2013	Summary of Telephone Conference Call Held on May 28, 2013, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests For Additional Information Pertaining to the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML13197A257)
July 19, 2013	Summary of Telephone Conference Call Held on April 24, 2013, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Requests For Additional Information Pertaining to the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML13196A460)
July 23, 2013	Letter from Lieb R. A., FENOC: Notification of Completion of a License Renewal Commitment Related to the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (Accession No. ML13206A382)
July 31, 2013	Letter to Lieb R. A., FENOC: Schedule Revision for the Environmental and Safety Review of the Davis-Besse Nuclear Power Station License Renewal Application (TAC Nos. ME4613 and ME4640). (Accession No. ML13205A036)
August 29, 2013	Summary of Telephone Conference Call Held on August 22, 2013, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning License Conditions Pertaining to the Davis-Besse Nuclear Power Station License Renewal Application (TAC No. ME4640). (Accession No. ML13240A070)

APPENDIX C

PRINCIPAL CONTRIBUTORS

This appendix lists the principal contributors for the development of this safety evaluation report (SER) and their areas of responsibility.

Name	Responsibility
Alley, D.	Reviewer—Mechanical
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Bloom, S.	Management Oversight
Buford, A.	Reviewer—Structural
Casto, G.	Management Oversight
Collins, J.	Reviewer—Materials
Cuadrado de Jesús, S.	Project Manager
Cunningham, M.	Management Oversight
Davidson, E.	Reviewer—Balance of Plant
Dennig, R.	Management Oversight
Dias, A.	Management Oversight
Diaz-Sanabria, Y.	Management Oversight
Doutt, C.	Reviewer—Electrical
Doyle, D.	Project Manager
Dozier, J.	Management Oversight
Elliot, B.	Reviewer—Mechanical
Evans, M.	Management Oversight
Fu, B.	Reviewer—Mechanical
Gall, J.	Reviewer—Mechanical
Galloway, M.	Management Oversight
Gavula, J.	Reviewer—Mechanical
Gettys, E.	Project Manager
Giitter, J.	Management Oversight
Gilanshahi, N.	Reviewer—Mechanical
Green, K.	Reviewer—Mechanical
Harris, B.	Project Manager
Hiland, P.	Management Oversight
Hiser, A.	Management Oversight
Hoang, D.	Reviewer—Mechanical

Name	Responsibility
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Homiack, M.	Reviewer—Mechanical
Hunt, C.	Reviewer—Chemical
Istar, A.	Reviewer—Structural
Iqbal, N.	Reviewer—Fire Protection
Jackson, C.	Management Oversight
Kalikian, R.	Reviewer—Mechanical
Khana, M.	Management Oversight
Kichline, M.	Reviewer—Mechanical
Klein, A.	Management Oversight
Klos, J.	Reviewer—Mechanical
Kulesa, G.	Management Oversight
Lee, B.	Reviewer—Mechanical
Lehman, B.	Reviewer—Structural
Li, R.	Reviewer—Electrical
Lubinski, J.	Management Oversight
Lupold, T.	Management Oversight
Marshall, M.	Management Oversight
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Nguyen, D.	Reviewer—Electrical
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Parks, B.	Reviewer—Mechanical
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Name	Responsibility
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Raval, J.	Reviewer—Mechanical
Razzaque, M.	Reviewer—Mechanical
Rhow, S.	Reviewer—Electrical
Rogers, B.	Reviewer—Scoping & Screening Methodology
Rosenberg, S.	Management Oversight
Ruland, W.	Management Oversight
Sahay, P.	Reviewer—Electrical
Sheikh, A.	Reviewer—Structural
Sheng, S.	Reviewer—Mechanical
Smith, E.	Reviewer—Scoping & Screening Methodology
Smith, W.	Reviewer—Mechanical
Sun, R.	Reviewer—Mechanical
Sydnor, C.	Reviewer—Mechanical
Taylor, R.	Management Oversight
Tsao, J.	Reviewer—Materials
Ulses, A.	Management Oversight
Uribe, J.	Reviewer—Mechanical
Wilson, G.	Management Oversight
Wise, J.	Reviewer—Mechanical
Wong, A.	Reviewer—Mechanical
Wong, E.	Reviewer—Chemical
Yee, O.	Reviewer—Mechanical
Contract Support	
Argonne National Laboratory	Technical Review
Center for Nuclear Regulatory Analysis	Technical Review
Oak Ridge National Laboratories	Technical Review
Ian, Evan, & Alexander Corporation	SER Support

APPENDIX 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 Davis-Besse Nuclear Power Station (Davis-Besse), Docket Number 50-346.

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11. ABSTRACT (200 words or less)
This safety evaluation report (SER) documents the technical review of the Davis-Besse Nuclear Power Station (Davis-Besse) license renewal application (LRA) by the U.S. Nuclear Regulatory Commission (NRC or the staff). By letter dated August 27, 2010, FirstEnergy Nuclear Operating Company (FENOC) 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." FENOC requests renewal of Davis-Besse operating license (Facility Operating License Number NPF-3) for a period of 20 years beyond the current license date of April 22, 2017.

Davis-Besse is located approximately 20 miles east of Toledo, Ohio. The NRC issued the construction permit on March 24, 1971. The NRC issued the operating license on April 22, 1977. The unit is a pressurized-water reactor design with a dry ambient containment. Babcock and Wilcox Corporation supplied the nuclear steam supply system, and Bechtel designed and constructed the balance of the plant. The licensed power output of the unit is 2,817 megawatt thermal with a gross electrical output of approximately 908 megawatt electric.

This SER presents the status of the staff's review of information submitted through June 4, 2013, the cutoff date for consideration in the SER. The staff has resolved all issues associated with requests for additional information and closed all open items since publishing the SER with Open Items. The staff did not identify any new open items that must be resolved before any final determination can be made on the LRA.

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