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Volume 2

Safety Evaluation Report

Related to the License Renewal of Columbia Generating Station

Volume 2

Docket Number 50-397

Energy Northwest

Office of Nuclear Reactor Regulation

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ABSTRACT

This safety evaluation report (SER) documents the technical review of the Columbia Generating Station (Columbia), license renewal application (LRA) by the U.S. Nuclear Regulatory Commission (NRC) staff (the staff). By letter dated January 19, 2010, Energy Northwest (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." Energy Northwest requests renewal of the operating license (Facility Operating License Number NPF-21) for a period of 20 years beyond the current license period of December 20, 2023. Columbia is located approximately 12 miles north of Richland, WA. The NRC issued the construction permit on March 19, 1973, and the operating license for Columbia on April 13, 1984. The unit is a Mark II boiling-water reactor (BWR) design. General Electric Company supplied the nuclear steam supply system. Burns and Roe, Inc., designed the balance of plant, and Bechtel Power Corporation constructed the plant. The licensed power output of the unit is 3,886 megawatts thermal, with a gross electrical output of approximately 1,230 megawatts electric. This SER presents the status of the staff's review of information submitted through January 4, 2012. The staff closed six open items previously identified in the SER with open items. SER Section 1.5 summarizes the closure of the open items.

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ABBREVIATIONS

AAI	applicant action item
AC	alternating current
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
ACSR	aluminum conductor steel reinforced
ADAMS	Agencywide Documents Access and Management System
ADS	automatic depressurization system
AERM	aging effect requiring management
AMP	aging management program
AMR	aging management review
ANSI	American National Standards Institute
AQ	augmented quality
AR	Action Request
ART	adjusted reference temperature
ASME	American Society of Mechanical Engineers
ASTM	American Standards for Testing and Materials
ATWS	anticipated transient without scram
B&PV	boiler and pressure vessel
B_4C	boron carbide
BADGER	Boron-10 Areal Density Gage for Evaluating Racks
BTP	Branch Technical Position
BWR	boiling-water reactor
BWRVIP	Boiling-Water Reactor Vessel and Internals Project
C	Celsius
CAS	control air system
CASS	cast austenitic stainless steel
CB&I	Chicago Bridge and Iron
CCH	control room chilled water
CEA	control element assembly
CEP	containment exhaust purge
CF	chemistry factor
CFR	Code of Federal Regulations
CI	confirmatory item
CIA	containment instrument air
CLB	current licensing basis
CMS	containment monitoring system
CN	containment nitrogen

CO	condensate (auxiliary)
CO_2	carbon dioxide
Columbia	Columbia Generating Station
CPR	condensate processing radioactive
CR	condition report
CRA	containment return air
CRD	control rod drive
CRDRL	control rod drive return line
CSP	containment supply purge
CSR	cable spreading room
CST	condensate storage tank
CUF	cumulative usage factor
CVB	containment vacuum breaker
CVN	Charpy-V notch
CW	circulating water
DBA	design basis accident
DBE	design basis event
DCW	diesel cooling water
DE	diesel (engine) exhaust
DEH	digital electro-hydraulic control system
DG	diesel generator
DLO	diesel lubricating oil
DO	dissolved oxygen
DOT	Department of Transportation
ΔRT_{NDT}	reference nil-ductility temperature caused by irradiation
DSA	diesel starting air
DW	demineralized water
EAF	environmentally-assisted fatigue
ECCS	emergency core cooling system
EDR	equipment drain radioactive
EFPY	effective full power years
EMA	equivalent margin analysis
EPRI	Electrical Power Research Institute
EQ	environmental qualification
ESF	engineered safety feature
F	Fahrenheit
FD	floor drain
FDR	floor drain radioactive

F_{en}	environmental life correction factors
FERC	Federal Energy Regulatory Commission
FIV	flow-induced vibrators
FPC	fuel pool cooling
FR	Federal Register
FSAR	final safety analysis report
ft	foot
ft^2	square foot
FW	feedwater
GALL	generic aging lessons learned
GE	General Electric
GEIS	generic environmental impact statement
GL	Generic Letter
HAZ	heat-affected zone
HCO	heating steam condensate
HELB	high-energy line break
HPCS	high-pressure core spray
HS	heating steam
HSV	heating steam vent
HVAC	heating, ventilation, and air conditioning
HWC	hydrogen water chemistry
I&C	instrumentation and control
IASCC	irradiation-assisted stress-corrosion cracking
ID	inside diameter
IGA	intergranular attack
IGSCC	intergranular stress-corrosion cracking
IN	Information Notice
in.	inch
INPO	Institute of Nuclear Plant Operation
IPA	integrated plant assessment
IR	infrared
ISG	interim staff guidance
ISI	inservice inspection
ISP	Integrated Surveillance Program
K_i	applied stress intensity
K_{ic}	fracture toughness based on crack initiation
kV	kilovolt

lb	pound
LBB	leak-before-break
LD	leak detection
LER	licensee event report
LOCA	loss of coolant accident
LPCI	low pressure coolant injection
LPCS	low-pressure core spray
LRA	license renewal application
LRIC	License Renewal Implementation Coordinator
LRP	leak reduction program
LTR	licensing topical report
LWR	light-water reactor
M	margin term
MEB	metal-enclosed bus
MEL	master equipment list
MIC	microbiologically-influenced corrosion
MRSM	maintenance rule scoping matrix
MS	main steam
MSIV	main steam isolation valve
MSLC	main steam leakage control
MWR	miscellaneous waste radioactive
MWt	megawatt thermal
NACE	National Association of Corrosion Engineers
NDE	non-destructive examination
NEI	Nuclear Energy Institute
NESC	National Electrical Safety Code
NFPA	National Fire Protection Association
NMCA	noble metal chemical application
NPS	nominal pipe size
NRC	U.S. Nuclear Regulatory Commission
NSAC	Nuclear Safety Analysis Center
NSAS	nonsafety affecting safety
NSSS	nuclear steam supply system
nvt	measure of fluence in n/cm^2
NWC	normal water chemistry
OBE	operating-basis earthquake
OD	outside diameter
OEM	original equipment manufacturer

OI	open item
OQAPD	operational quality assurance program description
OTSG	once-through steam generator
P&ID	piping and instrumentation drawing
PDI	performance demonstrative initiative
PER	problem evaluation request
PFSS	post-fire safe shutdown
PGCC	Power Generation Control Cabinet
PM	preventive maintenance
ppb	parts per billion
ppm	parts per million
PS	process sampling
PSD	plant sanitary drain
PSR	process sampling radioactive
P-T	pressure-temperature
PVC	polyvinyl chloride
PWC	potable cold water
PWH	potable hot water
PWR	pressurized-water reactor
PWSCC	primary water stress-corrosion cracking
QA	quality assurance
QIDs	qualification information documents
RAI	request for additional information
RCC	reactor closed cooling water
RCIC	reactor core isolation cooling
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RFO	refueling outage
RFW	reactor feedwater
RG	regulatory guide
RH	relative humidity
RHR	residual heat removal
RI-ISI	risk-informed inservice inspection
RP	regulatory position
RPS	reactor protection system
RPV	reactor pressure vessel
RRC	reactor recirculation
RT_{NDT}	reference nil-ductility temperature

RV	reactor vessel
RVI	reactor vessel internals
RVID	Reactor Vessel Integrity Database
RWCU	reactor water cleanup
SA	service air
SBO	station blackout
SC	structure and component
SCC	stress-corrosion cracking
SDV	scram discharge volume
SER	safety evaluation report
SFA	steam/feedwater application
SFP	spent fuel pool
SGT	standby gas treatment
SLC	standby liquid control
SPTM	suppression pool temperature monitoring
SRP-LR	Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants
SRRF	stress range reduction factor
SRV	safety relief valve
SSC	structure, system, and component
SSE	safe shutdown earthquake
SSP	Supplemental Surveillance Program
SSW	standby service water
TIP	traversing incore probe
TLAA	time-limited aging analysis
TMU	tower makeup water
TS	technical specification
TSW	plant service water
UFSAR	updated final safety analysis report
USE	upper-shelf energy
UT	ultrasonic testing
UV	ultraviolet
V	volt
VAC	volts alternating current
VIP	Vessel Internals Program
WCH	radwaste building chilled water
Zn	zinc

SECTION 4

TIME-LIMITED AGING ANALYSES

4.1 Identification of Time-Limited Aging Analyses

Certain plant-specific safety analyses involve time-limited assumptions defined by the current operating term. Pursuant to Section §54.21(c)(1) of Title 10 of the *Code of Federal Regulations* (10 CFR 54.21(c)(1)), applicants must list those analyses in the current licensing basis (CLB) that meet the definition of a time-limited aging analysis (TLAA), as defined in 10 CFR 54.3.

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

This section of the safety evaluation report (SER) provides the staff's evaluation of the applicant's basis for identifying those plant-specific or generic analyses that need to be identified as TLAAs for the license renewal application (LRA). This section of the SER also provides the staff's evaluation of the applicant's basis for concluding that its LRA identifies all exemptions in its CLB that are based on a TLAA.

4.1.1 Summary of Technical Information in the Application

LRA Section 4.1 provides the basis for identifying the applicant's analyses 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 complied with the six criteria for defining an analysis as a TLAA, as specified in 10 CFR 54.3. The list of TLAAs provided in LRA Table 4.1-1 meet the six criteria of a TLAA. The applicant stated that it reviewed the list of common TLAAs in NUREG-1800, Revision 1, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR), dated September 2005. The applicant also stated that its review of the CLB included a review of the updated final safety analysis report (UFSAR), fire protection evaluation, Quality Assurance Program, Inservice Inspection Program, docketed licensing correspondence, operating license (including technical specifications (TSs)), Code exemptions and relief requests, and design calculations and design reports.

Pursuant to 10 CFR 54.21(c)(2), the applicant stated that it did not identify exemptions granted under 10 CFR 50.12 based on a TLAA as defined in 10 CFR 54.3.

4.1.2 Staff Evaluation of the Applicant's Identification of TLAAs

As defined in 10 CFR 54.3, an analysis in the CLB meets the definition of a TLAA if it complies with all of the following six criteria:

(1) involves systems, structures, and components within the scope of license renewal, as described in 10 CFR 54.4(a)

(2) considers the effects of aging

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

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

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

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

The staff's Statement of Considerations (SOC) on 10 CFR Part 54—provided in Section III.g.(i) of *Federal Register* Notice, Volume 60, Number 88 (FRN Volume 60, No. 88, dated May 8, 1995)—provides additional clarification on when an analysis in the CLB needs to be identified as a TLAA. SRP-LR Section 4.1 provides additional guidance on when an analysis in the CLB needs to be identified as a TLAA. The staff noted that LRA Table 4.1-1 identifies the analyses in the CLB that meet the definition of a TLAA in 10 CFR 54.3. The staff's evaluations of the applicant's disposition for these TLAAs are documented in the applicable subsections of SER Section 4.

The staff also noted that LRA Tables 4.1-1 and 4.1-2 identify the following analyses in the CLB that do not meet the definition of a TLAA:

- 4.2 Reactor Vessel (RV) Neutron Embrittlement Analysis
  - Neutron Fluence
- 4.3 Metal Fatigue Analysis
  - Flow-Induced Vibration (FIV) Endurance Limit for the RV Internals
  - Ductility Reduction of Fracture Toughness for the RV Internals
  - Leak-Before-Break (LBB) Analysis
- 4.5 Concrete Containment Tendon Prestress Analysis
- 4.6 Containment Liner Plate, Metal Containment, and Penetration Fatigue Analysis
  - Fatigue Analysis of Containment Liner Plate
- 4.7 Plant-Specific TLAAs
  - Intergranular Separation of Heat-Affected Zone (HAZ) of RV Low-Alloy Steel Under Austenitic Stainless Steel Cladding
  - Low-Temperature Overpressure Protection (LTOP) Analyses
  - Fatigue Analysis of Reactor Coolant Pump Flywheel
  - Fatigue Analysis of Polar Crane
  - Metal Corrosion Allowance
  - Inservice Local Metal Containment Corrosion Analyses

For each of these analyses, the staff reviewed the applicant basis for claiming the analysis was not a TLAA and compared it to the applicant's CLB and the six criteria for TLAAs. The staff also used the guidance in SRP-LR Sections 4.1.2 and 4.1.3 and the clarifications in Section III.g.(i) of the SOC on 10 CFR Part 54 (FRN Volume 60, No. 88).

4.1.2.1 Neutron Fluence

Pursuant to 10 CFR 50.60, applicants are required to define the end-of-life fluence. LRA Section 4.2.1 states that the neutron fluence values for 51.6 effective full power years (EFPY) of reactor operation are addressed in UFSAR Section 4.3.2.8 and UFSAR Table 4.3-1. The applicant clarified that these fluence analyses are based on the original licensed thermal power of 3,323 megawatt thermal (MWt) through fuel cycle 10, and the currently licensed thermal power uprated to 3,486 MWt from cycle 11 through the end of operation. However, LRA Table 4.1-1 shows that the neutron fluence analysis is not a TLAA. The staff's evaluation of the neutron fluence analysis is documented in SER Section 4.2.1 and includes an assessment on whether the applicant's neutron fluence analysis meets the definition of a TLAA and is identified in accordance with 10 CFR 54.21(c)(1).

4.1.2.2 Flow-Induced Vibration Endurance Limit for the Reactor Vessel Internals

SRP-LR Table 4.1-3 identifies "Flow-Induced Vibration Endurance Limit for the Reactor Vessel Internals" as an analysis that may be generically applicable to an applicant's CLB. LRA Table 4.1-2 states that no analyses were identified within the CLB for the RV internals related to flow-induced vibration (FIV) endurance limit. UFSAR Section 3.9.2.3 states that the major reactor internal components within the vessel were subjected to extensive testing coupled with dynamic systems analyses to properly describe the resulting FIV phenomena incurred from normal operation and from anticipated operational transients. UFSAR Section 3.9.2.4 states that the reactor internals were tested in accordance with the provisions of Regulatory Guide (RG) 1.20, Revision 2, for non-prototype Category IV plants using Tokai-2 as the limited valid prototype. The applicant further stated that the test procedure involved taking vibration measurements to determine the vibration characteristics of reactor internals during the initial approach to full power operation. In addition, vibratory responses were recorded at various power levels and recirculation flow rates.

The applicant's justification for not considering the RV internals FIV analysis as a TLAA is based on the determination that the analysis does not involve a time-limited aging effect related to FIV for the licensed operating period. The staff reviewed the CLB and determined that the reactor internals FIV analysis does not consider a time-limited aging effect and does not involve time-limited assumptions defined by the current operating term of 40 years. The staff also noted that the staff approved a 4.9 percent stretch power uprate (SPU) in a staff evaluation (SE) dated May 2, 1995 (Agencywide Document Access and Management System (ADAMS) Accession No. ML022120154). In its SPU amendment request, the applicant indicated that the stretch power uprate did not have any impact on the FIV loads assumed for the design of the RV internals.

The staff reviewed the SPU SE and noted that Section 3.2.3 of the SE concludes that the 4.9 percent stretch power uprate would have no or little effect on the FIV assumptions for the RV internals because the uprated conditions did not create any change to the maximum allowable core flow. The staff also noted that the staff's evaluation of the RV and RV internal components under SPU loads did not include an assessment of any age-related degradation in the SPU SE.

Based on this review, the staff concludes that the RV internals FIV analysis does not meet the definition of a TLAA and does not need to be identified as a TLAA, in accordance with 10 CFR 54.21(c)(1). The analysis does not consider the effects of aging (Criterion 2 of

10 CFR 54.3(a)) and does not involve time-limited assumptions defined by the current operating term (Criterion 3 of 10 CFR 54.3(a)).

4.1.2.3 Ductility Reduction of Fracture Toughness for the Reactor Vessel Internals

SRP-LR Table 4.1-3 identifies “Ductility Reduction of Fracture Toughness for RV Internals" as an analysis that may be generically applicable to an applicant’s CLB. LRA Table 4.1-2 states that no analyses were identified within the CLB for the RV internals related to ductility reduction of fracture toughness. The staff reviewed the CLB, including the UFSAR, and confirmed that an analysis of ductility reduction of fracture toughness for the reactor internals is not contained or incorporated by reference in the CLB. The staff also confirmed that the applicant addressed the potential for reduction in fracture toughness properties for the RV internals through the implementation of the Boiling Water Reactor (BWR) Vessel and Internals Program (BWRVIP), including applicable augmented BWR Vessel and Internals Program inspections and flaw evaluation reports. The staff confirmed that the applicant’s BWR Vessel and Internals Program evaluates the impact that a reduction of fracture toughness will have on flaw acceptance.

Based on this review, the staff concludes that the generic reduction of fracture toughness analysis in SRP-LR Table 4.1-3 is not applicable to the applicant. It does not need to be identified as TLAA, in accordance with 10 CFR 54.21(c)(1), because the applicant does not use a time-dependent analysis to manage potential reduction of fracture toughness in its RV internal components, and it is not contained or incorporated in the CLB (Criterion 6 of 10 CFR 54.3(a)). The staff noted that the applicant credits its BWR Vessel and Internals Program or its Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program to manage potential reduction of fracture toughness in the RV internals. The staff’s evaluations of these programs are documented in SER Sections 3.0.3.1.6 and 3.0.3.1.30, respectively.

4.1.2.4 Leak-Before-Break Analysis

SRP-LR Table 4.1-3 identifies "Leak Before Break" (LBB) as an analysis that may be generically applicable to an applicant’s CLB. LRA Table 4.1-2 states that the applicant does not credit LBB.

The staff noted that it has approved LBB analyses for high-energy and large-bore piping systems in the reactor coolant pressure boundaries (RCPBs) of PWR facilities. The staff confirmed that it currently has not approved LBB analyses for any BWR nuclear plants. The staff reviewed the applicant’s UFSAR and confirmed that the plant is a BWR nuclear plant.

Based on this review, the staff concludes that the generic LBB analysis does not meet the definition of a TLAA. It does not need to be identified as a TLAA, in accordance with 10 CFR 54.21(c)(1), for the following reasons:

- The plant is a BWR design and the NRC has not approved the use of an LBB analysis for any BWR facility.
- The staff confirmed that an LBB analysis is not contained or incorporated by reference in the applicant’s CLB (Criterion 6 of 10 CFR 54.3(a)).

4.1.2.5 Concrete Containment Tendon Pre-stress Analysis

SRP-LR Table 4.1-2 identifies "Concrete Containment Tendon Prestress" as an analysis that may be generically applicable to an applicant’s CLB. LRA Table 4.1-2 and LRA Section 4.5

identify that this analysis is not a TLAA for Columbia because it has a General Electric (GE) Mark II primary containment and this structure does not include pre-stressed tendons.

The staff reviewed UFSAR Section 3.8 and confirmed that the applicant's primary containment is a steel containment structure, which does not use tendons. Thus, the staff confirmed that the generic concrete containment tendon analysis listed in SRP-LR Table 4.1-2 is not applicable to the applicant's CLB or design basis.

Based on this review, the staff concludes that the concrete containment tendon pre-stress analysis listed in SRP-LR Table 4.1-2 is not a TLAA for the applicant. This analysis is not a TLAA, in accordance with 10 CFR 54.21(c)(1), because the applicant has a Mark II containment that does not include containment tendons. The staff confirmed that a concrete containment tendon pre-stress analysis is not contained or incorporated by reference in the applicant's CLB (Criterion 6 of 10 CFR 54.3(a)).

4.1.2.6 Fatigue Analysis of Containment Liner Plate

SRP-LR Table 4.1-3 identifies "Fatigue Analysis of Containment Liner Plate" as an analysis that may be generically applicable to an applicant's CLB. LRA Table 4.1-2 identifies that this analysis does not meet the definition of a TLAA because the plant does not have a liner plate. The applicant further stated that the fatigue analysis of the metal containment shell is described in LRA Section 4.6.1.

The staff reviewed UFSAR Section 3.8 and confirmed that the applicant's primary containment is a steel containment structure, which does not not have a liner plate. Thus, the staff confirmed that the generic containment liner plate analysis listed in SRP-LR Table 4.1-3 is not applicable to the applicant's CLB or design basis.

Based on this review, the staff concludes that the containment liner plate fatigue analysis listed in SRP-LR Table 4.1-3 is not a TLAA for the applicant. This analysis is not a TLAA, in accordance with 10 CFR 54.21(c)(1), because the staff confirmed that the plant does not have a liner plate and a fatigue analysis of containment liner plate is not contained or incorporated by reference in the applicant's CLB (Criterion 6 of 10 CFR 54.3(a)).

4.1.2.7 Intergranular Separation in the Heat-Affect-Zone (HAZ) of Reactor Vessel Low-Alloy Steel under Austenitic Stainless Steel Cladding

SRP-LR Table 4.1-3 identifies "Intergranular Separation in the HAZ of RV Low-Alloy Steel under Austenitic SS Cladding" as an analysis that may be generically applicable to an applicant's CLB. LRA Table 4.1-2 states that no such analysis was identified within the CLB for Columbia.

The staff noted that SRP-LR Section 3.1.2.2.5 states that RV underclad cracking is only applicable to RVs whose designs include SA-508 Class 2 or 3 forging shells or forging nozzles that were welded to the vessel using a high heat input welding process. The staff confirmed in BWRVIP-74-A that the applicant's RV is fabricated from SA-533 low-alloy plate materials and does not include SA-508 Class 2 or 3 low-alloy shell or nozzle forging materials.

Based on this review, the staff concludes that the generic RV underclad cracking analysis listed in SRP-LR Table 4.1-3 is not a TLAA for the applicant. This analysis is not a TLAA in accordance with 10 CFR 54.21(c)(1) because the applicant's RV design does not include SA-508 Class 2 or 3 forging shells or forging nozzles that were welded to the vessel using a high heat input welding process. Additionally, the staff confirmed that an analysis for intergranular separation in the HAZ of RV low-alloy steel under austenitic stainless steel cladding is not contained or incorporated by reference in the applicant's CLB (Criterion 6 of 10 CFR 54.3(a)).

4.1.2.8 Low-Temperature Overpressure Protection Analyses

SRP-LR Table 4.1-3 identifies "Low Temperature Overpressurization Protection Analysis" as an analysis that may be generically applicable to a plant's CLB. LRA Table 4.1-2, stated that the CLB does not include a low temperature overpressurization protection (LTOP) analysis.

The staff noted that the generic LTOP analysis in SRP-LR Table 4.1-3 is only applicable to the LTOP systems that are included in designs of pressurized water reactor (PWR) facilities. As noted previously, the Columbia plant is a BWR facility.

Based on this review, the staff finds that the applicant has provided an acceptable basis for concluding that the generic LTOP analysis in the SRP-LR does not need to be identified as a TLAA because the staff has confirmed that the generic analysis is only applicable to PWR design facilities and an LTOP analysis is not contained or incorporated by reference in the applicant's CLB (Criterion 6 of 10 CFR 54.3(a)).

4.1.2.9 Fatigue Analysis of Reactor Coolant Pump Flywheel

SRP-LR Table 4.1-3 identifies "Fatigue Analysis of Reactor Coolant Pump Flywheel" as an analysis that may be generically applicable to an applicant's CLB. LRA Table 4.1-2 states that the fatigue analysis of reactor coolant pump flywheel does not meet the definition of a TLAA because the recirculation system pumps are not designed with flywheels.

The staff noted the applicant's basis, that the applicant's recirculation system pumps are not designed with flywheels, and thus there is no fatigue analysis of reactor coolant pump flywheel in the plant's CLB. The staff reviewed the applicant's UFSAR and confirmed that the plant is a BWR plant, and the inclusion of the generic TLAA for reactor coolant pump flywheels in SRP-LR Table 4.1-3 is only applicable to RCP pumps in PWR plants.

Based on this review, the staff concludes that the generic RCP flywheel analysis listed in SRP-LR Table 4.1-3 is not a TLAA for the applicant. This analysis is not a TLAA in accordance with 10 CFR 54.21(c)(1) because the staff confirmed that the plant recirculation system pumps are not designed with flywheels and a fatigue analysis of reactor coolant pump flywheel is not contained or incorporated by reference in the applicant's CLB (Criterion 6 of 10 CFR 54.3(a)).

4.1.2.10 Fatigue Analysis of Polar Crane

SRP-LR Table 4.1-3 identifies "Fatigue Analysis of Polar Crane" as an analysis that may be generically applicable to an applicant's CLB. LRA Table 4.1-2 states that the analysis of the polar crane does not meet the definition of a TLAA because it does not involve any time-limited assumptions defined by the current 40-year operating term.

The staff believes the analysis of the polar crane does meet the definition of a TLAA. The staff reviewed the applicant's Material Handling System Inspection Program, its program basis documents during the aging management programs (AMP) audit, and UFSAR Section 9.1.4.2.2 and noted that the reactor building crane is defined as a Class A1 nuclear fuel handling crane by the Crane Manufacturers Association of America Specification 70 (CMAA No. 70) for electric overhead traveling cranes. The staff notes that the polar crane has a design limit of cycles in the CMAA specification, and an "assumed design assessment" of the number of lifts compared to the CMAA specification.

This issue was open item (OI) 4.7.5-1 in the SER with open items. By letter dated October 5, 2011, and supplemented by letter dated November 16, 2011, the applicant responded to OI 4.7.5-1. The staff's evaluation and closure of OI 4.7.5-1 is documented in SER Section 4.7.5.2.

4.1.2.11 Metal Corrosion Allowances

SRP-LR Table 4.1-3 identifies "Metal Corrosion Allowance" as an analysis that may be generically applicable to an applicant's CLB. LRA Table 4.1-2 stated that the CLB does not include any metal corrosion allowance that meet the definition of a TLAA, and no explicit 40-year basis is applicable.

The staff conducted a search of the applicant's UFSAR, TSs, and AMPs for managing loss of material due to general corrosion in systems and components exposed to reactor coolant or treated or raw water. The staff also considered additional documents, such as NRC generic communications and ASME code requirements, which could incorporate a requirement in the CLB for a corrosion allowance TLAA.

In its review, the staff noted that the corrosion allowance in steel components in the reactor recirculation (RRC) system, the service water system, the residual heat removal (RHR) system, the condensate storage tanks, and the diesel generator fuel oil tanks, range from 0.062–0.120 inches. The staff also noted that the corrosion allowances that were included in the initial design of these components did not include any time-dependent analyses of a postulated aging effect.

Based on this review, the staff concludes that the generic metal corrosion allowance analysis listed in SRP-LR Table 4.1-3 is not a TLAA for the applicant. This analysis is not a TLAA in accordance with 10 CFR 54.21 because the inclusion of a corrosion allowance in the design of specific components did not involve any time-dependent assessment of a postulated aging effect (Criterion 2 and 3 of 10 CFR 54.3(a)).

4.1.2.12 Inservice Local Metal Containment Corrosion Analyses

SRP-LR Table 4.1-2 identifies "Inservice Local Metal Containment Corrosion Analyses" as an analysis that may be generically applicable to an applicant's CLB. LRA Table 4.1-2 stated that the CLB does not include inservice local metal containment corrosion analyses that meet the definition of a TLAA, and no explicit 40-year basis is applicable.

The staff also noted that UFSAR Section 3.8 indicates that the applicant's containment is a Mark II containment, which is a steel containment structure. The staff also noted that the applicant's CLB does not rely on an analysis to manage corrosion in the steel Mark II containment structure. Instead, the applicant relies on its Inservice Inspection (ISI) Program—IWE, for managing corrosion in the steel Mark II containment structure. The staff's evaluation of the applicant's ISI—IWE Program is documented in SER Section 3.0.3.1.20.

Based on this review, the staff concludes that the inservice local metal containment corrosion analyses listed in SRP-LR Table 4.1-2 is not a TLAA for the applicant. This analysis is not a TLAA in accordance with 10 CFR 54.21 because the analysis does not consider the effects of aging and therefore does not meet Criterion 6 of 10 CFR 54.3(a). Instead, the staff has verified that the applicant manages loss of material due to corrosion and in the steel Mark II containment structure using the applicant's ISI—IWE Program.

4.1.2.13 TLAAs related to BWRVIP Report Applicant Action Items (AAIs)

Several BWRVIP documents credited for license renewal have NRC safety evaluation reports (SERs) that have associated license renewal applicant action items (AAIs)). A plant-specific response for each of these AAIs is provided in Appendix C of the LRA. The staff's evaluation of the responses to license renewal AAIs associated with TLAAs is documented below.

4.1.2.13.1 TLAA AAI that is Generically Applicable to Multiple BWRVIP Report

The staff noted that AAI No. 2 is generically applicable to the following BWRVIP reports:

- BWRVIP-18-A BWR Core Spray Internals
- BWRVIP-25 BWR Core Plate
- BWRVIP-26-A BWR Top Guide
- BWRVIP-27-A BWR Standby Liquid Control System/Core Plate DP Inspection and Flaw Evaluation Guidelines[1]
- BWRVIP-42-A Low-Pressure Coolant Injection (LPCI) Coupling
- BWRVIP-47-A BWR Lower Plenum
- BWRVIP-38 BWR Shroud Support
- BWRVIP-41 BWR Jet Pump Assembly
- BWRVIP-48-A Vessel ID Attachment Weld
- BWRVIP-49-A Instrument Penetration
- BWRVIP-74-A BWR Reactor Pressure Vessel (RPV)

AAI No. 2 states the following for BWRVIP-74 (where the BWRVIP report number and affected components are specific to AAI No. 2 for each report):

> 10 CFR 54.21 (d) requires that an UFSAR supplement for the facility contain a summary description of the programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation. Those LR applicants referencing the BWRVIP-74 report for the reactor pressure vessel (RPV) components shall ensure that the programs and activities specified as necessary in the BWRVIP-74 report are summarily described in the UFSAR supplement.

In LRA Table C-11, the applicant stated that the UFSAR supplement, contained in LRA Appendix A, includes a summary description of the programs and activities, as required by this AAI. The staff confirmed that the applicant included the applicable UFSAR supplements for each of the TLAAs that need to be identified, in accordance with 10 CFR 54.3(a), and is included in the LRA, in accordance with 10 CFR 54.21(c)(1)(d). These UFSAR supplements—in LRA Appendix A Section A.1.3 and its subsections—include the following:

- UFSAR Supplement A.1.3.1 and the following subsections for each of the five TLAAs on neutron irradiation embrittlement of RV beltline components:

[1] NOTE: The applicant does not rely on the guidance of BWRVIP-27-A because, if necessary under a design basis event, the applicant's design injects the borated standby liquid control coolant into the RCPB using a nozzle to the high pressure core spray line and not through a SLC nozzle that is welded to the reactor vessel shell or lower head.

- A.1.3.1.2 on the upper-shelf energy (USE) assessment
- A.1.3.1.3 on the adjusted reference temperature (ART) assessment
- A.1.3.1.4 on the pressure-temperature (P-T) limits assessment
- A.1.3.1.5 on the RV circumferential weld probability of failure analysis
- A.1.3.1.6 on the RV axial weld probability of failure analysis

- UFSAR Supplement A.1.3.2 and its subsections for each of the five TLAAs on cumulative usage factor (CUF) analyses for Class 1 RCPB components, including the following:

  - A.1.3.2.1 on the CUF analyses for the RV components

  - A.1.3.2.2 on the CUF analyses for RV internal components, which are defined as ASME Code, Section III, Subsection NG core support structure components, and for internal jet pump assembly components

- UFSAR Supplement A.1.3.4 on the applicant's environmentally-assisted fatigue (EAF) analyses for specific Class 1 components in the RCPB, including selected RV components

Based on this review, the staff finds that the applicant resolved the generic applicability of AAI No. 2 because it has included the appropriate UFSAR supplement sections for the TLAAs that are associated with the RV and its internal components in LRA Appendix A.

4.1.2.13.2 Specific AAIs Associated with BWRVIP-74-A, BWR RPV

In LRA Appendix C, Table C-11, the applicant stated that "[t]he BWRVIP requires the inspection and evaluation guidelines of this BWRVIP report to be implemented at Columbia. Site procedures require a technical justification to be documented for any deviation from the guidelines. Columbia has not identified any deviation from the BWRVIP-74-A guidelines. Therefore, Columbia is bounded by the BWRVIP-74-A report." In its review, the staff noted that two AAIs associated with TLAAs are identified in the SER for BWRVIP-74-A. These AAIs and the applicant's response in LRA Table C-11 are as follows.

AAI No. 8 on BWRVIP-74-A states the following:

> LR applicants should verify that the number of cycles assumed in the original fatigue design is conservative to assure that the estimated fatigue usage for 60 years of plant operation is not underestimated. The use of alternative actions for cases where the estimated fatigue is projected to exceed 1.0 will require case-by-case staff review and approval. Further, a LR applicant must address environmental fatigue for the components listed in the BWRVIP-74 report for the LR period.

In LRA Table C-11, the applicant stated that metal fatigue (including discussion of cycles, projected cumulative usage factors, and environmental fatigue effects) is addressed in Section 4.3 of the LRA. The staff confirmed that the applicant included the applicable CUF-based TLAAs for the RV and RV internals components in LRA Sections 4.3.1 and 4.3.2, respectively. Based on this confirmation, the staff finds that the applicant's response to the first part of AAI No. 8 is acceptable and resolves the AAI item because the applicant has included the applicable CUF-based TLAAs for these components in LRA Sections 4.3.1 and 4.3.2. The staff's evaluations of the applicant's disposition of the TLAAs for the RV and its internal components are documented in SER Sections 4.3.1 and 4.3.2, respectively.

However, the staff noted that the applicant did not respond to second half of AAI No. 8, in which the staff asked the applicant to address environmental fatigue for the components listed in the BWRVIP-74 report for the license renewal period. The staff confirmed that the applicant included its EAF analyses for its plant-specific components in LRA Section 4.3.5 and provided its CUF_{en} values for its corresponding NUREG/CR-6260 locations in LRA Table 4.3-6. The staff's evaluation of the applicant's EAF analyses is documented in SER Section 4.3.5 and considers the inclusion of information in Section 4.3.5 of the LRA to be sufficient for resolving the second half of AAI No. 8 on BWRVIP-74A.

AAI No. 14 on BWRVIP-74-A states the following: "Components that have indications that have been previously analytically evaluated in accordance with Subsection IWB-3600 of Section XI to the ASME Code until the end of the 40-year service period shall be reevaluated for the 60 year service period corresponding to the LR term."

The applicant responded to the AAI item and clarified that it has two indications in the RV welds, BM and BG, which have been previously evaluated in accordance with Subsection IWB-3600 of Section XI to the ASME Code until the end of the 40-year service period. In LRA Table C-11, the applicant stated that these two RV shell indications were evaluated, and cracking of these indications will be managed by the ISI Program during the period of extended operation. The applicant further added that details of the evaluation are in LRA Section 4.7.1. The staff confirmed that the applicant included these ASME Section XI flaw evaluations as TLAAs in LRA Section 4.7.1. Based on this review, the staff finds that the applicant resolved AAI No. 14 because it included the appropriate flaw evaluation TLAA for the two RV shell indications in LRA Section 4.7.1. The staff's evaluation of the applicant's disposition for this TLAA is documented in SER Section 4.7.1. The staff's evaluation of the applicant's AMR items for these welds is documented in SER Section 3.1.2.3.1.

The staff noted that the applicant performed TLAAs for the period of extended operation related to RV neutron embrittlement. The staff confirmed that the applicant included these TLAAs to address the BWRVIP-74-A license renewal AAIs related to plant-specific TLAAs of USE, ART, P-T limits, RV circumferential weld examination relief, and RV axial weld failure probability. The staff also confirmed that the applicant discussed these TLAAs, along with its neutron fluence methodology, and provided its dispositions for these items in LRA Sections 4.2.1 through 4.2.6, respectively. The staff also confirmed that the applicant provided acceptable summary responses to the BWRVIP-74-A AAIs in LRA Appendix C. The staff's evaluation of the applicant's TLAAs, and the corresponding dispositions, is documented in the applicable subsections of SER Section 4.2.

4.1.2.13.3 Other Specific Fatigue TLAA AAIs Associated With BWRVIP Reports

In LRA Appendix C, the applicant stated that BWRVIP requires the inspection and evaluation guidelines of the following six BWRVIP report to be implemented, and it has not identified any deviation from the following BWRVIP guidelines:

- BWRVIP-18-A, BWR Core Spray Internals
- BWRVIP-25, BWR Core Plate
- BWRVIP-26-A, BWR Top Guide
- BWRVIP-27-A, BWR Standby Liquid Control (SLC) System and Core Plate ΔP
- BWRVIP-42-A, LPCI Coupling
- BWRVIP-47-A, BWR Lower Plenum

The following AAI items are specific to these BWRVIP reports.

BWRVIP-18-A, AAI No. 4 states the following: "Applicants referencing the BWRVIP-18 report for license renewal should identify and evaluate any potential TLAA issues which may impact the structural integrity of the subject RPV internal components."

LRA Table C-1 states that the only TLAA issues identified for the RPV internal core spray components were the CUFs in LRA Table 4.3-4 for the core spray sparger and core spray piping. The applicant further stated that disposition of these TLAAs is discussed in LRA Section 4.3.2.1. The staff confirmed that the applicant included its TLAA on the CUF analysis for the core spray sparger in LRA Section 4.3.2.1 and LRA Table 4.3-4. Based on this review, the staff finds that the applicant resolved AAI No. 4 on BWRVIP-18-A because it included the appropriate CUF TLAA for the core spray sparger in LRA Section 4.3.2.1. The staff's evaluation of the applicant's disposition for this TLAA is documented in SER Section 4.3.2.1.

BWRVIP-25, AAI No. 4 states the following: "Due to the susceptibility of the rim hold-down bolts to stress relaxation, applicants referencing the BWRVIP-25 report for license renewal should identify and evaluate the projected stress relaxation as a potential TLAA issue."

LRA Table C-2 states that stress relaxation of the core plate rim hold-down bolts is not a TLAA. The applicant added that, during original fabrication of the reactor internals, wedges were installed to prevent lateral motion of the core plate, and core plate rim hold-down bolts are not required for this function. The staff noted that BWRVIP-25 states that core plate wedges were used as the design feature for protecting some BWR plant core plates against lateral movement during normal operations, transient operations, and postulated design basis events. The staff confirmed that BWRVIP-25 clarifies that stress relaxation is not an aging effect of concern for those plants that were designed with core plate wedges.

In a conference call on May 6, 2011, the applicant stated that it had discovered that there were no core plate wedges located around the periphery of the core plate within the shroud. If core plate wedges are not installed, core plate rim hold-down bolts perform the function of preventing lateral motion of the core plate. However, core plate rim hold-down bolts are susceptible to stress relaxation and as described in the staff's license renewal SER for BWRVIP-25, dated December 7, 2000, "due to susceptibility of the rim hold-down bolts to stress relaxation, applicants referencing the BWRVIP-25 report for license renewal should identify and evaluate the projected stress relaxation as a potential TLAA issue."

By letter dated June 2, 2011, the staff issued RAI B.2.10-2 due to its concerns that there were no core plate wedges located around the periphery of the core plate. The staff's evaluation and resolution of RAI B.2.10-2 is documented in SER Section 3.0.3.1.6. Based on its review and resolution of RAI B.2.10-2, the staff finds that the applicant resolved AAI No. 4 on BWRVIP-25.

BWRVIP-26-A, AAI No. 4 states the following: "Due to [irradiation-assisted stress corrosion cracking] IASCC susceptibility of the subject safety-related components, applicants referencing the BWRVIP-26 report for license renewal should identify and evaluate the projected accumulated neutron fluence as a potential TLAA issue."

LRA Table C-2 states that accumulated neutron fluence for the top guide is not a TLAA for the applicant because the top guide has exceeded the threshold fluence levels for IASCC identified in BWRVIP-26-A. The applicant also stated that the aging effect is managed per the inspection recommendations in BWRVIP-183, which includes the inspections recommended by NUREG-1801 for the period of extended operation. The staff confirmed that, in LRA

Table 3.1.2-1, the applicant identified that cracking is an applicable aging effect requiring management for the top guide and its subcomponents, including the top guide grid-to-grid and grid-to-beam junctures, and credits its BWR Vessel and Internals Program for aging management. The staff finds that the applicant does not need to treat the fluence level for the top guide as a TLAA because the applicant postulates cracking as an applicable aging effect for the top guide components and credits its BWR Vessel and Internals Program and its BWRVIP-183 inspections for aging management. The staff noted that this includes management of IASCC, which may be induced when the neutron fluence exceeds the threshold defined in BWRVIP-26-A. The staff's evaluation of the BWR Vessel and Internals Program is documented in SER Section 3.0.3.1.6. Based on this review, the staff finds that the applicant resolved AAI No. 4 of BWRVIP-26-A.

BWRVIP-27-A, AAI No. 4 states the following: "Due to the susceptibility of the subject components to fatigue, applicants referencing the BWRVIP-27 report for license renewal should identify and evaluate the projected fatigue CUFs as a potential TLAA issue."

LRA Table C-4 states that BWRVIP-27-A is not applicable because the applicant does not inject SLC through the SLC and core differential pressure (core ΔP) nozzle. The staff noted that the applicant's basis for claiming that this AAI is not applicable was based on a determination that Columbia is not consistent with the background description in Section 1.1 of BWRVIP-27-A, which manages the effects of aging on the functionality of the ΔP and SLC vessel penetration and nozzle and safe-end extensions during the period of extended operation. Instead, the applicant injects SLC through the high-pressure core spray (HPCS) line. The applicant added that the only TLAA identified for the SLC and core ΔP nozzle is the cumulative usage factor (CUF) for the core ΔP nozzle stub tube, and this TLAA is addressed in LRA Section 4.3.1. The staff reviewed the BWRVIP report and confirmed that the design of the SLC system, if actuated, injects the SLC borated coolant into the RCS through the core spray line and not through a SLC and core ΔP nozzle.

Based on this review, the staff concludes that the applicant's response AAI No. 4 of BWRVIP-27-A is acceptable. The applicant does not need to include a CUF assessment of a SLC and core ΔP penetration nozzle to the RV because the plant design does not include a SLC nozzle that injects the directly into the RV and instead injects through a nozzle that is joined to the plant's core spray line. Based on this review, the staff finds that the applicant resolved AAI No. 4 of BWRVIP-27-A.

BWRVIP-42-A, AAI No. 4 states the following: "Applicants referencing the BWRVIP-42 report for license renewal should identify and evaluate any potential TLAA issues which may impact the structural integrity of the subject RPV internal components."

LRA Table C-4 states that the only TLAA identified for the LPCI coupling is the associated CUF analysis, which is addressed in LRA Section 4.3.2. The staff confirmed that BWRVIP-42-A does not specifically identify the types of TLAAs that may be applicable to the LPCI couplings. The staff also confirmed that the applicant includes the applicable CUF analysis for the LPCI coupling in LRA Section 4.3.2.1 and in LRA Table 4.3-4. The staff's evaluation of the applicant's disposition for the LPCI coupling CUF analysis is documented in SER Section 4.3.2.1.2. Based on this review, the staff finds that the applicant resolved AAI No. 4 of BWRVIP-42-A.

BWRVIP-47-A, AAI No. 4 states the following: "Due to fatigue of the subject safety-related components, applicants referencing the BWRVIP-47 report for LR should identify and evaluate the projected CUF as a potential TLAA issue."

LRA Table C-4 states that the TLAAs identified for the lower plenum are the CUFs for the control rod drive (CRD) housings, CRD stub tubes, and incore housing (instrument) penetrations, and it is addressed in LRA Section 4.3.1. The staff confirmed that the applicant includes the applicable CUF analysis for the CRD stub tubes, CRD housings, and incore housing (instrument) penetrations in LRA Section 4.3.1 and in LRA Table 4.3-3. The staff's evaluation of the applicant's disposition for these CUF analyses is documented in SER Section 4.3.1.2. Based on this review, the staff finds that the applicant resolved AAI No. 4 of BWRVIP-47-A.

4.1.2.13.4 TLAA-Related AAI Response Conclusion

Based on this review, the staff concludes that the applicant either responded to or resolved all of the staff's requests raised in applicable TLAA-related AAIs of the BWRVIP reports referenced in the SER. The applicant's responses to the TLAA-related AAIs are resolved.

4.1.3 Staff Evaluation of the Applicant's Identification of Those Exemptions in the CLB That Are Based on TLAAs

As required by 10 CFR 54.21(c)(2), an applicant must list all plant-specific exemptions, granted pursuant to 10 CFR 50.12, that are in effect and based on TLAAs, and provide an evaluation that justifies the continuation of these exemptions through the period of extended operation. LRA Section 4.1.3 states that the applicant's CLB documentation was reviewed for exemptions and the applicant stated that the CLB does not include any exemptions that are based on a TLAA.

The staff reviewed the following types of documents to verify if there were any exemptions in the CLB that were granted in accordance with the exemption criteria of 10 CFR 50.12 and that were based on a TLAA:

- Columbia Generating Station Operating License No. NPF-21
- applicable exemptions on neutron irradiation embrittlement analyses requested pursuant to the requirements in 10 CFR 50.60(b) and granted under the requirements in 10 CFR 50.12

The staff noted that the applicant's Operating License, No. NPF-21, issued December 20, 1993, states, in part, the following:

> Exemptions from certain requirements of Appendices G, H, and J to 10 CFR Part 50 are described in the Safety Evaluation Report. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore these exemptions are hereby granted pursuant to 10 CFR 50.12. With the granting of this exemption the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

The staff was not able to determine if any exemptions to the requirements of Appendices G, H, and J to 10 CFR Part 50 exist or whether these exemptions are still in effect and are based on a TLAA that will be needed for the period of extended operation. By letter dated August 26, 2010, the staff issued RAI 4.1-1, asking the applicant to clarify the exemptions to the requirements of Appendices G, H, and J to 10 CFR Part 50 and to clarify whether these exemptions are still in effect and are based on a TLAA. If it is in effect and based on a TLAA, the applicant was asked to justify continuation of the exemptions for the period of extended operation.

In its response, by letter dated November 11, 2010, the applicant stated that the exemptions to Appendices G, H, and J of 10 CFR Part 50, as discussed in the original Columbia (WNP-2) SER, are still in effect but are not based on a TLAA. The applicant summarized five exemptions that were requested under 10 CFR 50.60(b) with respect to the requirements for P-T limits or USE assessments in 10 CFR Part 50, Appendix G. The applicant stated that four of these exemptions relate to the determination of the initial USE or reference temperature for nil-ductility transition properties for the RV. The applicant added that these exemptions and their justifications are based on technical criteria that have no time-related parameter relationships and, therefore, are not based on TLAA and need not be reported for license renewal.

The staff noted that the exemptions to the requirements of 10 CFR Part 50, Appendix G, apply to the applicant's method for determining the initial USE property value in the applicant's USE analysis or the initial adjusted nil-ductility reference temperature value (i.e., initial RT_{NDT} value) for a given RV beltline component, as related to the applicant's determination of its P-T limit curves. The staff noted that the methods for deriving these parameters in the exemptions do not have a time-limited analysis assumption because it is based on the use of alternative testing methods or generic industry data for deriving the initial USE property or initial RT_{NDT} reference temperature for a given RV beltline material type. The staff noted that the applicant did not derive these types of material property parameters based on a TLAA.

Based on its review, the staff finds this portion of the response to RAI 4.1-1 acceptable. These four exemptions to the requirements of 10 CFR Part 50, Appendix G, do not need to be identified as exemptions based on a TLAA because it does not meet Criterion 3 of 10 CFR 54.3(a) or involve time-limited assumptions defined by the current operating term. Therefore, it does not meet the exemption identification criterion in 10 CFR 54.21(c)(2).

In its response by letter dated November 11, 2010, the applicant also stated that the fifth exemption pertains to the fracture toughness testing of material for the main steam isolation valves (MSIVs), as required by Paragraph IV.A.3 of 10 CFR Part 50, Appendix G. The applicant stated that this exemption does not involve any time-dependent aspects and, therefore, does not need to be identified as an exemption under 10 CFR 54.21(c)(2). The staff noted that, according to 10 CFR Part 50, Appendix G, Paragraph IV.A.3, this exemption involved either an exemption of the calibration of testing equipment, the qualification requirements for test personnel, or record retention requirements in the applicant's edition of record of the ASME Code, Section XI, Boiler and Pressure Vessel (B&PV) Code, Division 1. The staff confirmed that these requirements are not based on any time-dependent analysis criteria.

Based on its review, the staff finds this portion of the response to RAI 4.1-1 acceptable. This exemption to 10 CFR Part 50, Appendix G, does not need to be identified as an exemption based on a TLAA because it does not meet Criterion 3 of 10 CFR 54.3(a) or involve time-limited assumptions defined by the current operating term. Therefore, it does not meet the exemption identification criterion in 10 CFR 54.21(c)(2).

In its response by letter dated November 11, 2010, the applicant also clarified an exemption that was previously granted from meeting the RV Surveillance Program requirements of 10 CFR Part 50, Appendix H. The applicant clarified that the exemption on the RV Surveillance Program requirements of 10 CFR Part 50, Appendix H, relates to compliance with ASTM E 185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessel." The staff notes that the exemption was granted in NUREG-0892, Safety Evaluation

Report Related to the Operation of [Washington Public Power Supply System] WPPSS Nuclear Project No. 2," dated March 1982, because the applicant provided an alternative method so that the Charpy-V notch impact specimen orientation and limiting reactor vessel material are properly evaluated. The staff also notes that the NRC-granted exemption is not based on a TLAA because the exemption is not based on any time-dependent analysis or its assumptions.

Based on its review, the staff finds this portion of the response to RAI 4.1-1 acceptable. The exemption to the requirements of 10 CFR Part 50, Appendix H, does not need to be identified as an exemption based on a TLAA because it does not meet Criterion 3 of 10 CFR 54.3(a) or involve time-limited assumptions defined by the current operating term. Therefore, it does not meet the exemption identification criterion in 10 CFR 54.21(c)(2).

In the response by letter dated November 11, 2010, the applicant also clarified an approved exemption from meeting the Containment Leak Rate Testing Program requirements of 10 CFR Part 50, Appendix J. The applicant clarified that the exemption involves the method for performing leak rate testing of the main steam line isolation valves. The staff noted that this exemption only pertains to the applicant's testing method and the details for performing the required leak-rate testing of the MSIVs, which is not based on any time-dependent analysis or assumptions.

Based on its review, the staff finds this portion of the response to RAI 4.1-1 acceptable. This exemption to 10 CFR Part 50, Appendix J, on the leak-rate testing for the MSIVs does not need to be identified as an exemption based on a TLAA because it does not meet Criterion 3 of 10 CFR 54.3(a) or involve time-limited assumptions defined by the current operating term. Therefore, it does not meet the exemption identification criterion in 10 CFR 54.21(c)(2).

Based on its review, the staff finds the applicant's response to RAI 4.1-1 acceptable in its entirety. The exemptions identified above are not based on a TLAA and, therefore, do not meet the exemption identification criterion in 10 CFR 54.21(c)(2), as described above.

4.1.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided a complete and accurate 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 are 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) for light-water nuclear power reactors. Areas of review to ensure that the RV beltline materials have adequate fracture toughness to prevent brittle failure during normal and off-normal operating conditions are as follows:

- RV neutron fluence
- RV materials Charpy upper-shelf energy reduction due to neutron embrittlement
- ART for RV materials due to neutron embrittlement
- operating P-T limits for heatup and cooldown operations, as well as hydrostatic and leak-testing conditions

- RV circumferential weld examination relief
- RV axial weld failure probability

Fracture toughness requirements for ferritic pressure-retaining components that make up the RCPB of light water nuclear reactors are specified in 10 CFR Part 50, Appendix G. This rule states that RV beltline material properties, including the reference nil-ductility temperature (RT_{NDT}) values and USE values, must account for the effects of neutron radiation.

The adjusted RT_{NDT} (ART) value — defined as the sum of the initial RT_{NDT} value for the material in the unirradiated condition, the mean value of the adjustment in reference nil-ductility temperature caused by irradiation (ΔRT_{NDT}), and a margin term (M) — is one of the parameters used to account for the effects of neutron radiation, in accordance with 10 CFR Part 50, Appendix G requirements. The ART value forms the basis for determining the allowable pressure loadings on the beltline region of the RV, as a function of RCS temperature.

ΔRT_{NDT} is the shift in the reference nil-ductility temperature produced by irradiation and is an increasing function of the material's copper and nickel content and the neutron fluence to which the material is exposed as those values increase. ΔRT_{NDT} may be calculated as the product of a chemistry factor (CF) and a fluence factor (FF), based on the NRC staff guidance for radiation embrittlement calculations in RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

The CF is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Revision 2, or from surveillance data.

The FF 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 is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence, and the calculation methods. RG 1.99, Revision 2, describes the methodology to be used in calculating the M term.

The mean RT_{NDT} value, which is used for the analyses of the RV circumferential weld examination relief and the RV axial weld failure probability, is defined as the sum of the initial RT_{NDT} and the ΔRT_{NDT}.

Both the mean RT_{NDT} and ART calculations meet the criteria of 10 CFR 54.3(a). Therefore, the mean RT_{NDT} and ART are TLAAs. The ART values for the RV beltline materials are used for the P-T limits analysis. The mean RT_{NDT} values are used in the analyses of the RV circumferential weld examination relief and the RV axial weld failure probability. The TLAAs of the ART and mean RT_{NDT} for RV beltline materials are based on the use of projected neutron fluence inputs at specific locations in the RV wall. In accordance with 10 CFR Part 50, Appendix G requirements, the ART analysis is based on a flaw with a depth equal to one-quarter of the vessel wall thickness (¼ T), with the neutron fluence at the ¼ T depth location in the RV wall. In contrast, the mean RT_{NDT} values used for the analyses of the RV circumferential weld examination relief and the RV axial weld failure probability are calculated using neutron fluence values at the clad-to-base metal interface of the RV wall.

Appendix G of 10 CFR Part 50 provides the staff's criteria for maintaining acceptable levels of USE for the RV beltline materials of operating reactors throughout the licensed operating periods of the facilities. The Rule requires RV beltline materials to have a minimum USE value of 75 ft-lb in the unirradiated condition and to maintain a minimum USE value above 50 ft-lb throughout the life of the facility, unless it can be demonstrated through analysis that lower values of USE would provide acceptable margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. The Rule also mandates that the methods used to calculate USE values must account for the effects of neutron irradiation on the USE values for the materials, including any relevant RV surveillance capsule data that are reported through implementation of a plant's 10 CFR Part 50, Appendix H, RV Material Surveillance Program.

RG 1.99, Revision 2, describes two methods for determining USE values for RV beltline materials, depending on whether or not a given RV beltline material is represented in the plant's RV Material Surveillance Program in accordance with 10 CFR Part 50, Appendix H. If surveillance data is not available for a particular material, the USE value is determined in accordance with Position 1.2 in RG 1.99, Revision 2. If surveillance data is available, the USE should be determined in accordance with Position 2.2 in RG 1.99, Revision 2. These methods refer to Figure 2 in RG 1.99, Revision 2, which describes how the percentage drop in USE is dependent upon the amount of copper in the material and the neutron fluence. Since the analyses performed in accordance with 10 CFR Part 50, Appendix G, are based on a flaw with a depth of ¼ T, the neutron fluence used in the USE analysis is the neutron fluence at the ¼ T depth location in the RV wall.

The applicant described its evaluation of these TLAAs in LRA Section 4.2, "Reactor Vessel Neutron Embrittlement." The applicant described its evaluation of a TLAA for several flaws found in the RV shell in LRA Section 4.7.1, "Reactor Vessel Shell Indications," and the staff's evaluation of that LRA section is provided in SER Section 4.7.1.

4.2.1 Neutron Fluence Values

4.2.1.1 Summary of Technical Information in the Application

LRA Section 4.2.1 summarizes the reactor vessel neutron fluence determination, which the applicant performed to support the neutron embrittlement analyses. The applicant described neutron fluence projections for 54 effective full power years (EFPY) of operation, which is a bounding representation of 60 calendar years of operation, because reaching such exposure would require a plant capacity factor in excess of 95-percent from the date of the application until the end of the period of extended operation.

The applicant's current licensing basis projected fluence values, as provided in its UFSAR, represent 51.6 EFPY of operation, which are based on facility operation at the original licensed thermal power level of 3323 megawatts-thermal for the first ten cycles of operation, and on uprated operation at 3486 megawatts-thermal from fuel cycle 11 through the end of operation. The application states that the method used to perform the fluence calculations is described in GE report NEDC-32983P-A, Revision 2, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," which was previously approved by the NRC staff because it was found to be consistent with the guidance set forth in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."

The application describes that the CLB fluence projections for 51.6 EFPY were extrapolated to 54 EFPY, to conservatively cover the highest fluence that will be accrued during the period of extended operation.

The application describes that NUREG-1801 indicates that ferritic materials for RV beltline shells, welds, and other components are to be evaluated for neutron irradiation embrittlement if the projected high energy neutron fluence for these materials 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. Table 4.2-1 lists the 54 EFPY fluence values for the materials that meet this criterion. The application states that the only RV components, other than RV shell plates and welds, that would experience fluence levels greater than 1 x 10^{17} n/cm^2 (E > 1.0 MeV) at the end of the period of extended operation are the N12 instrumentation nozzles and the three N6 RHR/LPCI nozzles. The application states that the N12 instrumentation nozzles have a thickness less than 2.5 inches. According to the applicant, these nozzles require no fracture toughness evaluation due to this thickness criterion, per the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Appendix G, Paragraph G-2223 specification for fracture toughness evaluation requirements for nozzles. Accordingly, these nozzles are not listed in Table 4.2-1. The 54 EFPY fluence values for the N6 RHR/LPCI nozzles are listed in Table 4.2-1 along with the fluence values for the RV beltline plates and welds. According to the applicant, all RV beltline components listed in Table 4.2-1 are evaluated for neutron embrittlement, in accordance with 10 CFR Part 50, Appendix G, requirements.

The applicant concluded that neutron fluence is not a TLAA, but rather a time-limited assumption used in various neutron embrittlement TLAAs.

Subsequent to submittal of the LRA, the applicant identified a TLAA disposition for reactor vessel neutron fluence in its RAI response to RAI 4.2.1-2, by letter dated January 27, 2011. This response stated that the analyses have been projected to the end of the period of extended operation, consistent with 10 CFR 54.21(c)(1)(ii).

4.2.1.2 Staff Evaluation

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

RG 1.190 describes acceptable ways to calculate reactor vessel neutron fluence. RG 1.190 states that fluence calculations should adhere to NRC-approved methodology and provides acceptable qualification criteria.

Fluence calculations performed using NEDC-32983P-A utilize a relatively fine (r, θ, z) spatial mesh and are carried out using an S12 angular quadrature. Although cross sections are generally based on ENDF/B-V nuclear data, corrections have been made to include ENDF/B-VI-based cross sections for oxygen, hydrogen, and individual iron isotopes. These corrections address the differences between ENDF/B-V and ENDF/B-VI data identified in RG 1.190, and are discussed in the NRC safety evaluation report approving NEDC-32983P-A. Scattering cross sections are represented using a P3 Legendre expansion.

RG 1.190 specifies that acceptable fluence calculations should employ, at a minimum, S8 angular quadrature, cross sections based on the most recent nuclear data, and P3 Legendre expansion. As discussed above, the method described in NEDC-32983P-A addresses these recommendations acceptably, and thus the Columbia fluence calculations are acceptable with respect to use of an approved methodology.

The applicant stated that fluence values representing 51.6 EFPY of operation are provided in its UFSAR, and these values were extrapolated to 54 EFPY of operation in order to provide a bounding representation of 60 calendar years of facility operation. The flux used in the fluence calculation was a representation of the original licensed thermal power level through the end of fuel cycle 11, at which point an uprated flux value was used to predict the post-fuel cycle 11 fluence values. The fluence projection was extended from 51.6 EFPY to 54 EFPY by means of a linear extrapolation. The NRC staff finds the applicant's flux values through 51.6 EFPY acceptable because it is representative of actual and planned facility operation.

By letter dated July 15, 2010, the staff issued RAI 4.2.1-1 to confirm that the flux used to extrapolate from 51.6 EFPY to 54 EFPY was also based on uprated facility operation. The applicant confirmed, in response to RAI 4.2.1-1, by LRA supplement dated September 13, 2010, that the flux used for the 54 EFPY extrapolation was that assumed for post-fuel cycle 11 uprated operation. The NRC staff finds the flux values used to extrapolate fluence from 51.6 EFPY to 54 EFPY acceptable because it is representative of planned facility operation.

The applicant stated that fluence projections for 54 EFPY are bounding for 60 calendar years of operation because the facility would have to operate at a capacity factor exceeding 95-percent to reach 54 EFPY by the end of the renewed license period. The NRC staff finds the 54 EFPY fluence projection acceptable because the 95-percent capacity factor required to reach it by the end of the license period is sufficiently conservative. The staff's concern described in RAI 4.2.1-1 is resolved.

By letter dated October 20, 2010, the staff issued RAI 4.2.1-2 requesting the applicant to reconsider its determination that neutron fluence calculations are not a TLAA, since this position appears inconsistent with the six criteria used to define a TLAA in 10 CFR 54.3. The applicant revised its application by letter dated January 27, 2011, providing a disposition that, for reactor vessel neutron fluence, the analyses are a TLAA, and as such,have been projected to the end of the period of extended operation. This disposition is consistent with 10 CFR 54.21(c)(1)(ii). The staff finds the response acceptable because the reactor vessel neutron fluences have been projected to the end of the period of extended operation, using a methodology acceptable to the staff. The staff's concern described in RAI 4.2.1-2 is resolved.

The staff reviewed the information in LRA Section 4.2.1 for determining RV beltline components requiring neutron embrittlement evaluation in accordance with 10 CFR Part 50, Appendix G. 10 CFR Part 50, Appendix G defines the beltline as the region of the RV that directly 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 regard to radiation damage. The GALL Report (NUREG-1801) states that neutron embrittlement is a TLAA for all ferritic materials (including RV beltline shells, welds, and other components) which are exposed to high energy neutron fluence greater than 1×10^{17} n/cm^2 ($E > 1.0$ MeV) at the end of the period of extended operation. Based on this statement in the GALL Report, the applicant applied the 1×10^{17} n/cm^2 ($E > 1.0$ MeV) fluence threshold for identifying the beltline components subject to neutron embrittlement evaluation, as described in LRA Section 4.2-1.

According to the applicant, the N12 instrumentation nozzles will be exposed to a projected neutron fluence greater than 1×10^{17} n/cm^2 ($E > 1.0$ MeV) at the end of the period of extended operation. However, the N12 instrumentation nozzles and the associated N12 nozzle-to-vessel welds were not included in the list of RV beltline components in LRA Table 4.2-1 subject to neutron embrittlement analysis. Furthermore, although the three N6 RHR/LPCI nozzles were included in the list of RV beltline components in LRA Table 4.2-1 subject to neutron embrittlement analysis, the N6 RHR/LPCI nozzle-to-RV welds were not included, despite the fact that these nozzle-to-RV welds would experience neutron fluence levels that are similar to the fluence levels experienced by the N6 RHR/LPCI nozzles. LRA Section 4.2.1 stated that the N12 RV instrumentation nozzles require no fracture toughness evaluation, and hence no consideration of neutron embrittlement, in accordance with ASME Code, Section XI, Appendix G, Subparagraph G-2223(c), because it has a thickness of less than 2.5 inches. Therefore, the application did not include these nozzles in the analyses for the neutron fluence in LRA Section 4.2.1, the USE in LRA Section 4.2.2, and the ART in LRA Section 4.2.3, despite the fact that these are beltline nozzles that would be exposed to a projected neutron fluence greater than 1×10^{17} n/cm^2 (E > 1.0 MeV) at the end of the period of extended operation. Subparagraph G-2223(c) of the ASME Code, Section XI, Appendix G states that, "fracture toughness analysis to demonstrate protection against nonductile failure is not required for portions of nozzles and appurtenances having a thickness of 2.5 in. (63 mm) or less, provided the lowest service temperature is not lower than RT_{NDT} plus 60°F (33°C)."

Since the RV N12 instrumentation nozzles and the associated N12 nozzle-to-RV welds will be exposed to a projected neutron fluence greater that 1×10^{17} n/cm^2 (E > 1.0 MeV) at the end of the period of extended operation, the effects of radiation on the material properties of these nozzles and associated nozzle-to-RV welds must be considered in determining whether these nozzles meet the lowest service temperature criterion. Therefore, an ART value (i.e., RT_{NDT} adjusted to account for the effects of radiation) must be determined for the N12 instrumentation nozzles and the associated N12 nozzle-to-RV welds to determine if the lowest service temperature criterion will be met through the end of the extended operating period. If not, the N12 instrumentation nozzles and the associated N12 nozzle-to-RV welds must be considered when the applicant develops pressure-temperature limits for Columbia in accordance with Title 10 of the Code of Federal Regulations Part 50, Appendix G (10 CFR Part 50, Appendix G) and the ASME Code, Section XI, Appendix G.

Appendix G of 10 CFR Part 50, Paragraph IV.A.1.a., states that, "reactor vessel beltline materials must have Charpy upper-shelf energy in the transverse direction for the base material of no less than 75 ft-lb (102 J) initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68 J)…" 10 CFR Part 50, Appendix G, Paragraph II.F, defines beltline materials to include those "that are predicted to experience sufficient radiation damage to be considered in the selection of the most limiting material with regard to radiation damage." Without additional evaluation of the effects of radiation on the USE of the N12 instrumentation nozzles and the associated N12 nozzle-to-RV welds, it cannot be determined whether these materials are, or are not, limiting with respect to USE for the Columbia RV. Furthermore, without evaluation of the 54 EFPY USE values for the N12 instrumentation nozzles and the associated N12 nozzle-to-RV welds, it cannot be determined whether these nozzles and associated nozzle-to-RV welds will remain in compliance with the USE requirements of 10 CFR Part 50, Appendix G, through the end of the period of extended operation.

The staff issued RAI 4.2.1-a, by letter dated August 26, 2010, requesting that the applicant supplement LRA Sections 4.2.1, 4.2.2, and 4.2.3 (including Tables 4.2-1, 4.2-2, 4.2-3, or 4.2-4, as applicable), and Table 4.2-5, to include data for the analyses of the neutron fluence, ART, and USE for the Columbia RV N12 instrumentation nozzles.

By letter dated November 23, 2010, the applicant stated in RAI Response 4.2.1-a that the exact fluence for the N12 instrumentation nozzles (nozzles N12A, N12B, N12C, and N12D) has not been calculated. However, the applicant stated that the N6 nozzles are closer to the active core region and the bottom of the 12-inch N6 nozzles is more than 12 inches below the centerline of the N12 nozzles; thus, the bottom of the N6 nozzle is more exposed to the active core than the bottom of the N12 nozzle. Therefore, the applicant determined that a bounding fluence for the N12 nozzles is the fluence for the N6 nozzle given in LRA Table 4.2-1, specifically 4.48 x 10^{17} n/cm^2 (E > 1.0 MeV) at the ¼ T location in the RV wall. The applicant amended LRA Section 4.2.1, Table 4.2-1 and Section A.1.3.1.1 to address the fluence value for the N12 instrumentation nozzles (provided in LRA Amendment 12).

The staff reviewed the applicant's response to RAI 4.2.1-a and determined that the use of the Nozzle N6 fluence as a bounding value for the Nozzle N12 fluence is acceptable. The staff's concern described in RAI 4.2.1-a is resolved.

In addition to providing neutron fluence values for the N12 instrumentation nozzles, the staff found, in reviewing the applicant's response to RAI 4.2.1-a, that the applicant must also provide fluence values for the N12 nozzle-to-RV welds and the N6 RHR/LPCI nozzle-to-RV welds.

By letter dated December 20, 2010, the staff issued RAI 4.2-1 requesting that the applicant provide for the N12 nozzles and associated nozzle-to-RV welds: (a) the type of material it is composed of; (b) additional material specifications; (c) ART values at 54 EFPY; and (d) equivalent margin analysis (EMA).

In its response to RAI 4.2-1(a) dated January 28, 2011, the applicant stated that the N12 nozzle forgings are ferritic and the N12 nozzle-to-vessel welds are austenitic.

In its response to RAI 4.2-1(b) and 4.2-1(c) the applicant provided an LRA supplement that included addition of the material specifications and the ART values for the N12 nozzles.

In its response to RAI 4.2-1(d) the applicant stated that the initial USE for the N12 forgings is unknown, therefore, the applicant provided an LRA supplement that projected percent drop in USE to demonstrate equivalent margin. The applicant also stated that while the percent projected drop does meet the acceptance criterion of GE report NEDO-32205, "10 CFR 50 Appendix G Equivalent Margin Analysis for Low Upper-Shelf Energy in BWR-2 through BWR-6 Vessels," discussions between the staff, the applicant, and the orginal equipment manufacturer have confirmed that the NEDO-32205 acceptance criteria cannot be applied to forgings without further study. Therefore, the applicant included a commitment (Commitment 70) to perfom the necessary equivalent margin analysis for the N12 nozzle forgings prior to the period of extended operations.

The staff reviewed the applicant's response and notes that the requirement of 10 CFR Part 50, Appendix G applies to, "ferritic material of pressure-retaining components of the reactor coolant pressure boundary." The 10 CFR Part 50, Appendix G requirement do not apply to austenitic-phase materials, such as nickel-based alloys and austenitic stainless steel. Therefore, since the associated N12 nozzle-to-RV welds are composed of an austenitic material, the welds do not need to be analyzed for neutron fluence, USE, or ART, consistent with 10 CFR Part 50, Appendix G requirement.

The staff concludes that the responses to RAIs 4.2-1(a) and 4.2-1(b) are acceptable because:

(a) The N12 nozzle-to-RV welds are Alloy 182, an austenitic material. As austenitic alloys are not subject to 10 CFR Part 50, Appendix G requirements, no neutron embrittlement analysis is required, and fluence projections are not necessary for the N12 nozzle-to-RV welds.

(b) The applicant provided an LRA supplement that included addition information of the material specifications for the N12 nozzles that the staff had requested, so that the staff may further evaluate the issue.

The staff's concerns described in RAIs 4.2-1(a) and 4.2-1(b) are resolved. The staff's further evaluation of the applicant's response to RAI 4.2-1(c) and 4.2-1(d) are documented in SER Section 4.2.3 and 4.2.2, respectfully.

To address the fluence values for the N6 RHR/LPCI nozzle-to-RV welds, the staff issued RAI 4.2-2 by letter dated December 20, 2010, requesting the applicant to provide the following information for the N6 RV beltline RHR/LPCI nozzle forging and the associated nozzle-to-RV weld materials: (a) additional material specification; (b) calculation of the 54 EFPY ART; and (c) EMA to demonstrate that the 54 EFPY USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

In its responses to RAI 4.2-2(a) and 4.2-2(b) dated January 28, 2011, the applicant provided an LRA supplement that included additional material specifications and calculations of the 54 EFPY ART. These responses indicate that the ART for the N6 nozzle forging and the associated nozzle-to-RV weld are less limiting than those for other portions of the beltline.

The staff finds the applicant's response to RAI 4.2-2(a) acceptable because the applicant provided an LRA supplement that included addition information of the material specifications for the N6 nozzles, which included the fluence value for the N6 nozzle-to-RV welds as 5×10^{17} n/cm^2 (E>1.0 MeV). The fluence value was determined based on the known fluence for the N6 nozzle forgings. The staff determined that the fluence value reported for the N6 nozzle-to-RV welds is acceptable, based on the location of the nozzle weld in relation to the beltline forging. The staff's concern described in RAI 4.2-2(a) is resolved.

The staff's evaluation of the applicant's response to RAI 4.2-2(b) and 4.2-2(c) are documented in SER Section 4.2.3 and 4.2.2, respectfully.

In summary, the staff finds acceptable the 54 EFPY fluence values in LRA Table 4.2-1 for use in evaluating neutron embrittlement for the RV material.

4.2.1.3 UFSAR Supplement

LRA Section A.1.3.1.1 provides the UFSAR supplement for the neutron fluence analysis TLAA evaluation. Based on its review of the UFSAR supplement, the staff concludes that the information in the UFSAR 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 has demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that for reactor vessel neutron fluence and beltline evaluation, the analyses have been projected to the end of the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d), and, therefore, is acceptable.

4.2.2 Upper-Shelf Energy

4.2.2.1 Summary of Technical Information in the Application

Appendix G of 10 CFR Part 50 requires that the USE for all RV beltline materials, accounting for the effects of neutron irradiation, remain above 50 ft-lb at all times during plant operation. If the USE cannot be shown to remain above this limit, then an EMA must be performed to show that the margins of safety against fracture are equivalent to those required by Appendix G of Section XI of the ASME Code.

The USE calculation of record for the current 40-year licensed operating period (33.1 EFPY) is provided in Appendix F of GE Report NEDO-33144, "Pressure-Temperature Curves for Energy Northwest, Columbia," April 2004. The initial (unirradiated) USE is not known for all the Columbia RV beltline plates and welds. For those plates and welds for which the initial USE is known, USE was projected using Regulatory Guide 1.99, Revision 2 methods. For the vessel plates and welds for which the initial USE is not known, EMAs were performed using the Boiling Water Reactor Owners Group EMA methodology from the NRC staff-approved BWRVIP-74-A report, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines for License Renewal (BWRVIP-74-A)," June 2003. Results from the testing and analysis of surveillance materials were used in the EMA analyses.

USE values projected to 54 EFPY and supporting input data are listed in LRA Table 4.2-2 for the RV beltline materials with initial USE values. All of the projected USE values listed in LRA Table 4.2-2 are projected to remain above 50 ft-lbs through the end of the period of extended operation (54 EFPY).

The projected EMAs are listed in LRA Tables 4.2-3 and 4.2-4. The projected EMAs in LRA Tables 4.2-3 and 4.2-4 used the projected 54 EFPY fluence values listed in LRA Table 4.2-1, and the curves provided in Figure 2 of RG 1.99, Rev. 2 for calculating the percentage decrease in USE. The predicted values for the percentage decrease in USE at 54 EFPY were compared to the 54 EFPY USE EMA limits specified BWRVIP-74-A.

For the RV beltline plates, the maximum decrease in USE was found to be 13.2 percent (LRA Table 4.2-3). This is less than the 23.5 percent decrease in the USE from the applicable RV beltline plate EMA. Therefore, the maximum predicted decrease in USE at 54 EFPY for the limiting RV beltline plate is bounded by the generic 54 EFPY EMA for beltline plates documented in BWRVIP-74-A. The projected USE for the RV beltline plates is therefore acceptable for the period of extended operation.

For the RV beltline welds, the maximum decrease in USE was found to be 21.6 percent (LRA Table 4.2-4). This is less than the 39 percent decrease in the USE from the applicable EMA for RV beltline welds. Therefore, the maximum predicted decreases in USE at 54 EFPY for the limiting RV beltline weld is bounded by the generic 54 EFPY EMA documented in BWRVIP-74-A. The projected USE for the RV beltline welds is therefore acceptable for the period of extended operation.

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

4.2.2.2 Staff Evaluation

The staff reviewed LRA Section 4.2.2 and the TLAAs for the USE to verify, pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis have been projected to the end of the period of extended operation.

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 USE will ensure margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

According to RG 1.99, Rev. 2, the predicted decrease in USE due to neutron embrittlement during plant operation is dependent upon the amount of copper in the material and the projected neutron fluence for the material. Position 1 of the RG specifies methods for calculating the predicted decrease in USE for materials that do not have sufficient credible surveillance data available. The applicant provided calculations of the projected USE values at 54 EFPY for those RV beltline materials for which the initial (unirradiated) USE values are known, in LRA Table 4.2-2. The staff determined that the applicant correctly used Position 1.2 of RG 1.99, Rev. 2, (Figure 2 from the RG) for calculating the projected percentage decrease in USE at 54 EFPY for these RV beltline materials. The staff confirmed that the initial USE values were consistent with those listed in the staff's Reactor Vessel Integrity Database (RVID) for those RV beltline materials for which the initial USE values are known. The staff found that the applicant correctly determined the projected 54 EFPY USE values for these RV beltline materials by applying the predicted percentage decrease in USE from RG 1.99, Rev. 2, to the initial USE values. All of the 54 EFPY USE values 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.

The applicant was unable to directly calculate the predicted 54 EFPY USE values for several of the Columbia RV beltline materials specified in LRA Table 4.2-1 because initial (unirradiated) values for USE were unavailable. Specifically, the applicant utilized Figure 2 from RG 1.99, Rev. 2, to calculate the projected percentage decreases in the USE values for the period of extended operation to demonstrate that the values for the percentage USE decrease for these materials were bounded by the EMA acceptance criteria from BWRVIP-74-A.

The results of the applicant's application of the EMA from BWRVIP-74-A, for those RV beltline materials for which initial USE data is unavailable, were provided in LRA Tables 4.2-3 (RV beltline plate heats C1337-1 and C1337-2) and 4.2-4 (RV beltline weld heat 624039/D205A27A). In reviewing these tables, the staff determined that the applicant needed to clarify how it utilized Position 2.2 of RG 1.99, Rev. 2, for applying the BWRVIP Integrated Surveillance Program (ISP) data in the determination of the percentage decrease in the USE values (EMA data) for these materials because these tables also provide EMA data for several ISP surveillance capsules.

By letter dated August 3, 2010 the staff issued RAI 4.2.2-1, requesting that the applicant state whether the EMA/USE data for the ISP surveillance materials listed in LRA Table 4.2-3 and 4.2-4 was used for adjusting the EMA data for the corresponding RV beltline materials, in accordance with BWRVIP-74-A, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74)," Appendix B, and Regulatory Position (RP) 2.2 from RG 1.99, Rev. 2.

The applicant provided its response to RAI 4.2.2-1 by letter dated September 27, 2010. The applicant stated that for the beltline plates listed in LRA Table 4.2-3 (RV beltline plate heats C1337-1 and C1337-2), surveillance data was not available. Therefore, the EMA data for these plates is based on the direct calculation of the percentage decrease in the USE (13.2 percent at 54 EFPY based on RP 1.2 from RG 1.99, Rev. 2) without any adjustment based on surveillance data. The USE decrease for RV beltline weld heat 624039/D205A27A (analyzed in LRA Table 4.2-4) was adjusted based on the measured and RG 1.99-predicted values of the USE decrease for the representative surveillance capsule weld (heat 5P6756) from the BWRVIP ISP. The EMA data for RV beltline weld heat 624039/D205A27A was extrapolated by applying the bounding correction of the four weld heat 5P6756 surveillance capsules in accordance with Position 2.2 from RG 1.99, Rev. 2, and BWRVIP-74-A.

The staff noted that the application of the percentage USE decrease for surveillance capsule weld heat 5P6756 from the BWRVIP ISP for adjusting the EMA data for RV beltline weld heat 624039/D205A27A was valid because this RV weld heat identification was changed to 5P6756 in accordance with BWRVIP-86, "BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan," Rev. 1, September 2008. Therefore, submerged arc weld heat 5P6756 is the correct heat identification for both the representative ISP capsule weld and the RV beltline axial weld, identified in the LRA as heat no. 624039/D205A27A.

The staff found the applicant's response to RAI 4.2.2-1 acceptable because the applicant explained that the EMA data for RV beltline plate heats C1337-1 and C1337-2 was not adjusted based on ISP surveillance data because ISP data was not available. Conversely, valid ISP surveillance data was used for adjusting the EMA data for RV beltline weld heat 624039/D205A27A (now heat 5P6756), in accordance with BWRVIP-74-A, Appendix B, and Position 2.2 of RG 1.99, Rev. 2. The staff's concern described in RAI 4.2.2-1 is resolved.

The staff noted that LRA Table 4.2-4 provides the results of the USE EMA for the limiting beltline weld (Heat 624039/D205A27A) at 54 EFPY. This table depicts two percentage decreases in the USE for this weld – a "RG 1.99 predicted decrease" of 13.2 percent and an "adjusted decrease" of 21.6 percent. Therefore, by letter dated August 3, 2010, the staff issued RAI 4.2.2-2, requesting the applicant to clarify which of the values represents the accurate value for the actual reported percentage USE decrease for this weld.

In its response to RAI 4.2.2-2 by letter dated September 27, 2010, the applicant stated that the EMA data for RV beltline weld heat 624039/D205A27A was adjusted based on the USE decrease for the representative weld (heat 5P6756) from the BWRVIP ISP in accordance with Position 2.2 from RG 1.99, Rev. 2, and BWRVIP-74-A. As such, the adjusted percentage decrease in the USE for this weld of 21.6 percent was used for comparison with the EMA acceptance criterion of 39 percent from BWRVIP-74-A.

The staff found the applicant's response to RAI 4.2.2-2 acceptable because the applicant explained that the adjusted decrease in the USE for RV beltline weld heat 624039/D205A27A (21.6 percent) represented the accurate EMA value for this weld, and this value was used for comparison with the 39 percent EMA acceptance criterion from BWRVIP-74-A. The staff's concern described in RAI 4.2.2-2 is resolved.

Based on the applicant's responses to RAIs 4.2.2-1 and 4.2.2-2, the staff determined that the applicant correctly utilized the percentage decrease in the USE for the BWRVIP ISP representative weld (5P6756) to adjust the USE decrease value for Columbia RV beltline weld Heat 624039/D205A27A, in accordance with BWRVIP-74-A, Appendix B and Position 2.2 from

RG 1.99, Rev. 2. In addition, the applicant correctly utilized the results of the EMA from BWRVIP-74-A to demonstrate that the percentage decrease in the USE for the RV plate material, and adjusted percentage decrease in the USE for the RV weld material, were bounded by the results of the EMA for BWRVIP-74-A.

The staff independently verified the reduction in the USE values resulting from neutron irradiation using the methodology in RG 1.99, Rev. 2, and verified that these values are bounded by the EMA acceptance criteria of BWRVIP-74-A. The staff found that the applicant demonstrated that, with the exception of the RV beltline N6 RHR/LPCI nozzles, N12 instrumentation nozzles, and their associated nozzle-to-RV welds, all other RV beltline materials are projected to meet the USE requirements of 10 CFR Part 50, Appendix G at the end of the period of extended operation (54 EFPY).

As discussed in Section 4.2.1.2 of this SER, the applicant provided its response to RAI 4.2.1-a by letter dated November 23, 2010. Part of the applicant's response to RAI 4.2.1-a addresses the USE evaluation of the N12 instrumentation nozzles.

The applicant stated in RAI Response 4.2.1-a that the N12 instrumentation nozzles are not thick-walled forgings inserted in the RV wall and welded to the full penetration of the RV wall. Rather, these nozzles forgings are essentially pipes with a maximum outer diameter of 3.320 inches and a constant inner diameter of 1.938 inches. At the end of the nozzles, outside the RV, the wall thickness is only 0.309 inches, as the inside diameter is increased to 2.406 inches to accept the 2-inch schedule 80 instrument piping. The forgings are inserted into a slightly larger hole in the RV shell and welded at the RV inside diameter. The nozzles are located in the lower intermediate shell, which has a projected USE of 86.1 ft-lb at 54 EFPY, per LRA Table 4.2-2, well above the 10 CFR 50 required USE of 50 ft-lb.

The applicant further stated in RAI Response 4.2.1-a that the unirradiated USE for Columbia's N12 instrumentation nozzles is unknown; consequently a direct calculation of the 54 EFPY USE value is not possible. The applicant compared the N12 instrumentation nozzles to the EMA for plate material in BWRVIP-74A. A search of the records located the Certified Material Test Reports (CMTRs) for the four N12 nozzles; however, only one contained the analyzed weight percentage (wt. percent) copper content required to calculate a projected percentage decrease in the USE. The applicant projected the percentage decrease in the USE for the one N12 nozzle for which the wt. percent copper content was known, based on that copper content and the projected fluence described in RAI Response 4.2.1-a. The applicant's calculated value for the projected percentage decrease in the USE for this one N12 instrumentation nozzle is 16.3 percent at 54 EFPY. The applicant stated that this projected USE decrease is less than the 23.5 percent USE decrease acceptance criterion from the BWRVIP-74-A EMA for plate material and is therefore bounded by that EMA for plate material. Although the acceptance criterion of 23.5 percent from BWRVIP-74-A is for plate material, the applicant stated that GE records confirm that this EMA bounding value was derived from data that included both rolled plate and nozzle forgings, and is thus an appropriate acceptance criterion for these forged nozzles.

The staff reviewed the applicant's response to RAI 4.2.1-a and determined that the applicant had not demonstrated that the use of the BWRVIP-74-A EMA for rolled plate material is a valid EMA acceptance criterion for nozzle forgings. Furthermore, the staff determined that the applicant had not adequately demonstrated that the use of the wt. percent copper content for the one N12 nozzle (for which the wt. percent copper content is known) is valid for calculating the projected percentage decrease in the USE values for the other three N12 nozzles.

Therefore, the staff found that the applicant had not demonstrated that the USE for the N12 instrumentation nozzles will meet the requirements of 10 CFR Part 50, Appendix G, through the end of the period of extended operation. In order to conclusively demonstrate that the USE for the N12 instrumentation nozzles will meet the requirements of 10 CFR Part 50, Appendix G through the end of the period of extended operation, the applicant must submit information demonstrating the validity of their assumptions, with respect to (1) the use of the BWRVIP-74-A EMA acceptance criterion for plate material and (2) the application of the known wt. percent copper content for the one N12 nozzle (for which the wt. percent copper content is known) for calculating the projected percentage decrease in the USE values for the other three N12 nozzles.

The applicant's RAI response dated November 23, 2011, also provided Amendment 12 to the Columbia LRA. Amendment 12 included revisions to LRA Sections 4.2.2 and A.1.3.1.2 to address discussion of the EMA for the N12 instrumentation nozzles and added Table 4.2-9 to the LRA. Table 4.2-9 includes calculations for the N12 instrumentation nozzles EMA. After reviewing the applicant’s response to RAI 4.2.1-a, the staff also determined that the applicant must submit a USE evaluation for the N12 instrumentation nozzle-to-RV welds.

The staff issued RAI 4.2-1, dated December 20, 2010, requesting, in part (d), that the applicant provide an EMA, as described in SER Section 4.2.1.2, for the N12 nozzles and associated nozzle-to-RV welds.

In its response to RAI 4.2-1(d), the applicant stated that the initial USE for the N12 forgings is unknown, therefore, the applicant provided an LRA supplement that projected precent drop in USE to demonstrate equivalent margin. The applicant also stated that while the percent projected drop does meet the acceptance criterion of NEDO-32205, discussions between the staff, the applicant, and the orginal equipment manufacturer have confirmed that the NEDO-32205 acceptance criteria cannot be applied to forgings without further study. Therefore, the applicant included a commitment (Commitment No. 70) to perfom the necessary equivalent margin analysis for the N12 nozzle forgings prior to the period of extended operation.

The staff reviewed the applicant's response to RAI 4.2-1(d) and was concerned that the USE for the N12 nozzles may drop below 50 ft-lbs prior to the period of extended operation.

By letter dated March 23, 2011, the staff issued RAI 4.2-6 requesting the applicant to clarify its commitment to submit an EMA for NRC staff review and approval either (i) at least 2 years prior to the estimated date the N12 nozzles’ USE would drop below 50 ft-lbs, or (ii) at least 2 years prior to the period of extended operation.

In its response by letter dated April 22, 2011, the applicant stated that the N12 nozzles are fabricated from SA-508 Class 1 material. The applicant provided the projected USE to 54 EFPY for the N12 nozzle forgings in its LRA supplement. The unirradiated (initial) transverse USE of 62 ft-lbs and copper content of 0.27 percent used in the calculation of projected USE for the N12 nozzles are based on the results of a statistical analysis of data by the original equipment manufacturer for SA-508 Class 1 forging material. The RG 1.99 Rev. 2 decrease in USE projected to 54 EFPY for the N12 nozzles is 18 percent. The applicant stated that the USE projected to 54 EFPY results in 51 ft-lbs for the limiting N12 nozzle forging. The applicant concludes that the requirement for 50 ft-lbs minimum USE at the end of vessel life is met for the current license period and for the period of extended operation for the N12 nozzle forgings. Therefore, the USE will not drop below 50 ft-lbs prior to the period of extended operation. The applicant revised it commitment to perform the necessary equivalent margin analysis for the N12 nozzle forgings no later than 2 years prior to the period of extended operation.

The staff reviewed the applicant's response and had concerns that the applicant did not provide a technical basis on the unirradiated (initial) transverse USE of 62 ft-lbs and copper content of 0.27 percent used in the calculation of projected USE for the N12 nozzles. This basis should be provided by the applicant and verified by the staff in order for the applicant to demonstrate that the USE value for the N12 nozzle forgings will not fall below 50 ft-lbs prior to the period of extended operation. This issue was open item OI 4.2-1 in the SER with open items.

By letter dated September 26, 2011, the staff issued RAI 4.2-7 requesting the applicant to provide a technical basis for the initial USE values of 62 ft-lb and copper content of 0.27 percent for SA-508 Class 1 forging materials for the N12 nozzle forgings.

In its response by letter dated November 1, 2011, the applicant provided the unirradiated Charpy USE and Cu content data for SA-508, Class 1 forgings. This data was used to justify the applicant’s selection of 62 ft-lbs and 0.27 percent Cu for the initial USE and Cu content of the N12 nozzle forgings. The staff reviewed the data and determined that 0.27 percent represents an appropriately conservative value for the Cu content of the N12 nozzles at Columbia. However, the staff determined that the submitted data was not sufficient to support the applicant’s claim of 62 ft-lbs for the N12 nozzles’ initial USE. Therefore, in a teleconference discussion with the applicant on November 28, 2011, the staff requested that the applicant provide additional information to support its selection of an initial USE value of 62 ft-lbs for the N12 nozzles at Columbia.

In a supplemental letter dated December 6, 2011, the applicant provided heat number-specific unirradiated transverse Charpy test data from certified material test reports (CMTRs) for the four N12 nozzles at Columbia. One of the four N12 nozzles was fabricated from Heat No. 718259, and the other three N12 nozzles were fabricated from Heat No. 219972. In its December 6, 2011, supplemental letter, the applicant stated that the most limiting heat number, with respect to absorbed energy values during the Charpy test, is Heat No. 219972. The applicant noted that the two lowest values for the energy absorbed for the Heat No. 219972 sample set are 60 ft-lbs and 90 ft-lbs, corresponding to 34 percent and 38 percent shear fracture area, respectively. The applicant stated that the 34 percent shear value corresponding to the 60 ft-lbs of absorbed energy indicates that this absorbed energy value is well below the USE for this specific heat number of material. The applicant noted that the remaining four values of the energy absorbed for the Heat No. 219972 sample set are greater than or equal to 230 ft-lbs, corresponding to 100 percent shear. For the Heat No. 718259 sample set, the applicant noted that all of the absorbed energy values were 240 ft-lbs, which is the maximum applied energy, as limited by the Charpy impact test equipment, and the percent shear values could not be determined because these test specimens did not fail at this maximum applied energy. The applicant concluded that the heat-specific Charpy test data for the N12 nozzles at Columbia demonstrate that Columbia’s N12 nozzle forging heat numbers would be expected to have initial USE values well above 62 ft-lbs, and therefore 62 ft-lbs represents a conservative lower bound initial USE value for Columbia’s N12 nozzle forgings.

The staff reviewed the heat number-specific transverse Charpy test data from the CMTRs for the N12 nozzle forgings and noted that the Charpy impact test resulted in an absorbed energy value of 240 ft-lbs for the Heat No. 718259 test specimens. Although none of the Charpy test data for Heat No. 718259 could be used to define a USE value for this heat number because none of the specimens failed at 240 ft-lbs (which was the maximum applied energy level for the test equipment), the staff concludes that the USE for Heat No. 718259 is no lower than 240 ft-lbs. Therefore, the staff determined that an initial USE of 62 ft-lbs represents a conservative and therefore, acceptable value for the Heat No. 718259 nozzle forgings because the actual USE for this heat number is likely greater than 240 ft-lbs. For the Heat No. 219972 test specimens, the staff noted that the lowest absorbed energy value from the Charpy tests is 60 ft-lbs with 34 percent shear. The staff determined that a 34 percent shear value indicates that this specimen is exhibiting predominately brittle fracture characteristics, and therefore, the actual USE value for this specimen is significantly greater than 60 ft-lbs. The staff noted that the remaining absorbed energy values for the Heat No. 219972 test specimens are 90 ft-lbs at 38 percent shear (one specimen), 230 ft-lbs at 100 percent shear (two specimens), and 240 ft-lbs at 100 percent shear (two specimens). The staff noted that the American Society for Testing and Materials (ASTM) Standard Practice E 185-82, "Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," Section 4.17 defines the USE on the Charpy transition curve as absorbed energy with greater than 95 percent shear. Based on this definition, the staff determined that an initial USE of 62 ft-lbs represents a conservative and therefore, acceptable initial USE value for the Heat No. 219972 nozzle forgings because the lowest absorbed energy value (230 ft-lbs) at 100 percent shear indicates that the USE for this heat number is likely greater than or equal to 230 ft-lbs.

Based on its review of the applicant's November 1, 2011, RAI response, as supplemented by information provided on December 6, 2011, the staff determined that the applicant provided an adequate technical basis to support its selection of an initial USE value of 62 ft-lbs and content value of 0.27 percent Cu for the N12 nozzle forgings at Columbia. Specifically, the staff determined that, based on its review of the CMTR Charpy test data for Columbia's N12 nozzle forgings, the initial USE value for these nozzles is at least 230 ft-lbs, based on the definition of USE provided in ASTM E 185-82. Therefore, the staff finds that the applicant's selection of 62 ft-lbs as the initial USE value for the N12 nozzle forgings is conservative and therefore acceptable.

The staff determined that the applicant's 54 EFPY projected USE value of 51 ft-lbs was correctly calculated in accordance with RG 1.99 Revision 2, based on an initial USE value of 62 ft-lbs, 0.27 percent Cu, and 1/4T fluence of 4.48×10^{17} n/cm^2. The initial USE value of 62 ft-lbs and projected USE value of 51 ft-lbs exceed the 50 ft-lbs minimum value specified in 10 CFR Part 50, Appendix G for acceptance of RV beltline USE values without further analysis. Therefore, the staff determined that the applicant adequately demonstrated that the N12 nozzles are projected to remain in compliance with the USE requirements specified in 10 CFR Part 50, Appendix G. Accordingly, the staff determined that OI 4.2-1 is closed.

The staff noted that the applicant's USE evaluation did not address the three N6 RHR/LPCI nozzles or the associated N6 RHR/LPCI nozzle-to-RV welds. As discussed above in Section 4.2.1 of this SER, the applicant determined that these nozzles will be exposed to a projected neutron fluence greater than 1×10^{17} n/cm^2 (E > 1.0 MeV) at the end of the period of extended operation (54 EFPY). Specifically, the applicant determined that these nozzles will be exposed to a projected 54 EFPY neutron fluence of 4.48×10^{17} n/cm^2 (E > 1.0 MeV) at the 1/4T location in the RV wall. Therefore, the N6 RHR/LPCI nozzles are beltline components requiring neutron embrittlement evaluation in accordance with 10 CFR Part 50, Appendix G, and the GALL report.

As discussed in SER Section 4.2.1.2, the staff issued RAI 4.2.1-a, dated August 26, 2010, requesting that the applicant supplement Section 4.2.2, including Table 4.2-2, 4.2-3, or 4.2-4 (as applicable), of the Columbia LRA to include data for the analysis of the USE for the three RV N6 RHR/LPCI nozzles because the USE for these beltline nozzles must be projected to the end of the period of extended operation to determine whether the nozzles will remain in compliance with 10 CFR Part 50, Appendix G, requirements.

By letter dated November 23, 2010, the applicant provided its response to RAI 4.2.1-a, addressing the USE for RHR/LPCI nozzles. The applicant stated in this RAI response that the unirradiated USE for Columbia's N6 nozzles is not known; consequently a direct projection of the USE value is not possible. The applicant calculated the projected percentage decrease in the USE value for these nozzles based on the known copper content and projected 54 EFPY neutron fluence. The applicant determined that the projected percentage decrease in the USE for these nozzles at 54 EFPY is 9.6 percent. The applicant stated that this value is below the 23.5 percent acceptance criterion established for rolled plate material in BWRVIP-74 and is also below the projected USE decrease from the EMAs for Columbia's limiting plate and weld materials (LRA Tables 4.2-3 and 4.2-4). The applicant further stated that, although the acceptance criterion of 23.5 percent from BWRVIP-74-A is for rolled plate material, GE records confirm that this value was derived from data that included both rolled plate and nozzle forgings, and is thus an appropriate acceptance criterion for these forged nozzles.

The staff reviewed the applicant's response to RAI 4.2.1-a and determined that the applicant had not demonstrated that the use of the BWRVIP-74-A EMA for plate material is a valid EMA acceptance criterion for nozzle forgings. Therefore, the staff found that the applicant had not demonstrated that the USE for the N6 RHR/LPCI nozzles will meet the requirements of 10 CFR Part 50, Appendix G, through the end of the period of extended operation. Furthermore, the staff found that the applicant had not provided a USE evaluation for the N6 nozzle-to-RV welds. In order to conclusively demonstrate that the USE for the N6 RHR/LPCI nozzles will meet the requirements of 10 CFR Part 50, Appendix G, through the end of the period of extended operation, the applicant must submit information demonstrating that their assumption, with respect to the use of the BWRVIP-74-A EMA acceptance criterion for plate material to determine the acceptability of the EMA data for the N6 RHR/LPCI nozzles, is valid. The applicant must also submit a USE evaluation for the N6 RHR/LPCI nozzle-to-RV welds.

As described in SER Section 4.2.1.2, the staff issued RAI 4.2-2, dated December 20, 2010, requesting, in part (c), the applicant to provide an EMA for the N6 RV beltline RHR/LPCI nozzle forgings and the associated nozzle-to-RV weld materials to demonstrate that the 54 EFPY USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

In its response to RAI 4.2-2(c) dated January 28, 2011, the applicant stated that the projection of USE requires an initial (unirradiated) value of USE for the subject material. As the N6 nozzle forgings are SA-508 Class 2, an initial USE of 70 ft-lbs was used by the applicant.

The applicant provided the calculation of the projected USE for the N6 RHR/LPCI nozzle forgings in its LRA supplement. The applicant also stated that the projected USE exceeds 50 ft-lbs and thus no equivalent margin analysis is required by Appendix G of 10 CFR 50.

For the N6 RHR/LPCI nozzle-to-vessel welds, the applicant stated that the initial USE is unknown and therefore cannot be projected. An equivalent margin analysis is demonstrated by projecting the percent reduction in USE (7.8 percent) and comparing it to the weld material acceptance criteria in BWRVIP-74-A (<39 percent). The applicant states the acceptance criteria are met.

The staff reviewed the applicant's response to RAI 4.2-2(c) and notes the applicant did not provide a basis to support the 70 ft-lbs initial USE value for the N6 nozzle forgings.

By letter dated March 23, 2011, the staff issued RAI 4.2-4 requesting the applicant provide a basis for the 70 ft-lbs initial USE value and justify that the value is based on the lower bounding value of available USE test data for SA-508 Class 2 forging material.

In its response to RAI 4.2-4 dated April 22, 2011, the applicant stated that the initial USE value of 70 ft-lbs is based on the review of available data from EPRI and NRC databases for SA-508 Class 2 forging material and previous utility submittals related to SA-508 Class 2 materials. The databases are referenced in Altran Technical Report 96124-TR-01, "N-16 Nozzles Upper Shelf Energy Evaluation," December 1995. The initial USE data in the Altran report was updated in a later GE-Hitachi report 0000-0114-0580-R0-NP, "Limerick Generating Station, Units 1 & 2, Upper Shelf Energy Evaluation for LPCI Nozzle Forging Material," August 2010, to include additional SA-508 Class 2 forging data from the current NRC database RVID2. The applicant has reviewed all the tabulated data for SA-508 Class 2 and concluded that the lowest value in the reports for initial USE is 72 ft-lbs. The applicant stated that it conservatively selected the initial USE value of 70 ft-lbs in its calculation in order to be consistent with the initial USE previously accepted by the NRC for this material.

The staff reviewed the applicant's response to RAI 4.2-4 and finds it acceptable because the staff independently reviewed the initial USE data for SA-508, Class 2 forging material provided in the August 2010 GEH technical report and confirmed that the lowest USE value in the database is 70 ft-lbs. Furthermore, the staff found that most of the initial USE values for SA-508, Class 2 forging material were well above 70 ft-lbs. Accordingly, the staff concluded that 70 ft-lbs is a valid conservative initial USE value for the N6 nozzle forgings at Columbia, and the applicant's calculated 54 EFPY USE value of 63.3 ft-lbs for the N6 nozzle forgings is acceptable. Therefore, the staff found that the USE evaluation for the N6 nozzle forgings is projected to remain in compliance with the requirements of 10 CFR Part 50, Appendix G through the period of extended operation. The applicant's 54 EFPY USE projection for the N6 nozzle forgings was included in Amendment 28 to the Columbia LRA. The staff concern described in RAI 4.2-4 is resolved.

The staff's review of the applicant's response to RAI 4.2-2(c) also notes the percent USE decrease acceptance criteria in BWRVIP-74-A are based on minimum USE requirements derived from EMAs performed in GE NEDO-32205, "10 CFR 50 Appendix G Equivalent Margin Analysis for Low Upper-Shelf Energy in BWR-2 through BWR-6 Vessels," for shell plates and shell welds, and a conservative estimate of initial USE values based on a statistically significant set of USE data for each type of RV shell material.

The NEDO-32205 EMAs developed minimum USE acceptance criteria for shell plates and shell welds based on the ASME Code Case N-512 procedures, which are now codified in Appendix K of the ASME Code, Section XI. The procedures include (1) the selection of an appropriate J-integral fracture resistance curve for the class of material being analyzed, (2) the calculation of J-integral values due to applied loads for RV shell components based on a postulated flaw configuration, and (3) the application of the acceptance criteria for (a) the applied J-integral at a ductile flaw extension of 0.1 inch and (b) flaw stability due to ductile tearing.

Calculations of J-integrals due to applied loads are very component specific. For example, the applied J-integrals for RV shell components differ significantly form the applied J-integral for nozzles, even if the two types of components are fabricated from the same class of materials.

In order to demonstrate the BWRVIP-74-A acceptance criteria for shell welds can be used to determine the acceptability of the N6 nozzle welds, it is necessary to confirm that (1) the N6 nozzle weld material is of the same class as the shell weld material analyzed in NEDO-32205, with respect to weld filler metal, welding flux, and weld fabrication material, and (2) the N6 nozzle weld configuration, postulated flaw configuration, and loading is identical to (or bounded by) the RV shell weld configuration, postulated flaw configuration, and loading, with respect to the applied J-integral values as calculated using ASME Code Case N-512 and Appendix K procedures.

By letter dated March 23, 2011, the staff issued RAI 4.2-5 requesting the applicant provide justification for the use of the BWRVIP-74-A shell weld EMA acceptance criteria to determine the acceptability of the N6 RHR/LPCI nozzle welds, based on (1) N6 nozzle weld material and weld fabrication technique; and (2) the N6 nozzle weld configuration, postulated flaw configuration, and loading, relative to the RV shell welds.

In its response to RAI 4.2-5(1) dated April 22, 2011, the applicant stated the percent decrease in USE acceptance criteria in BWRVIP-74-A is based on NEDO-32205. Columbia's reactor vessel beltline shell plate, weld materials, and welding processes were included in the development of NEDO-32205. The welding procedure specifications used to fabricate Columbia's reactor vessel beltline shell plate welds specifies the weld materials and welding processes. The N6 nozzle welds were fabricated using the same welding procedure specification that was used for the shell plate welds, thus the same weld materials and welding processes were used for the N6 nozzle welds. Therefore, the applicant states that the N6 nozzle welds are of the same class as the shell weld material analyzed in NEDO-32205 with respect to filler metal, welding flux and weld fabrication technique.

The staff confirmed that the N6 nozzle-to-RV welds at Columbia were among the types of shell welds analyzed for the NEDO-32205 EMAs—Linde 124 submerged arc welds.

In its response to RAI 4.2-5(2) dated April 22, 2011, the applicant provided the calculation of projected USE for the N6 nozzle welds in an LRA supplement. The applicant states the projected USE exceeds 50 ft-lbs and thus an EMA using the methodology of BWRVIP-74-A (NEDO-32205) is not required. NEDO-32205, Table 2 provides an initial USE for "non-Linde 80" weld material of 70 ft-lbs. This value is based on the staff's statistical analysis for predicting 95 percent of the entire population with a one-sided 95 percent confidence for the "non-Linde 80" type of beltline weld material. The N6 RHR/LPCI nozzle weld material is "non-Linde 80." The 54 EFPY ¼ T fluence for the N6 RHR/LPCI nozzle welds is 5.00 $\times 10^{17}$ n/cm^2. The BWRVIP integrated surveillance program (ISP) best estimate chemistry copper content for N6 weld material, Heat 5P6214B, is 0.019 percent. Using the data and RG 1.99 Rev. 2, the applicant projected a decrease in USE of 7.8 percent. The applicant states the result is a projected USE value of 64 ft-lbs for weld Heat 5P6214B at 54 EFPY. The applicant concludes that the 10 CFR 50 Appendix G requirement for 50 ft-lbs minimum USE at the end of vessel life is met for the current license period and through the end of the period of extended operation for the N6 weld.

The staff's review of the applicant's response to RAI 4.2-5 and finds it acceptable because the staff determined that an initial USE value as low as 54.2 ft-lbs would result in acceptance of the USE for N6 nozzle welds through 54 EFPY, given that the percentage decrease in USE projected through 54 EFPY is only 7.8 percent. In addition, all of the initial USE data in the non-Linde 80 submerged arc weld database are significantly greater than 54.2 ft-lbs. Therefore, the staff found that the USE evaluations for the N6 nozzle forgings and welds are projected to remain in compliance with the requirements of 10 CFR Part 50, Appendix G, through the period of extended operation. The applicant's 54 EFPY USE projections for the N6 nozzle forgings and welds were included in Amendment 28 to the Columbia LRA. The staff's concerns described in RAI 4.2-2(c) and RAI 4.2-5 are resolved.

4.2.2.3 UFSAR Supplement

In LRA Section A.1.3.1.2, the applicant provided a UFSAR Supplement summary description for the USE Evaluation, the TLAA for which is described in LRA Section 4.2.2.

The staff also noted that the applicant committed (Commitment No. 70) to, "Perform a 54 EFPY equivalent margin analysis for the embrittlement (upper shelf energy) of the reactor vessel N12 (instrumentation) nozzle forgings."

The staff reviewed the applicant's UFSAR Supplement summary description for the USE Evaluation and determined that it is consistent with the TLAA described in LRA Section 4.2.2.

4.2.2.4 Conclusion

On the basis of its review and closure of OI 4.2-1, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the effects of aging on the upper shelf energy analysis have been projected to the end of the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d), and, therefore, is acceptable.

4.2.3 Adjusted Reference Temperature

4.2.3.1 Summary of Technical Information in the Application

In addition to the USE, the other key parameter that characterizes the fracture toughness of a material is the RT_{NDT}. This reference temperature changes as a function of exposure to neutron radiation resulting in an adjusted reference temperature (ART). The initial RT_{NDT} is the reference temperature for the unirradiated material as defined in Paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code. The change in the RT_{NDT} value due to neutron radiation is referred to as ΔRT_{NDT}. The ART is calculated by adding the initial RT_{NDT}, the ΔRT_{NDT}, and a margin term to account for uncertainties in the initial RT_{NDT} and ΔRT_{NDT} values, as prescribed in Regulatory Guide 1.99, Revision 2.

The ART calculations of record for the RV beltline plates and welds for the current licensed operating period (33.1 EFPY), including power uprate conditions, are provided in NEDO-33144, which lists the initial RT_{NDT} and chemistry values for the Columbia RV materials obtained from the Columbia RV Certified Material Test Reports. Some chemistry factors were adjusted when Surveillance Capsule Data and Integrated Surveillance Program (ISP) best estimates were available, as described in NEDO-33144.

The methodology of RG 1.99, Rev. 2, and the projected 54 EFPY fluence values listed in LRA Table 4.2-1 were used to project ART values for RV beltline materials that are valid through the end of the period of extended operation (54 EFPY). The results of this projection are summarized in LRA Table 4.2-5 for the RV beltline plates and welds. The 54 EFPY ART values will be used to develop P-T limit curves, as discussed in Section 4.2.4. The applicant concludes that all of the projected ART values for 54 EFPY are well below the 200°F end of life ART recommended in Section 3 of RG 1.99, Rev.2, and are, therefore, acceptable for the period of extended operation.

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

4.2.3.2 Staff Evaluation

The staff reviewed LRA Section 4.2.3 and the TLAAs for the ART to verify, pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis have been projected to the end of the period of extended operation.

The fluence values for the Columbia RV beltline materials at 54 EFPY, listed in LRA Table 4.2-5 for calculating the ART values, correspond to the fluence values provided in LRA Section 4.2.1, that were found acceptable by the staff in SER Section 4.2.1.2.

The staff reviewed LRA Section 4.2.3 and LRA Table 4.2-5 to verify that the applicant used accurate input data (e.g., initial RT_{NDT} and chemistry data) for determining the 54 EFPY ART values for the Columbia RV beltline materials. In reviewing the initial RT_{NDT} data, chemistry data (percent Cu and percent Ni), and CF values for the RV beltline materials provided by the applicant in LRA Table 4.2-5, the staff found that, with the exception of several data points discussed below, these values were consistent with the corresponding data utilized by the applicant in calculating 33.1 EFPY ART values for the Columbia RV beltline materials, as described in GE Report NEDO-33144. The staff noted that these initial RT_{NDT}, chemistry (percent Cu and percent Ni), and CF values were previously submitted by the applicant to the NRC staff as part of its analysis for determining the current reactor coolant system P-T limit curves for 33.1 EFPY. These P-T limits were approved by the staff in License Amendment No. 193, dated May 12, 2005. With respect to the corresponding initial RT_{NDT}, chemistry, and CF values currently established in the staff's Reactor Vessel Integrity Database (RVID), the staff noted several inconsistencies between the values reported in LRA Table 4.2-5 and the corresponding RVID values. However, the staff determined that the applicant previously addressed the discrepancies between the LRA Table 4.2-5 initial RT_{NDT}, chemistry, and CF values (also contained in GE Report NEDO-33144) and the corresponding RVID values in its 33.1 EFPY P-T limits submittal to the NRC, as part of its license amendment request, dated June 9, 2004, to implement the current 33.1 EFPY P-T limits. The staff approved the use of these values as inputs for determining the limiting 33.1 EFPY ART value for calculating the 33.1 EFPY P-T limits in License Amendment No. 193, dated May 12, 2005.

As stated above, the staff noted several discrepancies between data points contained in GE Report NEDO-33144 for the 33.1 EFPY ART calculations and those provided in LRA Table 4.2-5 for the 54 EFPY ART calculations. With respect to the value for the uncertainty in the initial RT_{NDT} value (σ_i) for the RHR/LPCI N6 nozzles the staff noted a discrepancy between the value reported in LRA Table 4.2-5 and Tables 4-5a and 4-6a of GE Report NEDP-33144 for calculating Columbia's current technical specification (TS) P-T limit curves. LRA Table 4.2-5 lists the σ_i value as 1.4 for the RHR/LPCI N6 Nozzles. Tables 4-5a and 4-6a of GE NEDO-33144 list the σ_i value as zero for these nozzles.

By letter dated August 3, 2010, the staff issued RAI 4.2.3-1, requesting that the applicant explain the discrepancy between the value reported in LRA Table 4.2-5 and Tables 4-5a and 4-6a of GE Report NEDO-33144 for calculating Columbia's current technical specification (TS) P-T limit curves.

In its response dated September 27, 2010, the applicant stated that the σ_i value of 1.4 listed in LRA Table 4.2-5 is a typographical error. The correct σ_i value is zero. The applicant noted that the full margin term value of 21.1 listed in LRA Table 4.2-5 was correctly calculated based on a σ_i value of zero. Accordingly, the applicant revised LRA Table 4.2-5 to reflect the correct σ_i value of zero for the RHR/LPCI N6 nozzles. The staff determined that the applicant's response to RAI 4.2.3-1 was acceptable because the applicant resolved the discrepancy between LRA Table 4.2-5 and GE NEDO-33144, with respect to the σ_i value for the RHR/LPCI N6 nozzles, and revised its LRA Table 4.2-5 to reflect the correct σ_i value of zero for these nozzles, which is consistent with the CLB. The staff's concern described in RAI 4.2.3-1 is resolved.

The staff also noted that Table 4-3 of GE NEDO-33144 lists two initial RT_{NDT} data points for weld heat 5P6756/0342-3477, one for single wire and one for tandem wire. LRA Table 4.2-5 lists only a single data point for this weld heat.

By letter dated August 3, 2010, the staff issued RAI 4.2.3-2, requesting that the applicant clarify whether the single data point for this weld heat in LRA Table 4.2-5 is representative of both the single wire and tandem wire properties. In its response dated September 27, 2010, the applicant stated that the two data points (single wire and tandem wire) for weld heat 5P6756/0342-3477 in Table 4-3 of GE NEDO-33144 both have initial RT_{NDT} values of -50 °F, and weld chemistry is not affected by the single or tandem wire process. The fluence listed for this weld in LRA Table 4.2-5 bounds the entire weld. As such, both the single and tandem wire weld portions are represented by the same line entry in LRA Table 4.2-5. The staff determined that the applicant's response to RAI 4.2.3-2 was acceptable because the applicant adequately explained that the single line entry in LRA Table 4.2-5 for weld heat 5P6756/0342-3477 is representative of both single and tandem wire properties. The staff's concern described in RAI 4.2.3-2 is resolved.

The staff also determined that further information was required from the applicant concerning its application of surveillance data to the ART calculations in LRA Section 4.2.3 and Table 4.2-5. By letter dated August 3, 2010, the staff issued RAI 4.2.3-3, requesting that the applicant indicate which of the RV beltline material ART values from LRA Table 4.2-5 utilize chemistry factor (CF) values that are calculated based on the application of credible surveillance data from Columbia surveillance capsules or BWR integrated surveillance program (ISP) surveillance capsules, in accordance with Regulatory Position (RP) 2.1 of RG 1.99, Rev. 2. In addition, the staff requested that the applicant state which of the RV beltline material ART values utilize CF values that are calculated based on RP 1.1 from RG 1.99, Rev. 2. The staff also requested that the applicant provide references for any surveillance capsule test reports that were used for determining CF values for the RV beltline materials, because no Columbia or ISP surveillance capsule test reports are referenced in LRA Section 4.8.

In its response dated September 27, 2010, the applicant stated that the ART value for Columbia RV beltline weld heat 5P6756/0342-3477 was calculated utilizing a chemistry factor based on the application of credible surveillance data from the BWRVIP ISP in accordance with RG 1.99 Rev. 2, Position 2.1. The applicant listed the surveillance capsules and provided references for the applicable BWRVIP surveillance capsule test reports, which were used for calculating the CF value for weld heat 5P6756/0342-3477. The references for the applicable BWRVIP surveillance capsule test reports are listed below:

1. River Bend 183º Capsule, BWRVIP-113, "BWR Vessel and Internals Project, River Bend 183 Degree Surveillance Capsule Report," June 2003.

2. Supplemental Surveillance Program (SSP) Capsule F, BWRVIP-111, Rev. 1, “BWR Vessel and Internals Project, Testing and Evaluation of BWR Supplemental Surveillance Program Capsules E, F, and I,” September 2007.
3. SSP Capsule H, BWRVIP-87, Rev. 1, “BWR Vessel and Internals Project, Testing and Evaluation of BWR Supplemental Surveillance Program Capsules D, G, and H,” September 2007, and BWRVIP-128, “Updated Fluence Calculations for Supplemental Surveillance Capsules D, G, and H Using RAMA Fluence Methodology,” August 2004.
4. SSP Capsule C, BWRVIP-169, “BWR Vessel and Internals Project, Testing and Evaluation of BWR Supplemental Surveillance Program Capsules A, B, and C,” March 2007.

The applicant stated that all other ART values for the Columbia RV beltline plate, nozzle and weld materials were calculated based on CF values obtained from RG 1.99, Rev. 2, Position 1.1. Upon review, the applicant determined that the referencing of footnote 2 in LRA Table 4.2-5 is not correct. Footnote 2 is correctly attached to RV weld heat 5P6756/0342-3477 on page 4.2-10 of the LRA (2nd page of LRA Table 4.2-5); it should not have been attached to RV plate heat number B5301-1 (1st page of LRA Table 4.2-5). Weld heat 5P6756/0342-3477 is the only entry in LRA Table 4.2-5 with a CF adjusted by surveillance data. A revised LRA Table 4.2-5 with footnote 2 removed from the data entry for RV plate B5301-1 was included with the applicant’s RAI response as part of LRA Amendment 8, which was provided by letter dated September 27, 2010.

The staff found the applicant’s response to RAI 4.2.3-3 acceptable because the applicant clearly delineated the RV beltline material ART values that were calculated based on the application of credible surveillance data in accordance with RP 2.1 from RG 1.99, Rev. 2, versus the RV beltline material ART values that were calculated based on RP 1.1 from RG 1.99, Rev. 2 (the CF tables). The applicant also provided the requested references for the ISP surveillance capsule test reports that were used for determining these CF and ART values. The staff's concern described in RAI 4.2.3-3 is resolved.

Note (2) in LRA Table 4.2-5 states that the "adjusted chemistry factor" for lower-to-lower intermediate shell circumferential weld heat 5P6756/0342-3477 was determined per GE Report NEDO-33144, “Pressure-Temperature Curves for Energy Northwest Columbia,” April 2004, Section 4.2.1.1, which was approved by the NRC in an SE and updated per Columbia-specific ISP data.

By letter dated August 3, 2010, the staff issued RAI 4.2.3-4, requesting that the applicant clarify whether the CF value listed in LRA Table 4.2-5 for this weld heat (153.97 ºF) is based on the application of credible surveillance data from Columbia or another applicable ISP plant in accordance with RP 2.1 from RG 1.99, Rev. 2. The staff noted that Tables 4-5b and 4-6b in GE Report NEDO-33144 list a CF value of 157.68 ºF for this weld. Therefore, the staff also requested in RAI 4.2.3-4 that the applicant explain whether the discrepancy between the LRA CF value and the NEDO-33144 CF value for this weld heat is due to the application of Columbia-specific or other ISP surveillance data to the CF calculation subsequent to the issuance of the License Amendment No. 193 for the 33.1 EFPY P-T limit curves (based on the application of GE Report NEDO-33144).

In its response to RAI 4.2.3-4, the applicant stated that the adjusted CF of 153.97 °F listed in LRA Table 4.2-5 for beltline weld heat 5P6756/0342-3477 is based on the application of credible surveillance data from applicable BWRVIP ISP capsules in accordance with RP 2.1 from RG

1.99, Rev. 2. The applicant stated that the CF for this weld heat has been updated based on BWRVIP ISP data applicable to Columbia that became available subsequent to the issuance of NEDO-33144. The BWRVIP ISP surveillance capsule test reports that include the data used for determining the new adjusted CF are BWRVIP-128 and BWRVIP-169, as explained in the response to RAI 4.2.3-3. The applicant modified footnote 2 in revised LRA Table 4.2-5, provided in LRA Amendment 8, to more accurately reflect the explanation provided with the response to RAI 4.2.3-4.

The staff found the applicant's response to RAI 4.2.3-4 acceptable because the applicant explained that (1) the CF value listed in LRA Table 4.2-5 for weld heat 5P6756/0342-3477 (153.97 ºF) is based on the application of credible surveillance data from the ISP in accordance with RP 2.1 from RG 1.99, Rev. 2; and (2) the discrepancy between the LRA Table 4.2-5 CF value and the NEDO-33144 CF value for weld heat 5P6756/0342-3477 is due to the application of credible ISP surveillance data to the CF calculation, subsequent to the issuance of the License Amendment No. 193 in 2005.

The staff independently reviewed all ART calculations in LRA Table 4.2-5 based on the approved chemistry and fluence data and determined that, with the exception of the RV beltline N12 instrumentation nozzles and the N6 RHR/LPCI nozzle-to-RV welds, the applicant appropriately followed the guidance of RG 1.99, Rev. 2, in determining the projected 54 EFPY ART values for the Columbia RV beltline materials. The staff's concern described in RAI 4.2.3-4 is resolved.

As discussed in Section 4.2.1.2 of this SER, the applicant provided its response to RAI 4.2.1-a by letter dated November 23, 2010. Part of the applicant's response to RAI 4.2.1-a addresses the ART analysis of the N12 instrumentation nozzles. The applicant stated that the data necessary to determine the ART for the N12 instrumentation nozzles is unavailable. Licensing Topical Report (LTR) NEDO-33178P-A, "GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves," June 2009, Appendix J addresses the fracture mechanics analysis of the instrument (N12) nozzles. This LTR was approved by the NRC in a SE for LTR NEDO-33178P-A, dated April 27 2009. According to the applicant, a plant-specific assessment for Columbia, based on the analysis of NEDO-33178P-A, demonstrated that the N12 instrumentation nozzles have no impact on the current TS P-T limit curves. The applicant stated that this assessment specifically demonstrated that the water level instrument nozzle (N12) P-T curves are bounded by the RV beltline shell and upper vessel P-T curves. The current TS P-T curves remain valid until 33.1 EFPY and are identified as a TLAA in LRA Section 4.2.4.

The applicant agrees that the N12 instrumentation nozzles must be considered when the applicant develops future P-T limits for Columbia in accordance with 10 CFR Part 50, Appendix G and the ASME Code, Section XI, Appendix G. The applicant stated that it will continue to develop future P-T limit curves for the period of extended operation taking into consideration all beltline plates, welds, and nozzles.

The staff reviewed the applicant's response to RAI 4.2.1-a, pertaining to the ART analysis for the N12 instrumentation nozzles. While the staff acknowledges the possibility that the fracture toughness and applied stress intensity calculations for the N12 nozzles may demonstrate that these nozzles are not bounding relative to other RV components, with respect to the current TS P-T limits for 33.1 EFPY, this fact would not preclude the requirement for calculating projected ART and USE values for these beltline nozzles for the period of extended operation, as the GALL Report specifically recommends that ferritic material for RV beltline shells, welds, and other components be evaluated for neutron embrittlement if these materials are exposed to high energy neutron fluence greater than 1 x 10^{17} n/cm^2 (E > 1.0 MeV) at the end of the period of extended operation.

The staff also determined upon further review, that the applicant had not provided ART analyses of the N12 nozzle-to-RV beltline welds and the N6 RHR/LPCI nozzle-to-RV beltline welds. Therefore the staff determined that the applicant must provide analyses of the ART values for the N12 instrumentation nozzles, the N12 nozzle-to-RV welds, and the N6 RHR/LPCI nozzle-to-RV welds that is valid for the period of extended operation (54 EFPY).

As described in SER Section 4.2.1.2, the staff issued RAI 4.2-1, dated December 20, 2010, requesting, in part (c), the applicant provide ART values at 54 EFPY, for the N12 nozzles and associated nozzle-to-RV welds.

In its response to 4.2-1(c) the applicant provided an LRA supplement that included additional material specifications and ART values for N12 nozzles for the staff to review.

The staff reviewed the applicant's response to RAI 4.2-1 and notes that the requirement of 10 CFR Part 50, Appendix G, apply to, "ferritic material of pressure-retaining components of the reactor coolant pressure boundary." The 10 CFR Part 50, Appendix G requirement do not apply to austenitic-phase materials, such as nickel-based alloys and austenitic stainless steel. Therefore, since the associated N12 nozzle-to-RV welds are composed of an austenitic material then the welds do not need to be analyzed for neutron fluence, USE, or ART.

However, the N12 nozzles are composed of a ferritic material and must be analyzed for the USE and ART. The N12 nozzles are in the beltline region of the RV because it is projected to experience neutron fluence greater than 1 x 10^{17} n/cm^2 (E>1.0 MeV) at the end of the period of extended operation, corresponding to 54 EFPY. In the staff's review of the N12 nozzle data, it reviewed the heat number, chemistry, and initial RT_{NDT} data for the N12 forgings and found the data to be acceptable because the initial RT_{NDT} data for the forgings is consistent with the previously approved values listed in NEDO-33144, and the wt. percent Cu and wt. percent Ni values, while unknown for the limiting heat (Heat No. 219972) of the N12 nozzle forgings, was set at 0.35 percent Cu and 1.00 percent Ni, which are the conservative high values recommended in RP 1.1 of RG 1.99, Rev. 2, if actual heat-specific Cu and Ni contents are unknown.

Also, the staff reviewed the 54 EFPY ART values for the N12 nozzle forgings and found it acceptable because it was determined using the guidance from RP 1.1 of RG 1.99, Rev. 2, based on the initial RT_{NDT} and chemistry data, including a correct margin term. The staff noted that the N12 nozzle forgings are represented by two different heats and therefore have two different ART values. For the more limiting heat (Heat No. 219972) of the N12 nozzle forgings, the ART value was correctly determined to be 149°F, based on a wt. percent Cu and Ni content of 0.35 percent Cu and 1.00 percent Ni and an initial RT_{NDT} value of 40°F. Accordingly, the N12 nozzle forgings from Heat No. 219972 are the limiting RV beltline material, with respect to ART, because the ART value of 149°F exceeds that for all other RV beltline materials, including all plates and welds. The implications of the 149°F ART value for the N12 nozzle forgings, with respect to the P-T limits TLAA is addressed in Section 4.2.4 of this SER. The staff's concern described in RAI 4.2-1(c) is resolved.

As described in SER Section 4.2.1.2, the staff issued RAI 4.2-2 by letter dated December 20, 2010, requesting the applicant to provide calculation of the 54 EFPY ART for the N6 RV beltline RHR/LPCI nozzle-to-RV weld materials.

In its response to RAI 4.2-2(b) dated January 28, 2011, the applicant provided an LRA supplement that included calculations of the 54 EFPY ART for the staff to review.

The staff reviewed the applicant's 54 EFPY ART values for these welds and found them acceptable because it was correctly calculated using the methods in RP 2.1 from RG 1.99, Rev. 2, which specifies the procedure for determining the ART, based on a chemistry factor value derived from a linear fit to the equation for ∆RTNDT in RG 1.99, Rev. 2 using two or more sets of credible surveillance data. The 54 EFPY ART value of -8.9 ºF was calculated using credible BWRVIP ISP surveillance data for this weld. The staff notes that the ART analysis for the N6 nozzle forgings was satisfactorily addressed in the original LRA submittal. The staff's concern described in RAI 4.2-2 is resolved.

4.2.3.3 UFSAR Supplement

LRA Section A.1.3.1.3, as amended, provides the UFSAR supplement for the ART analysis TLAA evaluation. Based on its review of the UFSAR supplement, the staff concludes that the information in the UFSAR 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 has demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the effects of aging on the ART analysis have been projected to the end of the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d), and, therefore, is acceptable.

4.2.4 Pressure-Temperature Limits

4.2.4.1 Summary of Technical Information in the Application

To ensure that adequate margins of safety are maintained for various modes of reactor operation, 10 CFR Part 50, Appendix G, specifies pressure-temperature (P-T) limit requirements and minimum temperature requirements for the service life of the RV. The basis for these fracture toughness requirements is found the ASME Code, Section XI, Appendix G. 10 CFR Part 50, Appendix G requires that P-T limits be established for hydrostatic pressure tests and leak tests for operations with the core not critical during heatup and cooldown and for core critical operations. The Columbia P-T limit curves were revised in 2005 in License Amendment No. 193 to address the effects of an increase in the licensed core thermal power level to 3,486 MWt. The current P-T limits are valid through 33.1 EFPY, which bounds the end of the current 40-year licensed operating period. P-T limits for the period of extended operation will be calculated using the most accurate fluence projections available at the time of the recalculation. The projections may be adjusted if there are changes in core design or if additional surveillance capsule results show the need for an adjustment. The projected RV beltline ART values for the period of extended operation, discussed in LRA Section 4.2.3, provide confidence that future P-T curves will provide adequate operating margin. License amendment requests to revise P-T limits established in the Columbia Technical Specifications (TSs) will be submitted to the NRC for approval, when necessary, to comply with 10 CFR Part 50, Appendix G, as part of the Columbia Reactor Vessel Surveillance Program.

The applicant dispositioned the TLAAs for the P-T limits 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.

4.2.4.2 Staff Evaluation

The staff reviewed LRA Section 4.2.4 and the TLAAs for the P-T limits 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.

In a license amendment request provided by Energy Northwest Letter GO2-04-107 to the NRC, "License Amendment Request to Revise Technical Specification 3.4.11 Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits," dated June 9, 2004, Columbia requested NRC authorization to implement the present TS P-T limits developed using the methodology of GE Report NEDO-33144. By letter dated May 12, 2005, the NRC issued License Amendment No. 193 to Columbia, "Columbia Generating Station – Issuance of Amendment RE: Reactor Coolant System (RCS) Pressure and Temperature Limits (TAC No. MC3591)," authorizing the implementation of these P-T limits in the Columbia TSs. These P-T limit curves are valid through 33.1 EFPY of facility operation, which bounds operation of the reactor coolant system through the end of the current 40-year licensed operating period. The staff confirmed that the current TS P-T limit curves are valid for the current licensed core thermal power level of 3486 MWt. The applicant stated in LRA Section 4.2.4 that P-T limits for the period of extended operation (54 EFPY) will be calculated using the most accurate fluence projections available at the time of the recalculation. The projections may be adjusted if there are changes in core design or if additional surveillance capsule results show the need for an adjustment.

In Columbia LRA Amendment 12, dated October 18, 2010, the applicant revised LRA Section 4.2.4 to address the impact of the RV N12 beltline instrumentation nozzles on the current TS P-T limits that are valid for 33.1 EFPY. The applicant stated in this revision to LRA Section 4.2.4 that the current TS P-T limit curves were reviewed in 2009 to ensure that the N12 instrumentation nozzles did not impact the existing curves for 33.1 EFPY of facility operation. The applicant also added a statement in this LRA revision indicating that future P-T limit curves will be developed taking into consideration all RV beltline plates, welds, and nozzles, including the N12 instrumentation nozzles, the N12 nozzle-to-RV welds, and the N6 RHR/LPCI nozzle-to-RV welds. The staff agreed with the applicant's statement that future P-T limits must account for the irradiated properties of all RV beltline components, including the N12 instrumentation nozzles and all RV beltline nozzle-to-RV welds.

However, the staff found that, based on a 54 EFPY ART value of 149 ºF, the N12 nozzle forgings from Heat No. 219972 are the limiting RV beltline material, with respect to the 54 EFPY ART value. Furthermore, the staff determined that for all N12 nozzle fluence exposure levels greater than 1×10^{17} n/cm^2 (E > 1.0 MeV) the N12 nozzles are the limiting RV beltline material, with respect to ART, principally due to high assumed values of copper and nickel content in the nozzle, which are consistent with RG 1.99, Revision 2. Therefore, the staff identified a concern regarding the impact of N12 nozzles on the current P-T limit curves and the ability of the applicant to effectively manage the P-T limit curves during the period of extended operation, given that the current approved curves, which were calculated in NEDO-33144, did not account for the N12 nozzles' limiting ART for 33.1 EFPY.

In a teleconference discussion with the applicant and GE, the applicant stated that their 2009 determination that the N12 nozzle forgings would remain bounded by the current TS P-T limit curves for 33.1 EFPY was specifically based on a June 2009 linear elastic fracture mechanics (LEFM) analysis of the N12 nozzle forgings. The LEFM analysis is documented in the staff approved GEH Report, NEDO-33178-A, "GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves," Appendix J, "Water Level Instrumentation Nozzle LEFM Evaluation," June 2009, which was reviewed and approved by the staff by letter dated April 27, 2009. The staff confirmed that the report documents a LEFM evaluation of the water level instrument nozzles in BWRs based on bounding assumptions for RV and water level instrument nozzle geometry, postulated flaw configuration, operating pressures, and thermal transients. The report documents calculations of Mode I applied stress intensity factors (K_I) due to pressure loads and thermal transients. The report calculates bounding "T-RT_{NDT}" values for the BWR water level instrument nozzle using the acceptance criteria for total applied K_I values (including safety factors) that are based on the lower bound of the static critical (or reference) stress intensity factor curve, as specified in the ASME Code, Section XI, Appendix G.

The staff noted that the results of the LEFM analysis documented in NEDO-33178-A, Appendix J could be used to calculate P-T limit curves specifically for Columbia's N12 instrument nozzles. However, in order to determine how these methods can be applied for determining P-T limits specifically for Columbia's N12 nozzles, the applicant should provide additional information concerning the plant-specific applicability of the postulated flaw configuration used for calculating the applied KI values, as described in the subject report. The staff specifically noted that the NEDO-33178-A, Appendix J analysis postulated a 2.276-inch deep flaw that originates at the blend radius of the instrument nozzle and extends through the nozzle into the adjacent RV shell plate. [This flaw depth is consistent with 10 CFR Part 50 Appendix G requirements for use of a 1/4T deep flaw]. The tip of the postulated flaw in this analysis is apparently located in the adjacent RV shell plate. Accordingly, Section 5.0 of NEDO-33178-A, Appendix J states that for BWR instrument nozzles located in the beltline region of the RV, "the ART from the adjacent RPV shell material is used to create a component-specific P-T curve."

By letter dated March 23, 2011, the staff issued RAI 4.2-3 to determine to plant-specific applicability of the NEDO-33178-A, Appendix J LEFM evaluation. This RAI requested the applicant to (a) state whether the 2.276-inch deep postulated flaw for the N12 nozzle extends into and terminates in the adjacent RV shell plate material and (b) identify the RV beltline shell plate material that surrounds the nozzle and the ART value used for determining the component-specific P-T limits for the N12 nozzle.

In it response dated April 22, 2011, the applicant stated that (a) the 2.276-inch deep postulated flaw does extend into and terminates in the adjacent reactor vessel shell plate material and (b) the limiting reactor vessel shell plate material adjacent to the N12 nozzle is Heat No. C1336-1 with 33.1 EFPY and 54 EFPY and the corresponding ART are 44°F and 58.2°F, respectively. This plate heat is addressed in the LRA Table 4.2-5 ART calculations.

The staff reviewed the applicant's response and finds it acceptable because, for postulated cracks originating at the inside corner of the nozzle, the tip of the crack terminates in the surrounding plate. Further , this plate's ART values at 33.1 EFPY and 54 EFPY, which would represent the N12 nozzle component-specific P-T limit curves (based on a 1/4T deep flaw terminating in the plate), will remain well bounded by the P-T limit curves for the limiting RV shell plate material, Lower Shell Heat No. C1272-1, through 54 EFPY. The staff's concern described in RAI 4.2-3 is resolved.

In its review, the staff also considered the possible circumstances of a shallower flaw that is wholly contained within the nozzle baseline material. The staff also notes that, if such a crack was to propagate from the nozzle, it is expected that the high toughness nickel alloy weld would prevent the crack from traveling into the plate material.

License amendment requests to revise the TS P-T limits must be submitted to the NRC, pursuant to 10 CFR 50.90, for approval, to comply with 10 CFR Part 50, Appendix G. Future revisions to the TS P-T limit curves for plant operation beyond 33.1 EFPY must be approved by the NRC prior to the expiration of the current 40-year license term. The staff acknowledged the applicant's statement that future revisions to the Columbia TS P-T limits will be determined by the applicant, submitted to NRC, and implemented as part of Columbia's Reactor Vessel Surveillance Program.

Based on the applicant's January 20, 2011, and April 22, 2011, RAI responses on the N12 nozzle forgings, the N12 nozzle-to-RV welds, and the N6 nozzle-to-RV welds, the staff finds:

1) The N12 instrumentation nozzles do not need to be considered in the evaluation of the P-T limits due to the 1/4T deep flaw used for Appendix G of 10 CFR Part 50 P-T limit evaluations extending into the higher toughness adjacent plate material, and the presence of the nickel alloy weld that will likely prevent propagation of a postulated flaw into the RV plate metal.
2) The N12 nozzle-to-RV welds do not need to be considered in the evaluation of the P-T limits because it is fabricated of austenitic materials and not ferritic materials.
3) The N6 RHR/LPCI nozzle-to-RV welds do need to be considered in future ART and P-T limit evaluations. The staff believes that the current P-T limits are valid because the N6 RHR/LPCI nozzle-to-RV welds are currently bounded by the materials used to determine the current P-T limits.

The staff finds that the applicant's plan to manage the P-T limits is acceptable because changes to the P-T limit curves will be implemented by the license amendment process (i.e., through revisions of the plant TS) and will meet the requirements of 10 CFR 50.60 and 10 CFR Part 50, Appendix G. Future changes to the TS P-T limits will be managed by the applicant as part of the Columbia Reactor Vessel Surveillance Program, an existing Columbia Aging Management Program (AMP) described in LRA Section B.2.46.

4.2.4.3 UFSAR Supplement

LRA Section A.1.3.1.4 provides the UFSAR supplement for the P-T limits analysis TLAA evaluation.

The staff also noted that the applicant committed (Commitment No. 54) to:

> The Columbia P-T limit curves were revised in 2005 to include the effects of power uprate to 3486 MWt. The P-T limits are valid for 33.1 EFPY through the end of the currently licensed period. P-T limits for the period of extended operation will be calculated using the most accurate fluence projections available at the time of the recalculation. The projections may be adjusted if there are changes in core design or if additional surveillance capsule results show the

need for an adjustment. The projected ART for the period of extended operation gives confidence that future P-T curves will provide adequate operating margin. License amendment requests to revise the P-T limits will be submitted to the NRC for approval, when necessary to comply with 10 CFR 50 Appendix G, as part of the Reactor Vessel Surveillance Program.

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

4.2.5 Reactor Vessel Circumferential Weld Examination Relief

4.2.5.1 Summary of Technical Information in the Application

The BWRVIP-05, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations," (BWRVIP-05) report concludes that the conditional failure probability for RV circumferential welds is sufficiently low to justify the elimination of inservice inspections (ISIs) for the welds. In its July 28, 1998, SER for BWRVIP-05, the NRC agreed with these conclusions, provided that certain conditions pertaining to plant-specific circumferential weld conditional failure probabilities and the frequency of low temperature overpressure operating events are met. The NRC also concluded in this SER that individual plants must apply for an alternative to eliminate the ASME Code, Section XI requirements for RV circumferential weld examinations based on the satisfaction of these conditions. The request for an alternative to eliminate the ASME Code, Section XI requirements for these volumetric examinations should demonstrate that at the expiration of the 40-year license term, the RV circumferential welds satisfy the limiting conditional failure probability for circumferential welds specified in the SER for BWRVIP-05. This evaluation of circumferential weld mean adjusted reference temperature is a TLAA.

The applicant stated that its analysis of the conditional probability of failure for the Columbia RV circumferential welds for current 40-year licensed operating period is consistent with the position in the SER for BWRVIP-05 and NRC Generic Letter (GL) 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," November 10, 1998. By letter dated July 15, 2004, the applicant submitted a request for an alternative to the requirements of the ASME Code, Section XI in order to implement the provisions of BWRVIP-05 as the technical basis for the elimination of the required inservice inspections of the RV circumferential welds through the end of the current 40-year licensed operating term. In its June 1, 2005, SE for this request, the NRC concluded that the conditional probability of failure for the Columbia RV circumferential welds was sufficiently low to justify elimination of the volumetric examinations for these welds through 33.1 EFPY.

The NRC safety evaluation report (SER) for BWRVIP-74 determined that license renewal applicants for plants operating with NRC authorized relief from the ASME Code, Section XI RV circumferential weld examination requirements, based on the plant having satisfied the criteria specified in BWRVIP-05 SER and GL 98-05 for the current 40-year license term, shall demonstrate that these criteria will continue to be satisfied through the end of the extended license term (60 years), as part of a TLAA for RV circumferential weld inspection relief. In LRA Table 4.2-6 the applicant provided calculations for demonstrating that the Columbia RV circumferential weld parameters at 54 EFPY will remain within the NRC's (64 EFPY) bounding RV parameters from the BWRVIP-05 SER. As such, the applicant concluded that the conditional probability of failure for the circumferential welds will remain below that stated in the NRC's Final SER for BWRVIP-05 through the end of the period of extended operation (54 EFPY).

The applicant dispositioned the TLAA associated with RV circumferential weld inspection relief in accordance with 10 CFR 54.21(c)(1)(ii), that the analyses have been projected to the end of the period of extended operation.

4.2.5.2 Staff Evaluation

The staff reviewed LRA Section 4.2.5 and the TLAAs for the RV circumferential weld inspection relief to verify, pursuant to 10 CFR 54.21(c)(1)(ii), that the analyses have been projected to the end of the period of extended operation.

The technical basis for relief from the ASME Code, Section XI circumferential weld ISI requirements is discussed in the NRC staff's final SER concerning the BWRVIP-05 report, which is enclosed in a July 28, 1998, letter from Mr. G.C. Lanais, NRC, to Mr. C. Terry, the BWRVIP Chairman. In this letter, the staff concluded that, since the failure frequency for circumferential welds in BWR plants is significantly below the criterion specified in RG 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," and below the core damage frequency (CDF) of any BWR plant, the continued inspection would result in a negligible decrease in an already acceptably low RV failure probability. Therefore, elimination of the ISI requirements for RV circumferential welds is justified. The staff's letter indicated that BWR applicants may request relief from the ISI requirements of 10 CFR 50.55a(g) for volumetric examination of circumferential RV welds by demonstrating that (1) at the expiration of the license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds specified in the NRC staff's July 28, 1998 safety evaluation, and (2) the applicants have implemented operator training and established plant procedures that limit the frequency of cold overpressure events to the frequency specified in the staff's SER. The letter indicated that the requirements for inspection of RV circumferential welds during an additional 20-year license renewal period would need to be reassessed, on a plant-specific basis, as part of any BWR LRA. Furthermore, the applicant must request relief from the ISI requirements for volumetric examination of circumferential welds for the extended license term in accordance with the requirements of 10 CFR 50.55a(g).

In accordance with 10 CFR 50.55a(3)(iii), the NRC staff requires that a request for an alternative to the ASME Code, Section XI RV circumferential shell weld examination requirements be submitted for ISI intervals during the period of extended operation for plants who seek relief from these requirements during the extended license term.

By letter dated August 3, 2010, the staff issued RAI 4.2.5-1, requesting that the applicant indicate when it would apply for relief from the ASME Code, Section XI circumferential weld examination requirements for the extended period of operation.

In its response dated September 27, 2010, the applicant stated that it will submit the necessary request for RV circumferential weld examination relief for each ISI interval within 12 months after the completion of the previous ISI interval, as required by 10 CFR 50.55a(g). That staff finds the applicant's response to RAI 4.2.5-1 acceptable because the applicant stated that Columbia will apply for relief from (as an alternative to) the ASME Code, Section XI requirements for the RV circumferential weld examinations during the extended license term, in accordance with 10 CFR 50.55a requirements. The staff's concern described in RAI 4.2.5-1 is resolved.

Section A.4.5 of the BWRVIP-74-A report indicates that the staff's SER of the BWRVIP-05 report conservatively evaluated BWR RVs to 64 EFPY, which is 10 EFPY greater than what is realistically expected for the end of the license renewal period. Consequently, the BWRVIP-74-A report states that license renewal applicants operating with RV circumferential weld inspection relief for the current license term may use the results of staff's SER for the BWRVIP-05 report to evaluate the RV circumferential weld properties for the license renewal period as part of the RV circumferential weld inspection relief TLAA. The NRC staff used the mean RT_{NDT} value to evaluate the failure probability of BWR circumferential welds at 32 and 64 EFPY in the staff SER on the BWRVIP-05 report, dated July 28, 1998. The neutron fluence used in this evaluation was the neutron fluence at the RV inner diameter clad-weld interface.

Since the staff analysis discussed in the BWRVIP-05 report is a generic analysis, the applicant submitted plant-specific information to demonstrate that the limiting Columbia RV circumferential weld will meet the criteria specified in the report through the end of the period of extended operation. In order to demonstrate that the Columbia RV circumferential welds will not undergo neutron embrittlement beyond the basis for inspection relief (currently authorized for the 40-year license term) through the end of period of extended operation, LRA Table 4.2-6, shows a comparison of 54 EFPY material data for the Columbia limiting RV circumferential weld with that of the 64 EFPY reference case in Table 2.6-5 of the SER for the BWRVIP-05 report. In LRA Table 4.2-6 the applicant listed the copper and nickel content, the CF value, the 54 EFPY neutron fluence at the RV inner diameter clad-weld interface, the initial RT_{NDT} value; and the calculated ΔRT_{NDT} and mean RT_{NDT} values for the limiting circumferential weld at the end of the period of extended operation. The staff verified the validity of the data for the copper and nickel contents and the initial RT_{NDT} values for the Columbia RV beltline materials based on the evaluation in Section 4.2.3 of this SER. The applicant's calculated 54 EFPY mean RT_{NDT} value for the limiting RV beltline circumferential weld at Columbia is -6°F. The staff confirmed the applicant's calculation for the 54 EFPY mean RT_{NDT} value for the limiting RV circumferential weld was accurate. This 54 EFPY mean RT_{NDT} value for the limiting Columbia RV circumferential weld is less than the 64 EFPY mean RT_{NDT} value of 70.6°F used by the NRC for determining an acceptably low value for the conditional failure probability of a circumferential weld ($P(F|E) = 1.78 \times 10^{-5}$ per low-temperature overpressurization event). The 64 EFPY mean RT_{NDT} value of 70.6 ºF from Table 2.6-5 of the staff SER dated July 28, 1998, is representative of the circumferential welds for RV's fabricated by Chicago Bridge and Iron (CB&I), the RV supplier for Columbia. Since the Columbia 54 EFPY mean RT_{NDT} value is less than the applicable 64 EFPY mean RT_{NDT} value from the staff SER dated July 28, 1998, the staff concludes that the Columbia RV circumferential weld conditional failure probability will remain bounded by the NRC analysis through the end of the period of extended operation.

In the July 28, 1998, SER for BWRVIP-05, the NRC staff concluded that inservice inspections of the RV circumferential shell welds would need to be performed in accordance with ASME Code, Section XI requirements, if the volumetric examinations of the RV axial shell welds revealed the presence of an age-related degradation mechanism.

By letter dated August 3, 2010, the staff issued RAI 4.2.5-2, requesting that the applicant state whether or not previous volumetric examinations of the RV axial shell welds have shown any indication of age-related degradation in the welds.

In its response dated September 27, 2010, the applicant stated that previous examinations of the Columbia RV axial shell welds have not identified any age-related degradation in the welds. The applicant referenced the RAI responses for LRA Section 4.7.1 for a discussion of indications previously discovered in the RV axial shell welds and screened in accordance with the ASME Code, Section XI, Article IWB-3500.

That staff found the applicant’s response to RAI 4.2.5-2 acceptable because the applicant confirmed that previous examinations of the RV axial shell welds identified no evidence of age-related degradation. The staff’s evaluation of the applicant’s TLAA of the relevant indications found in the RV shell welds, including discussion of why these indications are not caused by age-related degradation, is discussed in SER Section 4.7.1. The staff's concern described in RAI 4.2.5-2 is resolved.

BWRVIP-74-A, Section A.4.5, “Circumferential Weld Inspection Relief,” states that in order to obtain relief from circumferential weld examination requirements, each applicant must submit a plant-specific relief request. In that submittal, applicants have to demonstrate that (1) at the expiration of the license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds specified in the July 28, 1998, SER for BWRVIP-05, and (2) the applicants have implemented operator training and established procedures that limit the frequency of cold overpressure events to the frequency specified in this SER. LRA Section 4.2.5 addressed condition (1) for this TLAA. However, LRA Section 4.2.5 did not address condition (2).

By letter dated August 3, 2010, the staff issued RAI 4.2.5-3, requesting that the applicant address condition (2), as it relates to the proposed extended period of operation.

In its response dated September 27, 2010, the applicant stated that the procedures and training used to limit low temperature overpressure events will be the same as those approved by the NRC when Columbia requested relief from the ASME Code, Section XI requirements for RV circumferential weld inspections for the current license period, in accordance with BWRVIP-05. The staff found the applicant’s RAI response acceptable because the procedures and training currently used by the applicant to limit low temperature overpressure events will continue to be used during the period of extended operation, in accordance with BWRVIP-05 criteria. The staff's concern described in RAI 4.2.5-3 is resolved.

Based on the above, the staff determined that the applicant’s evaluation for this TLAA is acceptable because the 54 EFPY conditional failure probability for the Columbia RV circumferential welds will remain bounded by the NRC analysis in the staff’s SER dated July 28, 1998, and the applicant will be using procedures and training to limit cold overpressure events during the period of extended operation. This analysis is consistent with the evaluation criteria in the staff’s SER for BWRVIP-05; however, the applicant is still required to request relief from RV circumferential weld examination requirements for ISI intervals over the extended period of operation, in accordance with 10 CFR 50.55a.

4.2.5.3 UFSAR Supplement

LRA Section A.1.3.1.5 provides the UFSAR supplement for the RV circumferential weld inspection relief analysis TLAA evaluation. Based on its review of the UFSAR supplement, the staff concludes that the information in the UFSAR 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.

4.2.5.4 Conclusion

On the basis of its review, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the effects of aging on the RV circumferential weld inspection relief analysis have been projected to the end of the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d), and, therefore, is acceptable.

4.2.6 Reactor Vessel Axial Weld Failure Probability

4.2.6.1 Summary of Technical Information in the Application

The SER for BWRVIP-74 evaluated the failure frequency of axially-oriented welds in BWR RVs and determined that this failure frequency is below 5.00 x 10^{-6} per reactor-year of operation, based on the limiting mean RT_{NDT} values for the Pilgrim and Clinton plants. Applicants for license renewal must evaluate axially-oriented RV welds to demonstrate the failure frequencies for the welds will remain below 5.00 x 10^{-6} per reactor-year of operation through the end of the period of extended operation. An acceptable method for this analysis is to demonstrate that the projected mean RT_{NDT} value for the plant's limiting RV axial weld will remain below the values specified in Table 1 of the BWRVIP-74 SER through the end of the period of extended operation. The mean RT_{NDT} value from Table 1 of the BWRVIP-74 SER that corresponds to an RV axial weld failure frequency of 5.00 x 10^{-6} per reactor-year of operation is 114 ºF for Pilgrim, a BWR Type 3 plant, with a RV manufactured by Combustion Engineering (CE). Columbia's RV was manufactured by CB&I. For the Clinton plant, where the RV was manufactured by CB&I, Table 1 of the BWRVIP-74 lists an axial weld mean RT_{NDT} value of 91 ºF, and a corresponding RV axial weld failure frequency of 2.73 x 10^{-6} per reactor-year of operation.

LRA Table 4.2-7 shows that the Columbia limiting axial weld mean RT_{NDT} value at 54 EFPY is 16.9 ºF. This value remains well below the bounding value of 114 ºF for Pilgrim and the value of 91 ºF for Clinton from the SER for BWRVIP-74. Therefore, the applicant states, the Columbia limiting axial weld failure frequency is well below the acceptable limit of 5.00 x 10^{-6} per reactor-year of operation.

The applicant dispositioned the TLAA associated with RV axial weld failure probability in accordance with 10 CFR 54.21(c)(1)(ii), that the analysis have been projected to the end of the period of extended operation.

4.2.6.2 Staff Evaluation

The staff reviewed LRA Section 4.2.6 and the TLAAs for the RV axial weld failure probability to verify, pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis have been projected to the end of the period of extended operation.

In its July 28, 1998, letter to Mr. C. Terry, the BWRVIP Chairman, the staff identified a concern regarding the failure frequency of axial welds in BWR RVs. In response to this concern, the BWRVIP supplied evaluations of axial weld failure frequency in letters dated December 15, 1998, and November 12, 1999. The staff's BWRVIP-05 supplemental SER on these analyses is enclosed in a March 7, 2000, letter from Mr. J. Strosnider (NRC) to Mr. C. Terry (BWRVIP). The staff performed a generic analysis of RV axial weld failure frequencies using Pilgrim and Clinton as models for BWR RVs manufactured by CE and CB&I, respectively. The analysis, which is also addressed in the BWRVIP-74 SER for use in axial weld failure probability TLAAs by license renewal applicants, demonstrated that for a variant of Pilgrim input data, a mean RT_{NDT} value of 114°F would result in a bounding axial weld failure frequency of 5.02 x 10^{-6} per reactor-year of operation. For the Clinton input data, the staff analysis demonstrated that a mean RT_{NDT} value of 91°F would result in an axial weld failure frequency of 2.73 x 10^{-6} per reactor-year of operation.

The applicant calculated, and the staff confirmed, that the limiting axial weld mean RT_{NDT} value for Columbia at 54 EFPY is 16.9°F, which supports the conclusion that the failure frequency for the Columbia RV axial welds will be less than 5 x 10^{-6} per reactor-year of operation at the end of the period of extended operation. Therefore, this analysis is acceptable.

The limiting axial weld failure probability calculated by the NRC staff in the March 7, 2000, BWRVIP-05 SER supplement is based on the assumption that "essentially 100 percent" (i.e., greater than 90 percent) examination coverage of all RV axial welds can be achieved in accordance with 10 CFR 50.55a requirements.

By letter dated August 3, 2010, the staff issued RAI 4.2.6-1, requesting that the applicant indicate whether Columbia's inservice examinations achieve "essentially 100 percent" (i.e., greater than 90 percent) overall examination coverage for the RV axial welds for the duration of the current licensed operating period. If less than 90 percent overall examination coverage is achieved for the RV axial welds, the staff requested that the applicant revise their TLAA of the RV axial welds to account for the effects of the limited scope examination coverage.

In its response dated September 27, 2010, the applicant stated that "essentially 100 percent" of the RV beltline axial welds are inspected at Columbia in accordance with 10 CFR 50.55a requirements. During the previous (second) ISI interval, the applicant achieved greater than 90 percent coverage for each of the RV axial welds. According to the applicant, no axial welds have yet been inspected during the current (third) ISI interval at Columbia; the RV axial welds are scheduled to be examined during the last inspection period of the current interval. The applicant stated that the RV axial weld examinations for the third 10-year ISI interval will achieve greater than 90 percent coverage for each of the axial welds. The staff found the applicant's RAI response acceptable because the applicant confirmed that inservice examinations at Columbia achieve "essentially 100 percent" (i.e., greater than 90 percent) overall examination coverage of the RV axial welds for the current licensed operating period. The staff's concern described in RAI 4.2.6-1 is resolved.

By letter dated August 3, 2010, the staff issued RAI 4.2.6-2, requesting that the applicant state whether inservice examinations of the RV axial welds cover all of the intersections with the RV circumferential welds.

In its response dated September 27, 2010, the applicant stated that all the intersections of the RV axial welds and circumferential welds are inspected, resulting in examination coverage of approximately two to three percent of the circumferential welds in the region of intersection with the axial welds. The staff found the applicant's RAI response acceptable because the applicant confirmed that all the intersections of the RV axial welds and circumferential welds are inspected at Columbia. The staff's concern described in RAI 4.2.6-2 is resolved.

The staff determined that the applicant's evaluation for this TLAA is acceptable because the Columbia 54 EFPY RV axial weld failure probability is bounded by the NRC analysis in the BWRVIP-74 SER, and the March 7, 2000, supplemental SER for BWRVIP-05.

4.2.6.3 UFSAR Supplement

LRA Section A.1.3.1.6 provides the UFSAR supplement for the RV axial weld failure probability analysis TLAA evaluation. Based on its review of the UFSAR supplement, the staff concludes that the information in the UFSAR 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.

4.2.6.4 Conclusion

On the basis of its review, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the effects of aging on the RV axial weld failure probability analysis have been projected to the end of the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d), and, therefore, is acceptable.

4.3 Metal Fatigue

LRA Section 4.3 provides the assessment of metal fatigue analyses in the CLB that are considered TLAAs for license renewal. The applicant divided this section into the following subsections:

- LRA Section 4.3.1, “Reactor Pressure Vessel Fatigue Analyses”
- LRA Section 4.3.2, “Reactor Vessel Internals”
- LRA Section 4.3.3, “Reactor Coolant Pressure Boundary Piping and Piping Component Fatigue Analyses”
- LRA Section 4.3.4, “Non-Class 1 Component Fatigue Analyses”
- LRA Section 4.3.5, “Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping”

LRA Table 4.3-1 provides the design cycles from the stress reports for the Class 1 components. The LRA states that the same information is provided in UFSAR Section 3.9 and UFSAR Table 3.9-1. The applicant stated that it counts all fatigue significant cycles using the Fatigue Monitoring Program (FMP), not only those associated with fatigue analyses of RPV and RCPB components but also with analyses of other plant components. The applicant added that additional transients, determined to be fatigue significant after the original design, have been added to the FMP. LRA Table 4.3-2 lists the projected number of design transient occurrences, which used a linear extrapolation from the beginning of plant life. The applicant further stated that it manages fatigue using the FMP to track transient cycles and requires corrective action before any analyzed number of cycles is reached.

4.3.1 Reactor Pressure Vessel Fatigue Analyses

4.3.1.1 Summary of Technical Information in the Application

LRA Section 4.3.1 describes the applicant's TLAAs for the RPV. These TLAAs are based on the CUF analyses in the applicant's current design for the RPV assembly, which consists of the pressure vessel, vessel support skirt, shroud support, nozzles, penetrations, stub tubes, head closure flanges, head closure studs, refueling bellows support, and stabilizer brackets. LRA Table 4.3-3 summarizes the design CUFs for the limiting RPV assembly locations, which were obtained from the original design reports. These CUFs were calculated based on the design transients listed in LRA Table 4.3-2.

The applicant stated that it manages fatigue using the FMP to track transient cycles, and it requires corrective action before any analyzed number of cycles is reached. The applicant dispositioned the TLAAs for the RPV assembly 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.

4.3.1.2 Staff Evaluation

The staff reviewed LRA Section 4.3.1 and the TLAAs for the RPV 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 TLAAs and the corresponding disposition consistent with the review procedures in SRP-LR Section 4.3.3.1.1.3. The SRP-LR states that the reviewer should verify that the applicant identified the appropriate program, as described and evaluated in the GALL Report, for monitoring and tracking the number of critical thermal and pressure transients for the selected RCS components. Furthermore, the reviewer should also ensure that the applicant has stated that its program contains the same program elements evaluated by the staff and relied upon in approving the corresponding generic program in the GALL Report.

The applicant stated in LRA Section 4.3.1 that faulted conditions listed in the UFSAR are not used in the fatigue analyses and are not counted. The staff finds its acceptable that faulted conditions are not counted by the applicant's Fatigue Monitoring Program because these transients are not used in the design basis fatigue calculations and consideration of the incremental fatigue due to these transients is not required by ASME Code, Section III.

The applicant stated that it had determined from the 60-year projected cycles for its design transients, as shown in Table 4.3-2, determined that the analyzed numbers of transients may be exceeded for some transients. The applicant explained that these projections were determined by using a linear extrapolation from the beginning of plant life and that recent operating experience suggests lower projections. 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 not only the time period with the recent improved operating history. The staff also noted that LRA Table 4.3-2 also provides the number of cycles that will be used for any future fatigue analyses, including its environmental fatigue analysis and that the applicant's Fatigue Monitoring Program tracks the number of transient cycles that occur at its site and requires corrective action before any analyzed number of cycles is reached. The staff noted that, as long as the number of cycles that occur at the plant does not exceed the number of cycles assumed in the applicant's fatigue calculations for each component location, the fatigue calculations will remain valid and the design limit will not be exceeded. The staff's evaluation of the applicant's Fatigue Monitoring Program is documented in SER Section 3.0.3.2.7, which determined that the program is acceptable because systematically counts transient cycles to ensure that the numbers of analyzed cycles in the calculation for each component location are not exceeded, thereby ensuring that component fatigue usage limits are not exceeded and is consistent with the recommendations in GALL AMP X.M1.

Based on its review, the staff finds the applicant has demonstrated that its methodology for projecting design transients to the end of the period of extended operation is conservative and that the applicant will monitor those 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 for any component location, consistent with GALL AMP X.M1.

The staff noted that UFSAR Section 3.9.1.1.1 provides the design basis transients and their associated limits that are applicable to the CRDs, and UFSAR Section 3.9.1.1.2 provides the design transients and their associated limits that are applicable to the CRD housings and incore housings. However, in its review, the staff found out that LRA Section 4.3 did not provide information for the transients and design limits that are applicable to the CRD and to the CRD housings and incore housings. By letter dated August 26, 2010, the staff issued RAI 4.3-04, requesting that the applicant provide the basis for not including design basis transient cycle and 60-year projected cycles for the CRDs, CRD housings and incore housings that are based on the design basis transients and design limits in UFSAR Sections 3.9.1.1.1 and 3.9.1.1.2.

In its response dated November 11, 2010, the applicant stated that the CUFs related to the CRDs are 0.083 for the stub tube and 0.196 for the housing, as listed in LRA Table 4.3-3. The applicant added that these CUFs are from the original equipment manufacturer (OEM) stress report and are based on the design transients in LRA Table 4.3-1 (UFSAR Table 3.9-1). The applicant also stated that the design basis transient cycles listed in UFSAR Section 3.9.1.1.1 were used by the OEM for a generic analysis of the CRDs, and those listed in UFSAR Section 3.9.1.1.2 were used by the OEM for a generic analysis of the incore housings. These analyses are not plant-specific analyses and, therefore, are not considered TLAAs. As such, the cycles in UFSAR Sections 3.9.1.1.1 and 3.9.1.1.2 do not require extrapolation to 60 years. The staff noted that the applicant's response is associated with RAI 4.3-02 and follow-up RAI 4.3-02, which is discussed below. The staff's evaluation of these three RAIs will be discussed together.

The staff noted that the CUF values for the CRD housings and CRD stub tubes are included in the fatigue analyses of the RPV components, and these values are listed in LRA Table 4.3-3. However, LRA Tables 4.3-3 and 4.3-4 do not provide a CUF value for the incore housing penetrations. By letter dated August 26, 2010, the staff issued RAI 4.3-02, asking the applicant to clarify if the CUF value for stub tubes listed in LRA Table 4.3-3 is the CUF value that is identified for the incore housings. If the CUF value in LRA Table 4.3-3 for the stub tubes is not the CUF value for the incore housings, the staff asked that the applicant identify the CUF value of record for the incore housing penetrations and reference the design basis document that provides the design CUF value for this component.

In its response dated November 11, 2010, the applicant stated that the CUF value listed in LRA Table 4.3-3 for the CRD stub tube is not the CUF value for the incore housings. The CUF for the incore housings was not listed in Table 4.3-3 because it was not considered to be a TLAA. The applicant added that the OEM stress report for the RV calculated a CUF for the CRD penetrations but did not include the incore housing penetration, and these penetrations were evaluated in a generic stress report that is not part of the Columbia CLB. Since this is a generic analysis and not a plant-specific analysis, the applicant does not consider this a CUF of record and, therefore, it is not a TLAA. The applicant also stated that it listed the generic incore penetration CUF analysis in earlier versions of the basis documents upon which the LRA was based but deleted it because it was not a plant-specific analysis cited in the Columbia CLB. The staff noted that the reference to the CUF for the incore housing penetrations was not deleted from LRA Appendix C, Table C-8. Therefore, it is not clear to the staff why it was deleted from the basis documents since the applicant did not provide a justification or technical basis for this action. By letter dated December 2, 2010, the staff issued followup RAI 4.3-02, asking the applicant to either provide a technical basis why the analysis does not conform to the definition of a TLAA or provide the reference of the fatigue CUF analysis and resultant CUF values for the incore housing penetrations.

In its response dated January 20, 2010, the applicant explained that the generic analysis of the incore housing penetration was found in the initial search for license renewal basis documents. The applicant stated that, during a subsequent review of the license renewal basis document, it was determined that the generic analysis was not part of a plant design basis document and the results of that analysis were deleted from the license renewal basis document. The applicant also stated that the CUF for the incore housing penetrations is not contained or incorporated by reference in its current licensing basis, which includes its UFSAR and any document docketed by Energy Northwest. Therefore, the applicant concluded that the generic incore housing penetration fatigue analysis is not a TLAA because it does not satisfy Criterion 4 of 10 CFR 54.3 (that the analysis was determined to be relevant by the applicant in making a safety determination) or Criterion 6 of 10 CFR 54.3 (that the analysis is contained or incorporated by reference in the CLB).

Based on its review, the staff finds the applicant’s responses to RAI 4.3-02, followup RAI 4.3-02 and RAI 4.3-04 acceptable for the following reasons:

- The applicant clarified that the transient cycles listed in UFSAR Sections 3.9.1.1.1 and 3.9.1.1.2 were used by the OEM for a generic analysis of the CRDs and incore housings.
- The applicant clarified that the generic fatigue analysis for the CRDs and incore housing penetration are not contained or incorporated by reference into the applicant's CLB.
- The generic fatigue analysis is not a TLAA in accordance with Criterion 4 and Criterion 6 of 10 CFR 54.3(a).

The staff’s concerns in RAI 4.3-02, follow up RAI 4.3-02 and RAI 4.3-04 are resolved.

The staff noted that the CUF value listed for feedwater (FW) nozzle safe end in LRA Table 4.3-3 is 0.696 while the UFSAR Table 3.9-2a lists the value as 0.966. It is not clear to the staff if the CUF of record have been reanalyzed and, if so, if a lower CUF was obtained by decreasing the projected number of load cycles or decreasing the severity of the transient or both. Also, if the severity of the transient were decreased, it is not clear if the revised transients were verified for plant-specific stress-based fatigue monitoring. By letter dated August 26, 2010, the staff issued RAI 4.3-01 requesting that the applicant justify why LRA Table 4.3-3 lists a different CUF value for the FW nozzle safe end from UFSAR Table 3.9-2a. If the CUF value listed for the FW nozzle in LRA Table 4.3-3 represents the most updated CUF value, the staff asked the applicant to reference the document in the CLB that provides the CUF of record.

In its response dated November 11, 2010, the applicant stated that the FW nozzle safe end CUF in UFSAR Table 3.9-2a was corrected in Amendment 61 of the UFSAR to 0.696 and to match the value in LRA Table 4.3-3.

Based on its review, the staff finds the applicant's response to RAI 4.3-01 acceptable because the applicant clarified that the CUF value of 0.696 for the FW nozzle safe end is correctly documented in the applicant's Amendment 61 of the UFSAR and is consistent with the CUF value in LRA Table 4.3-3. The staff's concern described in RAI 4.3-01 is resolved.

The staff noted that the applicant's Fatigue Monitoring Program includes an enhancement to correlate information relative to fatigue monitoring and provide more definitive verification that the transients monitored and their limits are consistent with or bound the FSAR and the supporting fatigue analyses, including the EAF analyses. The applicant committed (Commitment No. 24) to implement this enhancement prior to the period of extended operation. The staff noted that the implementation of this enhancement will ensure that actions are taken prior to the design code limit of 1.0 being exceeded or prior to the analyzed cycles in the fatigue analysis being exceeded. The staff's review of this enhancement and the applicant's Fatigue Monitoring Program is documented in SER Section 3.0.3.2.7.

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 RPV components will be adequately managed for the period of extended operation. Additionally, the TLAAs associated with metal fatigue of the RPV meets the acceptance criteria in SRP-LR Section 4.3.2.1.2.3 because the applicant's FMP tracks the number of transient cycles that occur and requires corrective actions to be taken prior to any analyzed number of cycles in the TLAA being reached, which ensures that the analyses remain valid.

4.3.1.3 UFSAR Supplement

LRA Section A.1.3.2.1 provides an UFSAR supplement summarizing the TLAAs associated with metal fatigue of the RPV. The staff reviewed LRA Section A.1.3.2.1, consistent with the review procedures in SRP-LR Section 4.3.3.3, which states that the reviewer should verify that the applicant provided information, to be included in the UFSAR supplement, which 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.

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

4.3.2 Reactor Vessel Internals

4.3.2.1 Summary of Technical Information in the Application

LRA Section 4.3.2 describes the applicant's TLAAs for the overall RV internals, based on the CUF analyses of the overall reactor core support structures and RV internals performed as part of the applicant's current design, as well as fatigue analyses of the jet pumps performed in response to operating conditions. The application states that core support structures include the shroud, shroud support (included as part of the RV for fatigue), core plate with wedges and hold-down bolts, top guide, fuel supports, and control rod guide tubes. The RV internals include the following:

- jet pump assemblies
- jet pump instrumentation
- FW spargers
- vessel head spray line
- differential pressure line
- incore flux monitor guide tubes
- initial startup neutron sources (removed)
- surveillance sample holders
- core spray lines (in-vessel) and spargers
- incore instrument housings
- LPCI coupling
- steam dryer
- shroud head and steam separator assembly
- guide rods
- CRD thermal sleeves

Design CUF values for the limiting reactor vessel internals locations are obtained from design reports and are summaried in LRA Table 4.3-4. These CUFs were calculated based on the design transients listed in LRA Table 4.3-2.

Regarding the jet pumps, LRA Section 4.3.2.2 states that the effect of the flow imbalance resulted in a 0.0035 increase in the CUF for the plant's jet pumps, and inspections of the jet pumps in 2001 identified gaps in the jet pump set screws. The LRA states that a fatigue analysis of the jet pump risers, as of end of cycle 16, indicated an additional 0.119 increase in the CUF value for risers 1/2 and 5/6 due to the gaps in the component configuration. Additionally, in 2005, the applicant installed clamps on the jet pump mixer and diffuser areas in order to minimize FIVs caused by leakage at the mixer-to-diffuser slip joint interface. The LRA identifies that the maximum 60-year CUF for any jet pump riser is 0.920.

The applicant stated that it manages fatigue using the FMP to track transient cycles and requires corrective action before any analyzed number of cycles is reached. The applicant dispositioned the TLAAs for the RV internals 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.

4.3.2.2 Staff Evaluation

The staff reviewed LRA Section 4.3.2 and the TLAAs for the RV internals and jet jump assemblies 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 TLAAs and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.3.3.1.1.3. The SRP-LR states that the reviewer should verify that the applicant identified the appropriate program, as described and evaluated in the GALL Report, for monitoring and tracking the number of critical thermal and pressure transients for the selected RCS components. Furthermore, the reviewer should also ensure that the applicant has stated that its program contains the same program elements evaluated by the staff and relied upon in approving the corresponding generic program in the GALL Report.

In its review of LRA Section 4.3.2.1, the staff noted that the design cumulative usage factors (CUF) for the limiting reactor vessel internals locations were obtained from design reports and are summarized in LRA Table 4.3-4. The staff reviewed the design CUFs for the applicant's core support structures and reactor internals and noted that it was all less than the design limit of 1.0. The applicant stated that it will manage the effects of cumulative fatigue damage by using the Fatigue Monitoring Program to track transient cycles and require corrective action before any analyzed number of cycles is reached. The staff noted that as long as the number of cycles that occur at the plant does not exceed the number of cycles assumed in the applicant's fatigue calculations, the fatigue calculations will remain valid and the design limit will not be exceeded. The staff's evaluation of the acceptability of the applicant's Fatigue Monitoring Program is documented in SER Section 3.0.3.2.7.

In its review of LRA Section 4.3.2.2, the staff noted that, in August 2000, the applicant operated for a period of time with the recirculation pumps in an unbalanced mode (i.e., the running speeds for the pumps differed by more than 50 percent). LRA Section 4.3.2.2 states that the effect of the flow imbalance resulted in a 0.0035 increase in the CUF for the plant's jet pumps, and inspections of the jet pumps in 2001 identified gaps in the jet pump set screws. A fatigue analysis of the jet pump risers was done at that time to justify operation through the end of cycle 16, indicated an additional 0.119 increase in the CUF value for risers 1/2 and 5/6 due to the gaps in the component configuration. Additionally, in 2005, the applicant installed clamps on the jet pump mixer and diffuser areas in order to minimize FIVs caused by leakage at the mixer-to-diffuser slip joint interface. The staff noted that, in LRA Section 4.3.2.2, the applicant credits its FMP to disposition these TLAAs in accordance with 10 CFR 54.21(c)(1)(iii). However, the staff noted that the BWR Vessel and Internals Program is used to inspect for cracking and gaps (changes in configuration) in applicable jet pump assembly components.

By letter dated August 26, 2010, the staff issued RAI 4.3-03 requesting that the applicant provide the basis for using the FMP to disposition the TLAA for the jet pump assembly components. The staff also asked the applicant to explain why it would not be more appropriate to credit the inspections of the BWR Vessel and Internals Program for these components to manage cumulative fatigue damage.

In its response dated November 11, 2010, the applicant stated that the jet pump fatigue analysis depends on two components — the number of thermal cycles incurred and the jet pump gaps. As described in LRA Section 4.3.2.2, the majority of the fatigue usage comes from transient cycles rather than jet pump gaps. For example, the CUF for risers 1/2 and 5/6 consists of 0.75 due to transients and only 0.12 due to gaps; for the other eight risers, there is no contribution from gaps. The applicant further stated that the basis for managing the fatigue usage of the jet pumps by the FMP is that it not only counts cycles but also incorporates the BWR Vessel and Internals Program to ensure that the jet pump gaps remain below the analyzed gap and, thereby, do not contribute to any fatigue usage. The staff noted that the applicant's FMP credits the use of the BWR Vessel Internal Program to manage fatigue of the jet pumps by checking the jet pump set screw gaps during each outage and, if any out-of-specification gaps are found the applicant will calculate the additional fatigue accumulated by the jet pumps due to those gaps. By letter dated August 10, 2011, the applicant amended LRA Section A.1.3.2.2 to clarify that the FMP credits the BWR Vessel and Internals Program to montior the jet pump gaps and that the actions from both programs will manage fatigue of the jet pumps through the period of extended operation.

Based on its review, the staff finds the applicant's response to RAI 4.3-03 acceptable for the following reasons:

- The applicant is using its FMP to ensure that the number of transient cycles incurred by the jet pump assemblies does not exceed the number of cycles assumed in the analysis.
- The applicant is using its BWR Vessel and Internals Program to confirm the gap in the jet pump risers is less than the analyzed gap.

Both activities taken by the applicant ensure that the assumptions in the analysis remain valid. The staff's concern described in RAI 4.3-03 is resolved.

The staff noted that the applicant's Fatigue Monitoring Program includes an enhancement to correlate information relative to fatigue monitoring and provide more definitive verification that the transients monitored and their limits are consistent with or bound the FSAR and the supporting fatigue analyses, including the EAF analyses. The applicant committed (Commitment No. 24) to implement this enhancement prior to the period of extended operation. The staff noted that the implementation of this enhancement will ensure that actions are taken prior to the design code limit of 1.0 being exceeded or prior to the analyzed cycles in the fatigue analysis being exceeded. The staff's review of this enhancement and the applicant's Fatigue Monitoring Program is documented in SER Section 3.0.3.2.7.

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 RV internal and jet pump assembly components will be adequately managed for the period of extended operation. Additionally, the TLAAs associated with metal fatigue of the RV internals and jet pump assembly meet the acceptance criteria in SRP-LR Section 4.3.2.1.2.3 because the applicant's FMP tracks the number of transient cycles that occur and requires corrective actions to be taken prior to any analyzed number of cycles in the TLAA being reached, which ensures that the analyses remain valid and the design limit of 1.0 is not exceeded.

4.3.2.3 UFSAR Supplement

LRA Section A.1.3.2.2, as amended by letter dated August 10, 2011, provides an UFSAR supplement summarizing the TLAAs associated with metal fatigue of the RV internals and jet pump assembly. The amended LRA Section A.1.3.2.2 clarifies that the FMP credits the BWR Vessel and Internals Program to montior the jet pump gaps and that the actions from both programs will manage fatigue of the jet pumps through the period of extended operation. The staff reviewed LRA Section A.1.3.2.2, consistent with the review procedures in SRP-LR Section 4.3.3.3, which states that the reviewer should verify that the applicant provided information, to be included in the UFSAR supplement, which 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.

Based on its review of the UFSAR 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 effect of reactor coolant environment on fatigue usage, as required by 10 CFR 54.21(d).

4.3.2.4 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)(iii), that the effects of aging related to fatigue analyses of the RV internals and jet pump assemblies will be adequately managed for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.3 Reactor Coolant Pressure Boundary Piping and Component Fatigue Analyses

4.3.3.1 Summary of Technical Information in the Application

LRA Section 4.3.3 describes the TLAAs for the RCPB piping and piping components. These TLAAs are based on the applicant’s current design basis CUF calculations for Class 1 piping (pipe and fittings) and in-line components subject to ASME Section XI, Subsection IWB, inspection requirements. The applicant stated that these components are designed in compliance with ASME Section III, Subsection NB-3600 (and NC-3600 for piping less than or equal to 1 in. diameter). In addition, the applicant stated that all Class 1 piping was reviewed for the power uprate and the evaluation scaled then existing fatigue analyses based on the changes in stress expected from the power uprate. This evaluation showed that there was adequate margin in each system to accommodate the power uprate (the increased CUF after the power uprate was approximated by the report) and the maximum CUFs for Class 1 piping are shown in LRA Table 4.3-5. The information in this table provides the design fatigue usage for 40 years of operation for the limiting reactor coolant pressure boundary components.

The applicant stated that, as indicated in UFSAR Section 3.6.2, potential intermediate HELB locations have been eliminated based on CUFs being less than 0.1 and other stress criteria being satisfied. The applicant stated that it uses FMP to track the number transients that occur, and the program will identify when the transients for piping systems are approaching their analyzed numbers of design cycles. Therefore, prior to any transient exceeding the analyzed number of cycles, the design calculations for the piping system will be reviewed to determine if any additional locations should be designated as postulated HELBs, under the original criteria of UFSAR Section 3.6.

The applicant stated that, during initial plant startup, an induction heating stress improvement (IHSI) process was used on various RPV nozzles to safe end welds and safe end to pipe welds. In the 1994 refueling outage, it also performed a mechanical stress improvement process (MSIP) for multiple RPV nozzles to safe end welds and safe end to pipe welds. The applicant stated that no credit is taken for MSIP or IHSI in the calculation of CUFs for the vessel nozzles and safe ends.

The applicant stated that it manages fatigue using the FMP to track transient cycles and requires corrective action before any analyzed number of cycles is reached. The applicant dispositioned the TLAAs for all Class 1 RCPB piping and in-line components 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.

4.3.3.2 Staff Evaluation

The staff reviewed LRA Section 4.3.3 and the TLAAs for the RCPB piping and components 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 TLAAs and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.3.3.1.1.3. The SRP-LR states that the reviewer should verify that the applicant identified the appropriate program, as described and evaluated in the GALL Report, for monitoring and tracking the number of critical thermal and pressure transients for the selected RCS components. Furthermore, the reviewer should also ensure that the applicant has stated that its program contains the same program elements evaluated by the staff and relied upon in approving the corresponding generic program in the GALL Report.

LRA Section 4.3.3 states that the RCPB piping has been designed in accordance with ASME Section III, Subsection NB-3600, which requires that the CUF for the RCPB components be less than 1.0. The staff noted that the applicant's criteria for identification of postulated high-energy line breaks (HELBs) (i.e., locations with high CUF values) in RCPB piping and in-line components are described in UFSAR Section 3.6. The applicant stated that, as indicated in UFSAR Section 3.6.2, potential intermediate HELB locations have been eliminated based on CUFs being less than 0.1 and other stress criteria being satisfied. The applicant uses its FMP to track the number of transients that occur, and the program will identify when the transients for piping systems are approaching their analyzed numbers of design cycles. The staff finds the use of the FMP reasonable because it may identify other locations that require consideration for postulated HELBs, such that actions will be taken to address the fatigue CUFs for any new break locations.

The applicant stated that the design CUF for the reactor coolant pressure boundary piping and in-line components are summarized in LRA Table 4.3-5. The staff reviewed the design CUFs and noted that it was all less than the design limit of 1.0. The applicant stated that it will manage the effects of cumulative fatigue damage by using the Fatigue Monitoring Program to track transient cycles and require corrective action before any analyzed number of cycles is reached. The staff noted as long as the number of cycles that occur at the plant does not exceed the number of cycles assumed in each of the applicant's fatigue calculations, the fatigue calculations will remain valid and the design limit will not be exceeded. The staff's evaluation of the acceptability of the applicant's Fatigue Monitoring Program is documented in SER Section 3.0.3.2.7.

The staff noted that the applicant's Fatigue Monitoring Program includes an enhancement to correlate information relative to fatigue monitoring and provide more definitive verification that the transients monitored and their limits are consistent with or bound the FSAR and the supporting fatigue analyses, including the EAF analyses. The applicant committed (Commitment No. 24) to implement this enhancement prior to the period of extended operation. The staff noted that the implementation of this enhancement will ensure that actions are taken prior to the design code limit of 1.0 being exceeded or prior to the analyzed cycles in the fatigue analysis being exceeded. The staff's review of this enhancement and the applicant's Fatigue Monitoring Program is documented in SER Section 3.0.3.2.7.

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 RCPB piping and piping components will be adequately managed for the period of extended operation. Additionally, the TLAAs associated with metal fatigue of RCPB piping and piping components meets the acceptance criteria in SRP-LR Section 4.3.2.1.2.3 because the applicant's FMP tracks the number of transient cycles that occur and requires corrective actions to be taken prior to any analyzed number of cycles in the TLAA being reached, which ensures that the analyses remain valid and the design limit of 1.0 is not exceeded.

4.3.3.3 UFSAR Supplement

LRA Section A.1.3.2.3 provides an UFSAR supplement summarizing the TLAAs associated with metal fatigue of RCPB piping and piping components. The staff reviewed LRA Section A.1.3.2.3, consistent with the review procedures in SRP-LR Section 4.3.3.3, which states that the reviewer should verify that the applicant provided information, to be included in the UFSAR supplement, which 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.

Based on its review of the UFSAR 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 effect of reactor coolant environment on fatigue usage, as required by 10 CFR 54.21(d).

4.3.3.4 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 related to fatigue analyses of the RCPB piping and piping components will be adequately managed for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.4 Non-Class 1 Component Fatigue Analyses

4.3.4.1 Summary of Technical Information in the Application

LRA Section 4.3.4 describes the TLAAs for non-Class 1 components. These TLAAs are based on the criteria for performing implicit fatigue analyses for the following components:

- American National Standards Institute (ANSI) B31.1 piping components, as given in the ANSI B31.1 design code
- ASME Code Class 2 and 3 components, as specified in ASME Section III, Article NC-3000, for components designed to ASME Section III Class 2 requirements, and Article ND-3000, for components designed to ASME Section III Class 3 requirements

LRA Section 4.3.4 states that the design of ASME III, Code Class 2 and 3 piping systems incorporates a cycle-based stress-range reduction factor (SRRF) for determining acceptability of piping design with respect to thermal stress range. The applicant added that components designated as quality group D (Class 3) are designed to ANSI B31.1, which also incorporates SRRFs based upon the number of thermal cycles. In general, a stress-range reduction factor of 1.0 in the stress analyses applies for up to 7,000 thermal cycles. The allowable stress range is reduced by the SRRF if the number of thermal cycles exceeds 7,000. The applicant further stated that if fewer than 7,000 cycles are expected through the period of extended operation, then the fatigue analysis (SRRF) of record will remain valid through the period of extended operation.

For these analyses, the applicant stated that the total number of occurrences for the full thermal transients that are applicable to these components is projected to be less than 7,000 through the end of the period of extended operation. The applicant dispositioned the TLAAs for non-Class 1 piping and in-line components in accordance with 10 CFR 54.21(c)(1)(i), that the analysis remains valid for the period of extended operation.

4.3.4.2 Staff Evaluation

The staff reviewed LRA Section 4.3.4 and the TLAAs for non-Class 1 components to verify, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses 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 Sections 4.3.3.1.2.1 and 4.3.3.1.4. These procedures state that the operating cyclic experience and a list of the assumed thermal cycles used in the existing allowable stress determination should be reviewed to ensure that the number of assumed thermal cycles would not be exceeded during the period of extended operation.

In LRA Section 4.3.4, the applicant states that the fatigue evaluation for non-Class 1 components determined if the associated operating temperature exceeded threshold values for the affected materials and, if so, evaluated the number of transient cycles expected. The applicant added that, in every case, the number of projected cycles for 60 years was found to be less than 7,000 for piping and in-line components. The staff reviewed LRA Table 4.3-2 and confirmed that there is significant margin between the total projected cycles for the design transients and the 7,000 cycle design limit.

The staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses for the non-Class 1 piping and piping components will remain valid during the period of extended operation. Additionally, the TLAAs associated with metal fatigue of non-Class 1 piping and piping components meets the acceptance criteria in SRP-LR Sections 4.3.2.1.2.1 and 4.3.2.1.4 because the total projected cycles for 60 years of operation for those design transients that impact the non-Class 1 piping and piping components TLAAs is significantly less than the 7,000 cycles originally considered in the analyses.

4.3.4.3 UFSAR Supplement

LRA Section A.1.3.3 provides an UFSAR supplement summarizing the TLAAs associated with metal fatigue of non-Class 1 piping and piping components. The staff reviewed LRA Section A.1.3.3 consistent with the review procedures in SRP-LR Section 4.3.3.3, which states that the reviewer should verify that the applicant provided information, to be included in the UFSAR supplement, which 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.

Based on its review of the UFSAR 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 effect of reactor coolant environment on fatigue usage, 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 has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue analyses remain valid for non-Class 1 piping and piping components, for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.5 Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping

4.3.5.1 Summary of Technical Information in the Application

LRA Section 4.3.5 describes the evaluation of the effect of reactor water coolant environment on fatigue usage for the period of extended operation. These EAF analyses are not incorporated in the applicant's CLB and existing design bases for the Class 1 components. Instead, the applicant identified that, although not part of the existing design basis, these EAF evaluations were performed for the 60-year operation period to conform to acceptance criteria and review procedure recommendations for assessing the effects of the reactor coolant environment on existing fatigue analyses for ASME Class 1 components, as stated in SRP-LR Sections 4.3.2.2 and 4.3.3.2.

The applicant stated that the minimum set of components for a BWR of its vintage is derived from NUREG/CR-6260 as follows:

- RV shell and lower head
- RV FW nozzle
- RRC piping (including inlet and outlet nozzles)
- core spray line RV nozzle and associated Class 1 piping
- RHR return line Class 1 piping
- FW line Class 1 piping

The applicant stated that the original fatigue usage calculations were reviewed, and the transient groupings and load pairs used in those analyses were carried over to the EAF analyses. Thus, the analyses for various locations ranged from a single transient grouping with a single load pair (e.g., the RRC inlet nozzle) to nearly a dozen load pairs and individual transients (e.g., FW nozzle and RRC piping).

With regards to carbon and low-alloy steels, austenitic stainless steels, and nickel-alloy components, the LRA states that the formulae used to calculate environmental life correction factors (F_{en}) are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels, and the nickel-alloy components were analyzed using the stainless steel correlations. Also, since the applicant has operated with hydrogen water chemistry (HWC) since November 28, 2004, and is assumed to continue operating with HWC until January 13, 2044, an effective F_{en} based on a time-weighted average of normal water chemistry (NWC) and HWC over 60 year of operation was used to incorporate the effects of coolant environment. The environmentally-adjusted cumulative usage factor (U_{en}) for fourteen plant-specific locations are summarized in LRA Table 4.3-6. The applicant stated that the U_{en} for all locations are less than 1.0

The applicant stated that it manages fatigue using the FMP to track transient cycles and requires corrective action before any analyzed number of cycles is reached. The applicant dispositioned the evaluations of the effect of reactor water coolant environment on fatigue usage 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.

4.3.5.2 Staff Evaluation

The staff noted that the applicant conservatively addressed the effects of the reactor coolant environment on component fatigue life as a TLAA, 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. Based on Commission Order No. CLI-10-17, the staff finds the applicant's evaluation of the effects of the reactor coolant environment on component fatigue life as a TLAA is conservative and is an acceptable practice consistent with the staff's recommendations in the SRP-LR and the closure of GSI-190.

The staff reviewed LRA Section 4.3.5 and the evaluations of the effect of reactor water coolant environment on fatigue usage 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 TLAAs and the corresponding dispositions, consistent with the review procedures in SRP-LR Section 4.3.3.2, which states 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. The SRP-LR also states that if the applicant has chosen to assess the impact of the reactor coolant environment on a sample of critical components, the reviewer should verify the following:

- The critical components include, as a minimum, those selected in NUREG/CR-6260.
- The sample of critical components is evaluated by applying environmental correction factors to the existing ASME Code fatigue analyses.
- The formulae for calculating the environmental life correction factors (F_{en}) match those contained in NUREG/CR–6583 for carbon and low-alloy steels, and in NUREG/CR-5704 for austenitic SSs, or an approved technical equivalent.

The staff reviewed LRA Section 4.3.5 and noted that design basis CUF values in air, listed for the limiting environmental fatigue components in LRA Table 4.3-6, are different from the design basis CUF values listed for the components in either LRA Table 4.3-3 or 4.3-5. The values listed for these components in LRA Table 4.3-6 were typically lower than the corresponding values listed for the components in LRA Table 4.3-3 or 4.3-5 by a factor of 2 to 10. Also, the revised CUF for the CRD housing and FW nozzle safe end (i.e., 0.0007 and 0.00126, respectively) are a factor of 280 and 770 lower than the present CUF of record.

By letter dated August 26, 2010, the staff issued RAI 4.3-05 requesting that the applicant justify, for each component location listed in LRA Table 4.3-6, why this table reports a design basis CUF value for the component location that is lower than the CUF value reported in LRA Tables 4.3-3 or 4.3-5. The staff also asked the applicant to clarify if the design basis CUF values, reported in LRA Table 4.3-6, represent an update of the design basis and, if so, to identify the document that establishes the CUF value reported in LRA Table 4.3-6 as the current design basis CUF value.

In its response dated November 11, 2010, the applicant stated that LRA Tables 4.3-3 and 4.3-5 list the maximum CUF values found for the listed RV locations and piping systems, as found in the analyses of record prior to any reanalysis activity to evaluate the effects of reactor water environment. The column labeled “Revised CUF in air” in LRA Table 4.3.6 includes the computed CUF in air for the wetted surface of interest selected for evaluation of the effects of reactor coolant environment on the component. The applicant further stated that this change to the LRA is provided in the enclosure as Amendment 13. The CUFs in the referenced tables are different because of the following changes or refinements implemented prior to determining EAF factors:

- The usage factors listed in Table 4.3-6 reflect the projected plant cycles for 60 years of operation, which includes additional startup and shutdown cycles, reduced bolt up and un-bolt cycles, reduced vessel hydro test cycles, increased turbine generator scram cycles with FW on, and reduced cycles for other scrams. While some cycles have increased, such as startup and shutdown, it tends to cause very low thermal transient stresses and do not significantly impact fatigue usage.
- The location for environmental fatigue usage determination must be on a wetted surface, whereas the maximum usage locations reported in Tables 4.3-3 and 4.3-5 are not necessarily on the inside wetted surface.
- To gain sufficient margin in the “in air” CUF when incorporating the effect of coolant environment, conservatism in the design analyses of record was removed. This was achieved by regrouping of conservative load pairs, taking credit for hardware changes during construction that were previously not credited in the analyses, replacing original enveloping design transients with current design specification transients, and reducing conservative stress concentration penalties.

The staff finds the changes and refinements, as described above, to be reasonable for the following reasons:

- Projected cycles, based on actual plant operating experience, were used in the analyses.
- Realistic operating condition of the components, based on actual plant operating experience, were used in the analyses.
- The applicant accounted for actual equipment configurations that were not considered in the original design analyses.

For each of the components and locations listed in LRA Table 4.3-6, the applicant provided, in its response to RAI 4.3-05, a table with the values of CUF of record and the revised CUF in air, and a general explanation about what factors contribute to a difference between the two CUF values.

Based on its review, the staff finds the applicant’s response to RAI 4.3-05 acceptable because, for all the components and locations listed in LRA Table 4.3-6, the applicant provided and justified the revised CUFs in air to support the EAF evaluations. The staff’s concern described in RAI 4.3-05 is resolved.

In its review of the locations in the six NUREG/CR-6260 components for which EAF analyses were performed, the staff noted that the applicant’s EAF analysis does not always apply the NUREG/CR-6260 methodology to the RPV or Class 1 piping components that have the highest design basis CUF values. For example, the CRD tube and CRD housing were selected as the representative locations for the RPV shell and lower head, and the design basis CUF values for these component locations were 0.083 and 0.196, respectively. However, in LRA Table 4.3-3, the shroud support (0.399), main steam nozzle shell (0.47), or LPCI thermal sleeve (0.430) all have existing design basis CUF values that are greater than those reported for the CRD tubes and CRD housings in LRA Table 4.3-3. By letter dated August 26, 2010, the staff issued RAI 4.3-06, asking the applicant to provide the basis for selecting RPV and Class 1 piping locations as the EAF analysis locations in the LRA. Also, justify the basis for not selecting core shroud supports, main steam shell nozzles, and LPCI nozzle thermal sleeves as additional EAF assessment locations.

In its response dated November 11, 2010, the applicant stated that, consistent with the GALL Report AMP X.M1, it addresses the effects of the coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components identified in NUREG/CR-6260. Section 4.1 of this report states that, for both PWR and BWR plants, these components are not necessarily the locations with the highest design CUFs in the plant, but it was chosen to give a representative overview of components that had higher CUFs or were important from a risk perspective or both. The applicant added that it analyzed 14 site-specific locations that represent the six components identified in NUREG/CR-6260 for a BWR of Columbia’s vintage, and these locations contain all the different materials used in the Columbia pressure vessel and attached piping. The applicant further stated that the main steam shell nozzle is exposed to dry steam. The environmental life correction factors apply to components exposed to reactor coolant and not to surfaces exposed to gaseous environments such as dry steam. Additionally, high usage location on the LPCI nozzle thermal sleeve was not evaluated because it was located on the thermal sleeve extension within the RPV nozzle, in a non-pressure boundary portion of the sleeve.

However, the staff noted that the applicant's plant-specific configuration may contain additional locations (including, but not limited to, those provided in LRA Tables 4.3-3 and 4.3-5) that may need to be analyzed for the effects of the reactor coolant environment other than those identified in NUREG/CR-6260. This may include locations that are limiting or bounding for the applicant's particular plant-specific configuration or that have calculated environmentally-adjusted CUF values that are greater than those calculated by the applicant for locations that correspond to those identified in NUREG/CR-6260.

By letter dated February 3, 2011, the staff issued RAI 4.3-09, asking the applicant to confirm and justify that the locations selected for EAF analyses in LRA Table 4.3-6 consist of the most limiting locations for the plant (beyond the generic components identified in the

NUREG/CR-6260 guidance). If these locations are not bounding, the staff asked the applicant to clarify the locations that require an EAF analysis and the actions that will be taken for these additional locations. If the identified limiting location consists of nickel alloy, the staff asked the applicant to state whether the methodology used to perform the EAF calculation for nickel alloy is consistent with NUREG/CR-6909 and, if not, to justify the method chosen. By letter dated March 3, 2011, the applicant responded to RAI 4.3-09 by stating that the analyses of additional locations for limiting CUF will be a significant undertaking and tentatively planned to submit the response to this RAI in September 2011. This issue was open item OI 4.3-1 in the SER with open items.

By letter dated October 6, 2011, the applicant supplemented its March 3, 2011, response and stated that the locations originally selected for EAF in LRA Table 4.3-6 were based on the locations identified in NUREG/CR-6260 and did not necessarily contain the most limiting locations for the plant. The applicant reviewed additional plant-specific locations to ensure that the limiting locations had been identified and evaluated. The response included a description of how it selected these additional locations from LRA Tables 4.3-3 and 4.3-5. Table 1 of the RAI response provided calculated 60-year environmental fatigue usage factors (CUF_{en}) for these additional locations, and showed the CUF_{en} to be less than 1.0 for all but four locations. The four locations that had a CUF_{en} greater than 1.0 are the HPCS nozzle safe end extension, the RPV head spray check valve, the HPCS inboard isolation check valve, and the low pressure core spray (LPCS) inboard isolation check valve.

By letter dated November 4, 2011, the applicant provided an additional supplement to its response to RAI 4.3-09 that presented additional information on the methodology used for selecting additional EAF locations. The response also provided the 60-year CUF_{en} values for the HPCS nozzle safe end extension, the RPV head spray check valve, the HPCS inboard isolation check valve, and the LPCS inboard isolation check valve. The staff noted that the calculated 60-year CUF_{en} values for these components were less than the ASME Code design limit of 1.0.

From its review, the staff concluded that the applicant had not provided sufficient and detailed information related to the methodology that it used to select these additional locations to address the effects of reactor water environment on fatigue life. Specifically, it was not clear to the staff how the applicant used the information from its RPV and piping stress reports to populate LRA Tables 4.3-3 and 4.3-5, nor was it clear how it subsequently selected the "additional locations" from these LRA tables. Therefore, on November 28, 2011, to December 1, 2011, the staff conducted an audit of the applicant's methodology for selecting additional locations to address the effects of reactor water environment on component fatigue life. At the audit, the staff selected a sample set of components and piping systems to perform its audit of the applicant's methodology. The objectives of the audit were the following:

1. To review the applicant's methodology for selecting limiting reactor vessel locations and reactor pressure boundary piping and piping component locations.
2. To confirm that the locations screened out for review of effects of reactor coolant were appropriate (e.g., non-wetted, non-pressure boundary, previously evaluated NUREG/CR-6260 location, and locations that bound other similar locations).
3. To review the applicant's methodology for a sample set of EAF calculations.

A summary of the staff's audit is detailed in its Audit Report dated February 16, 2012. By letters dated December 16, 2011, and January 4, 2012, the applicant provided responses to the staff's

Audit Questions. The staff's evaluation of the responses to these Audit Questions is documented below.

During the audit, the staff reiterated that the information provided by the applicant regarding its selection criteria for additional EAF locations in letters dated October 6, 2011, and November 4, 2011, was not clear. Therefore, the staff requested clarification from the applicant on the methodology used for selecting additional EAF locations as part of Audit Question #1. The applicant stated that the selection of additional EAF locations was based on identifying the highest air fatigue usage locations (i.e., CUF) for all of the Class 1 piping systems connected to the RPV and all of the remaining RPV components (i.e., those not already addressed by NUREG/CR-6260).

In its response dated December 16, 2011, the applicant discussed its methodology related to identifying the additional locations for the RPV. In particular, LRA Table 4.3-3 lists all CUF values from the "Columbia Generating Station RPV Stress Report." The staff noted that when selecting additional locations for evaluation of EAF, the applicant considered all locations evaluated in the RPV stress report to determine if any of these locations was more limiting than the NUREG/CR-6260 locations. The staff finds it conservative that the applicant started its evaluation by considering all CUF values evaluated in the RPV stress report because none of the components were eliminated from consideration for selecting additional EAF locations.

The staff noted that the list of locations was then screened to eliminate non-wetted locations such as nozzles exposed to dry steam, and components that were not exposed to reactor water, such as the vessel skirt, RPV flange, and RPV studs. The staff finds it appropriate that locations that are not actively exposed to reactor water or are non-wetted were screened out from consideration because the environmental effects on fatigue life depend on the location being exposed to a reactor water environment. The staff also noted that the applicant screened out non-pressure boundary components such as thermal sleeves, which the staff finds appropriate because it is consistent with the Fatigue Action Plan documented in SECY-95-245, "Completion of the Fatigue Action Plan," (ML031480210), as all reactor coolant pressure boundary components were considered for the effects of reactor water environment on fatigue life.

In its response dated December 16, 2011, the applicant stated that the remaining population of locations included all materials used in RPV components subject to the reactor water environment, with EAF evaluations performed for all of the remaining components using the design basis analyses as a starting point for the evaluation. During its audit, the staff confirmed that the applicant considered the effects of reactor water environment on component fatigue life for the various material types of the remaining RPV components. The applicant stated that the RPV stress report evaluated fatigue for various portions of the vessel nozzles (e.g., safe end, safe end extensions, nozzle forging, and thermal sleeves). In addition, the applicant assumed that the transients for the vessel nozzles are influenced by the vessel transients and the transients that occur within the attached piping. Since this approach is consistent with the original RPV stress report, the staff finds it reasonable that the applicant assumed that EAF effects from the vessel transients or the transients that occur within the attached piping are applicable to the entire nozzle (e.g., safe end, safe end extensions, nozzle forging and thermal sleeve).

In some cases, the design basis analysis of a nozzle was conservatively used to envelope a similar nozzle. For example, the HPCS and LPCS nozzles are the same size, material, and configuration; therefore, these nozzles were addressed as one nozzle (core spray) in the RPV stress report. The applicant stated that the HPCS nozzle has more transient cycles, the cycles are more extreme than those for the LPCS nozzle, and the HPCS nozzle has a greater range of temperature and pressure change than the LPCS nozzle. Therefore, the RPV stress report evaluated the HPCS transients and qualified the LPCS nozzle by comparison. The staff noted that the EAF evaluation used the same approach as the RPV stress report. Since the HPCS and LPCS nozzles are similar (same size, material, and configuration) and the F_{en} factor input parameters (metal temperature, sulfur content, dissolved oxygen, strain rate) for the HPCS nozzle are equal to or greater than those for the LPCS nozzle, the staff finds it reasonable that the F_{en} factor for the HPCS nozzle is bounding for the LPCS nozzle. The staff finds it reasonable that the EAF calculation for the HPCS nozzle qualifies the LPCS nozzle for EAF because the same assumptions in the RPV stress report for the HPCS and LPCS nozzles were used, such that the transient severity (temperature and pressure change) of the HPCS nozzles is greater than the LPCS nozzles and the F_{en} factor for the HPCS nozzle is bounding.

The applicant also discussed its methodology for identifying additional locations beyond NUREG/CR-6260 for its Class 1 piping systems. The applicant stated that LRA Table 4.3-5, which lists the maximum usages for all of the Class 1 piping systems, was developed from a tabulation of all system fatigue usages as part of the license renewal project basis document. Similar to the RPV nozzles, a screening was completed to eliminate piping systems that are exposed to dry steam, such as main steam, from further evaluation. As described above in the discussion for the RPV components, the staff finds this screening appropriate.

The staff noted that the applicant only performed the EAF calculation for loop A of the reactor feedwater (RFW) piping system. During its audit, as documented in its Audit Report dated February 16, 2012, the staff noted it was not clear why loop A bounds loop B; therefore the staff asked the applicant to provide justification as part of Audit Question #1. In its response to Audit Question #1 dated December 16, 2011, the applicant stated that the maximum fatigue usage from only one of the loops was evaluated for piping systems such as reactor recirculation cooling (RRC) and RFW that have multiple loops or trains with similar geometric configuration and materials. The applicant clarified that the thermal transients are the same for each loop or train for these systems, thus evaluation of a bounding location on one loop would envelope the conditions of the other loop. Although the staff noted small differences in fatigue usage associated with pipe support and restraint locations between the two loops, it also finds the applicant's explanation acceptable as the applicant has considered the maximum usage between multiple loops or trains of the same system, and the thermal transients considered between the loops were consistent with each other.

The applicant stated that the piping systems such as reactor core isolation cooling (RCIC), RFW, residual heat removal (RHR), reactor water clean-up (RWCU), HPCS, and LPCS, are comprised primarily of SA-106 Grade B carbon steel. The Audit Report describes that the staff verified, from the design specifications, the materials for those Class 1 piping reviewed by the staff.

The applicant stated that stainless steel Class 1 piping is primarily located in the RRC system, short segments of the RHR and RWCU systems that connect to the RRC, the standby liquid control (SLC) system, small-bore piping, and the reactor vessel level instrument condensing chambers. In addition, small-bore instrumentation piping is also stainless steel, but uses the ASME Code Class 1 exemption for fatigue design of piping sized at 1-inch-and-under. Therefore, the applicant concluded that its review of highest usage locations included all material types (carbon and stainless steel) used in its Class 1 piping.

The applicant stated that since large sections of piping systems are all affected by the same fluid flow conditions, the highest usage locations normally occur at structural discontinuities such as branch connections, tee's, reducers, and tapered transitions. Because of this, dissimilar metal weld joints were generally not used at these fittings with structural discontinuities, to keep fatigue usage low. Butt weld joints were used in straight pipe locations with low fatigue usage for dissimilar metal welds between carbon and stainless steel in the RRC to RHR, RWCU to RRC, and SLC to HPCS connections. These locations were screened out by the applicant because the usage factors were extremely low. During its audit, the staff confirmed that the CUF values for the butt weld joints for the aforementioned connections are very low. Thus, even multiplying with the maximum F_{en} values, the staff found that the EAF values for these dissimilar metal welds are less than 1.0.

The applicant stated that the piping systems tabulated for the EAF evaluation contain systems that provide injection to the vessel or draw supply from the vessel, which provides a variety of thermal transient conditions that can give slow and fast heat-up and cool-down of piping systems. Therefore, all transients experienced by the pressure boundary components were evaluated for their impact on environmental fatigue. As described in its Audit Report dated February 16, 2012, for those Class 1 piping systems reviewed, the staff confirmed in the piping system design specification that the applicant incorporated into its EAF evaluations all transients that were considered during the design of the system.

During the audit, the applicant specifically discussed its methodology related to the consideration of multiple material types. The applicant stated that several of the limiting locations selected for evaluation were part of a piping system that had a dissimilar metal weld and thus a portion of the piping was another material. During its audit, the staff asked Audit Question #8 regarding a situation where a high usage location was evaluated for one material (e.g., stainless steel), and whether a different material in that same area, such as carbon steel, could have a higher CUF_{en} although the CUF was not particularly high. The specific circumstances that led to this question are described below for the SLC to HPCS piping.

The applicant identified several locations with dissimilar metal welds between carbon and stainless steel piping in straight runs of piping. The applicant explained that the highest usage location was evaluated for the piping system thermal transient conditions and for other portions of the piping, including the dissimilar metal welds, the usages were reviewed to determine if an EAF assessment should be done. The following are the carbon to stainless steel interfaces in the plant's configuration:

- RRC to RHR on RRC Loops A and B: The RRC stainless steel usage was evaluated for environmental effect. The applicant reviewed the applicable Design Report for carbon steel CUFs that were not evaluated for environmental effects. The applicant determined that all CUF values were sufficiently low that when projected for 60 years and using a bounding EAF correction factor (F_{en}) penalty, the CUF_{en} would not be limiting.
- SLC to HPCS: The SLC piping is stainless steel and transitions to carbon steel before it connects into the HPCS system. The limiting usage evaluated for EAF was for the carbon steel portion of the piping system. As documented in its Audit Report dated February 16, 2012, the staff noted that the highest 60-year CUF for the stainless steel segment was at node 25 with a value of 0.054. The staff asked the applicant to clarify why the stainless steel segment was not evaluated for EAF (Audit Question #8). For the dissimilar metal weld and the stainless steel portion of this piping, the applicant determined that the CUF_{en} would not exceed that of the carbon steel portion, even with a bounding F_{en} value. The staff finds the applicant's response to Audit Question #8

acceptable since the stainless steel portion would not provide the highest CUF_{en} for the piping even with a bounding F_{en} factor.

- RWCU to RRC: The RWCU to RRC piping dissimilar metal weld connections were reviewed. The applicant determined that the limiting location was carbon steel and this review also determined that the stainless steel portion of the piping was subject to the same transients. The applicant concluded that the stainless steel 60-year CUF_{en} calculated with a conservative F_{en} would not be limiting. The staff noted that the applicant did consider the stainless steel portion of the RRC piping in LRA Table 4.3-6 for effects of reactor water environment on fatigue life.

Based on its review and audit, the staff finds it reasonable that, for piping systems with interfaces between carbon and stainless steel materials, the applicant did appropriately identify the limiting locations by directly considering the values of CUF_{en} for each material. Thus the response to Audit Question #8 is acceptable.

During its audit, as documented in its Audit Report dated February 16, 2012, the staff noted that some ASME Class 1 valves were not addressed in LRA Table 4.3-5 of the applicant's basis document for TLAAs. The staff asked the applicant to clarify whether any of these valves should have been considered when addressing the effects of reactor water environment on fatigue life (Audit Question #2). By letter dated December 16, 2011, the applicant stated that valves HPCS-V-51, LPCS-V-5, LPCS-V-51, RHR-V-112A and RHR-V-112B are all evaluated in the same design report and all of these other valves were bounded when evaluating HPCS-V-51 for EAF. Since these valves are similar (same size, material, and pressure rating) and the F_{en} factor input parameters (metal temperature, sulfur content, dissolved oxygen, strain rate) for these valves are equal to those for HPCS-V-51, the staff finds it reasonable that the F_{en} factor for the HPCS-V-51 bounds that for the remaining valves (LPCS-V-5, LPCS-V-51, RHR-V-112A and RHR-V-112B). The staff finds the applicant's response to this part of Audit Question #2 acceptable and reasonable that the EAF evaluation for the HPCS-V-51 valve qualifies the LPCS-V-5, LPCS-V-51, RHR-V-112A and 112B valves for EAF because the same assumptions in the RPV stress report for these valves were used and the F_{en} factor for the HPCS-V-51 valve is bounding. The staff noted that the applicant amended LRA Table 4.3-5 to clarify that these five valves are represented by the "12-inch containment isolation valves" with a cumulative usage factor of 0.6599.

Similarly, the applicant also stated that valves RHR-V-53A and 53B are bounded by the evaluation of HPCS-V-51 because it is similar material (carbon steel), have similar geometry (i.e., same size and pressure rating), and the transients for HPCS are more or equally severe compared to the RHR temperature change and pressure. Since the HPCS-V-51, RHR-V-53A and 53B valves are similar (same size, material, and pressure rating) and the F_{en} factor input parameters (metal temperature, sulfur content, dissolved oxygen, strain rate) for the HPCS-V-51 valve are equal or greater than those for the RHR-V-53A and 53B valves, the staff finds it reasonable that the F_{en} factor for the HPCS-V-51 bounds the two remaining valves. The staff finds the applicant's response to this part of Audit Question #2 acceptable and reasonable that the EAF calculation for the HPCS-V-51 valve qualifies the RHR-V-53A and 53B valves for EAF because these valves are similar (same size, material, and pressure rating), the transients for the HPCS-V-51 are equal to or greater than the RHR valves, and the F_{en} factor for the HPCS-V-51 valve is bounding. The staff noted LRA Table 4.3-5 has been amended and that these RHR valves are also represented by the "12-inch containment isolation valves" with a cumulative usage factor of 0.6599.

As a result of the staff's audit question, the applicant amended LRA Table 4.3-5 to include the 40-year usage factors for the RFW and RWCU valves. By letter dated January 4, 2012, the applicant provided the 60-year CUF_{en} values for the RFW and RWCU valves. It was explained that the Class 1 valve cyclic stresses and 60-year design (air) fatigue usages were calculated in accordance with the procedures specified in Subarticle NB-3550 of the ASME Section III Code. The staff noted that the applicant used the thermal and pressure conditions identified in the piping design specifications for the RWCU return piping to RFW and the RFW supply piping to the RPV, and the calculations are based on the 60-year projected cyclic loading identified in LRA Table 4.3-2. The staff finds it reasonable that the applicant used the thermal and pressure conditions identified in piping design specifications because these are the conditions used for the design of the plant. In addition, it is reasonable that the applicant used the 60-year cycle projections in these calculations because the applicant's Fatigue Monitoring Program systematically counts transient cycles to ensure that the numbers of analyzed cycles in the calculation for each component location are not exceeded, thereby ensuring that component fatigue usage limits are not exceeded, consistent with the recommendations in GALL Report AMP X.M1.

The applicant used the procedures and the carbon steel F_{en} in NUREG/CR-6583 to account for environmental effects on fatigue life. In its response dated January 4, 2011, the applicant clarified that the F_{en} was calculated for each ΔT and ΔP cyclic load set condition. The staff noted that the applicant defined the transformed temperature term used in the environmental factor based on average temperatures for each load set thermal modes, as permitted by NUREG/CR-6583. The applicant used the bounding value for the transformed environmental strain rate and sulfur factors, which the staff finds conservative. In addition, the applicant considered the time it operated under both NWC and HWC dissolved oxygen conditions, which are based on plant-specific operating chemistry data. The staff's review of the applicant's use of NWC and HWC dissolved oxygen conditions is discussed as part of RAI 4.3-07, which is documented below in this same SER section. The staff noted that the 60-year CUF_{en} for the RWCU valve and RFW valve are 0.196 and 0.920, respectively. In addition, both components are being managed by the applicant's Fatigue Monitoring Program, which ensures that the number of cycles assumed in these analyses will not be exceeded prior to corrective actions to repair, replace or reanalyze the component.

During its audit, as documented in its Audit Report dated February 16, 2012, the staff asked the applicant if it had been managing the number of transient cycles since initial plant start-up (Audit Question #4). By letter dated December 16, 2011, the applicant stated that plant cycle counting has been done since plant start-up and that plant Technical Specification 5.5.5 has required counting of the plant thermal cycles listed in UFSAR Table 3.9-1. The applicant clarified that this required cycle counting is implemented once every year per its plant procedure "Tracking of Fatigue Cycles." The applicant stated that the latest summary tabulation of plant cycles was updated on August 26, 2011, and the update includes all events/cycles that have occurred since initial plant start-up. The staff finds the applicant's response to Audit Question #4 acceptable because the applicant has complete records of the number of transient events that have occurred at its site since initial plant startup, which provides an accurate gauge of the margin between the assumptions in its fatigue evaluations and the calculated CUF values.

As documented in its Audit Report dated February 16, 2012, the staff noted that the applicant did not update LRA Sections A.1.2.24 and A.1.3.4 and B.2.24 to indicate that additional locations had been evaluated to address the effects of reactor water environment on fatigue life; the staff asked the applicant why these updates were not made (Audit Question #5). By letter dated December 16, 2011, the applicant amended these LRA sections to clarify that other limiting components beyond those locations identified in NUREG/CR-6260 had been evaluated for the effects of reactor water environment. The staff noted that the locations identified in LRA Tables 4.3-6 and 4.3-7 have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of EAF will be adequately managed for the period of extended operation using the Fatigue Monitoring Program. The staff finds the applicant's response to Audit Question #5 acceptable because the applicant amended its LRA to clearly identify the disposition of these additional EAF evaluations, as required by 10 CFR 54.21(c)(1), and the applicant is managing the effects of reactor coolant environment on component fatigue life with its Fatigue Monitoring Program, thereby ensuring that the assumptions in these evaluations will remain valid during the period of extended operation on an on-going basis.

During its audit, the staff noted that the applicant considered the effects of reactor water environment on its plant-specific locations that correspond to NUREG/CR-6260 and additional critical plant-specific locations beyond NUREG/CR-6260. It is not clear to the staff if the applicant, during the period of extended operation, will ensure that all critical locations in these ASME Class 1 components and piping systems will be evaluated for the effects of reactor water environment. The staff asked the applicant (Audit Question #3) how it ensures that the effects of reactor water environment will continue to be evaluated for the limiting locations in the plant, even if a limiting location for a system and material has changed due to a physical or operational change, or plant operating experience. By letter dated December 16, 2011, the applicant amended its UFSAR supplement in LRA Section A.1.3.4 to state: "For the period of extended operation, on an ongoing basis, ensure that all the limiting locations in class 1 components and class 1 systems have been evaluated for the effect of reactor water environment." The staff noted that, regardless of any modifications or changes to the applicant's site or operation that may occur in the future, the applicant's Fatigue Monitoring Program will ensure that, for Class 1 components and piping systems, the effects of reactor water environment will be evaluated for the limiting locations on an on-going basis during the period of extended operation. The staff finds the applicant's response to Audit Question #3 acceptable because, for these EAF evaluations that are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii), the applicant will ensure that the limiting locations for its site have been addressed for the effects of reactor water environment during the period of extended operation.

The staff noted in LRA Table 4.3-7 for vessel head spray nozzle, that the applicant's October 6, 2011, letter stated that the location is a "dry steam environment." However, in its November 4, 2011, letter, the applicant provided CUF and CUF_{en} values for this nozzle. During its audit, as documented in its Audit Report dated February 16, 2012, the staff asked the applicant to clarify the revision (Audit Question #9). By letter dated December 16, 2011, the applicant amended LRA Table 4.3-7 to indicate that the vessel head spray nozzle is exposed to dry steam and is not subject to environmental effects on fatigue life. By letter dated January 4, 2012, the applicant clarified that credit was taken for the thermal sleeve and spray nozzle inserted into the vessel nozzle to direct RCIC spray onto the steam dryer and that the nozzle is at the top of the RPV head, and thus is exposed to dry steam. During the audit, the staff reviewed UFSAR Figure 5.4-11 and confirmed that the vessel head spray nozzle is exposed to a dry steam environment and not a reactor water environment; therefore, it is not subject to the effect of reactor water environment on fatigue life, and the staff finds the applicant's revision and response to Audit Question #9 acceptable.

During its audit, as documented in its Audit Report dated February 16, 2012, the staff noted that the reported CUF value in LRA Table 4.3-3 for the feedwater nozzle-shell junction was not consistent with the reported value in the original vessel stress report. The staff asked the applicant to clarify the discrepancy (Audit Question #7). By letter dated December 16, 2011, the applicant amended LRA Table 4.3-3 to state that the CUF value for the feedwater nozzle-shell junction is 0.709. By letter dated January 4, 2012, the applicant clarified that the original value listed in the table for the nozzle-shell junction usage was taken from the original Chicago Bridge & Iron (CB&I) vessel stress report, with a value of 0.650. Since that time, General Electric issued a report in May 2009 that changed the usage value resulting from the 1995 Power Uprate to 0.709. During its audit, the staff reviewed NEDC 32153, Rev.1, and noted that the CUF for the feedwater nozzle-shell junction is 0.709 after the applicant's power uprate, which is its current licensing basis; therefore, the staff finds the applicant's revision and response to Audit Question #7 acceptable.

The staff noted that the revised CUF in air for the RFW/RWCU tee is 0.097 in LRA Table 4.3-6, as amended by letter dated November 4, 2011. However, during its audit, as documented in its Audit Report dated February 16, 2012, the staff noted that the applicant's calculation for the RFW/RWCU tee states that the revised CUF in air was 0.210. The staff asked the applicant to clarify the discrepancy between the LRA and the Energy Northwest Manual Calculation (ME-02-09-17), Appendix K, "Evaluation of Environmental Fatigue Effects for the Class 1 RWCU Piping," (Audit Question #6). By letter dated December 16, 2011, the applicant amended LRA Table 4.3-6 to identify the revised CUF in air for the RFW/RWCU tee as 0.210. By letter dated January 4, 2012, the applicant clarified that the original value of 0.097 represented a 40-year air usage value while the column required a 60-year value. Thus, the 60-year CUF is 0.210 and the 60-year CUF_{en} is 0.4333. The staff finds the applicant's revision and response to Audit Question #6 acceptable because the LRA has been revised to be consistent with the environmentally assisted calculation.

Based on its review of submittals by the applicant and the results of the staff's audit, the staff finds the applicant's response to RAI 4.3-09 acceptable because it is consistent with recommendations in the GALL Report and the SRP-LR, in that the applicant considered the effects of reactor water environment on component fatigue life for the sample locations identified in NUREG/CR-6260 and additional locations beyond NUREG/CR-6260 that are based on the applicant's plant-specific configuration. In addition, based on its audit on November 28, 2011, to December 1, 2011, the staff reviewed the applicant's methodology and EAF calculations for a sample set of RPV components and Class 1 piping systems and determined that all applicable material types and plant-specific system configurations were considered for RPV components and Class 1 piping systems, expect as justified above. Based on all of this information, open item OI 4.3-1 is closed.

The staff also noted that the applicant used an effective F_{en} based on a time-weighted average of NWC and HWC over 60 years of operation to determine environmentally assisted CUFs. However, the LRA does not give any details regarding the values of dissolved oxygen (DO) for NWC and HWC operation or the basis for selecting those values. The staff noted that, according to BWRVIP-130 "BWR Vessel and Internals Project BWR Water Chemistry Guidelines—2004 Revision," the operating range for DO is 30 to 200 parts per billion (ppb) for NWC and 30 to 100 ppb for HWC. However, LRA Section 4.3.5 does not give any details regarding the DO concentration values for implementation of NWC and HWC conditions that were derived and applied to the F_{en} calculation methodology or the basis for deriving the DO values. By letter dated August 26, 2010, the staff issued RAI 4.3-07, asking the applicant to provide, for each component location listed in LRA Table 4.3-6, the DO concentration inputs under implementation of NWC and HWC operating conditions that were used in the calculation of the F_{en} values for the components. The staff also asked the applicant to clarify how these DO inputs were derived and why it is considered to be conservative for application to the F_{en} methodology.

In its response dated November 11, 2010, the applicant stated that the carbon steel piping in the low-pressure core spray (LPCS), HPCS, and RHR systems is exposed to air saturated water environments from the suppression pool or the condensate storage tank or both during the thermal transient fatigue loading. Therefore, a bounding DO concentration of 500 ppb was assumed for these locations. The applicant also stated that, at the reactor feedwater (RFW) and reactor water cleanup (RWCU) system tee connection, the return water from the lower head region (at 153 ppb and 1 ppb DO, respectively, under NWC and HWC) mixes with FW having average DO concentrations of 58 ppb and 54 ppb, respectively, under NWC and HWC.

The applicant further stated that plant-specific operating water chemistry data showed that the average reactor water DO concentrations in the RPV shell and upper head regions and recirculation piping operating under NWC and HWC were 87 ppb and 1 ppb, respectively. In the RV lower head region, the average DO concentrations under NWC and HWC are 153 ppb and 1 ppb, respectively. The applicant added that since the root of the weld of the reactor feedwater and reactor water cleanup system tee connection is expected to see some mixture of these two conditions, the F_{en} calculations assumed 150 ppb and less than 40 ppb DO for NWC and HWC, respectively. The applicant further added that the FW flow goes through the thermal sleeve directly into the sparger, which directs the water away from the vessel wall. The applicant stated that although bulk reactor water DO is 1 ppb, some mixing of the FW at 54 ppb is assumed, and a conservative value of 40 ppb was used for the DO at the blend radius of the FW nozzle under HWC. Based on the bulk volume of the reactor water at 1 ppb DO , compared to the volume of feedwater at 54 ppb DO, the staff finds it reasonable for the applicant to assume the resultant DO of the mixture between the two to be less than 50 ppb. The staff noted that based on NUREG/CR-6583 for low-alloy steel, the transformed DO is zero when DO contentration is less than 50 ppb and the F_{en} value is unaffected. Therefore, based on the operating parameters at the applicant's site, as discussed above, the assumption for DO under HWC at the reactor vessel feedwater nozzle is acceptable. Also, for the locations other than those discussed above and the RV FW nozzle, the DO under NWC and HWC was considered to be 200 ppb and less than 40 ppb, respectively, for carbon steel and low-alloy steel components, and less than or equal to 50 ppb and less than 50 ppb under NWC and HWC, respectively, for stainless steel and nickel alloy components.

Based on its review, the staff finds the applicant's response to RAI 4.3-07 acceptable for the following reasons:

- The applicant provided the DO concentrations under NWC and HWC operating conditions that were used in the calculation of the F_{en} values for the components listed in LRA Table 4.3-6.
- The DO concentrations were based on plant-specific operating chemistry data.
- The use of a weighted F_{en} based on NWC and HWC provide realistic effects of reactor water on fatigue life that occur at the applicant's site.

The DO concentrations used in the F_{en} calculations for the RV FW nozzle and feed water piping are discussed below in the applicant's response to RAI 4.3-08. The staff's concern described in RAI 4.3-07 is resolved.

In its review, the staff presumed that the 150 ppb average DO concentration value listed in LRA Section 4.3.5 for the FW nozzle was the value under implementation of NWC. However, it is not clear to the staff if the value listed for the FW nozzle is based on implementation of NWC or HWC. By letter dated August 26, 2010, the staff issued RAI 4.3-08, asking the applicant to justify the basis for assuming a DO concentration value of 150 ppb for the FW nozzles and to clarify if this value represents the value for operations under NWC conditions or HWC conditions.

In its response dated November 11, 2010, the applicant stated that the average reactor water DO concentrations in the RPV shell and upper head regions and recirculation piping operating under NWC and HWC were 87 ppb and 1 ppb, respectively. The plant-specific low-alloy steel FW nozzle locations exposed to these DO conditions include the FW nozzle to shell junctions (i.e., nozzle to shell blend radius) and the FW nozzle forging. The DO concentrations for the F_{en} calculations for the low-alloy steel FW nozzle to shell blend radius location were conservatively assumed to be 150 ppb and 40 ppb, respectively, for NWC and HWC conditions. The applicant also stated that the limiting location for the FW nozzle forging is exposed to RV water in the gap between the low-alloy steel nozzle forging and the nozzle thermal sleeve. Since there is very low flow in this region, it was anticipated that, under HWC conditions, DO concentration at this location would be higher than 1 ppb. Therefore, the DO concentrations for F_{en} calculations for the FW nozzle forging were conservatively assumed to be 150 ppb and 100 ppb, respectively, for NWC and HWC conditions. The staff noted that for low-alloy steel, the use of a higher DO concentration (such as 100 ppb compared to 40 ppb) in the F_{en} calculation results in a higher and more conservative F_{en} value. The applicant further added that plant-specific operating chemistry data at Columbia showed that, under NWC and HWC conditions, average DO concentrations in the FW piping are 58 ppb and 54 ppb, respectively. Therefore, consistent with the operating data, the DO concentrations for F_{en} calculations for the FW nozzle nickel alloy safe-end were assumed to be the default value of less than 50 ppb specified in NUREG/CR-5704, under both NWC and HWC conditions.

Based on its review, the staff finds the applicant’s response to RAI 4.3-08 acceptable. For all locations listed in LRA Table 4.3-6, the DO concentrations considered in the F_{en} calculations are acceptable because the applicant used either conservative values or values consistent with the plant operating water chemistry data. The staff’s concern described in RAI 4.3-08 is resolved.

Based on the acceptance criteria in SRP-LR Section 4.3.2.1.1.3, closure of OI 4.3-1, and review procedures in SRP-LR Section 4.3.3.1.1.3 for dispositioning CUF-based TLAAs in accordance with 10 CFR 54.21(c)(1)(iii), the staff finds that the applicant provided valid bases for demonstrating that each of the CUF analyses for the effect of reactor coolant environment on fatigue life of components and piping would be acceptable, in accordance with 10 CFR 54.21(c)(1)(iii).

The staff noted that the applicant’s Fatigue Monitoring Program includes an enhancement to correlate information relative to fatigue monitoring and provide more definitive verification that the transients monitored and their limits are consistent with or bound the UFSAR and the supporting fatigue analyses, including the EAF analyses. The applicant committed (Commitment No. 24) to implement this enhancement prior to the period of extended operation. The staff noted that the implementation of this enhancement will ensure that actions are taken prior to the design code limit of 1.0 being exceeded or prior to the analyzed cycles in the fatigue analysis being exceeded. The staff’s review of this enhancement and the applicant’s Fatigue Monitoring Program is documented in SER Section 3.0.3.2.7.

The staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that, for the TLAAs that address the effects of reactor coolant environment on the fatigue life of piping and component, the effects of aging on the intended functions will be adequately managed for the period of extended operation. Additionally, with the closure of OI 4.3-1, the TLAA associated with the effects of reactor coolant environment on fatigue life meets the acceptance criteria in SRP-LR Section 4.3.2.2 because the applicant's FMP tracks the number of transient cycles that occur and requires corrective actions to be taken prior to any analyzed number of cycles in the TLAA being reached. This ensures that the analyses, when considering reactor water environmental effects, remain valid, and the design limit of 1.0 is not exceeded.

4.3.5.3 UFSAR Supplement

LRA Section A.1.3.4 provides an UFSAR supplement summarizing the evaluations for the effects of reactor water environment on fatigue life. The staff reviewed LRA Section A.1.3.4, consistent with the review procedures in SRP-LR Section 4.3.3.3, which states that the reviewer should verify that the applicant provided information, to be included in the UFSAR supplement, which 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.

Based on its review of the UFSAR supplement and closure of OI 4.3-1, the staff finds that 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 effect of reactor coolant environment on fatigue usage, as required by 10 CFR 54.21(d).

4.3.5.4 Conclusion

On the basis of its review and closure of OI 4.3-1, 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 is consistent with Commission Order No. CLI-10-17. The staff also concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of reactor coolant environment on component fatigue life will be adequately managed for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the evaluation, as required by 10 CFR 54.21(d).

4.4 Environmental Qualification (EQ) of Electrical Equipment

The environmental qualification 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 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 loss of coolant accident (LOCA), a high energy line break (HELB), or post-LOCA environment. EQ equipment is comprised of safety-related equipment, non-safety-related equipment whose failure could prevent satisfactory accomplishment of any safety-related function, and necessary post-accident monitoring equipment.

4.4.1 Summary of Technical Information in the Application

LRA Section 4.4 summarizes the evaluation of EQ of electrical equipment for the period of extended operation. The applicant stated that its review of Columbia EQ qualification information documents (QIDs) for electrical equipment showed that the majority are TLAAs. There are 113 QIDs for equipment covered by 10 CFR 50.49. Of these, 100 QIDs are identified in the LRA as TLAAs because it meets all six of the criteria established in the TLAA definition of 10 CFR 54.3. The remaining 13 QIDs are not identified in the LRA as TLAAs because the subject equipment has a qualified life of less than 40 years. The applicant also stated that the EQ TLAAs were dispositioned in accordance with 10 CFR 54.21(c)(1), and any updates of the QIDs will be performed in accordance with EQ Program processes. Updates of the QIDs are not a license renewal commitment. The license renewal commitment is that the EQ Program will be used to manage aging of EQ components. Ultimately any needed updates of the QIDs to extend qualified life prior to entering the period of extended operation will be driven by the EQ Program, using the same methodology as in the current license term to ensure components do not exceed their qualified life. The applicant further stated that updates may include re-analysis of the qualified life, refurbishment of the equipment, or replacement of the equipment. A re-analysis will be performed in a timely manner (that is, with sufficient time available to refurbish, replace, or re-qualify the component if the re-analysis is unsuccessful).

The applicant dispositioned the TLAAs for the EQ of electrical equipment 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.

4.4.2 Staff Evaluation

The staff reviewed LRA Sections 4.4 and B.2.22, plant basis documents, additional information provided to the staff, and interviewed plant personnel to verify whether the applicant provided adequate information to meet the requirements 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. In the GALL Report, plant EQ programs that implement the requirements of 10 CFR 50.49 are considered acceptable aging management programs. GALL 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 in accordance with 10 CFR 50.49. The staff conducted an audit of the information provided in LRA Sections 4.4 and B.2.22 and program basis documents. LRA Section 4.4 discusses the component reanalysis attributes, including analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria and corrective actions. On the basis of its audit (as described in SER Section 3.0.3.1.16), the staff found that the EQ program is in fact consistent with GALL AMP X.E1, “Environment Qualification (EQ) of Electric Components.” Therefore, the staff finds that the applicant’s EQ program demonstrates, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The applicant’s EQ program is therefore capable of programmatically managing the qualified life of components within the scope of the program for license renewal. The continued implementation of the EQ program provides assurance that the aging effects will be managed and that components within the scope of the EQ program will continue to perform their intended functions for the period of extended operation.

4.4.3 UFSAR Supplement

In LRA Appendix A, Section A.1.3.5, the applicant provides the UFSAR supplement summary description for the Environmental Qualification of Electrical Equipment TLAA. The staff reviewed LRA Section A.1.3.5 consistent with the review procedures in SRP-LR Section 4.4.3.2, which states that the reviewer verifies that the applicant has provided information to be included in the UFSAR supplement that includes a summary description on the TLAA evaluation of the environmental qualification of electric equipment consistent with LRA Section 4.4.

Based on its review of the UFSAR supplement, the staff finds it meets the acceptance criteria in SRP-LR Section 4.4.3.2. Additionally, the staff determined that the applicant provided an adequate summary description of its actions to address TLAAs 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 related to environmental qualification of electrical equipment will be adequately managed for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.5 Loss of Prestress in Concrete Containment Tendons

4.5.1 Summary of Technical Information in the Application

LRA Section 4.5, stated that Columbia containment does not have prestressed tendons. As such, loss of prestress in concrete containment tendons is not a TLAA.

4.5.2 Staff Evaluation

As discussed in SER Section 4.12.5, the staff reviewed the applicant’s UFSAR and confirmed that the containment is a steel primary containment vessel and does not have prestressed tendons and, therefore, finds the applicant’s statement acceptable.

4.5.3 UFSAR Supplement

The UFSAR supplement for the fatigue analyses of loss of prestress in concrete containment tendons is not needed.

4.5.4 Conclusion

Based on its review, the staff concludes that loss of prestress in concrete containment tendons is not a TLAA. The staff also concludes that the UFSAR supplement is not needed.

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

Summary of Technical Information in the Application

LRA Section 4.6 describes the applicant's TLAA for the Columbia primary containment. The applicant stated that the Columbia primary containment utilizes a GE Mark II over-under pressure-suppression configuration. The drywell is connected to the suppression pool by 99 downcomer pipes (3 of the 102 original pipes have been capped) that channel steam released during a LOCA for quenching and pressure suppression.

The applicant states that the cycles used in the fatigue evaluation of the containment components are listed in UFSAR Table 3A.4.1-3. The four events considered for the fatigue evaluation are operating basis earthquake, safe shutdown earthquake, safety relief valve (SRV) actuations, and chugging.

According to Section 4.6 of the LRA, no operating basis earthquake has occurred through 2007. On this basis, the applicant projects that it will remain within the 5 analyzed events through 60 years of operation. The applicant also states in LRA Section 4.6 that the safe shutdown earthquake and post-LOCA chugging are once in a lifetime events and thus will not exceed the one analyzed event through 60 years of operation. The applicant further stated that it reviewed the plant data and found that no more than 636 SRV cycles have occurred through 2007. Based on this data, the applicant has conservatively projected the number of SRV cycles to 2,400 through 60 years of operation, which is well below the 13,434 cycles that have been analyzed. The LRA further states that the fatigue analyses performed using 13,434 cycles will remain valid for the period of extended operation.

The applicant dispositions the TLAAs associated with fatigue of the containment in accordance with 10 CFR 54.21(c)(1)(i), that the analyses remain valid for the period of extended operation.

Staff Evaluation

The staff reviewed LRA Section 4.6 to verify, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses associated with load cycle limits of primary containment components remain valid for the period of extended operation. According to the applicant's UFSAR, the primary containment vessel and its appurtenances comply with the requirement of the ASME Code, Section III, Subsection NE, Class MC components. These components were designed for one safe shutdown earthquake and one post-LOCA chugging event. These events are considered once in a lifetime events so it is highly unlikely that the plant will exceed one occurrence of each event in 60 years of operation. According to the applicant, no operating basis earthquake has occurred through 2007. The containment and its appurtenances were designed for five operating basis earthquakes. Since the plant has yet to experience one operating basis earthquake, it is highly unlikely that the design limit of five operating basis earthquakes will be exceeded during the period of extended operation. The applicant stated that no more than 636 SRV cycles have occurred through 2007 and conservatively projected that number to 2,400 cycles through 60 years of operation. This is significantly less than the 13,434 cycles assumed by the applicant in the original fatigue analysis.

Based on its review, the staff finds the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue analyses for the containment remain valid during the period of extended operation because the containments are designed for more cycles than the maximum expected cycles during 60 years of operation.

UFSAR Supplement

LRA Section A.1.3.6.1 provides the UFSAR supplement summarizing the TLAA evaluation of the load cycle limits for the primary containment components for the period of extended operation. Based on its review of the UFSAR supplement, the staff concludes that the applicant provided an adequate summary description of its actions to address primary containment components fatigue analyses for the period of extended operation, as required by 10 CFR 54.21(d).

Conclusion

On the basis of its review, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses for load cycle limits of primary containment components remain valid during the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.6.1 ASME Class MC Components

4.6.1.1 Summary of Technical Information in the Application

LRA Section 4.6.1 states that the ASME Class MC components include the primary containment vessel shell, large openings (equipment hatch, personnel hatches, and access hatch), penetrations (all except the large openings), and attachments (pipe supports in the wetwell, welding pads in the drywell, supports for the stabilizer truss, seal and shear lugs at the drywell floor, supports for the downcomer bracing system, pipe whip supports, radial beam supports, cap truss supports, catwalks, monorail, and platforms). The LRA also states that the Class MC components were analyzed for fatigue using the transients listed in UFSAR Table 3A.4.1-3. In the LRA, the applicant further states that since these cycles will not be exceeded for 60 years of operation, this Class MC component fatigue analysis will remain valid for the period of extended operation.

The applicant performed a specific fatigue analysis for the main steam penetrations using the transients listed in UFSAR Table 3A.4.1-3. The applicant stated that the maximum revised CUF was 0.174.

The NRC staff granted the applicant an amendment to the operating license to allow an increase in the power level of the plant in 1995. According to the applicant, the loss of coolant accident (LOCA) containment dynamic loads are not affected by the power uprate and the SRV containment loads will remain below their design allowables.

The applicant dispositions the TLAAs associated with fatigue of the ASME Class MC components in accordance with 10 CFR 54.21(c)(1)(i), that the analyses remain valid for the period of extended operation.

4.6.1.2 Staff Evaluation

The staff reviewed LRA Section 4.6.1 to verify, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue analyses for the ASME Class MC components remain valid for the period of extended operation.

The staff's review of LRA Sections 4.6 and 4.6.1 indicate that the ASME Class MC components were analyzed using the transients listed in UFSAR Table 3A.4.1-3. As described in SER Section 4.6, the staff evaluated the fatigue analysis of the primary containment for the transients listed in UFSAR Table 3A.4.1-3 and found that the fatigue analyses remain valid for the period of extended operation because the existing analyses consider more cycles than the maximum expected cycles during 60 years of operation. Since the ASME Class MC components were also designed to the aforementioned transients, the staff has determined that the fatigue analyses for ASME Class MC components remain valid for the period of extended operation.

The staff also noted that the applicant has a specific analysis of the main steam penetrations. The applicant determined that the maximum CUF for the main steam penetrations for the transients listed in UFSAR Table 3A.4.1-3 was 0.174. This CUF of 0.174 is based on more cycles than the expected cycles during during the 60 years of operation. In accordance with ASME Section III, CUF values must be less than 1.0. The CUF for the main steam penetrations is significantly less than the 1.0 limit and, therefore, is acceptable.

The staff reviewed the Columbia UFSAR Appendix 3A and found that Energy Northwest requested an amendment to the operating license in July 1993 to allow an increase in the power level of the plant. The NRC granted the license amendment in May 1995. According to the Columbia UFSAR Section 3A, for the short-term containment pressure response, the peak pressure values are below design values and remain virtually unaffected by power uprate and extended load line limit. In addition, the LOCA containment dynamic loads are not affected by power uprate, and SRV containment loads will remain below their design allowables. Therefore, the staff has determined that the power uprate will not affect the fatigue analyses because the design loads used in the original fatigue analysis will not be exceeded, nor will the number of cycles be exceeded. Therefore, the fatigue analyses remain valid.

Based on its review, the staff finds that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue analyses for the ASME Class MC components remain valid for the period of extended operation because it is designed for more cycles than the maximum expected cycles during 60 years of operation.

4.6.1.3 UFSAR Supplement

The applicant provided a UFSAR supplement summary description of its TLAA evaluation of ASME Class MC Components in LRA Sections A.1.3.6.2. Based on its review of the UFSAR supplement, the staff concludes that the applicant provided an adequate summary description of its actions to address fatigue of ASME Class MC Components for the period of extended operation, 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 has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue analyses for ASME Class MC components remain valid for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.6.2 Downcomers

4.6.2.1 Summary of Technical Information in the Application

LRA Section 4.6.2 states that the plant has 84 24-inch diameter downcomers and 18 28-inch downcomers. Three of the downcomers are capped. The applicant states that the downcomer vent pipes are designed to contain and direct uncondensed drywell steam into the suppression pool following a pipe break accident. The LRA states that the upper portions of the downcomers are designed and constructed in accordance with ASME Section III Class 2 requirements, while the lower portions are designed and constructed to ASME Section III Class 3 requirements. The application provides the results of a fatigue analysis on the downcomers, even though it is not required by the ASME Code. The LRA states that the fatigue evaluation of the downcomer lines in the wetwell air volume was based on the number of cycles provided in LRA Table 3A.4.1-3. The application states that the maximum fatigue usage factor for the 24-inch downcomers is 0.0346 and the maximum usage factor for the 28-inch downcomers is 0.0629.

The applicant dispositions the TLAAs associated with fatigue of the downcomers in accordance with 10 CFR 54.21(c)(1)(i), that the analyses remain valid for the period of extended operation.

4.6.2.2 Staff Evaluation

The staff reviewed LRA Section 4.6.2 to verify, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue analyses for the downcomers remain valid for the period of extended operation. The staff noted that fatigue analyses were provided in the application although it is not required, since the downcomers were designed to ASME Section III Class 2 for the upper portion and ASME Section III Class 3 for the lower portion. The staff reviewed the applicant's UFSAR and found in Table 3A.4.2-4 that the CUF for the 24-inch downcomer anchor is 0.0346. The staff also found in UFSAR Table 3A.4.2-5 that the CUF for the 28-inch downcomer anchor is 0.0629. In accordance with ASME Section III, CUF values must be less than 1.0. The CUF for the downcomers is significantly less than the 1.0 limit.

As described in SER Section 4.6, the staff evaluated the fatigue analysis of the primary containment for the transients listed in UFSAR Table 3A.4.1-3 and found that the fatigue analyses remain valid for the period of extended operation because the existing analyses consider more cycles than the maximum expected cycles during 60 years of operation. Since the downcomers were also designed to the aforementioned transients, the staff has determined that the fatigue analyses for the downcomers remain valid for the period of extended operation.

Based on its review, the staff finds that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue analyses for the downcomers remain valid for the period of extended operation because the number of cyclic loads assumed in their design will not be exceeded during the period of extended operation.

4.6.2.3 UFSAR Supplement

The applicant provided a UFSAR supplement summary description of its TLAA evaluation of downcomers in LRA Sections A.1.3.6.3. Based on its review of the UFSAR supplement, the staff concludes that the applicant provided an adequate summary description of its actions to address fatigue of downcomers for the period of extended operation, 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 demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that fatigue analyses for the downcomers remain valid for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.6.3 SRV Discharge Piping

4.6.3.1 Summary of Technical Information in the Application

LRA Section 4.6.3 states that each of the 18 SRVs on the main steam lines in the drywell chamber have a discharge line into the wetwell that terminates at a quencher in the suppression pool. To pass through the drywell floor, the discharge lines are routed through downcomers. The applicant also stated that the fatigue evaluation used the number of cycles presented in UFSAR Table 3A.4.1-3. The maximum fatigue usage factor for all 18 SRV discharge lines in the wetwell air volume was identified in the SER to be 0.896, below the ASME allowable limit of 1.0.

The applicant dispositions the TLAA associated with fatigue of the SRV discharge piping in accordance with 10 CFR 54.21(c)(1)(i), that the analysis remains valid for the period of extended operation.

4.6.3.2 Staff Evaluation

The staff reviewed LRA Section 4.6.3 to verify, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue analysis for SRV discharge piping remains valid for the period of extended operation. The staff reviewed Columbia UFSAR Section 3A.4.2.4.6 and found that the fatigue evaluation on all 18 SRV lines in the wetwell air volume was performed using ASME Section III, Class 1 rule (NB-3600). All 18 SRV discharge lines in the wetwell region were analyzed for appropriate load combinations and their associated number of cycles as presented on Table 3A.4.1-3, and the maximum fatigue usage factor was found to be less than the ASME allowable limit of 1.0.

As described in SER Section 4.6, the staff evaluated the fatigue analysis of the primary containment for the transients listed in UFSAR Table 3A.4.1-3 and found that the fatigue analyses remain valid for the period of extended operation because the existing analyses consider more cycles than the maximum expected cycles during 60 years of operation. Since the SRV discharge piping was also designed to the aforementioned transients, the staff has determined that the fatigue analysis for the SRV discharge piping remains valid for the period of extended operation.

Based on its review, the staff finds that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue analysis for the SRV discharge piping remains valid for the period of extended operation because the number of cyclic loads assumed in their design will not be exceeded during the period of extended operation.

4.6.3.3 UFSAR Supplement

The applicant provided a UFSAR supplement summary description of its TLAA evaluation of SRV discharge piping in LRA Sections A.1.3.6.4. Based on its review of the UFSAR supplement, the staff concludes that the applicant provided an adequate summary description of its actions to address fatigue of SRV discharge piping for the period of extended operation, 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 has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that, the fatigue analysis for the SRV discharge piping remains valid for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.6.4 Diaphragm Floor Seal

4.6.4.1 Summary of Technical Information in the Application

LRA Section 4.6.4 states that the diaphragm floor seal is located at the inside surface of the primary containment vessel periphery. The LRA describes that this seal provides a flexible, pressure tight seal between the primary containment vessel and the diaphragm floor and is capable of accommodating differential thermal expansion between them. The applicant stated that the fatigue evaluation was performed using the cycle numbers noted in Section 4.6. The maximum CUF is 0.7 per UFSAR Table 3A.4.1-5. All events are projected to remain below the containment cyclic basis from UFSAR Table 3A.4.1-3 for 60 years of operation as discussed in LRA Section 4.6.

The applicant dispositions the TLAA associated with fatigue of the diaphragm floor seal in accordance with 10 CFR 54.21(c)(1)(i), that the analysis remains valid for the period of extended operation.

4.6.4.2 Staff Evaluation

The staff reviewed LRA Section 4.6.4 to verify, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue analysis for the diaphragm floor seal remains valid for the period of extended operation. The staff reviewed the Columbia UFSAR and found that the CUF of the diaphragm floor seal in UFSAR Table 3A.4.1-5 is 0.7, which is less than the ASME allowable CUF limit of 1.0.

As described in SER Section 4.6, the staff evaluated the fatigue analysis of the primary containment for the transients listed in UFSAR Table 3A.4.1-3 and found that the fatigue analyses remain valid for the period of extended operation because the existing analyses consider more cycles than the maximum expected cycles during 60 years of operation. Since the diaphragm floor seal was also designed to the aforementioned transients, the staff has determined that the fatigue analysis for the diaphragm floor seal remains valid for the period of extended operation.

Based on its review, the staff finds that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue analysis for the diaphragm floor seal remains valid for the period of extended operation because the number of cyclic loads assumed in their design will not be exceeded during the period of extended operation.

4.6.4.3 UFSAR Supplement

The applicant provided a UFSAR supplement summary description of its TLAA evaluation of the diaphragm floor seal in LRA Sections A.1.3.6.5. Based on its review of the UFSAR supplement, the staff concludes that the applicant provided an adequate summary description of its actions to address fatigue of the diaphragm floor seal for the period of extended operation, as required by 10 CFR 54.21(d).

4.6.4.4 Conclusion

On the basis of its review, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue analysis for the diaphragm floor seal remains valid for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.6.5 Emergency Core Cooling System Suction Strainers

4.6.5.1 Summary of Technical Information in the Application

LRA Section 4.6.5 states that the original Columbia emergency core cooling system (ECCS) suction strainers were replaced with a new strainer design constructed from cold-worked austenitic stainless steel. The LRA states that a linear elastic fracture mechanics analysis was performed to bound all the martensitic material in the suction strainer screens. In this analysis, a crack depth was assumed based on the depth of the Alpha Prime martensite in the strainer screen material. The applicant stated that the fatigue crack evaluation determined that the assumed cracks will not propagate to a critical size for the remaining life of the plant. The cyclic stresses used in the analysis included direct pressure and inertial components from SRV actuation, OBE loads, and SRV steam chugging.

The applicant dispositions the TLAA associated with fatigue crack growth of the ECCS suction strainers in accordance with 10 CFR 54.21(c)(1)(i), that the analysis remains valid for the period of extended operation.

4.6.5.2 Staff Evaluation

The staff reviewed LRA Section 4.6.5 to verify, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue crack growth analysis for the ECCS suction strainers remains valid for the period of extended operation. The staff reviewed UFSAR Section 6.1.1.1.3 and NRC's safety evaluation for License Amendment 153 for Columbia Generating Station, dated May 21, 1998, and confirmed that the existing fatigue analysis is conservative because plastic deformation was not considered, and the critical flaw sizes were large compared to the thickness of the strainer material. Consideration of plastic deformation would result in larger critical flaw sizes. The existing fatigue analysis concluded that the assumed cracks for the ECCS strainers will not propogate to a critical size for the remaining life of the plant. The number of cycles during the period of extended operation is projected to remain below the cycles used in the fatigue evaluation of the containment components, including ECCS strainers, listed in UFSAR Table 3A.4.1-3. Therefore, the staff has determined that the existing analysis for the ECCS suction strainers remains valid for the period of extended operation. The applicant stated that the stress value included direct pressure and inertial components from SRV actuation, OBE loads, and SRV steam chugging. All events are projected to remain below the containment cyclic basis from UFSAR Table 3A.4.1-3 for 60 years of operation.

The staff also noted that the ECCS strainers are not ASME pressure retaining components. In addition, according to Columbia UFSAR Section 6.3.2.2.6, the strainer materials and fabrication meet ASME Section III, Class 2 requirements. Therefore, fatigue evaluation of the suction strainers is not required to be performed in accordance with ASME Code.

Based on its review, the staff finds that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue crack growth analysis for the ECCS suction strainers remains valid for the period of extended operation because the number of cyclic loads assumed in their existing evaluation will not be exceeded during the period of extended operation.

4.6.5.3 UFSAR Supplement

The applicant provided a UFSAR supplement summary description of its TLAA evaluation of the ECCS suction strainers in LRA Sections A.1.3.6.6. Based on its review of the UFSAR supplement, the staff concludes that the applicant provided an adequate summary description of its actions to address the fatigue crack growth analysis for the ECCS suction strainers for the period of extended operation, as required by 10 CFR 54.21(d).

4.6.5.4 Conclusion

On the basis of its review, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the fatigue crack growth analysis for the ECCS suction strainers will remain valid for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7 Other Plant-Specific Time Limited Aging Analyses

There are certain plant-specific safety analyses that may have been based on an explicitly assumed 40-year plant life and may, therefore, be TLAAs. Pursuant to 10 CFR 54.21(c), the applicant is required to evaluate all TLAAs.

This subsection provides the staff's review of other plant-specific TLAAs that the applicant has evaluated in the LRA.

4.7.1 Reactor Vessel Shell Indications

4.7.1.1 Summary of Technical Information in the Application

LRA Section 4.7.1 describes the analysis of two flaws in the RV shell that were identified using ultrasonic testing methods during the 2005 ISIs. According to the applicant, these flaws were "present in past inservice inspection examinations, but became rejectable under current ASME Code, Section XI, IWB-3610[a] requirements [i.e., the flaws did not pass IWB-3500 flaw screening criteria]." The applicant stated that the rejected flaws were analytically evaluated in accordance with IWB-3600 flaw evaluation criteria and determined to be acceptable for continued service without repair, as reported to the NRC in a flaw evaluation report referenced in LRA Section 4.7.1. This flaw evaluation report was submitted to the NRC by Energy Northwest letter GO2-05-153, "W. Oxenford (Energy Northwest) Letter to NRC Document Control Desk, 'Columbia Generating Station, Docket No. 50-397 Analytical Evaluation of Inservice Inspection Examination Results,'" dated September 15, 2005. The flaws were evaluated per the guidelines of ASME Code Section XI, IWB-3610, which includes acceptance criteria based on comparison of the applied stress intensity factors determined using conservative assumptions in the applied stresses compared to the material fracture toughness (K_{Ic}) values.

This evaluation calculated fatigue crack growth of 0.0064 in. at the end of 33.1 EFPY, corresponding to a 40-year RV operating life. The applicant stated that this crack growth value is insignificant in comparison to the bounding initial crack size of 0.39 in. The applicant also determined that the applied stress intensity factor (about 30 [kilo force pound per square inch-square root inches] ksi√in) is below the bounding fracture toughness value (K_{Ic}) of 63.25 ksi√in.

The applicant's flaw evaluation, referenced in LRA Section 4.7.1, used two time-limited assumptions based on the original 40-year life of the plant; this is the basis for identification of this analysis as a TLAA in the LRA. The application identifies that these time-limited assumptions are:

- The ¼ T neutron fluence for weld BG at 33.1 EFPY (5.11×10^{17} n/cm^2 (E > 1.0 MeV) at 33.1 EFPY) was used for both welds. This fluence was used to calculate the material properties of the cracked area, hence, the crack propagation. The projected ¼ T fluence for Weld BG at 54 EFPY is 8.10×10^{17} n/cm^2 (E > 1.0 MeV).
- The applicant assumed 500 significant thermal transients (SRV blowdown cycles being the worst case thermal cycle). Based on LRA Table 4.3-2, no SRV blowdown cycles are expected through 60 years of operation. Based on LRA Table 4.3-2, 409 significant thermal transients are expected (233 heatup and cooldowns, 166 scrams, and 10 HPCS actuations) through 60 years of operation.

The LRA states that, although this calculation easily meets the ASME Code, Section XI, IWB-3600, acceptance criteria for analytical evaluation of flaws, it is based on a time-limited assumption for neutron fluence that will not remain valid for the period of extended operation. The applicant stated that "[t]his indication is currently scheduled for re-inspection in 2015. Columbia will re-evaluate the indication based on the results of the 2015 inspection and either project this analysis through the period of extended operation or continue augmented inspections as required by the ASME Code, [Section XI]."

The applicant dispositioned the TLAA associated with the RV shell indications 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.

4.7.1.2 Staff Evaluation

The staff reviewed LRA Section 4.7.1 and the TLAAs for the RV flaw indications 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.

By letter dated August 3, 2010, the staff issued RAI 4.7.1-1, requesting that the applicant state (a) whether these flaws were found in weld material, in plate material adjacent to welds, or in plate material away from any weld; (b) whether these flaws were found in or near the circumferential or axial welds; and (c) the Columbia RV weld and/or plate designations (e.g., welds "BG", "BM", etc.) where the flaws were found.

By letter dated September 27, 2010, the applicant stated, in response to RAI 4.7.1-1, that the two indications are planar subsurface indications. The first indication is located in the base material adjacent to RV beltline axial weld BG. The second indication is located in non-beltline axial weld BM. The staff found the applicant's response to RAI 4.7.1-1 acceptable because the applicant provided the necessary information concerning the location of the flaws in the Columbia RV.

By letter dated August 3, 2010, the staff issued RAI 4.7.1-2, requesting that the applicant state whether any other flaws, other than the subject flaws addressed in LRA Section 4.7.1, were discovered in the RV plates, welds, or forgings that required screening in accordance with the ASME Code, Section XI, IWB-3500. If any flaws requiring screening were discovered in these components, the staff requested that the applicant state whether any of these flaws were found to be unacceptable for continued operation in accordance with IWB-3500.

In its response dated September 27, 2010, the applicant stated that the two flaws addressed in LRA Section 4.7.1 are the only two RV weld indications that required screening in accordance with the ASME Code, Section XI, Article IWB-3500. It also represents the only unacceptable flaws (per the screening criteria of IWB-3500) that have been discovered in RV shell plate or weld material at Columbia.

The staff found the applicant's response to RAI 4.7.1-2 acceptable because the applicant confirmed that the flaws addressed in LRA Section 4.7.1 are the only RV weld indications that required screening in accordance with the ASME Code, Section XI, Article IWB-3500; it also represents the only unacceptable flaws (per the screening criteria of IWB-3500) that have been discovered in RV shell plate or weld material at Columbia.

By letter dated August 3, 2010, the staff issued RAI 4.7.1-3, requesting that the applicant state whether the subject flaws addressed in LRA Section 4.7.1 are subsurface flaws (i.e., completely embedded in the RV weld, plate, or forging material) or surface-breaking flaws.

In its response dated September 27, 2010, the applicant stated that the two indications are planar subsurface indications. Both indications are approximately located at the midpoint between the RV inside diameter (ID) and outside diameter (OD) surfaces. The indication adjacent to beltline vertical weld BG has a through-wall extent of 0.39 inch, a length of 3.0 inches, and a minimum surface separation of 2.68 inches. The indication located in non-beltline weld BM has a through-wall extent of 0.38 inch, a length of 3.75 inches, and a minimum surface separation of 2.78 inches.

The staff found the applicant's response to RAI 4.7.1-3 acceptable because the applicant provided information demonstrating that the subject flaws are subsurface flaws significantly separated from the RV ID and OD surfaces.

In order to make a determination that reactor coolant pressure boundary (RCPB) components with flaws are acceptable for continued service without repair, it is necessary to ensure that any relevant flaws previously discovered in RCPB components were not produced by service-induced aging degradation during plant operation.

By letter dated August 3, 2010, the staff issued RAI 4.7.1-4, requesting that the applicant state whether the analytical flaw evaluation referenced in LRA Section 4.7.1 determined that the subject flaws were caused by serviced-induced aging degradation or whether the subject flaws were found to be fabrication defects.

In its response dated September 27, 2010, the applicant stated that both flaws are approximately located in the middle of the RV wall thickness. Service-induced flaws usually initiate on the RV ID or OD surface; it does not usually initiate within the RV wall a significant distance from the RV surfaces. An analytical flaw evaluation was performed because the flaws did not meet the ASME Code, Section XI, IWB-3500 acceptance standards. This flaw evaluation concluded that both flaws are fabrication defects and are not service-induced.

The staff found the applicant's response to RAI 4.7.1-4 acceptable because the applicant provided the requested information, confirming the flaw evaluation report conclusion that the subject flaws are fabrication defects and are not caused by service-induced aging degradation.

The applicant stated in LRA Section 4.7.1 that the subject flaws were found during ISI conducted in 2005 and that the flaws were also identified during previous ISI examinations, but "became rejectable under current ASME Section XI, IWB-3610 requirements." The staff determined that the applicant must clarify this statement concerning when the flaws were determined to be unacceptable for continued service.

By letter dated August 3, 2010, the staff issued RAI 4.7.1-5, requesting that the applicant specify the year when these flaws were determined to be rejectable. If the flaws were determined to be rejectable in 2005, the staff requested that the applicant explain why these flaws did not become rejectable until 2005, given that it was identified during previous ISI examinations.

In its response dated September 27, 2010, the applicant stated that both flaws were first indentified during inservice inspections conducted during Columbia Refueling Outage R8 (1993). Under the ASME Code, Section XI flaw recording criteria in effect during the 1993 examinations, the flaws did not require further evaluation and were determined to be acceptable in accordance with the recording criteria. When welds BG and BM were examined during the 2005 outage, the ASME Code, Section XI recording and evaluation criteria had changed. This change required recording and evaluation of flaws at a lower ultrasonic signal level. In addition to the ASME Code changes, Columbia's flaw detection techniques had improved, contributing to the change in the acceptability status of the flaws between 1993 and 2005. The response stated that the flaws will be re-inspected in 2015 in accordance with the 2001 edition and 2003 addenda of the ASME Code, Section XI.

The staff found the applicant's response to RAI 4.7.1-5 acceptable because the applicant provided satisfactory explanation of their initial detection and acceptance of the subject flaws in 1993 and the reason for the change in the ASME Code, Section XI acceptability status for these flaws between the 1993 and 2005 ISIs.

The applicant stated in LRA Section 4.7.1 that the analytical evaluation of the subject flaws used two time-limited assumptions based on the original 40-year licensed operating period for the plant (33.1 EFPY). The first time-limited assumption is based on the projected neutron fluence used in the analytical flaw evaluation. Specifically, the analytical evaluation of the subject flaws assumes that the ¼ T neutron fluence at weld BG is 5.11×10^{17} n/cm^2 ($E > 1.0$ MeV), which is a fluence value that is valid for 33.1 EFPY of facility operation. The applicant stated that this ¼ T neutron fluence value was used to calculate material properties at the flaw location for both of the subject welds where the flaws were discovered. The staff determined that further information was required concerning this first time-limited assumption.

By letter dated August 3, 2010, the staff issued RAI 4.7.1-6, requesting that the applicant (a) state why the flaw evaluation referenced in LRA Section 4.7.1 used a projected neutron fluence value for the end of the original 40-year license operating period (33.1 EFPY), as opposed to the projected fluence value for this weld that is valid for the end of the period of extended operation (54 EFPY); (b) state why the flaw evaluation referenced in LRA Section 4.7.1 did not utilize a more conservative fluence value at the RV inside diameter (ID) location for determining the fracture toughness (K_{Ic}) value, as opposed to a neutron fluence value at the ¼ T location; and (c) state why the ¼ T neutron fluence value at weld BG was used to calculate fracture toughness at the flawed region for both welds.

In its response to RAI 4.7.1-6(a), the applicant stated that the flaw evaluation report referenced in LRA Section 4.7.1 did not utilize projected neutron fluence values for 54 EFPY because at the time of the analysis (2005) the design lifetime of Columbia was only 40 years. According to the applicant, referencing a flaw evaluation report that projects flaw acceptability only through 33.1 EFPY is valid because Columbia will re-evaluate the subject flaws based on the results of the 2015 inspection and either project this analysis through the period of extended operation or continue augmented inspections of the subject welds as required by the ASME Code, Section XI. Columbia will manage the aging of these flaws using the Columbia Inservice Inspection Aging Management Program, as described in LRA Section B.2.33, through the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii). The staff found the applicant's response to RAI 4.7.1-6(a) acceptable because the applicant provided an adequate explanation for the appropriate use of a projected neutron fluence that is valid through 33.1 EFPY in the flaw evaluation and the applicant will manage the aging of the reactor vessel shell flaws using the Inservice Inspection (ISI) Program in accordance with the requirements of 10 CFR 54.21(c)(1)(iii). Since the ISI in 2015 is within the initial 40-year license period, the applicant does not need to update the analysis for the renewed license operating period until after the ISI is completed.

In its response to RAI 4.7.1-6(b), the applicant stated that the flaw evaluation report referenced in LRA Section 4.7.1 utilized a neutron fluence value at the ¼ T location because it represents a conservative estimate of the actual fluence where the flaws are located. Weld BG (a beltline weld) is nominally 6.44 inches thick, and the flaw adjacent to weld BG (in the base metal) is located 3.37 inches from the RV ID and 2.68 inches from the RV OD, with a through-wall extent of 0.39 inch. Weld BM (a non-beltline weld) is nominally 6.56 inches thick, and the flaw in weld BM is located 3.40 inches from the RV ID and 2.78 inches from the RV OD, with a through wall extent of 0.38 inch. Therefore, according to the applicant, both of the flaws start at over one-half the thickness of the RV wall from the RV ID surface.

The staff found the applicant's response to RAI 4.7.1-6(b) acceptable because the applicant adequately demonstrated that the fluence value at ¼ T location, as used in the 2005 flaw evaluation, was a conservative estimate of the actual fluence at the location of the subject flaws.

In its response to RAI 4.7.1-6(c), the applicant stated that the ¼ T neutron fluence at weld BG was used for both welds because weld BG is located in the RV beltline and is therefore, exposed to a much higher fluence than weld BM, a non-beltline weld. Rather than perform an additional fluence analysis specific to weld BM, the bounding fluence associated with weld BG was used for both welds.

The staff found the applicant's response to RAI 4.7.1-6(c) acceptable because the applicant explained that the use of the ¼ T neutron fluence value at weld BG for calculating the fracture toughness at the flawed region for both welds BG and BM was done to avoid an unnecessary fluence calculation for weld BM. Furthermore, the use of the beltline weld BG fluence value for non-beltline weld BM is conservative with respect to the evaluation of the flaw in weld BM.

The applicant's second time-limited assumption used in the flaw evaluation is the number of transient cycles used in the flaw evaluation for projecting flaw growth. Specifically, the applicant assumed 500 significant thermal transient cycles in its projection of flaw growth for the flaw evaluation. The applicant stated that, based on the 60-year projected transient cycles from LRA Table 4.3-2, only 409 significant thermal cycles are projected through the end of the period of extended operation. The staff determined that further information was required concerning this second time-limited assumption.

By letter dated August 3, 2010, the staff issued RAI 4.7.1-7 requesting that the applicant state whether the flaw evaluation referenced in LRA Section 4.7.1 analyzed plant cycles for projecting the flaw acceptability out to the end of the current 40-year license operating period or the end of period of extended operation.

In its response to RAI 4.7.1-7 dated September 27, 2010, the applicant stated that the 2005 flaw growth evaluation for the subject flaws is neither an explicit "40-year" nor "60-year" analysis because it was not explicitly based on either the 40-year design cycle or the 60-year design cycle projections. As stated in LRA Section 4.7.1, this evaluation analyzed 500 significant thermal cycles. The most limiting thermal transient is both the lifting of the safety relief valves and the 500 cycles of that transient analyzed. At the time of the analysis (2005), the most limiting thermal transient was expected to bound all other transients that would be incurred for the life of the plant. Based on the 60-year thermal cycle projections for license renewal described in LRA Section 4.3, the applicant determined that the 500 cycle assumption used for the 2005 flaw growth analysis would remain bounding for 60 years of facility operation because only 409 significant thermal transients are expected (0 safety/relief valve actuations, 233 heat-ups/cool-downs, 166 scrams, and 10 high pressure core spray actuations) for the 60-year plant operating life.

The staff found that the applicant's response to RAI 4.7.1-7 was acceptable because the applicant adequately explained, with respect to the assumed 500 thermal cycles used in the 2005 flaw growth evaluation, this flaw evaluation would remain bounding through the end of the period of extended operation, irrespective of the fact that the flaw evaluation was not explicitly based on either 40-year or 60-year cycle projections.

The Columbia site corrective action and condition reporting program documents the identification of flaws discussed in LRA Section 4.7.1 and immediate corrective actions taken to address these flaws. The NRC staff identified a site condition report, Columbia Action Request Report (AR) No. 00031237, dated August 5, 2006, documenting an indication associated with RV axial weld BM, that was determined to be unacceptable for continued service (without repair or evaluation under IWB-3600) per the ASME Code, Section XI, Table IWB-3510-1 acceptance criteria. This condition report states that "the analytical evaluation path will be followed." The date of the flaw evaluation report submittal referenced in LRA Section 4.7.1 (September 15, 2005) precedes the date of the AR (August 5, 2006).

By letter dated August 3, 2010, the staff issued RAI 4.7.1-8(a), requesting that the applicant state whether the flaw documented in AR No. 00031237 is identical to one of the two flaws discussed in LRA Section 4.7.1. If this AR addresses another unacceptable flaw not discussed in LRA Section 4.7.1, the staff requested that the applicant revise LRA Section 4.7.1 to include documentation of a TLAA for this flaw, and provide a reference for an IWB-3600 analytical evaluation for this flaw. In RAI 4.7.1-8(b), the staff requested that the applicant explain why the date of the flaw evaluation report submittal (September 15, 2005) precedes the date of AR No. 00031237 (August 5, 2006).

In its response to RAI 4.7.1-8(a) dated September 27, 2010, the applicant stated that the flaw documented in Columbia AR 00031237 is one of the two flaws documented in LRA Section 4.7.1. Columbia corrective action program (CAP) reports CR 2-05-03679, PER 205-0348, and AR 00031237 all document the same corrective action activity for the indication in weld BM. In response to RAI 4.7.1-8(b), the applicant stated that the date of the flaw evaluation report submittal (September 15, 2005) precedes the date of AR 00031237 because of a change in the Columbia CAP data base. During the move to the new database, all previous electronic CRs and PERs from the old software database were migrated to the new database and assigned new AR numbers. Therefore, the date of AR 00031237 reflects the conversion date to the new database. AR 00031237 is the conversion of the original CR for this flaw from 2005. The staff finds that the applicant's responses to RAIs 4.7.1-8(a) and 4.7.1-8(b) are acceptable because the applicant (a) confirmed that the flaw documented in AR 00031237 is the same flaw documented in LRA Section 4.7.1 for RV axial weld BM; and (b) adequately explained why the date of the flaw evaluation report submittal (September 15, 2005) precedes the date of AR 00031237 (August 5, 2006).

Regarding the statement in LRA Section 4.7.1 on planned flaw inspection and re-evaluation activities in 2015, the staff requested in RAI 4.7.1-9(a) dated August 3, 2010, that the applicant state whether the statement applies to just one of the flaws discussed in LRA Section 4.7.1 or to both flaws. In its response to RAI 4.7.1-9(a) dated September 27, 2010, the applicant stated that all RV axial welds are scheduled for re-examination during the 2015 refueling outage at Columbia, and thus the statement from LRA Section 4.7.1 applies to both of the flaws discussed in the LRA. The staff found the applicant's response to RAI 4.7.1-9(a) acceptable because the applicant clarified the statement from LRA Section 4.7.1 to indicate that both of the subject flaws will be re-examined and re-evaluated in 2015.

In RAI 4.7.1-9(b) dated August 3, 2010, the staff requested that the applicant add the flaw inspection and re-evaluation statement to the Columbia LRA Commitment Table, with respect to the status of both flaws referenced in LRA Section 4.7.1, given that the flaw evaluation referenced in LRA Section 4.7.1 will only remain valid through the end of the current 40-year licensed operating period (33.1 EFPY). In its response to RAI 4.7.1-9(b) dated September 27, 2010, the applicant stated that the cited RV axial weld inspections in 2015, including portions of RV axial welds BG and BM with the flaws, are a part of the NRC-approved ISI program for the current 10-year ISI interval, as required by 10 CFR 50.55a. The applicant added that these examinations are required by 2015, well before the beginning of the period of extended operation. The applicant stated that these examinations are required for the current 40-year license term, regardless of whether or not the Columbia operating license receives a 20-year extension. Thus, the applicant concluded in response to RAI 4.7.1-9(b) that "it is not a license renewal commitment to repeat these inspections."

In reviewing the applicant's response to RAI 4.7.1-9(b), the staff acknowledged that the RV axial welds, including the subject flaws, are required to be re-examined prior to the end of the third 10-year ISI interval at Columbia, in accordance with ASME Code, Section XI, requirements for the current 40-year license term. However, the analysis of these RV shell indications is a license renewal TLAA that has not been projected to remain in compliance with ASME Code, Section XI flaw acceptance criteria through the end of the period of extended operation. Furthermore, the Columbia ISI Aging Management Program description in LRA Section B.2.33 does not specifically address re-evaluation of existing flaws in ASME Code Class 1 components. In order to ensure that the effects of aging for these flaws will be adequately managed in accordance with the requirements of 10 CFR 54.21(c)(1)(iii), the staff believes the applicant should include a license renewal commitment to re-evaluate the subject flaws for the period of extended operation, based on the results of the 2015 inservice inspection.

By letter dated December 20, 2010, the staff issued RAI 4.7.1-10 requesting the applicant to include a license renewal commitment to re-evaluate the subject flaws for the period of extended operation (54 EFPY), in accordance with the requirements of ASME Code, Section XI, IWB-3600 based on the results of the 2015 ISI.

In its response dated January 28, 2011, the applicant included a license renewal commitment to, "Re-evaluate the portions of the reactor pressure vessel beltline welds BG and BM for the period of extended operation (54 EFPY), in accordance with the requirements of the ASME Code, Section XI, IWB-3600 based on the results of 2015 inservice inspection."

The staff finds the applicant's response to RAI 4.7.1-10 acceptable because the staff is assured that the effects of aging for the flaws will be adequately managed, in accordance with 10 CFR 54.21(c)(1)(iii), with the re-evaluation of the subject flaws for the period of extended operation during the third 10-year ISI interval in 2015. The staff's concern described in RAI 4.7.1-10 is resolved.

The staff reviewed the applicant's flaw evaluation report in order to verify that the RV will remain acceptable for continued service through 33.1 EFPY, based on the ASME Code, Section XI, IWB-3612 analytical acceptance criteria for RV components containing flaws that do not meet initial screening requirements of IWB-3500.

The staff verified that the subject flaws were correctly characterized as subsurface flaws, in accordance with IWB-3610(b) requirements, based on the measured values for the flaw depth, and the separation distance between the flaw boundaries and the RV ID and OD surfaces.

Based the review of the applicant's flaw evaluation, referenced in LRA Section 4.7.1, the staff determined that the applicant adequately demonstrated that the subject RV shell weld flaws will meet the analytical acceptance criteria of the ASME Code, Section XI, IWB-3612, through 33.1 EFPY.

The staff finds the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the RV shell indications will be adequately managed for the period of extended operation because the flaws will be managed using the Inservice Inspection Aging Management Program, as described in LRA Section B.2.33.

4.7.1.3 UFSAR Supplement

The applicant provided a UFSAR supplement summary description for the TLAA related to the RV shell indications in LRA Section A.1.3.7.1. Based on its review of the UFSAR supplement, the staff concludes concludes that the applicant provided an adequate summary description of its actions to address the RV shell indications for the period of extended operation, as required by 10 CFR 54.21(d).

4.7.1.4 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 aging related to RV shell indications will be adequately managed for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.2 Sacrificial Shield Wall

4.7.2.1 Summary of Technical Information in the Application

LRA Section 4.7.2 identifies that the sacrificial shield wall (SSW) is discussed in UFSAR Section 3.8.3.6, which states that the outside face of the SSW will experience a neutron fluence of less than $2x10^{16}$ nvt in the 40-year life expectancy of the station. The applicant noted that, for the discussion in this section, nvt is equivalent to n/cm^2 with neutron energy greater than 1 MeV. The applicant also stated that the projected fluence at the SSW outer wall for 60 years of operation, including a margin to account for a power uprate, will remain below $2x10^{16}$ nvt.

The applicant dispositions the TLAA associated with the sacrificial shield wall fluence 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 and the TLAAs for the sacrificial shield wall fluence to verify, pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis has been projected to the end of the period of extended operation.

The UFSAR identified a TLAA in Section 3.8.3.6 pertaining to neutron fluence on the outside face of the SSW. The UFSAR states that the neutron fluence remains below a threshold value, which the applicant stated was equivalent to $2x10^{16}$ n/cm^2. The staff also reviewed the 60-year fluences and finds that the license renewal related fluences meet its criterion and that the projected fluence will remain below $2x10^{16}$ nvt. Because the projected fluence does not exceed the threshold identified in the CLB, the NRC staff finds the projection acceptable.

The staff finds the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis for the sacrificial shield wall fluence has been projected to the end of the period of extended operation because the projected fluence, considering 60 years of operation and the effects of power uprate, is below the fluence identified in the UFSAR for the sacrificial shield wall.

4.7.2.3 UFSAR Supplement

LRA Section A.4.7.2 provides the UFSAR supplement for the sacrificial shield wall fluence TLAA evaluation. The staff reviewed this UFSAR supplement description of the program and notes that it conforms to the recommended description for this type of program, as described in SRP-LR Section 4.7.3.2. Based on its review of the UFSAR supplement, the staff concludes that the applicant provided an adequate summary description of its actions to address the sacrificial shield wall fluence for the period of extended operation, 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 has demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis for the SSW outer wall neutron fluence has been projected to the end of the period of extended operation. The staff also concludes that the

UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.3 Main Steam Line Flow Restrictor Erosion Analyses

4.7.3.1 Summary of Technical Information in the Application

LRA Section 4.7.3 describes the applicant's TLAA for erosion of the main steam line flow restrictors. The applicant stated that UFSAR Section 5.4.4 indicates that a main steam line flow restrictor is provided for each of the four main steam lines, using a cast stainless steel material that has excellent resistance to erosion-corrosion from high velocity steam. The applicant states that the restrictor is a complete assembly that is welded into the main steam line between the last main steam line SRV and the inboard MSIV. The applicant states that UFSAR Section 5.4.4.4 indicates that very slow erosion of the main steam line flow restrictor is expected. The applicant views erosion of the flow restrictor as a safety concern because it could impair the ability of the flow restrictor to limit vessel blowdown following a main steam line break.

The applicant dispositions the main steam line flow restrictor erosion analysis TLAA 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.

4.7.3.2 Staff Evaluation

The staff reviewed LRA Section 4.7.3 and the main steam line flow restrictor erosion analysis TLAA to verify, pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis has been projected to the end of the period of extended operation.

The staff reviewed the applicant's TLAA and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.7.3.1.2, which states that the applicant may recalculate the TLAA using a 60-year period to show that the TLAA acceptance criteria continue to be satisfied for the period of extended operation. The SRP-LR also states that the applicant may revise the TLAA by recognizing and re-evaluating any overly conservative conditions and assumptions.

In the LRA, the applicant stated that the restrictor is designed to limit coolant flow rate from the RV (before the MSIVs are closed) to less than 200 percent of normal flow in the event a main steam line break occurs outside the containment. It was further stated that the projections conclude that, after 60 years of erosion, the main steam flow restrictors will continue to perform their intended function.

The LRA did not contain information regarding the analysis that demonstrates that the choked flow will remain less than the design limit of 200 percent of normal flow in the event of a main steam line break. Continued extended wear could cause erosion that may prevent the restrictor from continuing to perform its safety function during the period of extended operation. In RAI 4.7.3-1, dated August 3, 2010, the staff requested that the applicant provide the results of the analysis that demonstrates that the main steam line flow restrictor will perform satisfactorily for the period of extended operation.

In its response dated September 27, 2010, the applicant stated that the revised analysis used to conclude that the main steam flow restrictors will continue to perform its intended function uses more realistic wear rates based on technical reports and operating experience. The applicant revised its original TLAA analysis by recognizing and re-evaluating overly conservative conditions and assumptions. The applicant reported that the environment of the main steam lines, at the location of the flow restrictors, is treated water in the form of steam with only 0.1–0.2 percent moisture. The applicant also stated that Columbia operating experience indicates that the wear rate on a carbon steel elbow upstream of the main steam line flow restrictors between refueling outages 5 and 9 was an average of 0.00091 in. per year. The applicant stated that the wear rate of the throat diameter would not be expected to exceed this value for several reasons. First, the change in flow direction in the elbow is 90° whereas the flow restrictor throat is parallel to the flow direction. Secondly, the erosion resistance of stainless steel is at least twice that of carbon steel, negating the need to double the elbow wear rate to accommodate the flow restrictor geometry. Lastly, the applicant cited inspections at Quad Cities after 30 years of operation that identified no impact erosion of the flow restrictors, even after 34 days of operation with a significant carryover of moisture due to a damaged steam dryer. The applicant indicated that virtually no water droplets exist in the steam in the main steam line, to cause erosion. As such, the applicant stated that the flow rate used in the analysis is based on a 0.003 in. per year wear rate (over three times the observed rate of 0.00091 in. per year for the carbon steel 90° elbow), which gives a 60-year maximum flow rate of 199.4 percent. This flow rate meets the design limit of 200 percent of normal flow.

The staff finds the applicant’s response acceptable because the flow rate analysis uses a conservative wear rate of 0.003 in. per year, based on technical reports and operating experience, which projects the flow rate to be 199.4 percent of normal flow, which is less than 200 percent of normal flow in the event a main steam line break occurs outside containment. Therefore, the staff’s concern described in RAI 4.7.3-1 is resolved.

The staff finds the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis for the main steam line flow restrictor analysis TLAA has been projected to the end of the period of extended operation.

Additionally, the applicant's analysis meets the acceptance criteria in SRP-LR Section 4.7.3.1.2 because the applicant revised its original TLAA analysis by recognizing and re-evaluating overly conservative conditions and assumptions; therefore, it demonstrates that the flow rate will be less than 200 percent of normal flow in the event a main steam line break occurs outside containment.

4.7.3.3 UFSAR Supplement

The applicant provided a UFSAR supplement summary description for the TLAA related to the main steam line flow restrictor erosion analysis in LRA Section A.1.3.7.3. Based on its review of the UFSAR supplement, the staff concludes that the applicant provided an adequate summary description of its actions to address the main steam line flow restrictor erosion analysis 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 for the main steam line flow restrictor erosion analysis has been projected to the end of the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.4 Core Plate Rim Hold-Down Bolts

4.7.4.1 Summary of Technical Information in the Application

In its original LRA submitted on January 19, 2010, the applicant stated that Columbia had core plate wedges installed around the periphery of the core plate within the shroud. However, in a letter dated May 6, 2011, the applicant informed the staff that Columbia had no core plate wedges, which results in the bolt inspection of BWRVIP-25 being applicable. The applicant also stated that it would deviate from the BWRVIP-25 inspection guidance because it does not plan to inspect the hold down bolts for stress relaxation due to difficulties performing the inspection.

4.7.4.2 Staff Evaluation

In a conference call and in a letter dated May 6, 2011, the applicant stated that it had discovered that there were no core plate wedges located around the periphery of the core plate within the shroud. Having no core plate wedges results in the applicant having to perform bolt inspection as described in BWRVIP-25, “BWR Core Plate Inspection and Flaw Evaluation Guidelines.” However, the applicant also stated that the nuclear industry research organization, EPRI (Electric Power Research Institute), is currently working on developing revised guidance for the core plate hold-down bolts and that the applicant would deviate from BWRVIP-25 inspection guidance, until December 31, 2015, because it does not plan to inspect the hold down bolts for cracking due to difficulties performing the inspection. The staff reviewed the applicant's submittal letter regarding its intent to deviate from BWRVIP-25 inspection guidelines and had concerns that the effects of aging will not be adequately managed without performing the inspections. This was considered and included in the core plate hold-down bolts open item, OI 4.7.4-1, in the SER with open items.

Without core plate wedges, the core plate rim hold-down bolts perform the function of preventing lateral motion of the core plate. However, core plate rim hold-down bolts are susceptible to stress relaxation and as described in the staff's license renewal SER for BWRVIP-25, dated December 7, 2000, "due to susceptibility of the rim hold-down bolts to stress relaxation, applicants referencing the BWRVIP-25 report for license renewal should identify and evaluate the projected stress relaxation as a potential TLAA issue."

By letter dated June 29, 2011, in response to RAI B.2.10-2, the applicant provided LRA Amendment 36, which includes an LRA supplement for addressing the analysis of the core plate rim hold-down bolts. Included in the LRA supplement is LRA Section 4.7.4, which describes the applicant’s TLAA for loss of preload on the core plate rim hold-down bolts.

This letter states the applicant's intent to disposition the core plate rim hold-down bolts TLAA by performing either of the following, two years prior to the period of extended operation:

(1) Install wedges to prevent lateral motion of the core plate in the event of stress relaxation of the core plate rim hold-down bolts at least two years prior to the beginning of the period of extended operation, or

(2) Submit a plant-specific TLAA addressing stress relaxation of the core plate rim hold-down bolts to the NRC for review and approval at least two years prior to the beginning of the period of extended operation. This TLAA shall analyze stress relaxation of the core plate rim hold-down bolts due to exposure of the pre-loaded bolts to neutron radiation over the life of the plant, and the analysis methods shall be consistent with the generic BWR core plate analysis specified in Appendix B of the BWRVIP-25.

In LRA Amedment 36, the applicant provided a commitment in the UFSAR Supplement (Commitment No. 71) to perform one of the two actions described above.

The staff reviewed the applicant's response and noted that the applicant had submitted a TLAA for the core plate rim hold-down bolts but had not selected one of the three options of 10 CFR 54.21(c)(1) to demonstrate its evaluation of the TLAA. Also, the applicant did not provide an AMR line item for the core plate rim hold-down bolts with the aging effect of loss of preload due to stress relaxation. Further, the applicant stated that it intended to deviate from BWRVIP-25 inspection guidelines, which could result in inadequate management of the aging effect. This issue was open item OI 4.7.4-1 in the SER with open items.

By letter dated November 4, 2011, the applicant provided LRA Amendment 44, which included revisions to all LRA sections related to the aging management and the TLAA of the core plate rim hold-down bolts. These LRA revisions were provided to address the staff's concerns identified in OI 4.7.4-1, regarding the applicant's ability to manage aging of the core plate rim hold-down bolts during the period of extended operation.

LRA Amendment 44 revised the UFSAR supplement (Section, A.1.3.7.4) and Commitment No. 71. These revisions state an intent to install core plate wedges at least two years prior to the period of extended operation unless (1) a site-specific analysis is approved by the NRC that resolves core plate bolt loss of preload due to both stress relaxation and cracking; or (2) an NRC-approved method is developed to inspect the core plate bolts for cracking and a site-specifc analysis for loss of preload due to stress relaxation of the core plate bolts is approved by the NRC.

LRA Amendment 44 also revised the TLAA identified in LRA Section 4.7.4, "Core Plate Rim Hold-Down Bolts," to (1) identify this TLAA as dispositioned consistent with 10 CFR 54.21(c)(1)(iii), and (2) identify the revised commitment related to the core plate rim hold-down bolts, consistent with the amended UFSAR supplement Section, A.1.3.7.4.

In addition to the above, the applicant also revised items for the core plate rim hold-down bolts in LRA Table 3.1.1, Item 3.1.1-44 and LRA Table 3.1.2-2 to include (a) loss of preload as an aging effect that is addressed by a TLAA, (b) cracking as an aging effect that is managed by the BWR Vessel Internals Program, and (c) cracking as an aging effect that is managed by the BWR Water Chemistry Program.

Lastly, LRA Amendment 44 modified the LRA Appendix C, Table C-2 BWRVIP-25 action item responses to address (a) the BWRVIP-25 Deviation Disposition DD-09, as it applies to BWRVIP-25 Applicant Action Item Nos. (1) and (5); and (b) the revised commitment related to the core plate rim hold-down bolts (Commitment No. 71).

The staff reviewed the applicant's revised LRA sections related to the core plate rim hold-down bolts, as provided in LRA Amendment 44, and determined that the applicant's response was acceptable because:

(1) the applicant appropriately cited the requirements of 10 CFR 54.21(c)(1)(iii) to demonstrate its evaluation of the TLAA for the core plate rim hold-down bolts;

(2) the applicant provided an acceptable UFSAR supplement and commitment (Commitment No. 71) for ensuring that core plate wedges will be installed at least two years prior to the period of extended operation, unless the NRC approves specific analyses and/or inspection methodologies that would resolve issues regarding cracking

and loss of preload due to stress relaxation for the core plate rim hold-down bolts during the period of extended operation;

(3) the applicant provided the necessary AMR line items in LRA Table 3.1.2-2 for comprehensively identifying the aging effects, TLAA, and aging management programs related to the core plate rim hold-down bolts; and

(4) the applicant appropriately identified BWRVIP-25 Deviation Disposition DD-09 and addressed Commitment No. 71, as revised, in the LRA Appendix C, Table C-2 BWRVIP-25 action item responses.

Furthermore, the staff determined that the revised UFSAR supplement and Commitment No. 71, pertaining to the installation of core plate wedges at least two years prior to the period of extended operation, ensure that BWRVIP-25 Deviation Disposition DD-09 will not represent an aging management concern for the hold-down bolts during the period of extended operation because the installation of wedges would ensure adequate lateral restraint of the core plate even if the hold-down bolts undergo a significant loss of preload. The Deviation Disposition DD-09 is scheduled to end on December 31, 2015. Before the Deviation Disposition DD-09 schedule ends, the applicant will provide to the NRC its alternative to managing the hold down bolts for cracking, which may include inspecting the hold down bolts, following the new guidance established by EPRI, or submitting a new deviation. The staff notes that additional measures may be taken between December 31, 2015, and the date of installation of the core plate wedges (two years prior to entering the period of extended operation in 2023). Therefore, the staff finds that all concerns addressed by OI 4.7.4-1 have been resolved by the LRA revisions provided in LRA Amendment 44, and thus OI 4.7.4-1 is closed.

To ensure that core plate wedges will be installed to prevent lateral motion of the core plate, the staff will issue a license condition requiring the applicant to install wedges on or before December 20, 2021. The license condition will also require the applicant to submit a report to NRC staff summarizing the results of the installation of wedges and if applicable, corrective action.

4.7.4.3 UFSAR Supplement

The applicant provided a UFSAR supplement summary description for the TLAA related to the core plate rim hold-down bolts in LRA Section A.1.3.7.4. Based on its review of the UFSAR supplement and closure of OI 4.7.4-1, the staff concludes that the applicant provided an adequate summary description of its actions to address the core plate rim hold-down bolts 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, and closure of OI 4.7.4-1, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging related to core plate rim hold-down bolts will be adequately managed for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.5 Crane Load Cycle Limit

4.7.5.1 Summary of Technical Information in the Application

LRA Section 4.7.5, as amended by letter dated October 5, 2011, and November 16, 2011, describes the applicant's TLAA for crane load cycle limit. The applicant stated that all of the cranes and hoists in-scope of license renewal were designed to CMAA 70 "Specification for Top Running and Gantry Type Multiple Girder Electric Overhead Traveling Cranes." In addition, all but one of the cranes and hoists were designed to Service Class A (standby or infrequent service) and the remaining crane (MT-HOI-40 installed in 2009) was designed to class D (heavy service), which has a higher range of load cycles than class A. The applicant provided an evaluation of the TLAA for each crane within the scope of license renewal.

The applicant stated that it analyzed the crane load cycles in accordance with 10 CFR 54.21(c)(1)(i), and that these analyses will remain valid during the period of extended operation.

4.7.5.2 Staff Evaluation

As discussed in SER Section 4.1.2.10, the SER with open items identified open item OI 4.7.5-1, related to the need for a TLAA related to load lift limits for in-scope cranes and hoists. This issue was discussed with the applicant on August 22, 2011 (teleconference summary dated September 8, 2011).

By letter dated October 5, 2011, and supplemented on November 16, 2011, the applicant addressed OI 4.7.5-1 and revised the LRA to include LRA Sections 4.7.5 and A.1.3.7.5, titled "Crane Load Cycle Limit," to identify the analyses of the TLAA associated with crane load cycle limits.

The staff reviewed LRA Section 4.7.5 and TLAA for crane load cycle limits, to verify pursuant to 10 CFR 54.21(c)(1)(i), that the analyses remain 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 states that the existing analyses should be shown to be bounding even during the period of extended operation and the reviewer should assure that the applicant's activity is sufficient to confirm the calculation assumptions for the 60-year period. The staff reviewed CMAA No. 70 and confirmed that Service Class A cranes are designed for up to 100,000 load cycles and that Service Class D cranes can be designed for up to 500,000 load cycles.

The applicant stated that for the cranes and hoists associated with the ECCS pump rooms (MT-HOI-6, 7, 8, 9 and 10), the majority of the time the cranes and hoists are used for lifts are for the removal and reinstallation of the floor plugs during outages to provide access for work. One out of the three floor plugs is removed and reinstalled on a rotational basis every outage. The staff noted that besides these work activities, the hoist and cranes are used for the removal of the associated pump or motor, which are rebuilt approximately every 8-10 years. However, these work activities have not been required during every outage, and the applicant assumed all three plugs have been and will be removed and reinstalled during each outage through the period of extended operation (36 total refueling outages).

The resultant number of estimated load cycles for each hoist is approximately 220 cycles. The applicant doubled the estimated number of load cycles (approximately 440 load cycles) which is well below the allowance of 100,000 load cycles for Service Class A. Based on the actual usage of the hoist and cranes during outages, the staff finds it conservative that the applicant assumed these cranes will be used to remove and install each floor plug during each past and future outage and accounted for unexpected crane usage during outages. The applicant stated that the hoist and cranes associated with the reactor recirculation, service water and high pressure core spray pumps (MT-HOI-16 and MT-CRA-6A/6B) have been used even less frequently than the ECCS pump hoists and are bounded by the estimated 440 cycles through the period of extended operation. The staff finds it reasonable that the estimated 440 load cycles for the ECCS pump hoists bounds the MT-HOI-16 and MT-CRA-6A/6B because it is operated less frequently and there are sufficient margins between the estimated load cycles and the design cycles (100,000 load cycles) to account for unplanned crane usage. The staff noted that the applicant's estimated use of these Service Class A cranes are based on operations that occur during refueling outages and thus are routine and predictable; therefore, the staff finds the applicant's estimates for its crane usage to be reasonable. In addition, the staff finds it conservative that the applicant considered unanticipated usage of these cranes through the period of extended operation. For MT-HOI-6, 7, 8, 9 and 10, MT-HOI-16 and MT-CRA-6A/6B the staff noted that the applicant's estimate for crane usage through the period of extended operation was no more than 0.44 percent of the 100,000 design load cycles specified in CMAA No.70 for Service Class A and finds that there is a sufficient margin to account for any unexpected crane use through the period of extended operation.

The applicant stated that the reactor building refuel floor bridge crane (MT-CRA-2) is used extensively during refueling outages and during Independent Spent Fuel Storage Installation (ISFSI) off-loading campaigns. The staff noted that the applicant anticipates a total of 110 casks through the period of extended operation and each cask requires 20-25 lifts. For the evaluation of this TLAA, the applicant treated each lift as a load cycle and assumed 25 lifts for each cask; therefore, it was determined that for ISFSI-related work there will be approximately 2750 load cycles. The applicant reviewed its reactor disassembly/reassembly procedures and noted that during each outage there are approximately 40 lifts that utilize the main hook. The staff noted this will result in approximately 1440 load cycles due to vessel disassembly/reassembly during outages through the period of extended operation. The applicant combined and doubled the load cycles for the ISFSI and vessel disassembly and reassembly to account for other potential heavy loads, which results in approximately 8380 load cycles.

The applicant stated that the turbine deck bridge crane (MT-CRA-1) is also used extensively during outages, but not as often as the MT-CRA-2, and the design of this crane is associated with loads related to the overhaul of the turbine and generator. Since the reactor building refuel floor bridge crane is used more extensively then the turbine deck bridge crane, the staff finds it reasonable that the load cycles estimated through the period of extended operation for the reactor building refuel floor bridge crane bounds the turbine building bridge crane. The staff noted that the applicant's estimated use of these Service Class A cranes are based on operations that occur during refueling outages and ISFSI-related activities, and thus are routine and predictable; therefore, the staff finds the applicant's estimates for its crane usage to be reasonable. In addition, the staff finds it conservative that the applicant considered unanticipated usage of these cranes through the period of extended operation. For MT-CRA-2 and MT-CRA-1 the staff noted that the applicant's estimate for crane usage through the period of extended operation was no more than 9 percent of the 100,000 design load cycles specified in CMAA No.70 for Service Class A and finds that there is a sufficient margin to account for any unexpected crane use through the period of extended operation.

The applicant stated that the steam tunnel hoist (MT-HOI-18) is used predominantly during refueling outages for removal of the floor plugs above the pipe (steam) tunnel and movement of valves as necessary. There are sixteen floor plugs above the tunnel, but normal outage activities do not require removal of all plugs to gain access for the scheduled work. Therefore, to obtain a conservative estimate, the applicant assumed all sixteen floor plugs have been and will be removed and reinstalled during refueling outages through the period of extended operation (total of 36 outages) and doubled that number to account for other lifts that may occur during work activities in the steam tunnel, which resulted in approximately 2304 load cycles. The staff noted that the applicant's estimated use of this Service Class A crane is based on operations that occur during refueling outages and thus are routine and predictable; therefore, the staff finds the applicant's estimates for its crane usage to be reasonable. In addition, the staff finds it conservative that the applicant considered unanticipated usage of this crane through the period of extended operation. For MT-HOI-18 the staff noted that the applicant's estimate for crane usage through the period of extended operation was no more than 2.4 percent of the 100,000 design load cycles specified in CMAA No.70 for Service Class A and finds that there is a sufficient margin to account for any unexpected crane use through the period of extended operation.

The applicant stated that the gantry crane (MT-HOI-40) of the reactor building was installed in 2009 to support outage activities starting in R-19. This crane was designed and fabricated to CMAA Service Class D, which is defined as service with 10-20 lifts per hour with loads approaching 50 percent of capacity. The staff noted that this anticipated hourly lift rate was not reached during the first two outages following installation of the crane, and the applicant does not expect this rate to be reached during future outages. However, for the evaluation of this TLAA, the applicant used a rate of 20 lifts per hour to determine the estimated lifts through the period of extended operation. Based on installation and use for the R-19 outage, the applicant determined that this crane will see service in 18 outages through the period of extended operation with an expected average duration of 35 days or less for each outage. Therefore, use of this gantry crane around the clock during these outages would result in approximately 302,400 load cycles. Based on the applicant's past usage of this crane and expected usage of this crane, the staff finds it conservative that the applicant assumed the maximum of 20 lifts per hour and that the crane would operate non-stop for the duration of an outage for every outage through the period of extended operation. The staff noted that the applicant's estimated use of this Service Class D crane is based on operations that occur during refueling outages and thus are routine and predictable; therefore, the staff finds the applicant's estimates for its crane usage to be reasonable. For MT-HOI-40, the staff noted that the applicant's estimate for crane usage through the period of extended operation was approximately 60 percent of the 500,000 design load cycles specified in CMAA No.70 for Service Class D and finds that, even with the applicant's conservative assumption on crane usage, there is a sufficient margin to account for any unexpected crane use through the period of extended operation.

The applicant stated that the three jib cranes (MT-CRA-9A/9B and MT-CRA-11) are located on the refuel floor of the reactor building and are primarily used for work activities associated with receipt of new fuel. The average number of new fuel bundles handled for an outage is approximately one third of the core; however, the applicant assumed a full core reload during each outage to account for multiple handling of bundles and other miscellaneous loads. Therefore, the applicant's evaluation for the past 20 refueling outages and 16 additional outages through the period of extended operation resulted in approximately 27,500 loads cycles for each hoist. The staff noted that the applicant's estimated use of these Service Class A cranes are based on operations that occur during receipt of new fuel and thus are routine and predictable; therefore, the staff finds the applicant's estimates for its crane usage to be reasonable. In addition, the staff finds it conservative that the applicant assumed full core reload during each outage through the period of extended operation, since approximately one third of the core is reloaded during each outage. For MT-CRA-9A/9B and MT-CRA-11 the staff noted that the applicant's estimate for crane usage through the period of extended operation was no more than 28 percent of the 100,000 design load cycles specified in CMAA No.70 for Service Class A and finds that there is a sufficient margin to account for any 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, it meets 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 and 500,000 design load cycles specified in CMAA No. 70 for Service Class A and Service Class D cranes, respectively, and these analyses bound the crane usage through the period of extended operation. The staff's concern in OI 4.7.5-1 is closed.

4.7.5.3 UFSAR Supplement

LRA Section A.1.3.7.5, as amended by letter dated November 16, 2011, provides the UFSAR supplement summarizing the TLAA for crane load cycles of all cranes in the scope of license renewal, which were designed to CMAA 70. The staff reviewed LRA Section A.1.3.7.5 consistent with the review procedures in SRP-LR Section 4.7.3.2, which states that the reviewer verifies that the applicant has provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of the TLAA.

Based on its review of the UFSAR supplement, as amended, 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).

4.7.5.4 Conclusion

On the basis of its review and closure of open item OI 4.7.5-1, 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 for the Service Class A and D cranes and hoists within the scope of license renewal that were designed to CMAA 70 remain valid for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.8 Conclusion for Time-Limited Aging Analyses

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), or
- 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 UFSAR supplements for the TLAAs and finds that, the supplements contain 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

The NRC staff issued its safety evaluation report (SER) with open items related to the renewal of operating license for Columbia Generating Station on August 30, 2011. On October 19, 2011, the applicant presented its license renewal application, and the staff presented its review findings to the ACRS Plant License Renewal Subcommittee. The staff reviewed the applicant's comments on the SER and completed its review of the license renewal application. The staff's evaluation is documented in an SER that was issued by letter dated February 28, 2012.

During the 593rd meeting of the ACRS, April 12-14, 2012, the ACRS completed its review of the Columbia license renewal application and the NRC staff's SER. The ACRS documented its findings in a letter to the Commission dated April 19, 2012. A copy of this letter is provided on the following pages of this SER Section.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

April 24, 2012

The Honorable Gregory B. Jaczko
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE COLUMBIA GENERATING STATION

Dear Chairman Jaczko:

During the 593rd meeting of the Advisory Committee on Reactor Safeguards (ACRS), April 12-14, 2012, we completed our review of the license renewal application (LRA) for the Columbia Generating Station (CGS) and the final Safety Evaluation Report (SER) prepared by the NRC staff. Our Plant License Renewal Subcommittee also reviewed this matter during its meeting on October 19, 2011. During these reviews, we had the benefit of discussions with representatives of the NRC staff and Energy Northwest (EN or the applicant). We also had the benefit of the documents referenced. This report fulfills the requirement of 10 CFR 54.25 that the ACRS review and report on all license renewal applications.

CONCLUSION AND RECOMMENDATION

1. The programs established and committed to by the applicant to manage age-related degradation provide reasonable assurance that CGS can be operated in accordance with its current licensing basis (CLB) for the period of extended operation (PEO) without undue risk to the health and safety of the public.

2. The EN application for renewal of the operating license of CGS should be approved.

BACKGROUND AND DISCUSSION

CGS is a boiling-water reactor (BWR-5) designed by General Electric with a Mark II containment. CGS is located approximately 12 miles north of Richland, WA and 3 1/2 miles west of the Columbia River, on land leased from the Department of Energy on the Hanford Nuclear Site. The licensed power output of the unit is 3,886 megawatts thermal with a gross electrical output of approximately 1,230 megawatts electric. EN has requested renewal of the CGS operating license for 20 years beyond the current license term, which expires on December 20, 2023.

In the final SER, the staff documented its review of the license renewal application and other information submitted by the applicant or obtained from the staff audits and inspection at the plant site. The staff reviewed the completeness of the applicant's identification of the structures, systems, and components (SSCs) that are within the scope of license renewal; the integrated plant assessment process; the applicant's identification of the plausible aging mechanisms associated with passive, long-lived components; the adequacy of the applicant's Aging Management Programs (AMPs); and the identification and assessment of time-limited aging analyses (TLAAs) requiring review.

In the CGS license renewal application, EN identified the SSCs that fall within the scope of license renewal. For these SSCs, the applicant performed a comprehensive aging management review. The applicant will implement 55 AMPs for license renewal, of which 35 are existing programs and 20 are new programs. The EN application either demonstrates consistency with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, or documents deviations to the approaches specified in that Report. We have reviewed the exceptions and agree with the staff that they are acceptable.

The staff conducted two license renewal audits and an inspection at CGS. The audits verified the appropriateness of the scoping and screening methodology, aging management review, and associated AMPs. The inspection verified that the license renewal requirements are being appropriately implemented. Based on the audit and inspection, the staff concluded in the final SER that the proposed activities will reasonably manage the effects of aging of SSCs identified in the application and that the intended functions of these SSCs will be maintained during the period of extended operation. We agree with these conclusions.

Closure of the Open Items from the draft SER

At the conclusion of the ACRS Plant License Renewal Subcommittee meeting on October 19, 2011, there were six open items. These were closed as follows:

High Voltage Porcelain Insulators

The applicant indicated that it would include the 230 kV post insulators at the Ashe Substation as part of the High Voltage Porcelain Insulator Program with testing every eight years and cleaning if needed.

Use of Operating Experience

The staff reviewed several aspects associated with the applicant's activities for the ongoing review of operating experience and determined that the applicant will perform the appropriate review of operating experience related to aging.

Upper-Shelf Energy (USE)

The staff had concerns that the applicant did not provide a technical basis for the unirradiated transverse USE and copper content used in the calculation of the projected USE for the N12 nozzles. The applicant provided additional information to address the staff's concerns. The copper content was acceptable to the staff because it was an appropriately conservative value from the database. The applicant also identified Charpy data from the same heat as its N12 nozzles. The data indicated that the USE in the longitudinal orientation for this heat is on the order of 230 ft-lbs or more. Based on this data, the staff found the applicant's conservative data of 62 ft-lbs to be acceptable, and that the USE for the N12 nozzles will remain greater than 50 ft-lbs at the end of vessel life in accordance with 10 CFR Part 50 Appendix G.

Metal Fatigue

The staff noted that the applicant's plant-specific configuration contains additional locations that may need to be analyzed for the effects of the reactor coolant environment other than those identified in NUREG/CR-6260. The applicant provided additional information to address the staff's concern. Based on an audit, the staff was able to verify the applicant's approach in identifying locations that can be affected by environmentally assisted fatigue.

Core Plate Rim Hold-Down Bolts

In its original LRA submitted on January 19, 2010, the applicant stated that CGS had wedges installed around the periphery of the core plate within the shroud. Subsequently, the applicant was informed by General Electric that no core plate wedges were installed. CGS confirmed this to be accurate by in-vessel inspection. Lateral restraint was instead provided by hold-down bolts. Unlike hold-down bolts, core plate wedges prevent lateral motion of the core plate and are not subject to stress relaxation. The applicant has committed to follow the guidance in BWRVIP-25, "BWR Core Plate Inspection and Flaw Evaluation Guidelines," for the analysis and inspection of the hold-down bolts, which provides a justification for operation through the current license period. The staff will issue a license condition requiring the applicant to install core plate wedges on or before December 20, 2021.

Upon discovery that the vendor design information was inaccurate, the applicant conducted a review of the extent of condition of the vessel internals that are subject to inspection according to BWRVIP guidelines. This review demonstrated that the absence of the core plate wedges was the only deviation from the documented design of the components required to be inspected.

Crane Load Cycle Limit

In the LRA, the applicant did not address TLAAs of its in-scope cranes. However, the staff determined that the analyses of the cranes meet the definition of a TLAA because the cranes have a design limit on cycles. The applicant provided additional information to address the staff's concern and identified the analyses of its cranes as TLAAs.

The staff concluded that the applicant has provided an adequate list of TLAAs. Further, the staff concluded that the applicant has met the requirements of the License Renewal Rule by demonstrating that the TLAAs will remain valid for the PEO, or that the TLAAs have been projected to the end of the PEO, or that the aging effects will be adequately managed for the PEO.

The staff has concluded that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the PEO, as required by 10 CFR 54.21(a)(3). We concur with this conclusion.

We agree with the staff that there are no issues related to the matters described in 10 CFR 54.29(a)(1) and (a)(2) that preclude renewal of the operating license for CGS. The programs established and committed to by EN provide reasonable assurance that the CGS can be operated in accordance with its current licensing basis for the PEO without undue risk to the health and safety of the public. The EN application for renewal of the operating license for CGS should be approved.

Sincerely,

/RA/

J. Sam Armijo
Chairman

REFERENCES

1. NRC Safety Evaluation Report Related to the License Renewal of Columbia Generating Station, February 2012 (ML12059A357).

2. Safety Evaluation Report with Open Items Related to the License Renewal of Columbia Generating Station, August 2011 (ML11349A022).

3. Columbia Generating Station License Renewal Application, January 19, 2010 (ML100250656).

4. Energy Northwest Letter, Columbia Generating Station License Renewal Application First Annual Update, July 16, 2010 (ML102090559).

5. NRC Letter, NRC Scoping and Screening Audit Report Regarding the Columbia Generating Station License Renewal Application, August 19, 2010 (ML102160357).

6. NRC Letter, Columbia Generating Station NRC License Renewal Inspection Report 05000397/2010007, December 17, 2010 (ML103540496).

7. NRC Letter, Audit Report Regarding the Columbia Generating Station License Renewal Application, January 21, 2011 (ML102450757).

8. NRC Letter, Audit Report on the Metal Fatigue Calculations in the Columbia Generating Station License Renewal Application, February 16, 2012 (ML12033A058).

SECTION 6

CONCLUSION

The staff of the U.S. Nuclear Regulatory Commission (NRC) (the staff) reviewed the license renewal application (LRA) for Columbia Generating Station (Columbia) in accordance with NRC regulations and NUREG-1800, Revision 1, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” (SRP-LR), dated September 2005. 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 notes that the requirements of 10 CFR Part 51, Subpart A are documented in Supplement 47 to NUREG-1437, “Generic Environmental Impact Statement for License Renewal of Nuclear Plants (GEIS) Regarding Columbia Generating Station.”

APPENDIX A

COLUMBIA GENERATING STATION LICENSE RENEWAL COMMITMENTS

During the review of the Columbia Generating Station (Columbia) license renewal application (LRA) by the staff of the U.S. Nuclear Regulatory Commission (NRC) (the staff), Energy Northwest (EN) (applicant) made commitments related to Aging Management Programs (AMPs) to manage aging effects of structures and components (SCs) prior to the period of extended operation. The following table lists these commitments, along with the implementation schedules and the sources for each commitment.

Table A-1. Columbia License Renewal Commitments

Item Number	Commitment	Updated Final Safety Analysis Report (UFSAR) Supplement Section/LRA Section	Enhancement or Implementation Schedule	Source
1) Aboveground Steel Tanks Inspection	The Aboveground Steel Tanks Inspection is a new program. The Aboveground Steel Tanks Inspection detects and characterizes the conditions on the bottom surfaces of the condensate storage tanks. The program provides direct evidence as to whether, and to what extent, the relevant effects of aging have occurred in inaccessible areas.	A.1.2.1	Within the 10-year period prior to the period of extended operation. Then ongoing.	LRA Appendix B.2.1 January 19, 2010 Columbia Letter GO2-10-173 RAI B.2.1-3 December 7, 2010
2) Air Quality Sampling Program	The Air Quality Sampling Program is an existing program that will be continued for the period of extended operation.	A.1.2.2	Ongoing	LRA Appendix B.2.2 January 19, 2010
3) Appendix J Program	The Appendix J Program is an existing program that will be continued for the period of extended operation.	A.1.2.3	Ongoing	LRA Appendix B.2.3 January 19, 2010
4) Bolting Integrity Program	The Bolting Integrity Program is an existing program that will be continued for the period of extended operation.	A.1.2.4	Ongoing	LRA Appendix B.2.4 January 19, 2010
5) Buried Piping and Tanks Inspection Program	The Buried Piping and Tanks Inspection Program is an existing program that will be continued for the period of extended operation, with the following enhancements: • revise the site program document to include: – cracking, loss of material and loss of pre-load of bolting as aging effects managed by the program. – loss of material for (buried) stainless steel piping and piping components as an aging effect managed by the program. – components that are located underground (below grade) in areas, such as outdoor vaults, valve pits and inside guard pipes where access for inspection is restricted. – buried concrete and polymeric piping to confirm the absence of significant aging effect. • revise the site program document to: – confirm (prior to the period of extended operation) that all portions of buried piping and components, except for the diesel fuel oil system, in the scope of license renewal are provided with cathodic	A.1.2.5	Enhancement prior to the period of extended operation. Then ongoing.	LRA Appendix B.2.5 January 19, 2010 Columbia Letter GO2-10-094 First Annual Update July 16, 2010 Columbia Letter GO2-10-179 RAI 3.4.2.3-1 December 21, 2010 Columbia Letter GO2-11-028 RAI B.2.5-2

<table>
<tr><th>Item Number</th><th>Commitment</th><th>Updated Final Safety Analysis Report (UFSAR) Supplement Section/LRA Section</th><th>Enhancement or Implementation Schedule</th><th>Source</th></tr>
<tr><td></td><td>protection through the period of extended operation.
– ensure that cathodic protection is operable at least 90% of the time between inspections during the period of extended operation.
– ensure that deviations from cathodic protection criteria are less than 90 days duration during the period of extended operation.
• revise the site program document to:
– require that inspection of a representative sample of each buried and underground piping material and buried tank be performed within the 10-year period prior to entering the period of extended operation (i.e., between year 30 and end of year 40) and in each 10 year interval of the period of extended operation (i.e., between year 40 and year 50, and again between year 50 and year 60) per the following table:
<table>
<tr><th rowspan="2">Material/Location (buried or underground)</th><th colspan="2">Number or percentage of Inspections (per 10-Year interval through end of the period of extended operation)</th></tr>
<tr><th>Code-Class/ Safety-Related/ Other</th><th>Hazmat</th></tr>
<tr><td>Polymer/Buried</td><td>1</td><td>--</td></tr>
<tr><td>Concrete/Buried</td><td>1</td><td>--</td></tr>
<tr><td>Stainless Steel/Buried</td><td>1</td><td>--</td></tr>
<tr><td>Steel/Buried</td><td>1</td><td>2%</td></tr>
<tr><td>Stainless Steel/Underground</td><td>1</td><td>--</td></tr>
<tr><td>Steel/Underground</td><td>2</td><td>2%</td></tr>
<tr><td>Steel Tank/Buried</td><td>1</td><td>--</td></tr>
</table></td><td></td><td></td><td>January 28, 2011</td></tr>
</table>

Item Number	Commitment	Updated Final Safety Analysis Report (UFSAR) Supplement Section/LRA Section	Enhancement or Implementation Schedule	Source
	– require appropriate tactile (e.g., manual) examination of buried polymeric components to supplement visual inspections for confirmation that significant aging effect are not occurring. – require wall thickness measurement by a non-destructive examination technique such as ultrasonic testing (UT) and results documentation for further evaluation, if loss of material has been detected. – require confirmation that backfill is acceptable with regards to degradation of pipe coatings, and thereby, meets the objectives of NACE SP0169-2007. – include collection of trending information on cathodic protection system effectiveness (e.g., potential difference and current measurements) and adjustment of the program as needed based on the results. – include trending of the external surface condition or coating condition of buried and underground piping, piping components and buried tanks and adjustment of the program as needed based on the results. • revise the site program document for acceptance criteria associated with the inspections: – criteria for soil-to-pipe potential as listed in NACE Standard SP0169-2007 – backfill is acceptable with regard to degradation of pipe external coatings and, thereby, meets the objectives of NACE Standard SP0169-2007 – for coated piping or tanks, either no evidence of coating degradation or the type and extent of coating degradation determined to be insignificant as evaluated by an individual with the qualification to evaluate coatings. – If coated or uncoated metallic piping show evidence of corrosion, the remaining wall thickness in the affected area is determined to ensure that the minimum wall thickness is maintained.			

Item Number	Commitment	Updated Final Safety Analysis Report (UFSAR) Supplement Section/LRA Section	Enhancement or Implementation Schedule	Source
	– cracking or blistering of polymeric piping is evaluated. – concrete piping may exhibit minor cracking and spalling provided there is no evidence of leakage or exposed rebar or reinforcing “hoop” bands.			
6) BWR Feedwater Nozzle Program	The BWR Feedwater Nozzle Program is an existing program that will be continued for the period of extended operation.	A.1.2.6	Ongoing	LRA Appendix B.2.6 January 19, 2010
7) BWR Penetrations Program	The BWR Penetrations Program is an existing program that will be continued for the period of extended operation.	A.1.2.7	Ongoing	LRA Appendix B.2.7 January 19, 2010
8) BWR Stress Corrosion Cracking Program	The BWR Stress Corrosion Cracking Program is an existing program that will be continued for the period of extended operation.	A.1.2.8	Ongoing	LRA Appendix B.2.8 January 19, 2010
9) BWR Vessel ID Attachment Welds Program	The BWR Vessel ID Attachment Welds Program is an existing program that will be continued for the period of extended operation.	A.1.2.9	Ongoing	LRA Appendix B.2.9 January 19, 2010
10) BWR Vessel Internals Program	The BWR Vessel Internals Program is an existing program that will be continued for the period of extended operation.	A.1.2.10	Ongoing	LRA Appendix B.2.10 January 19, 2010
11) BWR Water Chemistry Program	The BWR Water Chemistry Program is an existing program that will be continued for the period of extended operation.	A.1.2.11	Ongoing	LRA Appendix B.2.11 January 19, 2010
12) Chemistry Program Effectiveness Inspection	The Chemistry Program Effectiveness Inspection is a new activity. The Chemistry Program Effectiveness Inspection detects and characterizes the condition of materials in representative low flow and stagnant areas of systems with water chemistry controlled by the BWR Water Chemistry Program or the Closed Cooling Water Chemistry Program, and with fuel oil chemistry controlled by the Fuel Oil Chemistry Program. The inspection provides direct evidence as to whether, and to what extent, the relevant effects of aging have occurred.	A.1.2.12	Within the 10- year period prior to the period of extended operation.	LRA Appendix B.2.12 January 19, 2010
13) Closed Cooling Water Chemistry	The Closed Cooling Water Chemistry Program is an existing program that will be continued for the period of extended operation, with the	A.1.2.13	Enhancement prior to the period of extended	LRA Appendix B.2.13 January 19, 2010

Item Number	Commitment	Updated Final Safety Analysis Report (UFSAR) Supplement Section/LRA Section	Enhancement or Implementation Schedule	Source
Program	following enhancement: • Ensure that at least one additional reactor closed cooling water corrosion rate measurement is performed and evaluated prior to entering the period of extended operation to provide direct information as to the effectiveness of the chemical treatments. If necessary, based on the results, establish a frequency for subsequent measurements.		operation. Then ongoing.	
14) Cooling Units Inspection Program	The Cooling Units Inspection Program is a new program. The Cooling Units Inspection Program manages the effects of loss of material of aluminum, steel, copper alloy, and stainless steel cooling unit components that are exposed to condensation. The inspection also manages the effects of a reduction in heat transfer due to fouling of heat exchanger tubes and fins and cracking due to SCC of aluminum components exposed to condensation. The Cooling Units Inspection Program consists of baseline inspections prior to the period of extended operation followed by opportunistic inspections during the period of extended operation. Following the baseline inspection, inspection findings will be reviewed periodically to ensure that each material exposed to condensation has been examined via opportunistic inspection or actions are taken to ensure inspections are performed. Initial interval for review of inspection findings is 5 years and may be adjusted based on operating experience.	A.1.2.14	Implementation prior to the period of extended operation and initial inspection within the 10-year period prior to the period of extended operation. Then ongoing.	LRA Appendix B.2.14 January 19, 2010 Columbia Letter GO2-11-025 RAI B.2.14-1 January 28, 2011
15) CRDRL Nozzle Program	The CRDRL Nozzle Program is an existing program that will be continued for the period of extended operation.	A.1.2.15	Ongoing	LRA Appendix B.2.15 January 19, 2010
16) Diesel Starting Air Inspection	The Diesel Starting Air Inspection is a new activity. The Diesel Starting Air Inspection detects and characterizes the condition of materials for the DSA system air dryers and downstream piping and components (excluding the DSA system air receivers). The inspection provides direct evidence as to whether, and to what extent, the relevant effects of aging have occurred.	A.1.2.16	Within the 10- year period prior to the period of extended operation.	LRA Appendix B.2.16 January 19, 2010

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17) Diesel Systems Inspection Program	The Diesel Systems Inspection Program is a new program. The Diesel Systems Inspection Program manages the effects of loss of material due to corrosion and cracking due to stress corrosion cracking of materials for the interior of the steel and stainless steel exhaust piping components for the Division 1, 2, and 3 diesels in the diesel engine exhaust system, including the loop seal drains from the exhaust piping. The Diesel Systems Inspection Program consists of baseline inspections prior to the period of extended operation followed by opportunistic inspections during the period of extended operation. Following the baseline inspection, inspection findings will be reviewed periodically to ensure that each material exposed to air-outdoor and raw water has been examined via opportunistic inspection or actions are taken to ensure inspections are performed. Initial interval for review of inspection findings is 5 years and may be adjusted based on operating experience.	A.1.2.17	Implementation prior to the period of extended operation and initial inspection within the 10-year period prior to the period of extended operation. Then ongoing.	LRA Appendix B.2.17 January 19, 2010 Columbia Letter GO2-10-094 First Annual Update July 16, 2010 Columbia Letter GO2-11-025 RAI B.2.14-1 January 28, 2011 Columbia Letter GO2-11-185 RAI B.2.14-1 November 16, 2011
18) Diesel-Driven Fire Pumps Inspection Program	The Diesel-Driven Fire Pumps Inspection Program is a new program. The Diesel-Driven Fire Pumps Inspection Program manages the effects of loss of material, due to corrosion or erosion, and reduction in heat transfer of the interior of the fire protection system diesel engine exhaust piping, and of fire protection system diesel heat exchangers exposed to a raw water environment. The program also manages cracking due to SCC of susceptible materials. The Diesel-Driven Fire Pumps Inspection Program consists of baseline inspections prior to the period of extended operation followed by opportunistic inspections during the period of extended operation. Following the baseline inspection, inspection findings will be reviewed periodically to ensure that each material exposed to air-outdoor and raw water has been examined via opportunistic inspection or actions are taken to ensure inspections are performed. Initial interval for review of inspection findings is 5 years and may be adjusted based on operating experience.	A.1.2.18	Implementation prior to the period of extended operation and initial inspection within the 10-year period prior to the period of extended operation. Then ongoing.	LRA Appendix B.2.18 January 19, 2010 Columbia Letter GO2-11-025 RAI B.2.14-1 January 28, 2011 Columbia Letter GO2-11-185 RAI B.2.14-1 November 16, 2011

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19) Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements Program	The Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements Program is a new program. The Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements Program is an inspection program that detects degradation of electrical cables and connections that are not environmentally qualified and are within the scope of license renewal. The program provides for the periodic visual inspection of accessible, non-environmentally qualified cables and connections in order to determine if age-related degradation is occurring, particularly in plant areas with adverse localized environments.	A.1.2.19	Implementation prior to the period of extended operation. Then ongoing.	LRA Appendix B.2.19 January 19, 2010
20) Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements Used in Instrumentation Circuits Program	The Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements Used in Instrumentation Circuits Program is a new program. The Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements Used in Instrumentation Circuits Program is a monitoring program that detects degradation of electrical cables and connections that are not environmentally qualified and used in circuits with sensitive, low-current applications. The program provides for a review of calibration records for low-current instruments, in order to detect and identify degradation of the cable system insulation resistance. The program retains the option to perform direct cable testing.	A.1.2.20	Implementation prior to the period of extended operation. Then ongoing.	LRA Appendix B.2.20 January 19, 2010
21) Electrical Cable Connections Not Subject to 10 CFR 50.49 EQ Requirements Inspection	The Electrical Cable Connections Not Subject to 10 CFR 50.49 EQ Requirements Inspection is a new activity. The Electrical Cable Connections Not Subject to 10 CFR 50.49 EQ Requirements Inspection detects and characterizes the material condition of metallic electrical connections within the scope of license renewal. The inspection uses thermography (augmented by contact resistance testing) to detect loose or degraded connections that lead to increased resistance for a representative sample of metallic electrical connections in various plant locations.	A.1.2.21	Within the 10- year period prior to the period of extended operation.	LRA Appendix B.2.21 January 19, 2010
22) EQ Program	The EQ Program is an existing program that will be continued for the period of extended operation.	A.1.2.22 A.1.3.5	Ongoing	LRA Appendix B.2.22 January 19, 2010
23) External Surfaces Monitoring	The External Surfaces Monitoring Program is an existing program that will be continued for the period of extended operation, with the following	A.1.2.23	Enhancement prior to the period of extended	LRA Appendix B.2.23 January 19, 2010

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Program	enhancements: • add aluminum, copper alloy, copper alloy >15 % Zn, gray cast iron, stainless steel (including CASS), and elastomers to the program scope. • add cracking as an aging effect for aluminum components. • add visual (VT-1 or equivalent) or volumetric examination techniques to detect cracking. • add hardening and loss of strength as aging effects for elastomer-based mechanical sealants in HVAC systems. • add physical examination techniques in addition to visual inspection to detect hardening and loss of strength for elastomer-based mechanical sealants in HVAC systems.		operation. Then ongoing.	Columbia Letter GO2-10-094 First Annual Update July 16, 2010 Columbia Letter GO2-11-025 RAI B.2.14-1 January 28, 2011
24) Fatigue Monitoring Program	The Fatigue Monitoring Program is an existing program that will be continued for the period of extended operation, with the following enhancements: • Columbia has analyzed the effects of the reactor coolant environment on fatigue for the six locations recommended by NUREG\CR-6260. These analyses are based on the projected cycles for 60 years of operation (plus some conservatism) rather than the original design cycles in FSAR Table 3.9-1. The Fatigue Monitoring Program will be enhanced to ensure that action will be taken when the lowest number of analyzed cycles is approached. – For each location that may exceed a CUF of 1.0 (due to projected cycles exceeding analyzed, or due to as-yet undiscovered industry issues), the Fatigue Monitoring Program will implement one or more of the following: (1) Refine the fatigue analyses to determine valid CUFs less than 1.0, (2) Manage the effects of aging due to fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC, or (3) Repair or replace the affected locations before exceeding a CUF of 1.0. • Correlate information relative to fatigue monitoring and provide more definitive verification that the transients monitored and their limits are consistent with or bound the FSAR and the supporting fatigue analyses, including the environmentally-assisted fatigue analyses.	A.1.2.24 A.1.3.2 A.1.3.4	Enhancement prior to the period of extended operation. Then ongoing.	LRA Appendix B.2.24 January 19, 2010

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25) Fire Protection Program	The Fire Protection Program is an existing program that will be continued for the period of extended operation.	A.1.2.25	Ongoing	LRA Appendix B.2.25 January 19, 2010

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26) Fire Water Program	The Fire Water Program is an existing program that will be continued for the period of extended operation, with the following enhancements: • perform either ultrasonic testing or internal visual inspection of representative portions of above ground fire protection piping that are exposed to water, but do not normally experience flow, after the issuance of the renewed license, but prior to the end of the current operating term and at reasonable intervals thereafter, based on engineering review of the results. • either replace sprinkler heads that have been in place for 50 years or submit representative samples to a recognized laboratory for field service testing in accordance with NFPA 25 recommendations. Perform subsequent replacement or field service testing of representative samples at 10-year intervals thereafter or until there are no sprinkler heads installed that will reach 50 years of service life during the period of extended operation. • perform hardness testing (or equivalent) of the sprinkler heads as part of their NFPA sampling, to determine whether loss of material due to selective leaching is occurring. • perform ultrasonic testing of representative portions of above ground fire protection piping that is exposed to flowing water during periodic tests, and susceptible to a loss of material due to erosion (wall thinning). • perform visual inspection of a representative sample of copper alloy >15% Zn fire protection components exposed to water for evidence of cracking (presence of ammonia) within the 10-year period prior to entering the period of extended operation (i.e., between year 30 and 40). • perform an additional visual inspection of a representative sample of copper alloy >15% Zn fire protection component exposed to water for evidence of cracking (presence of ammonia) within the 10-year period after entering the period of extended operation (i.e., between year 40 and 50). • address loss of material due to cavitation erosion with activities such as scheduled inspections of the fire protection piping locations that have had indications of cavitation erosion in the past.	A.1.2.26	Enhancement prior to the period of extended operation. Then ongoing.	LRA Appendix B.2.26 January 19, 2010 Columbia Letter GO2-10-117 RAI B.2.26-1 August 19, 2010 Columbia Letter GO2-11-029 RAI B.2.26-6 January 28, 2011

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27) Flexible Connection Inspection Program	The Flexible Connection Inspection Program is a new program. The Flexible Connection Inspection Program manages degradation, including the effects of loss of material due to wear and hardening and loss of strength of elastomer components exposed to treated water, dried air, gas, and indoor air environments. The program consists of base line inspections prior to the period of extended operation followed by opportunistic inspections during the period of extended operation. Following the baseline inspection, inspection findings will be reviewed periodically to ensure that each material and environment combination has been examined via opportunistic inspection or actions are taken to ensure inspections are performed. Initial interval for review of inspection findings is 5 years and may be adjusted based on operating experience.	A.1.2.27	Implementation prior to the period of extended operation and initial inspection within the 10-year period prior to the period of extended operation. Then ongoing.	LRA Appendix B.2.27 January 19, 2010 Columbia Letter GO2-11-025 RAI B.2.14-1 January 28, 2011
28) Flow-Accelerated Corrosion (FAC) Program	The Flow-Accelerated Corrosion (FAC) Program is an existing program that will be continued for the period of extended operation, with the following enhancements: • add the containment nitrogen system components supplied with steam from the auxiliary steam system to the scope of the program. • Add gray cast iron as a material identified as susceptible to FAC.	A.1.2.28	Enhancement prior to the period of extended operation. Then ongoing.	LRA Appendix B.2.28 January 19, 2010
29) Fuel Oil Chemistry Program	The Fuel Oil Chemistry Program is an existing program that will be continued for the period of extended operation.	A.1.2.29	Ongoing	LRA Appendix B.2.29 January 19, 2010
30) Heat Exchangers Inspection	The Heat Exchangers Inspection is a new activity. The Heat Exchangers Inspection detects and characterizes the surface conditions with respect to fouling of heat exchangers and coolers that are in the scope of the inspection and exposed to treated water, closed cooling water, or indoor air. The inspection provides direct evidence as to whether, and to what extent, the relevant effects of aging have occurred.	A.1.2.30	Within the 10- year period prior to the period of extended operation	LRA Appendix B.2.30 January 19, 2010
31) High-Voltage Porcelain Insulators Aging Management Program	The High-Voltage Porcelain Insulators Aging Management Program is an existing program that will be continued for the period of extended operation, with the following enhancement: For the in-scope station post insulators located at the Ashe substation, add testing for contamination, and cleaning if required, every 8 years.	A.1.2.31	Enhancement prior to the period of extended operation. Then ongoing.	LRA Appendix B.2.31 January 19, 2010 Columbia Letter GO2-11-139

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				RAI B.2.31-1 August 18, 2011
32) Inaccessible Power Cables Not Subject to 10 CFR 50.49 EQ Requirements Program	The Inaccessible Power Cables Not Subject to 10 CFR 50.49 EQ Requirements Program is a new program. The Inaccessible Power Cables Not Subject to 10 CFR 50.49 EQ Requirements Program manages the aging of inaccessible medium-voltage and lower service voltage cables that are not environmentally qualified and are within the scope of license renewal. The program provides for testing to identify the conditions of the conductor insulation, and also provides for periodic inspection and drainage (if necessary) of electrical manholes. The frequency of the cable testing portion of the program will be once every 6 years, with the first test to be performed prior to the period of extended operation. The frequency of the manhole inspections will be at least annually, with the first inspections to be performed prior to the period of extended operation. The scope of the AMP will be enhanced to include inaccessible power cables (400V to 2kV), in response to industry and plant-specific operating experience.	A.1.2.32	Implementation prior to the period of extended operation. Then ongoing.	LRA Appendix B.2.32 January 19, 2010 Columbia Letter GO2-10-173 RAI B.2.32-4 December 7, 2010
33) Inservice Inspection (ISI) Program	The Inservice Inspection (ISI) Program is an existing program that will be continued for the period of extended operation.	A.1.2.33	Ongoing	LRA Appendix B.2.33 January 19, 2010
34) Inservice Inspection (ISI) Program – IWE	The Inservice Inspection (ISI) Program – IWE is an existing program that will be continued for the period of extended operation.	A.1.2.34	Ongoing	LRA Appendix B.2.34 January 19, 2010
35) Inservice Inspection (ISI) Program – IWF	The Inservice Inspection (ISI) Program - IWF is an existing program that will be continued for the period of extended operation.	A.1.2.35	Ongoing	LRA Appendix B.2.35 January 19, 2010
36) Lubricating Oil Analysis Program	The Lubricating Oil Analysis Program is an existing program that will be continued for the period of extended operation, with the following enhancements: – Include the following fire protection system components that are exposed to lubricating oil within the scope of the program: (1) fire protection diesel engine heat exchangers (lube oil coolers), (2) fire protection diesel engine lube oil piping, and (3) fire protection diesel engine lube oil pump casings.	A.1.2.36	Enhancement prior to the period of extended operation. Then ongoing.	LRA Appendix B.2.36 January 19, 2010

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37) Lubricating Oil Inspection	The Lubricating Oil Inspection is a new activity. The Lubricating Oil Inspection detects and characterizes the condition of materials in systems and components for which the Lubricating Oil Analysis Program is credited with aging management. The inspection provides direct evidence as to whether, and to what extent, the relevant effects of aging have occurred.	A.1.2.37	Within the 10- year period prior to the period of extended operation.	LRA Appendix B.2.37 January 19, 2010
38) Masonry Wall Inspection	The Masonry Wall Inspection is an existing program that will be continued for the period of extended operation, with the following enhancements: • Specify that for each masonry wall, the extent of observed masonry cracking or degradation of steel edge supports and bracing are 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).	A.1.2.38	Enhancement prior to the period of extended operation. Then ongoing.	LRA Appendix B.2.38 January 19, 2010 Columbia Letter GO2-10-094 First Annual Update July 16, 2010
39) Material Handling System Inspection Program	The Material Handling System Inspection Program is an existing program that will be continued for the period of extended operation.	A.1.2.39	Ongoing	LRA Appendix B.2.39 January 19, 2010 Columbia Letter GO2-10-094 First Annual Update July 16, 2010
40) Metal-Enclosed Bus Program	The Metal-Enclosed Bus Program is a new program. The Metal-Enclosed Bus Program is an inspection program that detects degradation of metal-enclosed bus within the scope of license renewal. The program provides for the visual inspection of interior sections of bus, and an inspection of the elastomeric seals at the joints of the duct sections. The program also makes provision for thermographic inspection of bus bolted connections. The thermography portion of the program will be performed once every 10 years, with the initial inspections to be performed prior to the period of extended operation. The visual inspection portion of the program will also be performed once every 10 years, with the first inspections to be performed prior to the period of extended operation. Infra-red window installation at bolted connections of the in-scope bus will be completed prior to the initial	A.1.2.40	Implementation prior to the period of extended operation. Then ongoing.	LRA Appendix B.2.40 January 19, 2010 Columbia Letter GO2-11-020 RAI B.2.40-1 January 27, 2011

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	thermography inspection, discussed above.			
41) Monitoring and Collection Systems Inspection Program	The Monitoring and Collection Systems Inspection Program is a new program. The Monitoring and Collection Systems Inspection Program manages the effects of loss of material due to corrosion or erosion for the internal surfaces of subject mechanical components that are exposed to equipment or area drainage water and other potential contaminants and fluids. The program also manages cracking due to stress corrosion cracking of susceptible materials. The program consists of baseline inspections prior to the period of extended operation followed by opportunistic inspections during the period of extended operation. Following the baseline inspection, inspection findings will be reviewed periodically to ensure that each material and environment combination has been examined via opportunistic inspection or actions are taken to ensure inspections are performed. Initial interval for review of inspection findings is 5 years and may be adjusted based on operating experience.	A.1.2.41	Implementation prior to the period of extended operation and initial inspection within the 10-year period prior to the period of extended operation. Then ongoing.	LRA Appendix B.2.41 January 19, 2010 Columbia Letter GO2-11-025 RAI B.2.14-1 January 28, 2011
42) Open-Cycle Cooling Water Program	The Open-Cycle Cooling Water Program is an existing program that will be continued for the period of extended operation, with the following enhancements: • Address loss of material due to cavitation erosion (for the standby service water (SW), circulating water (CW), plant service water (TSW), and tower make-up (TMU) systems) with activities such as opportunistic inspections of portions of the systems that have had indications of cavitation erosion in the past. • Include the non-safety related components within the license renewal scope in the SW, CW, TSW, and TMU systems, and the non-safety related components served by or connected to the TSW system that are in the process sampling, process sampling radioactive, radwaste building mixed air, radwaste building return air, reactor building return air, and reactor closed cooling water systems.	A.1.2.42	Enhancement prior to the period of extended operation. Then ongoing.	LRA Appendix B.2.42 January 19, 2010
43) Potable Water Monitoring Program	The Potable Water Monitoring Program is an existing program that will be continued for the period of extended operation, with the following enhancements:	A.1.2.43	Enhancement and inspection within the 10-year period prior to the period	LRA Appendix B.2.43 January 19, 2010

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	• Include periodic inspection activities for evidence of a loss of material, or to confirm lack thereof. Based on operating experience, it is necessary that inspections be conducted at least once every five years, and include components of the potable cold water and potable hot water systems that are located in the reactor building, and components associated with the reactor building outside air (ROA) air washer (ROA-AW-1), including the air washer housing. • Include engineering evaluation of inspection results and adjustment of inspection frequencies. At least one inspection will be conducted within the 10-year period prior to the period of extended operation.		of extended operation. Then ongoing.	Columbia Letter GO2-10-124 RAI B.2.43 August 26, 2010
44) Preventive Maintenance – RCIC Turbine Casing	The Preventive Maintenance – RCIC Turbine Casing is an existing program that will be continued for the period of extended operation.	A.1.2.44	Ongoing	LRA Appendix B.2.44 January 19, 2010
45) Reactor Head Closure Studs Program	The Reactor Head Closure Studs Program is an existing program that will be continued for the period of extended operation.	A.1.2.45	Ongoing	LRA Appendix B.2.45 January 19, 2010
46) Reactor Vessel Surveillance Program	The Reactor Vessel Surveillance Program is an existing program that will be continued for the period of extended operation.	A.1.2.46	Ongoing	LRA Appendix B.2.46 January 19, 2010
47) Selective Leaching Inspection	The Selective Leaching Inspection is a new activity. The Selective Leaching Inspection detects and characterizes the conditions on internal and external surfaces of subject components exposed to raw water, treated water, fuel oil, soil, and moist air (including condensation) environments. The inspection provides direct evidence through a combination of visual examination and hardness testing, or NRC-approved alternative, as to whether, and to what extent, the relevant effects of aging have occurred.	A.1.2.47	No earlier than 5 year prior to the period of extended operation.	LRA Appendix B.2.47 January 19, 2010 Columbia Letter GO2-11-074 Second Annual Update April 5, 2011
48) Service Air System Inspection	The Service Air System Inspection Program is a new-program. The Service Air System Inspection Program manages the effect of loss	A.1.2.48	Implementation prior to the period of extended	LRA Appendix B.2.48 January 19, 2010

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Program	of material due to corrosion of steel piping and valve bodies exposed to an "air (internal)" (i.e., compressed air) environment within the license renewal boundary of the service air system. The program consists of baseline inspections prior to the period of extended operation followed by opportunistic inspections during the period of extended operation. Following the baseline inspection, inspection findings will be reviewed periodically to ensure that each material and environment combination has been examined via opportunistic inspection or actions are taken to ensure inspections are performed. Initial interval for review of inspection findings is 5 years and may be adjusted based on operating experience.		operation and initial inspection within the 10-year period prior to the period of extended operation. Then ongoing.	Columbia Letter GO2-11-025 RAI B.2.14-1 January 28, 2011
49) Small Bore Class 1 Piping Program	The Small Bore Piping Program will inspect a sample of ASME Code Class 1 piping LT NPS 4 to GE to NPS 1. The inspection will include 10% of the weld population or a maximum of 25 welds of each weld type (e.g., full penetration or socket weld) using a methodology to select the most susceptible and risk-significant welds. For socket welds, opportunistic destructive examination can be performed in lieu of volumetric examination. Because more information can be obtained from a destructive examination than from nondestructive examination, Columbia may take credit for each weld destructively examined equivalent to having volumetrically examined two welds.	A.1.2.49	Implemented prior to the period of extended operation. Inspection activities will start during the fourth 10-year inservice inspection interval, then ongoing.	LRA Appendix B.2.49 January 19, 2010 Columbia Letter GO2-10-135 RAI B.2.49-1 September 13, 2010 Columbia Letter GO2-11-020 RAI B.2.49-2 January 27, 2011
50) Structures Monitoring Program	The Structures Monitoring Program is an existing program that will be continued for the period of extended operation, with the following enhancements: • Include and list the structures within the scope of license renewal that credit the Structures Monitoring Program for aging management. • Specify that if a below grade structural wall or structural component becomes accessible through excavation; a follow-up action is initiated for the responsible engineer to inspect the exposed surfaces for age-related degradation prior to backfilling. • Identify that the term "structural component" for inspection includes	A.1.2.50	Enhancement prior to the period of extended operation. Then ongoing.	LRA Appendix B.2.50 January 19, 2010 Columbia Letter GO2-10-128 RAI B.2.50-2 September 3, 2010 Columbia Letter GO2-11-017

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	component types that credit the Structures Monitoring Program for aging management. • Include the potential degradation mechanism checklist in the procedural documents. The checklist also requires enhancement to include aging effect terminology (e.g., loss of material, cracking, change in material properties, and loss of form). • Specify that the responsible engineer shall review site groundwater and raw water testing results for pH, chlorides, and sulfates prior to inspection to validate that the below-grade or raw water environments remain non-aggressive during the period of extended operation. Chemistry data shall be obtained from Columbia's chemistry and environmental departments. Groundwater chemistry data shall be collected at least once every four years. The time of data collection shall be staggered from year to year (summer-winter-summer) to account for seasonal variations in the environment. • Specify additional direction for quantifying, monitoring and trending of inspection results. • Provide better alignment with referenced Industry codes, standards and guidelines regarding terminology and evaluation. • Revise to add sufficient acceptance criteria and critical parameters to trigger level of inspection and initiation of corrective action. ACI 349.3R-96 provides an acceptable basis for developing acceptance criteria for concrete structural elements, steel liners, joint, coatings, and waterproofing membranes. Plant specific quantitative degradation limits, similar to the three-tier hierarchy acceptance criteria from Chapter 5 of ACI 349.3R-96, will be developed and added to the inspection procedure. • Conduct a base line inspection of the structures within the scope of license renewal plus a minimum of one additional inspection prior to entering the period of extended operation.			RAI B.2.50-1, RAI B.2.50-2, RAI B.2.50-3 January 20, 2011 Columbia Letter GO2-11-112 June 23, 2011
51) Supplemental Piping/Tank Inspection	The Supplemental Piping/Tank Inspection is a new activity. The Supplemental Piping/Tank Inspection detects and characterizes the material condition of steel, gray cast iron, and stainless steel components exposed to moist air environments. The inspection provides direct evidence as to whether, and to what extent, the relevant	A.1.2.51	Within the 10- year period prior to the period of extended operation.	LRA Appendix B.2.51 January 19, 2010

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	effects of aging have occurred.			
52) Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program	The Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program is a new program. The Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program will manage loss of fracture toughness due to thermal aging and neutron irradiation embrittlement of CASS reactor vessel internals. • The program includes: (a) identification of susceptible components determined to be limiting from the standpoint of thermal aging or neutron irradiation embrittlement (neutron fluence), (b) a componentspecific evaluation to determine each identified component's susceptibility to loss of fracture toughness, and (c) a supplemental examination of any component not eliminated by the component specific evaluation.	A.1.2.52	Implementation prior to the period of extended operation. Then ongoing.	LRA Appendix B.2.52 January 19, 2010
53) Water Control Structures Inspection	The Water Control Structures Inspection is an existing program that will be continued for the period of extended operation, with the following enhancements: • Include and list the water control structures within the scope of license renewal. Include the RG 1.127 Revision 1 inspection elements for the water control structures, including submerged surfaces. Ensure descriptions of concrete conditions conform with the appendix to the American Concrete Institute (ACI) publication, ACI 201, "Guide for Making a Condition Survey of Concrete in Service." Add a recommendation to use photographs for comparison of previous and present conditions. Add a requirement for the documentation of new or progressive problems as a part of the inspection program. • Specify additional direction for quantifying, monitoring and trending of inspection results. • Provide better alignment with referenced Industry codes, standards and guidelines regarding terminology and evaluation. • Revise to add sufficient acceptance criteria and critical parameters to trigger level of inspection and initiation of corrective action. ACI 349.3R-96 provides an acceptable basis for developing acceptance criteria for concrete structural elements, steel liners, joints, coatings, and waterproofing membranes. Plant specific quantitative degradation limits, similar to the three-tier hierarchy acceptance criteria from Chapter 5 of ACI 349.3R-96, will be developed and added to the	A.1.2.53	Enhancement prior to the period of extended operation. Then ongoing.	LRA Appendix B.2.53 January 19, 2010 Columbia Letter GO2-10-128 RAI B.2.50-2 September 3, 2010 Columbia Letter GO2-11-014 RAI B.2.50-6 January 18, 2011

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	inspection procedure. • Energy Northwest will conduct a baseline inspection of the spray ponds (including submerged portions) plus a minimum of one additional inspection prior to entering the period of extended operation. Inspection will use quantitative acceptance criteria in accordance with ACI 349.3R.			
54) Pressure-Temperature Limits	The Columbia P-T limit curves were revised in 2005 to include the effects of power uprate to 3486 MWt. The P-T limits are valid for 33.1 EFPY through the end of the currently licensed period. P-T limits for the period of extended operation will be calculated using the most accurate fluence projections available at the time of the recalculation. The projections may be adjusted if there are changes in core design or if additional surveillance capsule results show the need for an adjustment. The projected ART for the period of extended operation gives confidence that future P-T curves will provide adequate operating margin. License amendment requests to revise the P-T limits will be submitted to the NRC for approval, when necessary to comply with 10 CFR 50 Appendix G, as part of the Reactor Vessel Surveillance Program.	A.1.3.1.4	Ongoing	LRA Appendix B.2.54 January 19, 2010
55) Incorporate FSAR Supplement	Energy Northwest will incorporate the FSAR Supplement into the Columbia FSAR as required by 10 CFR 54.21(d).	A.1 A.1.1	Following issuance of the renewed operating license.	LRA Appendix B.2.55 January 19, 2010
56) Operational Quality Assurance Program Description	The elements of corrective actions, confirmation process, and administrative controls in the OQAPD will be applied to required aging management programs for both safety-related and non-safety related structures and components determined to require aging management during the period of extended operation.	A.1.2	Prior to the period of extended operation.	LRA Appendix B.2.56 January 19, 2010
57) License Renewal Commitment List	The commitments identified in association with Columbia license renewal will be tracked within the Columbia Regulatory Commitment Management Program.	A.1.5	Upon submittal of the license renewal application to the NRC.	LRA Appendix B.2.57 January 19, 2010
58) BWRVIP-42-A, AAI#5	In accordance with the BWR Vessel Internals Program, Columbia will implement the additional inspection requirements of BWRVIP-42-A once those requirements are approved by the NRC staff.	LRA Appendix C	Upon issuance of approved revision to BWRVIP-42 that recommends inspection of	LRA Appendix C January 19, 2010

Item Number	Commitment	Updated Final Safety Analysis Report (UFSAR) Supplement Section/LRA Section	Enhancement or Implementation Schedule	Source
			inaccessible welds	
59) BWRVIP-116	Energy Northwest will submit a licensing basis change request to implement the BWRVIP ISP(E) at least two years prior to the period of extended operation. Columbia will implement the ISP(E) as amended by the BWRVIP letter of January 11, 2005, including the new capsule test schedule in Table 1 of that letter.	LRA Appendix C	Two years prior to the period of extended operation.	LRA Appendix C January 19, 2010
60) BWRVIP-116	Implementation of the BWRVIP ISP(E) for Columbia will include the following details in support of the contingency plan: (1) Energy Northwest will include the requirement to keep all tested material (irradiated or unirradiated) for possible future reconstitution and testing. (2) The Columbia site procedure, as modified, will continue to require any capsules removed from the reactor vessel to be stored in a manner that would support future re-insertion of these capsules in the reactor vessel. (3) Energy Northwest will notify the BWRVIP prior to any change in the storage of on-site materials. NRC approval will be obtained prior to any change in the storage of surveillance materials that would affect the potential use of the materials under the contingency plan.	LRA Appendix C	Ongoing	LRA Appendix C January 19, 2010
61) Boron Carbide Monitoring Program	The Boron Carbide Monitoring Program is an existing program that will be continued for the period of extended operation. Initial in situ testing of the spent fuel rack neutron absorbing material will be performed prior to the period of extended operation to determine the current state of the racks. Additional in situ testing will be based on the results of this initial testing, but at an interval not to exceed ten years.	A.1.2.54	Initial in situ testing prior to the period of extended operation, then ongoing.	Columbia Letter GO2-10-117 RAI 3.3.2.2.6-1 August 19, 2010 Columbia Letter GO2-11-011 RAI B.2.54-1 January 14, 2011
62) Service Level 1 Protective Coatings Program	The Service Level 1 Protective Coatings Program is an existing program that will be continued for the period of extended operation.	A.1.2.55	Ongoing	Columbia Letter GO2-10-180 RAI XI.S8-1 December 21, 2010
63) Inservice Inspection (ISI)	Columbia currently relies on acceptable enhanced visual technique (EVT-1) inspections in accordance with ASME Code Section XI.	A.1.2.33	Upon availability of acceptable UT	Columbia Letter GO2-10-179

Item Number	Commitment	Updated Final Safety Analysis Report (UFSAR) Supplement Section/LRA Section	Enhancement or Implementation Schedule	Source
Program	Ultrasonic Testing (UT) examination of creviced shroud support plate access hole cover weld, top hat configuration, will be performed, in addition to EVT-1, once a demonstrated acceptable UT technique becomes available.		technique. Then ongoing.	RAI 3.1.1.x-1 December 21, 2010
64) Inservice Inspection (ISI) Program - IWE	Verify leakage is not entering the annular space between the containment vessel and the concrete shield wall from the outer refueling bellows seal. Inspection of the portions of the outer containment vessel shell made accessible by opening all eight inspection ports in the containment vessel at 570 foot elevation will be performed to check for evidence of leakage. These inspections will be performed during a refueling outage while the reactor cavity is flooded.	A.1.2.34	During the fourth Inservice Inspection (ISI) interval prior to the period of extended operation (PEO) and again in the fifth ISI interval after entering the PEO.	Columbia Letter GO2-11-014 RAI 3.5.2.2.1.4-1 January 18, 2011 Columbia Letter GO2-11-112 June 23, 2011
65) Inservice Inspection (ISI) Program	Columbia will prepare and submit the ISI Program Plan for the fourth 10-year interval no later than 2015. (The third 10-year ISI interval extends from December 2005 until December, 2015.) The Small Bore Piping Program will be included in the fourth 10-year interval ISI Program Plan as an augmented inspection. The locations to be inspected, the sample size, the inspection methodology will be included in the program plan.	A.1.2.33	Upon submittal of the ISI Program Plan for the fourth 10-year interval.	Columbia Letter GO2-11-020 RAI B.2.49-2 January 27, 2011 Columbia Letter GO2-11-074 Second Annual Update April 5, 2011
66) Structure Monitoring Program	Perform a one-time internal inspection of the spent fuel pool tell tale drain lines prior to the period of extended operation to confirm the drain lines are free of obstructions. Unexpected inspection results of clogged lines will require a condition report be documented and further engineering evaluation of adverse impacts to the spent fuel pool structure and to identify the periodicity of drain cleaning and maintenance process.	A.1.2.50	Prior to the period of extended operation	Columbia Letter GO2-11-017 RAI B.2.50-5 January 20, 2011
67) Structure Monitoring Program	Perform a one-time boroscope inspection of the containment sand pocket drain lines to confirm the absence of clogged drain lines and that a flow path exists for identification of any potential leakage into the sand pocket region. Unexpected inspection results (clogged drain lines) will be documented under corrective action process.	A.1.2.50	Prior to 12/31/15	Columbia Letter GO2-11-017 RAI B.2.34-1 January 20, 2011

Item Number	Commitment	Updated Final Safety Analysis Report (UFSAR) Supplement Section/LRA Section	Enhancement or Implementation Schedule	Source
				Columbia Letter GO2-11-029 RAI B.2.26-6 January 28, 2011
68) Flow-Accelerated Corrosion (FAC) Program	Ensure that the condensate (COND) and reactor feedwater (RFW) systems are screened and evaluated for cavitation prior to entering the period of extended operation (PEO). If the in-scope portion of either system is determined to be susceptible to loss of material due to cavitation erosion, then a program(s) will be modified or created to manage the loss of material.	A.1.2.28	Prior to the period of extended operation.	Columbia Letter GO2-11-029 RAI B.2.26-6 January 28, 2011 Columbia Letter GO2-11-074 Second Annual Update April 5, 2011
69) Inservice Inspection (ISI) Program	Re-evaluate the portions of the reactor pressure vessel beltline welds BG and BM for the period of extended operation (54 EFPY), in accordance with the requirements of the ASME Code, Section XI, IWB-3600 based on the results of 2015 inservice inspection.	A.1.2.33	Prior to the period of extended operation.	Columbia Letter GO2-11-031 RAI 4.7.1-1 January 28, 2011 Columbia Letter GO2-11-074 Second Annual Update April 5, 2011
70) TLAA – Embrittlement of reactor vessel	Perform a 54 EFPY equivalent margin analysis for the embrittlement (upper shelf energy) of the reactor vessel N12 (instrumentation) nozzle forgings.	A.1.3.1.2	No later than 2 years prior to the period of extended operation.	Columbia Letter GO2-11-031 RAI 4.2-1 January 28, 2011 Columbia Letter GO2-11-084 RAI 4.2-6

Item Number	Commitment	Updated Final Safety Analysis Report (UFSAR) Supplement Section/LRA Section	Enhancement or Implementation Schedule	Source
				April 22, 2011 Columbia Letter GO2-11-195 RAI 4.2-6 December 14, 2011
71) BWR Vessel Internals Program	At least two years prior to the period of extended operation, Columbia will install core plate wedges unless: 1) A site-specific analysis is approved by the NRC that resolves core plate bolt loss of preload due to both stress relaxation and cracking, or 2) An NRC approved method is developed to inspect the core plate bolts for cracking and a site-specific analysis for loss of preload due to stress relaxation of the core plate bolts is approved by the NRC.	A.1.2.10	2 years prior to period of extended operation.	Columbia Letter GO2-11-113 RAI B.2.10-2 June 29, 2011 Columbia Letter GO2-11-176 RAI B.2.10-2 November 4, 2011

APPENDIX B

CHRONOLOGY

This appendix contains a chronological listing of the routine correspondence between the staff of the U.S. Nuclear Regulatory Commission (NRC) (the staff) and Energy Northwest (EN) (the applicant) and other correspondence regarding the staff's reviews of the Columbia Generating Station (Columbia), Docket Number 50-397, license renewal application (LRA).

Table B-1. Chronology

Date	Subject
January 19, 2010	Columbia Generating Station - License Renewal Application. (Accession No. ML100250656)
January 19, 2010	Columbia Generating Station - License Renewal Application, Technical Information, Cover Page - 3.3-400. (Accession No. ML100250658)
January 19, 2010	Columbia Generating Station - License Renewal Application, Technical Information, Pages 3.4-1 to D-2. (Accession No. ML100250654)
January 19, 2010	Columbia Generating Station - License Renewal Application, Applicant's Environmental Report Operating License Renewal Stage, Appendix E. (Accession No. ML100250666)
January 26, 2010	Receipt and Availability of the License Renewal Application for Columbia Generating Station. (Accession No. ML100220037)
January 26, 2010	Federal Register Notice: Notice of Receipt and Availability of Application for Renewal of Columbia Generating Station. (Accession No. ML100220041)
February 3, 2010	Press Release-10-025: NRC Announces Availability of License Renewal Application for Columbia Nuclear Power Plant. (Acccession No. ML100340369)
March 4, 2010	Determination of Acceptability and Sufficiency for Docketing, Proposed Review Schedule, and Opportunity for a Hearing regarding the Application From Energy Northwest, for Renewal of the Operating Licenses for the Columbia Generating Station. (Accession No. ML100541619)
March 4, 2010	Federal Register Notice: Notice of Acceptance for Docketing of the Application, Notice of Opportunity for Hearing for Facility Operating License No. NPF-21 for and Additional 20-Year Period Energy Northwest Columbia Generating Station. (Accession No. ML100550728)
March 5, 2010	Notice of Intent to Prepare an Environmental Impact Statement and Conduct the Scoping Progress for License Renewal for the Columbia Generating Station. (Accession No. ML100570266)
March 5, 2010	Federal Register Notice: Notice of Intent to Prepare an Environmental Impact Statement and Conduct the Scoping Process for Columbia Generating Station Docket No. 50-397 (FRN). (Accession No. ML100570282)
March 8, 2010	Press Release-10-043: NRC Announces Opportunity for Hearing on Application to Renew Operating License for Columbia Generating Station Nuclear Power Plant. (Accession No. ML100670526)
March 25, 2010	Notice of Forthcoming Meeting on April 6, 2010, to Discuss the License Renewal Process and Environmental Scoping for Columbia Generating Station License Renewal Application Review. (Accession No. ML100810403)
March 26, 2010	Press Release-IV-10-010: NRC Seeks Public Input on Environmental Review of Columbia Generating Station License Renewal; Meetings April 6, 2010. (Accession No. ML100850318)

Date	Subject
April 6, 2010	Transcript of Columbia Generating Station License Renewal Process and Environmental Scoping Public Meeting, Afternoon Session, April 6, 2010, Pages 1-39. (Accession No. ML101241002)
April 6, 2010	Transcript of Columbia Generating Station License Renewal Process and Environmental Scoping Public Meeting, Evening Session, April 6, 2010, Pages 1-30. (Accession No. ML101241037)
May 10, 2010	Meeting Summary-CGS License Renewal Overview and Environmental Scoping Meetings on April 6, 2010. (Accession No. ML101250314)
May 10, 2010	Summary of Public License Renewal Overview and Environmental Scoping Meetings Related to the Review of the Columbia Generating Station License Renewal Application on April 6, 2010 (TAC Nos. ME3058 and ME3121). (Accession No. ML101250519)
May 20, 2010	Division of License Renewal's Transition from Paper Distribution to Electronic Distribution of Outgoing Correspondence. (Accession No. ML101310138)
June 9, 2010	Letter to W.S. Oxenford, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station License Renewal Application, Scoping And Screening Methodology. (Accession No. ML101530226)
June 21, 2010	Letter to W.S. Oxenford, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station License Renewal Application. (Accession No. ML101660665)
June 24, 2010	Letter to W.S. Oxenford, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station License Renewal Application, Scoping And Screening Methodology. (Accession No. ML101650276)
June 24, 2010	Letter to W.S. Oxenford, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station License Renewal Application. (Accession No. ML101660030)
June 30, 2010	Letter to W.S. Oxenford, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station License Renewal Application. (Accession No. ML101720623)
July 7, 2010	Letter to W.S. Oxenford, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station License Renewal Application concerning Structures. (Accession No. ML101730468)
July 7, 2010	Letter to W.S. Oxenford, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station License Renewal Application concerning Electrical. (Accession No. ML101730271)
July 13, 2010	Letter to W.S. Oxenford, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station License Renewal Application concerning Electrical. (Accession No. ML101660166)
July 15, 2010	Letter to W.S. Oxenford, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station License Renewal Application concerning Electrical. (Accession No. ML101820636)
July 15, 2010	Letter to W.S. Oxenford, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station License Renewal Application concerning Electrical. (Accession No. ML101900125)
July 16, 2010	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information, regarding the Scoping and Screening Methodology.of the License Renewal Application. (Accession No. ML102020260)
July 16, 2010	Letter from S. K. Gambhir, Energy Northwest: Columbia Generating Station License Renewal Application First Annual Update. (Accession No. ML102090559)

Date	Subject
July 19, 2010	Summary of Telephone Conference Call Held on June 17, 2010, between the U.S. Nuclear Regulatory Commission and Energy Northwest, concerning the Request for Additional Information pertaining to the Columbia Generating Station, LRA. (Accession No. ML101890311)
August 3, 2010	Letter to S. K. Gambhir, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station License Renewal Application concerning Section 2.4. (Accession No. ML102020129)
August 5, 2010	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information, Dated July 2, 2010, regarding the License Renewal Application. (Accession No. ML102300503)
August 6, 2010	Letter to S. K. Gambhir, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station License Renewal Application. (Accession No. ML101960640)
August 10, 2010	Columbia Ltr. Informing NRC that Mr. Mark E. Reddemann has been Selected as New Chief Executive Officer. (Accession No. ML102380030)
August 10, 2010	Letter to S. K. Gambhir, Energy Northwest: Schedule Revision for the Environmental Review of the Columbia Generating Station License Renewal Application. (Accession No. ML102100303)
August 16, 2010	Letter to S. K. Gambhir, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station License Renewal Application. (Accession No. ML102080506)
August 16, 2010	Letter to S. K. Gambhir, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station License Renewal Application. (Accession No. ML102230369)
August 19, 2010	Letter to S. K. Gambhir, Energy Northwest: Scoping and Screening Audit Report regarding the Columbia Generating Station, License Renewal Application. (Accession No. ML102160357)
August 19, 2010	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML102440342)
August 26, 2010	Letter to S. K. Gambhir, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application for Fatigue Monitoring Program, TLAA Exemptions, Metal Fatigue TLAA, Cumulative Fatigue Damage, CASS, and Structural (TAC No. ME3058). (Accession No. ML102220373)
August 26, 2010	Letter to S. K. Gambhir, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML102300229)
August 26, 2010	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML102430205)
August 30, 2010	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML102450055)
September 3, 2010	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML102520048)
September 3, 2010	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML102520049)
September 13, 2010	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML102590047)
September 14, 2010	Summary of Telephone Conference Call Held on August 10, 2010, between the U.S. Nuclear Regulatory Commission and Energy Northwest, concerning the Request For Additional Information pertaining to the Columbia Generating Station, License Renewal Application. (Accession No. ML102450571)

Date	Subject
September 14, 2010	Summary of Telephone Conference Call Held on August 26, 2010, between the U.S. NRC and Energy Northwest, concerning the Request for Additional Information pertaining to the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML102450621)
September 15, 2010	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML102660205)
September 16, 2010	Letter to S. K. Gambhir, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML102450727)
September 16, 2010	Summary of Telephone Conference Call Held on August 12, 2010, between the U.S. Nuclear Regulatory Commission and Energy Northwest, concerning the RAI pertaining to the Columbia Generating Station, LRA (TAC No. ME3058). (Accession No. ML102450756)
September 21, 2010	Letter to S. K. Gambhir, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML102530645)
September 21, 2010	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML102660029)
September 24, 2010	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML102720030)
September 27, 2010	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML102740028)
October 4, 2010	Summary of Telephone Conference Call Held on September 13, 2010, between the U.S. Nuclear Regulatory Commission and Energy Northwest, concerning the Request for Additional Information pertaining to the Columbia Generating Station, License Renewal Application. (Accession No. ML102700433)
October 14, 2010	Letter to S. K. Gambhir, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML102800426)
October 20, 2010	Letter to S. K. Gambhir, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML102730355)
October 20, 2010	Letter to S. K. Gambhir, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application concerning Structures (TAC No. ME3058). (Accession No. ML102850735)
October 25, 2010	Summary of Telephone Conference Call Held on October 5, 2010, between the U.S. Nuclear Regulatory Commission and Energy Northwest, concerning the Draft Request for Additional Information pertaining to the Columbia Generating Station, License Renewal Application. (Accession No. ML102790223)
October 25, 2010	Summary of Telephone Conference Call Held on September 22, 2010, between the U.S. Nuclear Regulatory Commission and Energy Northwest concerning the Draft Request for Additional Information pertaining to the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML102870193)
October 25, 2010	Summary of Teleconference Held on October 13, 2010, between the U.S. Nuclear Regulatory Commission and Energy Northwest, concerning the Draft Request for Additional Information pertaining to the Columbia Generating Station License Renewal Application. (Accession No. ML102870245)
October 25, 2010	Letter to M. E. Reddeman, Energy Northwest: Columbia Generating Station - Project Manager Assignment to Balwant Singal Effective November 7, 2010. (Accession No. ML102980515)

Date	Subject
October 27, 2010	Summary of Telephone Conference Calls Held on September 20, 2010, between the U.S. Nuclear Regulatory Commission and Energy Northwest, concerning the Response to Request for Additional Information pertaining to the Columbia Generating Station, LRA. (Accession No. ML102850103)
November 1, 2010	Letter to S. K. Gambhir, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML102930593)
November 5, 2010	Letter to S. K. Gambhir, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML103010080)
November 11, 2010	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML103160425)
November 19, 2010	Summary of Telephone Conference Call Held on October 26, 2010, between the U.S. Nuclear Regulatory Commission and Energy Northwest, concerning the Draft Request for Additional Information pertaining to the Columbia Generating Station, LRA (TAC No. ME3058). (Accession No. ML103000479)
November 19, 2010	Summary of Teleconference Held on September 22, 2010, between the U.S. Nuclear Regulatory Commission and Energy Northwest, concerning the Responses to the Request for Additional Information pertaining to the Columbia Generating Station, License Renewal Application. (Accession No. ML103090566)
November 19, 2010	Letter to S. K. Gambhir, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML103130548)
November 19, 2010	Summary of Teleconference Held on November 8, 2010, between the U.S. Nuclear Regulatory Commission and Energy Northwest, concerning the Responses to Request for Additional Information pertaining to the Columbia Generating Station, LRA. (Accession No. ML103200338)
November 19, 2010	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML103280371)
November 23, 2010	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML103280370)
December 3, 2010	Summary of Teleconference Held on November 11, 2010, between the U.S. Nuclear Regulatory Commission and Energy Northwest, concerning Schedule Change for Columbia Generating Station regarding the License Renewal Application. (Accession No. ML103160226)
December 3, 2010	Summary of Telephone Conference Call Held on October 7, 2010, between the U.S. Nuclear Regulatory Commission and Energy Northwest concerning the Draft Request for Additional Information pertaining to the Columbia Generating Station, LRA (TAC No ME3058). (Accession No. ML103210396)
December 3, 2010	Letter to S. K. Gambhir, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML103260155)
December 7, 2010	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML103420568)
December 17, 2010	Letter to M. E. Reddeman, Energy Northwest: Columbia Generating Station – NRC License Renewal Inspection Report (IR 0500397-10-007). (Accession No. ML103540496)
December 20, 2010	Letter to S. K. Gambhir, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML103540022)

Date	Subject
December 21, 2010	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application for Monitoring and the Maintenance of Protective Coatings. (Accession No. ML103620325)
December 21, 2010	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML103620326)
December 27, 2010	Letter to S. K. Gambhir, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML103550603)
January 5, 2011	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML110070353)
January 6, 2011	Summary of Telephone Conference Call Held on January 3, 2011, between the U.S. Nuclear Regulatory Commission and Energy Northwest, concerning the Request for Additional Information pertaining to the Columbia Generating Station, License Renewal Application (TAC Number ME3058). (Accession No. ML110050018)
January 10, 2011	Letter to S. K. Gambhir, Energy Northwest: Schedule Revision for the Review of the Columbia Generating Station License Renewal Application (TAC Nos. ME3058, ME3121). (Accession No. ML103430526)
January 11, 2011	Letter to S. K. Gambhir, Energy Northwest: Safety Project Manager Change for the License Renewal of Columbia Generating Station (Tac No. ME3058). (Accession No. ML103630739)
January 13, 2011	Summary of Telephone Conference call Held on January 5, 2011, between the U.S. Nuclear Regulatory Commission and Energy Northwesy, concerning the Request for Additional Information pertaining to the Columbia Generating Station License Renewal Application (Tac No. ME3058). (Accession No. ML110060438)
January 14, 2011	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML110180457)
January 18, 2011	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML110190657)
January 20, 2011	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML110270135)
January 20, 2011	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML110270236)
January 20, 2011	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML110270242)
January 21, 2011	Letter to S. K. Gambhir, Energy Northwest: Audit Report regarding the Columbia Generating Station, License Renewal Application. (Accession No. ML102450757)
January 27, 2011	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML110310010)
January 28, 2011	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML110320340)
January 28, 2011	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML110320419)
January 28, 2011	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML110320504)
January 28, 2011	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML110320505)
January 28, 2011	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application to Support RAI 3.1.2.3.1-2. (Accession No. ML110320538)

Date	Subject
January 28, 2011	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the License Renewal Application. (Accession No. ML110330134)
January 31, 2011	Summary of Telephone Conference Call Held on January 12, 2011, between the U.S. Nuclear Regulatory Commission and Energy Northwest, concerning the Responses to Request for Additional Information pertaining to the Columbia Generating Station, License Renewal Application (TAC NO. ME3058). (Accession No. ML110140588)
February 3, 2011	Summary of Teleconference Call Held on January 13, 2011 between the U.S. Nuclear Regulatory Commission and Energy Northwest, concerning the Draft Request for Additional Information pertaining to the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML110200374)
February 3, 2011	Summary of Telephone Conference Call Held on January 11, 2011, between the U.S. Nuclear Regulatory Commission and Energy Northwest concerning the Request for Additional Information pertaining to the Columbia Generating Station, License Renewal Application. (Accession No. ML110200710)
February 3, 2011	Summary of Telephone Conference Held on January 20, 2011 between the U.S. Nuclear Regulatory Commission and Energy Northwest, concerning the Request for Additional Information pertaining to the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML110240202)
February 3, 2011	Letter to S. K. Gambhir, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application for Metal Fatigue (TAC No. ME3058). (Accession No. ML110240426)
February 16, 2011	Summary of Telephone Conference Call Held on January 24, 2011, between the U.S. Nuclear Regulatory Commission and Energy Northwest, concerning the Request for Additional Information pertaining to the Columbia Generating Station, License Renewal Application. (Accession No. ML110260380)
February 23, 2011	Summary of Telephone Conference Call Held on February 14, 2011, between the U.S. Nuclear Regulatory Commission and Energy Northwest, concerning the Responses to the Request for Additional Information pertaining to the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML110470215)
February 24, 2011	Letter to M. E. Reddeman, Energy Northwest: Columbia Generating Station - Project Manager Assignment Effective March 13, 2011 from Balwant Singal to Mohan Thadani. (Accession No. ML110540579)
February 25, 2011	Email from A. A. Mostala, Energy Northwest: Clarification Question regarding the Response to RAI 3.1.2.3-01 in Columbia Letter Dated September 21, 2010. (Accession No. ML110601208)
February 25, 2011	Summary of Telephone Conference Call Held on February 17, 2011, between the U.S. Nuclear Regulatory Commission and Energy Northwest, concerning the Responses to the Request for Additional Information pertaining to the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML110540126)
March 3, 2011	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the Columbia Generating Station, License Renewal Application. (Accession No. ML110690022)
March 9, 2011	Letter to S. K. Gambhir, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application for Buried Piping and Tanks Inspection (TAC No. ME3058). (Accession No. ML110610712)
March 15, 2011	Summary of Telephone Conference Call Held on March 8, 2011, between the U.S. Nuclear Regulatory Commission and Energy Northwest concerning the Responses to the Request for Additional Information pertaining to the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML110690997)
March 18, 2011	Letter to S. K. Gambhir, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application for Drywell Floor Peripheral Seal Assembly (TAC No. ME3058). (Accession No. ML110680670)

Date	Subject
March 23, 2011	Letter to S. K. Gambhir, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application for Time-Limited Aging Analyses Of Reactor Vessel Neutron Embrittlement (TAC Number ME3058). (Accession No. ML110630360)
April 5, 2011	Summary of Telephone Conference Call Held on March 23, 2011, between the U.S. Nuclear Regulatory Commission and Energy Northwest concerning the Responses to the Request for Additional Information pertaining to the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML110871495)
April 5, 2011	Letter from S. K. Gambhir, Energy Northwest: Second Annual Changes Update regarding the Columbia Generating Station, License Renewal Application. (Accession No. ML110970354)
April 5, 2011	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the Columbia Generating Station, License Renewal Application. (Accession No. ML110970355)
April 13, 2011	Letter from S. K. Gambhir, Energy Northwest: Response to NRC Request for Additional Information regarding the Columbia Generating Station, License Renewal Application. (Accession No. ML11104A049)
April 21, 2011	Letter from David A. Swank, Energy Northwest: Response to NRC Request for Additional Information regarding the Columbia Generating Station, License Renewal Application. (Accession No. ML11115A098)
April 22, 2011	Letter from David A. Swank, Energy Northwest: Response to NRC Request for Additional Information regarding the Columbia Generating Station, License Renewal Application. (Accession No. ML11116A169)
May 12, 2011	Email from A. A. Mostala, Energy Northwest: Request for a list of documentation to determine there are no CASS ASME III Class 1 valve less than four inches installed at Columbia. (Accession No. ML11137A043)
May 24, 2011	Letter to David A. Swank, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application for Operating Experience (TAC No. ME3058). (Accession No. ML11138A323)
May 26, 2011	Letter from David A. Swank, Energy Northwest: Response to Request for Additional Information on License Renewal Application Table 3.1.2-3, Row Numbers 182 and 183. (Accession No. ML11147A157)
June 2, 2011	Letter to David A. Swank, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application Regarding Core Plate Assembly (TAC No. ME3058). (Accession No. ML11140A161)
June 8, 2011	Summary of Telephone Conference Call Held on May 6, 2011, between the U.S. Nuclear Regulatory Commission and Energy Northwest Concerning the Draft Request for Additional Information Pertaining To The Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML11137A044)
June 22, 2011	Summary of Telephone Conference Call Held on June 1, 2011, Between the U.S. Nuclear Regulatory Commission and Energy Northwest, Concerning the Request for Additional Information Pertaining to the Columbia Generating Station, LRA (TAC No. ME3058). (Accession No. ML11165A243)
June 23, 2011	Letter from David A. Swank, Energy Northwest: Response to Request for Additional Information License Renewal Application. (Accession No. ML11180A013)
June 29, 2011	Letter from David A. Swank, Energy Northwest: Response to Request for Additional Information License Renewal Application. (Accession No. ML11182C038)
July 11, 2011	Letter from David A. Swank, Energy Northwest: Response to Request for Additional Information License Renewal Application. (Accession No. ML11195A145)

Date	Subject
July 12, 2011	Summary of Telephone Conference Call Held on July 6, 2011, Between the U.S. Nuclear Regulatory Commission and Energy Northwest, Concerning the Request for Additional Information Pertaining to the Columbia Generating Station, LRA (TAC No. ME3058). (Accession No. ML11188A238)
July 19, 2011	Letter to David A. Swank, Energy Northwest: Request For Additional Information For The Review Of The Columbia Generating Station, License Renewal Application Regarding High Voltage Porcelain Insulators (TAC No. ME3058). (Accession No. ML11195A240)
July 29, 2011	Letter from David A. Swank, Energy Northwest: Response to Request for Additional Information License Renewal Application. (Accession No. ML11215A010)
August 2, 2011	Summary of Telephone Conference Call Held on July 19, 2011, Between the U.S. Nuclear Regulatory Commission and Energy Northwest, Concerning the Request for Additional Information Pertaining to the Columbia Generating Station, LRA (TAC No. ME3058). (Accession No. ML11208B049)
August 10, 2011	Letter from David A. Swank, Energy Northwest: Response to Request for Additional Information License Renewal Application. (Accession No. ML11227A010)
August 10, 2011	Letter from David A. Swank, Energy Northwest: Response to Request for Additional Information License Renewal Application. (Accession No. ML11227A011)
August 11, 2011	Summary of Telephone Conference Call Held on August 3, 2011, Between the U.S. Nuclear Regulatory Commission and Energy Northwest, Concerning the Response to the Request for Additional Information Pertaining to the Columbia Generating Station, LRA (TAC No. ME3058). (Accession No. ML11216A253)
August 11, 2011	Summary of Telephone Conference Call Held on August 3-4, 2011, Between the U.S. Nuclear Regulatory Commission and Energy Northwest, Concerning the Response to the Request for Additional Information Pertaining to the Columbia Generating Station, LRA (TAC No. ME3058). (Accession No. ML11217A022)
August 11, 2011	Summary of Telephone Conference Call Held on February 3, 2011, Between the U.S. Nuclear Regulatory Commission and Energy Northwest, Concerning the Response to the Request for Additional Information Pertaining to the Columbia Generating Station, LRA (TAC No. ME3058). (Accession No. ML11220A010)
August 18, 2011	Letter from David A. Swank, Energy Northwest: Response to Request for Additional Information License Renewal Application. (Accession No. ML11242A018)
August 23, 2011	Summary Of Telephone Conference Call Held on August 11, 2011, Between the U.S. Nuclear Regulatory Commission and Energy Northwest, Concerning The Response To The Request For Additional Information Pertaining To The Columbia Generation Station, LRA (TAC No. ME3058). (Accession No. ML11223A351)
August 23, 2011	Summary Of Telephone Conference Call Held on August 17, 2011, Between the U.S. Nuclear Regulatory Commission And Energy Northwest, Concerning the Response to the Request For Additional Information Pertaining to the Columbia Generating Station, LRA (TAC No. ME3058). (Accession No. ML11230B089)
August 30, 2011	Safety Evaluation Report with Open Items Related To The License Renewal Of Columbia Generating Station (TAC ME3058). (Accession No. ML11172A092). Revised. See Safety Evaluation Report with Open Items Related to the License Renewal of Columbia Generating Station with Proprietary Information Removed, dated December 21, 2011. (Accession No. ML11349A017)
August 30, 2011	Safety Evaluation Report with Open Items Related To The License Renewal Of Columbia Generating Station Docket No. 50-397. (Accession No. ML11242A121). Revised. See Safety Evaluation Report with Open Items Related to the License Renewal of Columbia Generating Station, dated December 21, 2011. (Accession No. ML11349A022)
August 30, 2011	Advisory Committee on Reactor Safeguards Review Of The Columbia Generating Station, License Renewal Application - Safety Evaluation Report With Open Items. (Accession No. ML11242A094)

Date	Subject
September 6, 2011	Columbia Generating Station Docket No. 50-397 Public Meeting. (Accession No. ML11256A157)
September 8, 2011	Summary Of Telephone Conference Call Held on August 22, 2011, Between the U.S. Nuclear Regulatory Commission And Energy Northwest, Concerning Topics Pertaining to the Columbia Generating Station, License Renewal Application. (Accession No. ML11250A015)
September 23, 2011	Summary Of Telephone Conference Call Held on September 12, 2011, Between the U.S. Nuclear Regulatory Commission And Energy Northwest, Concerning the Draft Request for Additional Information Pertaining to the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML11256A300)
September 26, 2011	Letter to David A. Swank, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application Regarding Upper Shelf Energy (TAC No. ME3058). (Accession No. ML11269A014)
September 27, 2011	Email from A. A. Mostala, Energy Northwest: Clarification Question Regarding Columbia Lubirating Oil Analysis Program discussed in the inspection report. (Accession No. ML11273A002)
September 29, 2011	Letter from David A. Swank, Energy Northwest: Response to Request for Additional Information License Renewal Application. (Accession No. ML11278A187)
September 30, 2011	Letter to David A. Swank, Energy Northwest: Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application Regarding Regarding Operating Experience (TAC No. ME3058). (Accession No. ML11272A124)
October 4, 2011	Summary Of Telephone Conference Call Held on September 28, 2011, Between the U.S. Nuclear Regulatory Commission And Energy Northwest, Concerning the Draft Request for Additional Information Pertaining to the Columbia Generating Station, License Renewal Application. (TAC No. ME3058). (Accession No. ML11273A001)
October 4, 2011	Summary of the Telephone Conference Call Held on September 29, 2011, Between the U.S. Nuclear Regulatory Commission And Energy Northwest, Concerning the Inspection Report Pertaining to the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML11273A066)
October 5, 2011	Letter from David A. Swank, Energy Northwest: Response to Request for Additional Information License Renewal Application. (Accession No. ML11285A042)
October 6, 2011	Letter from David A. Swank, Energy Northwest: Response to Request for Additional Information License Renewal Application. (Accession No. ML11285A046)
October 19, 2011	Transcript of the Advisory Committee on Reactor Safeguards Plant License Renewal Subcommittee Meeting, October 19, 2011. (Accession No. ML11311A233)
October 27, 2011	Letter from David A. Swank, Energy Northwest: Response to Request for Additional Information License Renewal Application Related to Reactor Pressure Vessel (RPV) Nozzle Upper Shelf Energy (USE) U.S. Nuclear Regulatory Commission Request for Additional Information RAI 4.2-7 Support: Data and Affidavit. (Accession No. ML11308A023)
November 1, 2011	Letter from David A. Swank, Energy Northwest: Response to Request for Additional Information License Renewal Application. (Accession No. ML11308A022)
November 3, 2011	Summary of the Telephone Conference Call Held on October 27, 2011, Between the U.S. Nuclear Regulatory Commission And Energy Northwest, Concerning the Metal Fatigue Open Item in the Columbia License Renewal Safety Evaluation with Open Items (TAC No. ME3058). (Accession No. ML11305A176)
November 4, 2011	Letter from Bradley J. Sawatzke, Energy Northwest: Response to Request for Additional Information License Renewal Application. (Accession No. ML11312A245)
November 4, 2011	Letter from Bradley J. Sawatzke, Energy Northwest: Response to Request for Additional Information License Renewal Application. (Accession No. ML11312A247)

Date	Subject
November 10, 2011	Summary of Telephone Conference Call Held on November 2, 2011, Between the U.S. Nuclear Regulatory Commission And Energy Northwest, Concerning the Core Plate Hold-Down Bolts Open Item in the Columbia License Renewal Safety Evaluation with Open Items (TAC No. ME3058). (Accession No. ML11311A266)
November 10, 2011	Letter from David A. Swank, Energy Northwest: Response to Request for Additional Information License Renewal Application. (Accession No. ML11318A280)
November 16, 2011	Letter from Alex L. Javorik, Energy Northwest: Columbia Generating Station - License Renewal Application Supplement. (Accession No. ML11325A056)
November 17, 2011	Letter from Alex L. Javorik, Energy Northwest: Response to Request for Additional Information License Renewal Application. (Accession No. ML11325A067)
November 21, 2011	Summary of Telephone Conference Call Held on August 29, 2011, Between the U.S. Nuclear Regulatory Commission And Energy Northwest, Concerning the Response to the Request for Additional Information Pertaining to the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML11249A013)
December 6, 2011	Letter from Alex L. Javorik, Energy Northwest: Response to Request for Additional Information License Renewal Application. (Accession No. ML11341A116)
December 12, 2011	Summary of Telephone Conference Call Held on November 8, 2011, Between the U.S. Nuclear Regulatory Commission and Energy Northwest, Concerning the Metal Fatigue Open Item in the Columbia License Renewal Safety Evaluation Report with Open Items (TAC No. ME3058). (Accession No. ML11339A085)
December 12, 2011	Summary of Telephone Conference Call Held on November 14, 2011, Between the U.S. Nuclear Regulatory Commission and Energy Northwest, Concerning the Metal Fatigue Open Item in the Columbia License Renewal Safety Evaluation Report With Open Items (TAC No. ME3058). (Accession No. ML11336A150)
December 12, 2011	Summary of Telephone Conference Call Held on November 28, 2011, Between the U.S. Nuclear Regulatory Commission and Energy Northwest, Concerning the Upper Shelf Energy Open Item in the Columbia License Renewal Safety Evaluation Report With Open Items (TAC No. ME3058). (Accession No. ML11339A087)
December 12, 2011	Letter from Alex L. Javorik, Energy Northwest: Comments on Safety Evaluation Report for License Renewal Application. (Accession No. ML11350A037)
December 14, 2011	Letter from Alex L. Javorik, Energy Northwest: Response to Request for Additional Information License Renewal Application. (Accession No. ML11354A097)
December 16, 2011	Summary of Telephone Conference Call Held on December 6, 2011, Between the U.S. Nuclear Regulatory Commission and Energy Northwest, Concerning the Metal Fatigue and Operating Experience Open Items in the Columbia License Renewal Safety Evaluation Report With Open Items (TAC No. ME3058). (Accession No. ML11339A087)
December 16, 2011	Letter from Alex L. Javorik, Energy Northwest: Response to U.S. Nuclear Regulatory Commission Audit Questions, License Renewal Application. (Accession No. ML11356A078)
December 16, 2011	Letter from Alex L. Javorik, Energy Northwest: Response to Request for Additional Information License Renewal Application. (Accession No. ML11356A076)
December 21, 2011	Summary of Telephone Conference Call Held on December 15, 2011, Between the U.S. Nuclear Regulatory Commission and Energy Northwest, Concerning the Metal Fatigue and Operating Experience Open Items in the Columbia License Renewal Safety Evaluation Report With Open Items (TAC No. ME3058). (Accession No. ML11350A056)
December 21, 2011	Letter to Alex L. Javorik, Energy Northwest: Safety Evaluation Report with Open Items Related to the License Renewal of Columbia Generating Station with Proprietary Information Removed. (Accession No. ML11349A017)
December 21, 2011	Safety Evaluation Report with Open Items Related to the License Renewal of Columbia Generating Station. (Accession No. ML11349A022)

Date	Subject
January 4, 2012	Letter from Alex L. Javorik, Energy Northwest: Response to Information Request, License Renewal Application. (Accession No. ML12006A211)
February 16, 2012	Letter to Alex L. Javorik, Energy Northwest: Audit Report on the Metal Fatigue Calculations in the Columbia Generating Station, License Renewal Application (TAC No. ME3058). (Accession No. ML12033A058)
February 16, 2012	Letter from Alex L. Javorik, Energy Northwest: Columbia Generating Station License Renewal Application Third Annual Update. (Accession No. ML12052A005)
February 27, 2012	Memoranda from Elmo E. Collins, U.S. Nuclear Regulatory Commission to Eric Leeds, U.S. Nuclear Regulatory Commission: Regional Administrator's Letter. (Accession No. ML12058A496)
February 28, 2012	Letter to Alex L. Javorik, Energy Northwest: Safety Evaluation Report Related to the License Renewal of Columbia Generating Station (TAC No. ME3058). (Accession No. ML11263A001)
February 28, 2012	Safety Evaluation Report Related to the License Renewal of Columbia Generating Station. (Accession No. ML12059A357)
April 23, 2012	Letter from Alex L. Javorik, Energy Northwest: Response to NRC Information Request, License Renewal Application. (Accession No. ML12116A150)
April 24, 2012	Letter to Gregory B. Jaczko, Chairman, USNRC: Report on the Safety Aspects of the License Renewal Application for the Columbia Generating Station. (Accession No. ML12108A211)

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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Buford, A.	Reviewer— Structural
Casto, G.	Management Oversight
Cunanan, A.	Project Manager
Davidson, E.	Reviewer—Balance of Plant
Dennig, R.	Management Oversight
Doutt, C.	Reviewer—Electrical
Evans, M.	Management Oversight
Fu, B.	Reviewer—Mechanical
Gall, J.	Reviewer—Mechanical
Galloway, M.	Management Oversight
Gavula, J.	Reviewer—Mechanical
Gettys, E.	Project Manager
Gilanshahi, N.	Reviewer—Mechanical
Hiser, A.	Management Oversight
Hoang, D.	Reviewer—Mechanical
Holian, B.	Management Oversight
Holston, W.	Reviewer—Mechanical
Iqbal, N.	Reviewer—Fire Protection
Kalikian, R.	Reviewer—Mechanical
Khana, M.	Management Oversight
Kichline, M.	Reviewer—Mechanical
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Klos, J.	Reviewer—Mechanical
Lee, B.	Reviewer—Mechanical
Lehman, B.	Reviewer—Structural
Li, R.	Reviewer—Electrical
Mathew, R.	Management Oversight

Name	Responsibility
Medoff, J.	Reviewer—Mechanical
Miller, K.	Reviewer—Electrical
Min, S.	Reviewer—Mechanical
Mitchell, M.	Management Oversight
Morey, D.	Management Oversight
Nguyen, D.	Reviewer—Electrical
Nickell, C.	Reviewer—Mechanical
Obodoako, A.	Reviewer—Mechanical
Parks, B.	Reviewer—Mechanical
Pelton, D.	Management Oversight
Pham, B.	Management Oversight
Prinaris, A.	Reviewer— Structural
Raval, J.	Reviewer—Mechanical
Razzaque, M.	Reviewer—Mechanical
Rogers, B.	Reviewer—Scoping & Screening Methodology
Ruland, W.	Management Oversight
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
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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 Columbia Generating Station (Columbia), Docket Number 50-354.

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Arthur Cunanan, NRC Project Manager

11. ABSTRACT (200 words or less)

This safety evaluation report (SER) documents the technical review of the Columbia Generating Station (Columbia), license renewal application (LRA) by the U.S. Nuclear Regulatory Commission (NRC) staff (the staff). By letter dated January 19, 2010, Energy Northwest (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." Energy Northwest requests renewal of the operating license (Facility Operating License Number NPF-21) for a period of 20 years beyond the current license period of December 20, 2023.

Columbia is located approximately 12 miles north of Richland, WA. The NRC issued the construction permit on March 19, 1973, and the operating license for Columbia on April 13, 1984. The unit is a Mark II boiling-water reactor (BWR) design. General Electric Company supplied the nuclear steam supply system. Burns and Roe, Inc., designed the balance of plant, and Bechtel Power Corporation constructed the plant. The licensed power output of the unit is 3,886 megawatts thermal, with a gross electrical output of approximately 1,230 megawatts electric.

This SER presents the status of the staff's review of information submitted through January 4, 2012. The six open items previously identified by the staff from the SER with open items have been closed (see SER Section 1.5); the staff did not identify any open items before the staff made a final determination. SER Section 6.0 provides the staff's final conclusion of the LRA review.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

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