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NUREG-2190, Vol. 2

Safety Evaluation Report

Related to the License Renewal of Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2

Docket Nos. 50-454, 50-455, 50-456, and 50-457

Exelon Generation Company, LLC

Sections 4 to 6 and Appendices

Office of Nuclear Reactor Regulation

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ABSTRACT

This safety evaluation report (SER) documents the technical review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, (BBS) license renewal application (LRA) by the United States (U.S.) Nuclear Regulatory Commission (NRC) staff (the staff). By letter dated May 29, 2013, Exelon Generation Company, LLC (Exelon or the applicant), submitted the LRA in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." Exelon requests renewal of the BBS operating licenses (Operating License Nos. NPF-37, NPF-66, NPF-72, and NPF-77, respectively) for a period of 20 years beyond the current expiration at midnight October 31, 2024; November 6, 2026; October 17, 2026; and December 18, 2027, respectively.

Byron is located in north central Illinois, near the town of Byron, Illinois, and near the Rock River approximately 95 miles from Chicago, Illinois. The Braidwood Station is located in northeastern Illinois, near the town of Braidwood, Illinois, and near the Kankakee River approximately 60 miles from Chicago, Illinois. The NRC issued the Byron construction permit on December 31, 1975, and operating licenses on February 14, 1985 (Unit 1), and January 30, 1987 (Unit 2). The NRC issued the Braidwood construction permit on December 31, 1975, and operating licenses on July 2, 1987 (Unit 1), and May 20, 1988 (Unit 2). Each BBS unit has a Westinghouse Electric Corporation (Westinghouse) four-loop pressurized water reactor (PWR) and a turbine-generator furnished by Westinghouse. For both stations, Babcock & Wilcox supplied the steam generators for Unit 1, and Westinghouse supplied the steam generators for Unit 2. Sargent & Lundy was the architect-engineer for both stations. Each containment is a PWR dry ambient containment structure. The BBS licensed power outputs are about 3,645 megawatts thermal with a gross electrical output of approximately 1,260 megawatts electric.

Unless otherwise indicated, this SER presents the status of the staff's review of information submitted through April 17, 2015, the cutoff date for consideration in the SER. The two open items previously identified in the SER with Open Items, issued October 30, 2014, have been closed (see Section 1.5); therefore, no open items remain to be resolved before the final determination is reached by the staff on the LRA.

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ABBREVIATIONS AND GLOSSARY TERMS

°F/hr	degree(s) Fahrenheit per hour
μm/yr	micrometer(s) per year
A/LAI	Applicant/Licensee Action Item
AA	all aluminum
AAC	alternate AC
AC	alternating current
ACAR	aluminum conductor aluminum alloy reinforced
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
ACSR	aluminum conductor steel reinforced
ADAMS	Agencywide Documents Access and Management System
AERM	aging effect requiring management
AFW	auxiliary feedwater
ALARA	as low as is reasonably achievable
AMP	aging management program
AMR	aging management review
AOO	anticipated operational occurrence
applicant	Exelon Generation Company, LLC
ART	adjusted reference temperature
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient(s) without scram
AWWA	American Water Works Association
B&PV	Boiler and Pressure Vessel
B&W	Babcock & Wilcox
BBS	Byron and Braidwood Stations
BMI	bottom-mounted instrumentation
Braidwood	Braidwood Station, Units 1 and 2
BWR	boiling-water reactor
Byron	Byron Station, Units 1 and 2
CAF	containment access facility
CAP	corrective action program
CASS	cast austenitic stainless steel
CCA	common cause analysis/analyses
CE	Combustion Engineering
CFR	*Code of Federal Regulations*
CLB	current licensing basis/bases
CLSM	controlled low strength material
cm^3	cubic centimeter(s)
CMTR	certified material test report
CO_2	carbon dioxide
CPVC	chlorinated polyvinyl chloride
CRDM	control rod drive mechanism
CRGT	control rod guide tube

CSS	containment spray system
CST	condensate storage tank
cSt	centistoke(s)
CUF	cumulative usage factor
CUF_{en}	environmentally adjusted cumulative usage factor
CVCS	chemical and volume control system
DBA	design-basis accident
DBE	design-basis event
DG	diesel generator
DO	dissolved oxygen
dpa	displacements per atom
E	energy
EAF	environmentally assisted fatigue
ECCS	emergency core cooling system
EDG	emergency diesel generator
EFPY	effective full-power year(s)
EPDM	ethylene propylene diene monomer
EPR	ethylene propylene rubber
EPRI	Electric Power Research Institute
EQ	environmental qualification
EQP	Environmental Qualification Program
ESF	engineered safety feature
ETA	ethanolamine
Exelon	Exelon Generation Company, LLC
FASA	Focused Area Self-Assessment
F_{en}	environmental fatigue life correction factor
FMECA	failure modes, effects, and criticality assessment
FR	*Federal Register*
FSAR	final safety analysis report
ft	foot/feet
GALL	Generic Aging Lessons Learned
GDC	general design criterion/criteria
GEIS	Generic Environmental Impact Statement
GL	generic letter
gpm	gallon(s) per minute
HAZ	heat affected zone(s)
HDPE	high-density polyethylene
HELB	high-energy line break
HPSI	high-pressure safety injection
HVAC	heating, ventilation, and air conditioning
I&C	instrumentation and control(s)
I&E	inspection and evaluation
IASCC	irradiation-assisted stress-corrosion cracking
IEEE	Institute of Electrical and Electronics Engineers
IGSCC	intergranular stress-corrosion cracking

ILRT	integrated leak rate test
IN	information notice
in.	inch(es)
INPO	Institute of Nuclear Power Operations
IPA	integrated plant assessment
ISG	interim staff guidance
ISI	inservice inspection
ksi	kilogram(s) per square inch
kV	kilovolt(s)
LAS	low-alloy steel
LBB	leak-before-break
LCO	limiting condition(s) for operation
LER	Licensee Event Report
LLRT	local leakage rate test
LOCA	loss-of-coolant accident
long-lived	not subject to periodic replacement based on a qualified life or specified time period
LR-ISG	license renewal interim staff guidance
LRA	license renewal application
LTOP	low temperature overpressure protection
LWR	light-water reactor
MC	metal containment
MEB	metal-enclosed bus
MEQ	mechanical environmental qualification
MeV	megaelectron volt
MIC	microbiologically influenced corrosion
MoS_2	molybdenum disulfide
MPA	methoxypropylamine
mpy	mil per year
MRP	Materials Reliability Program
MRV	minimum required prestressing force or value
MSIP®	Mechanical Stress Improvement Process
MSIV	main steam isolation valve
MSLB	main steamline break
MUR	measurement uncertainty recapture
n/cm^2	neutrons per square centimeter
NACE	National Association of Corrosion Engineers
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NEPA	National Environmental Policy Act
NFPA	National Fire Protection Association
NPS	nominal pipe size
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSSS	nuclear steam supply system

OBE	operating basis earthquake
ODSCC	outer-diameter/outside-diameter stress-corrosion cracking
OE	operating experience
OI	open item
OPEX	[Exelon] Operating Experience
OSG	original steam generator
P-T	pressure-temperature
P&ID	piping and instrumentation diagram
passive	without moving parts or a change in configuration or properties
Pb	lead
PEO	period(s) of extended operation
pH	potential of hydrogen
PLL	predicted lower limit
ppm	part(s) per million
PSARV	pressurizer safety and relief valve
psid	pound(s) per square inch differential
PTFE	polytetrafluoroethylene
PTLR	pressure-temperature limits report
PTS	pressurized thermal shock
PVC	polyvinyl chloride
PVCO	oriented polyvinyl chloride
PVDF	polyvinylidene fluoride
PWR	pressurized-water reactor
PWSCC	primary water stress corrosion cracking
PWST	primary water storage tank
QA	quality assurance
RAI	request for additional information
RCCA	rod cluster control assembly
RCFC	reactor containment fan cooling
RCL	reactor coolant loop
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RCSC	Research Council on Structural Connections
RG	regulatory guide
RHR	residual heat removal
RI-ISI	risk-informed inservice inspection
RIS	Regulatory Issue Summary
RPV	reactor pressure vessel
RSG	replacement steam generator
Rule	10 CFR Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants”
RVI	reactor vessel internal
RVLIS	reactor vessel level instrumentation system
RWST	refueling water storage tank
SAT	system auxiliary transformer
SBO	station blackout

SC	structure and component
SCC	stress-corrosion cracking
scoping	within the scope of license renewal
screening	subject to an AMR
SER	safety evaluation report
SFP	spent fuel pool
SIS	safety injection system
SR/IR	source range/intermediate range
SRP-LR	Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants
SS	stainless steel
SSC	system, structure, and/or component
staff	U.S. NRC staff
SWOL	structural weld overlay
SX	[essential] service water
SXCT	essential service water cooling tower
TAC	Technical Assignment Control
TF	tendon force
TLAA	time-limited aging analysis
TMI	Three Mile Island
TOC	total organic carbon
TR	technical report
TS	technical specification
U.S.	United States
UFSAR	updated final safety analysis report
UHS	ultimate heat sink
USE	upper-shelf energy
UT	ultrasonic testing
UV	ultraviolet
V	volt(s)
Vac	volt(s) alternating current
VT	Visual Testing (method, e.g., VT-1)
WCAP	Westinghouse Commercial Atomic Power
Westinghouse	Westinghouse Electric Corporation
WOG	Westinghouse Owners Group
XLPE	cross-linked polyethylene

SECTION 4

TIME-LIMITED AGING ANALYSES

4.1 Identification of Time-Limited Aging Analyses

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

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

(1) The evaluation, analysis, or calculation must involve a system, structure, or component (SSC) that is within the scope of license renewal, as mandated in 10 CFR 54.4(a).

(2) The evaluation, analysis, or calculation must consider the effect or effects of aging.

(3) The evaluation, analysis, or calculation must be based on time-limited assumptions that are defined by the current operating term (for example, 40 years).

(4) The evaluation, analysis, or calculation must have been determined to be relevant by the applicant in making a safety determination.

(5) The evaluation, analysis, or calculation must involve conclusions, or provide the basis for conclusions, related to the capability of the SSC to perform its intended function(s), as described in 10 CFR 54.4(b).

(6) The evaluation, analysis, or calculation must be contained or incorporated by reference in the CLB.

For each evaluation, analysis, or calculation that is a TLAA, the applicant must demonstrate that the TLAA will be acceptable for the period of extended operation in accordance with one of the following three acceptance criteria for TLAAs in 10 CFR 54.21(c)(1):

(i) demonstration that the evaluation, analysis, or calculations of record will remain valid for the period of extended operation

(ii) demonstration that the evaluation, analysis, or calculation has been projected to the end of the period of extended operation

(iii) demonstration that the impact of the effects of aging on the intended function(s) will be adequately managed during the period of extended operation

In the LRA, the applicant dispositioned each TLAA (i.e., identified the criterion satisfied) based on one of the above three acceptance criteria per 10 CFR 54.21(c)(1). The staff reviewed the applicant's disposition for each TLAA against the requirements per 10 CFR 54.21(c)(1).

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

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

Pursuant to 10 CFR 54.22, applicants must identify any facility technical specification (TS) changes or additions that are necessary to manage the effects of aging during the period of extended operation, along with a justification for those TS changes or additions.

4.1.1 Summary of Technical Information in the Application

4.1.1.1 Identification of TLAAs

The applicant stated that the list of TLAAs for the LRA was identified using methods that are consistent with those provided in the SRP-LR and 10 CFR Part 54, "Requirements for Renewal of Operating License for Nuclear Power Plants." The applicant stated that a list of potential TLAAs was assembled from the following sources: (a) the SRP-LR, (b) the Generic Aging Lessons Learned Report (GALL Report), (c) Nuclear Energy Institute (NEI) Report NEI-95-10, Revision 6, (d) NRC statement of considerations on 10 CFR Part 54, and (e) prior LRAs and associated NRC requests for additional information (RAIs) and SERs.

The applicant also stated that the following CLB and design basis documentation sources were searched to identify potential TLAAs: (a) the updated final safety analysis report (UFSAR) for the Byron and Braidwood Stations (BBS), (b) plant TSs and TS bases documents, (c) the plant Technical Requirements Manuals, (d) docketed licensing correspondence, (e) NRC SERs, (f) design basis documents (DBDs), (g) Westinghouse Electric Corporation (Westinghouse) design analyses and reports, (h) vendor design analyses and reports, and (i) environmental qualification (EQ) binders.

LRA Table 4.1-1 provides the applicant's comparison of the BBS TLAAs to those analyses that are listed as potential TLAAs in the SRP-LR. LRA Table 4.1-2 provides a summary listing of the TLAAs that the applicant has identified as being applicable to BBS and the criteria that are used to accept these TLAAs in accordance with either 10 CFR 54.21(c)(1)(i), (ii), or (iii).

4.1.1.2 Identification of Exemptions

In LRA Section 4.1.5, the applicant stated that the exemptions for BBS were identified through a review of the UFSAR, the operating licenses, the TSs, NRC SERs, ASME Section XI program documentation, fire protection documents, the staff's Agencywide Documents Access and Management System (ADAMS) database, and docketed correspondence. The applicant stated that it identified the following regulatory exemptions that are based on a TLAA:

(a) Those exemptions that were granted on July 13, 1995, November 29, 1996, December 12, 1997, and January 16, 1998, which collectively permit the applicant to use ASME Code Case N-514 as the basis for establishing the low-temperature overpressure protection (LTOP) system enable temperature setpoints and for establishing the LTOP pressure lift setpoints at 110 percent of that which would be established using the methods of analysis in the ASME Code, Section XI, Appendix G.

(b) An exemption that was granted on August 8, 2001, allowing the applicant to use methodologies in ASME Code Cases N-588 and N-640 as alternative bases for generating the pressure-temperature (P-T) limit curves for BBS.

(c) An exemption that was granted on November 22, 2006, allowing the applicant to use the alternative methodology in Westinghouse Proprietary Report No. Westinghouse Commercial Atomic Power (WCAP)-16143-P for establishing the minimum temperature requirements for the P-T limit curves for BBS.

The applicant stated that all of these exemptions are based on the P-T limit curves that are in effect for 32 effective full-power years (EFPY) of operation. The applicant stated that, based on the EFPY projections described in LRA Section 4.2.1, the BBS units are expected to exceed 32 EFPY prior to entering the period of extended operation, thereby necessitating updates to the P-T limit curves in accordance with 10 CFR Part 50, Appendix G, prior to the period of extended operation. The applicant stated that it anticipates that these exemptions will not be required for the period of extended operation. The applicant clarified that if the BBS reactors do not reach 32 EFPY prior to the period of extended operation, the exemptions are acceptable for the period of extended operation because the staff did not place a limitation on the time of applicability for the exemptions.

4.1.1.3 Identification of Technical Specification Changes or Additions Needed to Manage Aging during the Period of Extended Operation

LRA Appendix D provides the applicant's evaluation regarding whether the LRA would need to include any facility TS changes or additions in order to manage the effects of aging during the period of extended operation. The applicant stated that it performed a review of the information in the LRA and the TS and determined that the LRA did not need to include any TS changes or additions to manage the effects of aging during the period of extended operation.

4.1.2 Staff Evaluation

4.1.2.1 Identification of TLAAs

The staff reviewed the applicant's methodology and results for identifying the TLAAs for the LRA against the six criteria for TLAA identification in 10 CFR 54.3 and the generic list of TLAAs in SRP-LR Section 4.1, including those in SRP-LR Tables 4.1-2 and 4.1-3 as applicable to the

CLB for the reactor units. The staff used the “acceptance criteria” in SRP-LR Section 4.1.2 and the “review procedures” in SRP-LR Section 4.1.3 as the basis for its review.

4.1.2.1.1 Evaluations, Analyses, and Calculations That Conform to the Definition of a TLAA, as Defined in 10 CFR 54.3

The staff noticed that LRA Table 4.1-2 identifies that the following analyses in the CLB meet the definition of a TLAA in 10 CFR 54.3:

- LRA Section 4.2 – Reactor Vessel Neutron Embrittlement Analysis
  - LRA Section 4.2.1, Neutron Fluence Projections
  - LRA Section 4.2.2, Upper-Shelf Energy
  - LRA Section 4.2.3, Pressurized Thermal Shock
  - LRA Section 4.2.4, Adjusted Reference Temperature
  - LRA Section 4.2.5, Pressure-Temperature Limits
  - LRA Section 4.2.6, Low Temperature Overpressure Protection (LTOP) Analyses
- LRA Section 4.3 – Metal Fatigue
  - LRA Section 4.3.1, Transient Inputs to Fatigue Analyses
  - LRA Section 4.3.2, ASME Code Section III, Class 1, Class 2, and Class 3 Fatigue Analyses
  - LRA Section 4.3.3, ASME Code Section III, Class 2 and 3 and ANSI B31.1 Allowable Stress Analyses
  - LRA Section 4.3.4, Class 1 Component Fatigue Analyses Supporting GSI-190 Closure
  - LRA Section 4.3.5, Reactor Vessel Internals Fatigue Analyses
  - LRA Section 4.3.6, High-Energy Line Break (HELB) Analyses Based on Fatigue
  - LRA Section 4.3.7, NRC Bulletin 88-11 Revised Fatigue Analysis of the Pressurizer Surge Line for Thermal Cycling and Stratification
  - LRA Section 4.3.8, ASME Code Section III, Subsection NF, Class 1 Component Supports Allowable Stress Analyses
  - LRA Section 4.3.9, Fatigue Design of Spent Fuel Pool Liner and Spent Fuel Storage Racks for Seismic Events
  - LRA Section 4.3.10, Pressurizer Heater Sleeve Structural Assessment
- LRA Section 4.4 – Environmental Qualification (EQ) of Electric Components
- LRA Section 4.5 – Concrete Containment Tendon Prestress Analysis
- LRA Section 4.6 – Containment Liner Plate, Metal Containments, and Penetrations Fatigue Analyses
  - LRA Section 4.6.1, Containment Liner Plates Fatigue
  - LRA Section 4.6.2, Containment Airlocks and Hatches Fatigue
  - LRA Section 4.6.3, Containment Electrical Penetrations Fatigue

- LRA Section 4.6.4, Containment Piping Penetrations Fatigue
- LRA Section 4.6.5, Fuel Transfer Tube Bellows Fatigue
- LRA Section 4.6.6, Recirculation Sump Guard Piping Bellows Fatigue

- LRA Section 4.7 – Other Plant-Specific TLAAs
  - LRA Section 4.7.1, Leak-Before-Break
  - LRA Section 4.7.2, Crane Load Cycle Limits
  - LRA Section 4.7.3, Mechanical Environmental Qualification
  - LRA Section 4.7.4, Residual Heat Removal Heat Exchangers Tube Side Inlet and Outlet Nozzles Fracture Mechanics Analysis
  - LRA Section 4.7.5, Reactor Coolant Pump Flywheel Crack Growth Analysis
  - LRA Section 4.7.6, Byron Unit 2 Pressurizer Seismic Restraint Lug Flaw Evaluation
  - LRA Section 4.7.7, Braidwood Unit 2 Feedwater Pipe Elbow Crack Growth Evaluation
  - LRA Section 4.7.8, Analyses Supporting Flaw Evaluations of Primary System Components

The staff determined that the applicant's identification of these TLAAs is consistent with the staff's list of generic TLAAs in SRP-LR Table 4.1-2, "Generic Time Limited Aging Analyses," and list of potential plant-specific TLAAs in SRP-LR Table 4.1-3, "Examples of Potential Plant-Specific TLAAs." Based on this review, the staff finds that the identification of these TLAAs is acceptable because it is in accordance with 10 CFR 54.21(c)(1). The staff's evaluation of the applicant's basis for accepting these TLAAs in accordance with either 10 CFR 54.21(c)(1)(i), (ii), or (iii) is documented in SER Sections 4.2 to 4.7 and their subsections.

4.1.2.1.2 Evaluation of Applicant's List of Evaluations, Analyses, and Calculations That Do Not Conform to the Definition of a TLAA, as Defined in 10 CFR 54.3

SRP-LR Table 4.1-2, "Generic Time Limited Aging Analyses," and SRP-LR Table 4.1-3, "Examples of Potential Plant-Specific Time Limited Aging Analyses," provide a collective list of analyses that may be part of an applicant's CLB and that may need to be identified as TLAAs in the LRA. Of the 14 potential plant-specific TLAAs listed in SRP-LR Table 4.1-3, the applicant identified 9 that were TLAAs for the LRA. These are the TLAAs listed in SER Section 4.1.2.1.1, which reference the TLAA in LRA Section 4.2.6 (i.e., the TLAA on LTOP) and the eight plant-specific TLAAs in LRA Sections 4.7.1 – 4.7.8. The staff reviewed the information in LRA Table 4.1-3 against the list of potential plant-specific TLAAs in SRP-LR Table 4.1-3 in order to evaluate the validity of the applicant's bases for not identifying the remaining analyses in SRP-LR Table 4.1-3 as TLAAs in the LRA. In addition, the applicant also identified that the containment corrosion analysis in LRA Table 4.1-2 is not applicable to the BBS CLB and does not need to be identified as a TLAA for the LRA. For the analyses that the applicant claimed were not TLAAs for the CLB, the staff's evaluations of the applicant's "absence of a TLAA" bases are given in the subsections that follow.

Lack of an Inservice Local Metal Containment Corrosion Analysis. SRP-LR Table 4.1-2 identifies inservice local metal containment (MC) corrosion analyses as a generic TLAA. The

SRP-LR identifies that these analyses may conform to the definition of a TLAA in 10 CFR 54.3(a) and need to be identified as TLAAs in accordance with 10 CFR 54.21(c)(1).

LRA Table 4.1-1 states that the CLB does not include any inservice local MC corrosion analyses that meet the definition of a TLAA in 10 CFR 54.3(a) or would need to be identified as TLAAs in accordance with the requirement in 10 CFR 54.21(c)(1).

The staff reviewed UFSAR Section 6.2.1 to evaluate the validity of the applicant's basis for its conclusion. UFSAR Section 6.2.1 identifies that the containment structures are designed as prestressed-concrete shell structures, with each structure being made up of a cylinder with a shallow dome roof and a flat foundation slab. The UFSAR states that each containment structure is lined on the inside with steel plate, which acts as a leak tight membrane. The staff noticed that the reactor units are not designed with metallic containment structures. As such, the staff concludes that the CLB does not include this type of plant-specific TLAA because the staff has confirmed the reactor designs do not rely on metallic containment structures as the basis for maintaining containment integrity during normal operating conditions, transient operating conditions, or postulated loss-of-coolant accident (LOCA) conditions.

<u>Lack of a TLAA for Evaluating Intergranular Separations in the Heat Affected Zones (HAZ) of Reactor Vessel Low-Alloy Steel Forging Components</u>. SRP-LR Section 3.1.2.2.5 identifies that SA-508, Class 2 forging components in Babcock & Wilcox (B&W)-designed reactor vessels may be susceptible to cracking in the welds that join the reactor vessel cladding to the reactor vessel forging components. SRP-LR Table 4.1-3 identifies that the CLB for pressurized-water reactors (PWRs) may include plant-specific reactor vessel underclad cracking analyses. The SRP-LR identifies that theses analyses may conform to the definition of a TLAA in 10 CFR 54.3(a) and need to be identified as TLAAs in accordance with the requirement in 10 CFR 54.21(c)(1).

In LRA Section 3.1.2.2.5 and Footnote 2 of LRA Table 4.1-1, the applicant stated that the phenomenon of reactor vessel underclad cracking is not applicable to the design of the BBS reactor vessels because the reactor vessels were not designed by B&W. Therefore, the applicant stated that the CLB does not include any analysis on reactor vessel underclad cracking that would need to be identified as a TLAA in accordance with 10 CFR 54.21(c)(1). Instead, the applicant stated that procedural protocols were implemented to control the heat input that was used to join the reactor vessel cladding to any SA-508, Class 2 low-alloy steel (LAS) components in the reactor vessels.

The staff evaluated the applicant's basis for claiming that the CLB does not include any reactor vessel underclad cracking TLAAs in SER Section 3.1.2.2.5, where the staff found that the applicant does not rely on an analysis to demonstrate that potential underclad cracks are acceptable for the period of extended operation. Instead, the staff has confirmed that the applicant relies on procedural welding heat controls (as described in the UFSAR) to preclude or underclad cracking from occurring on the BBS reactors.

<u>Lack of a Fatigue Analysis for the Main Steam Supply Lines to Turbine-Driven Auxiliary Feedwater Pumps</u>. SRP-LR Table 4.1-3 identifies that the CLB for PWRs may include fatigue analyses of the lines that provide steam to the turbine-driven auxiliary feedwater (AFW) pumps. The SRP-LR identifies that these analyses may conform to the definition of a TLAA in 10 CFR 54.3(a) and need to be identified as TLAAs in accordance with the requirement in 10 CFR 54.21(c)(1).

In LRA Table 4.1-1, the applicant stated that the CLBs for the units do not include a TLAA related to main steam supply line fatigue analyses for the AFW pumps because the units do not include steam driven AFW pumps.

The staff reviewed UFSAR Section 10.4.9.2 to evaluate the validity of the applicant's basis for its conclusion. The staff noticed that UFSAR Section 10.4.9.2 identifies that the AFW systems at BBS consist of the following two subsystems: (1) one subsystem that is designed with a motor-driven AFW pump powered by one of the emergency onsite power systems supplied from a diesel generator (DG), and (2) a second subsystem that is designed with a motor-driven AFW pump powered by a diesel engine through a gear increaser. The staff noticed that the CLBs do not rely on turbine-driven AFW pumps as a source of emergency AFW.

As such, the staff concludes that the applicant has provided an adequate basis for concluding that the CLBs do not include this type of plant-specific TLAA because the staff has confirmed the plant designs do not utilize a turbine-driven AFW system as a potential source of AFW into the secondary sides of the steam generators.

Lack of a TLAA on Flow-Induced Vibrations for Reactor Vessel Internal Components. SRP-LR Table 4.1-3 identifies that the CLB for PWRs may include flow-induced vibration analyses for the reactor vessel internals (RVIs) components. The SRP-LR identifies that these analyses may conform to the definition of a TLAA in 10 CFR 54.3(a) and need to be identified as TLAAs in accordance with the requirement in 10 CFR 54.21(c)(1).

In LRA Table 4.1-1, the applicant stated that the CLBs do not include any RVI flow-induced vibration analyses that conform to the definition of a TLAA in 10 CFR 54.3(a) or would need to be identified as TLAAs in accordance with the requirement in 10 CFR 54.21(c)(1). Specifically, LRA Section 4.3.5 states that the analyses associated with flow-induced vibration of the RVIs are not based on any time-dependent assumptions that would cause them to be considered a TLAA in accordance with 10 CFR 54.3(a), Criterion 3. The applicant stated that these analyses concluded that the stress ranges for the RVI components remain below the endurance limit of 10^{11} cycles on the applicable ASME fatigue curves. The applicant stated that the endurance limit is the stress range below that which the material will not experience fatigue failure. The applicant stated that, since the stress ranges remain below the endurance limit, the number of the stress range cycles is not limited over the current operating life and, therefore, the analyses are not based on any time-dependent assumptions defined by the current operating terms.

The staff reviewed the UFSAR for relevant information on flow-induced vibrations of the RVI components. The applicant's basis for evaluating the impacts of flow induced vibrations on the structural integrity and intended functions of RVI components is given in UFSAR Section 3.9. UFSAR Section 3.9.2.3 indicates that the design bases rely on previous RVI flow-induced vibration models and tests that were performed at the Indian Point Unit 2 and Trojan nuclear power plants and that these models and tests are the basis for assessing flow-induced vibrations of the RVI components at BBS.

These models and tests are summarized in the following Westinghouse technical reports (TRs):

- WCAP-8317-A, "Prediction of the Flow-Induced Vibration of Reactor Internals by Scale Model Tests," July 1975
- WCAP-8780, "Verifications of Neutron Pad and 17 X 17 Guide Tube Designs by Preoperational Tests on the Trojan Unit 1 Plant," May 1976

In UFSAR Section 3.9.5.2, the applicant stated that the design of the RVI components is based on the design basis loading conditions for normal operating, upset, emergency, and faulted condition transients listed on UFSAR pages 3.9-96 and 3.9-97. The staff noticed that, for the RVI components, vibratory loads (including those that would occur during postulated operational basis earthquake conditions) are listed as normal operating condition loads for the RVI components. The LRA states that the magnitude of the vibration loads for the RVI components are lower than the stress endurance limits for inducing fatigue in components. However, the staff could not determine whether this type of technical basis was established in either WCAP-8317-A or WCAP-8780. On February 26, 2014, the staff issued RAI 4.1-1, requesting that the applicant clarify whether WCAP-8317-A or WCAP-8780 establishes the basis for concluding that the RVI vibration stress loads are lower than the endurance limit for the initiation of high-cycle fatigue. If not, the staff asked to applicant to identify and justify the document in the CLB that establishes and is relied upon for this position.

The applicant responded to RAI 4.1-1 by letter dated March 28, 2014. In its response, the applicant stated that both WCAP-8317-A and WCAP-8780 establish the basis for the RVI flow-induced vibration analysis in the CLB, as discussed in UFSAR Section 3.9.2.3. The applicant stated that the initial basis for high-cycle vibratory analyses and scale model testing is provided in WCAP-8317-A and that the conclusions of WCAP-8317-A were confirmed by instrumented plant hot functional test results performed at Trojan Unit 1, as discussed in WCAP-8780.

The applicant stated that WCAP-8780 demonstrates that the stress levels due to flow-induced vibration on the RVI critical structural components were well below the endurance limits for the component materials and therefore will not experience fatigue failure.

The staff concluded that the applicant's response demonstrates that the assessment of flow-induced vibrations in the RVI components is not within the scope of any fatigue growth parameters defined by the current operating term because (a) the vibrational stresses on the components are lower than the stress threshold for initiating fatigue cracks and (b) this demonstrates that the assessment of flow-induced vibrations in the RVI components does not involve time-limited assumptions defined by the current operating term. Therefore, based on this review, the staff finds that the applicant has provided a valid basis for concluding that the LRA does not need to include any TLAA for flow-induced vibrations because the applicant demonstrated that the assessment of flow-induced vibration does not involve time-limited assumptions defined by the current operating term such that Criterion 3 in 10 CFR 54.3(a) is not met. The staff's concerns described in RAI 4.1-1 are resolved.

<u>Lack of a Ductility Reduction Analysis (TLAA) for Reactor Vessel Internal (RVI) Components</u>. SRP-LR Section 3.1.2.2.3, Subsection 3, and SRP-LR Table 4.1-3 both identify that the CLB for PWRs designed by B&W may include reduction of ductility analyses for the RVI components in

the plant design. The SRP-LR identifies that the applicable analysis is given in B&W TR BAW-2248 and may conform to the definition of a TLAA in 10 CFR 54.3(a) and need to be identified as a TLAA in accordance with the requirement in 10 CFR 54.21(c)(1).

In LRA Section 3.1.2.2.3, Subsection 3 and in LRA Table 4.1-1, the applicant stated that the CLB do not include these types of analyses. The staff's evaluation of the applicant's basis for claiming that the ductility reduction analyses in BAW-2248 is not a TLAA for the CLB is given in SER Section 3.1.2.2.3, item 3, where the staff found that the RVI components at BBS are not within the scope of the generic analysis that was evaluated in TR BAW-2248.

Lack of Metal Corrosion Allowance TLAA. SRP-LR Table 4.1-3 identifies that some plant CLB may include metal corrosion analyses for metallic components in the plant designs. The SRP-LR identifies that these analyses may conform to the definition of a TLAA in 10 CFR 54.3(a) and need to be identified as TLAAs in accordance with 10 CFR 54.21(c)(1).

In LRA Table 4.1-1, the applicant stated that the CLBs do not include any component-specific metal corrosion allowance analyses applicable to 40-year operation for BBS.

The staff reviewed the UFSAR for relevant information and noticed that it does make reference to one metal corrosion allowance. Specifically, the staff noticed that UFSAR Section 5.4.2.5.4 refers to B&W TR 222-7720-PR05, Revision 3, "Replacement Steam Generators Secondary Side Corrosion Allowance Values for Design of Analysis." However, the UFSAR does not state whether this report is being relied upon as part of the CLB or design bases for the reactor units.

By letter dated February 26, 2014, the staff issued RAI 4.1-2, requesting that the applicant clarify whether B&W TR 222-7720-PR05, Revision 3, is being relied upon for the CLB or BBS design bases. If so, the staff asked the applicant to justify why the metal corrosion allowance analysis in this report would not need to be identified as a TLAA for the secondary side of the steam generators at BBS.

The applicant responded to RAI 4.1-2 by letter dated March 28, 2014. In its response, the applicant stated that B&W TR 222-7720-PR05, Revision 3 is relied on for the CLB and that the scope of the report is used to assess general corrosion losses in the secondary side surfaces of the steam generators during normal operations and chemical cleaning activities. The applicant stated that the report includes technical bases for adding an additional metal corrosion allowance to the design thickness of these steam generator surfaces based on vendor guidance and industry experience. The applicant stated that the report is not a TLAA because it does not involve conclusions or provide the basis for drawing conclusions related to the capability of the steam generators to perform their intended functions, as defined in 10 CFR 54.4(b). The applicant concluded that the corrosion allowance basis in B&W TR No. 222-7720-PR05, Revision 3, does not meet Criterion 5 in 10 CFR 54.3(a).

The term "metal corrosion allowance" refers to and represents an additional amount of metal that was included in the original design of a metallic component beyond the amount of metal that was required to be included in the design and fabrication of the component by its design code. Licensees that previously opted to include a metal corrosion allowance in the original design of a particular metallic component did so as an additional mitigative design measure for protecting the component against loss of material effects that could be induced by potential corrosive aging mechanisms (e.g., loss of material induced by general, pitting, or crevice corrosion). SRP-LR Section 4.1 is based, in part, on assumption that, for metallic components that were designed with metal corrosion allowances, the CLB may have included time-dependent analyses that determined how much additional metal was to be included in the design and fabrication of the components. However, the staff noticed that the amount of additional metal (i.e., corrosion allowances) may also have been based on other design factors, such as operating experience (OE), simple vendor recommendations, or a design decision by the plant owner that was forwarded to the fabricator and vendor of the particular component prior to component fabrication. The staff also noticed that the amount of additional metal that was added as a design feature goes beyond the design requirements for the components, unless the corrosion allowance was specifically required to be included in the component design by the design code for the component.

The applicant's response to RAI 4.1-2 demonstrates that the additional corrosion allowance added to the wall thickness of the secondary side steam generator surfaces is based on vendor recommendations and industry experience and is not based on any analysis that would need to be identified as a TLAA, as defined by the six criteria in 10 CFR 54.3(a). The staff also noticed that this demonstrates that the additional metal corrosion allowance that was added into the design of the steam generator secondary side shell surfaces is not based on any analysis that involves conclusions or provides the basis for drawing conclusions related to the capability of the steam generators to perform their intended functions, as defined in 10 CFR 54.4(b). Therefore, based on this review, the staff finds that the applicant has provided a valid basis for concluding that the metal corrosion allowance for the steam generators does not involve a TLAA because: (a) the applicant demonstrated that the metal corrosion allowance is not based on an analysis that involves conclusions or provides the basis for drawing conclusions related to the capability of the steam generators to perform their intended functions, as defined in 10 CFR 54.4(b), and (b) this demonstrates that the assessment of the metal corrosion allowance for the steam generators does not conform to Criterion 5 in 10 CFR 54.3(a). The staff's concern described in RAI 4.1-2 is resolved.

<u>Other Potential Plant-Specific TLAAs Not Referenced in the SRP-LR</u>. The staff reviewed the information in the UFSAR to determine whether the design bases include any additional plant analyses, evaluations, calculations, or reports that would need to be identified as plant-specific TLAAs for the LRA. The staff did not identify any other plant-specific or generic analyses, evaluations, calculations, or reports that would need to be identified as plant-specific TLAAs for the LRA.

4.1.2.2 Identification of Exemptions

In LRA Section 4.1.5, the applicant identified six exemptions that were based on a TLAA and were granted in accordance with the staff's regulatory exemption acceptance requirements in 10 CFR 50.12. The staff noticed that all of these exemptions are based on the P-T limit curves that are in effect for 32 EFPY. The applicant made the following statement with respect to whether these exemptions would be applied during the period of extended operation:

> All six of the above exemptions are associated with Pressure-Temperature (P-T) limits that are applicable for 32 effective full-power years (EFPY). Based on EFPY projections described in LRA Section 4.2.1, it is expected that Byron Units 1 and 2 and Braidwood Units 1 and 2 will exceed 32 EFPY prior to the period of extended operation (PEO), thereby necessitating replacement of the P-T limit curves in accordance with 10 CFR 50, Appendix G, prior to the PEO. It is therefore anticipated that these exemptions will not be required to be in effect during PEO. If however, 32 EFPY is not reached prior to the PEO for any reason for any of the BBS units, continuation of these exemptions into the PEO, if

necessary, is acceptable because the use of the exemptions as a basis for the 32 EFPY P-T limits was approved by the NRC without a limitation with respect to plant operation beyond the original license term. The above exemptions and their acceptability are not tied to or limited by the original license term.

The staff determined that the four exemptions granting permission for use of ASME Code Case N-514 and the establishment of the LTOP system setpoints are relevant to the staff's evaluation of the applicant's basis for accepting the TLAA on LTOP in accordance with 10 CFR 54.21(c)(1)(iii). The staff evaluated whether these exemptions will be needed for the period of extended operation as part of the staff's review of the UFSAR supplement for the LTOP TLAA in SER Section 4.2.6.3.

The staff also determined that the exemptions granting permission for use of ASME Code Cases N-640 and N-588, and the minimum temperature requirements methodology in WCAP-16143-P are relevant to the staff's evaluation of the applicant's basis for accepting the P-T limits TLAA in accordance with 10 CFR 54.21(c)(1)(iii). The staff evaluated whether these exemptions will be needed for the period of extended operation as part of its review of the UFSAR supplement for the P-T limits TLAA in SER Section 4.2.5.3.

The staff did not identify any other regulatory exemptions in the CLB that were granted in accordance with 10 CFR 50.12 and were based on a TLAA. Based on this review, the staff finds that the applicant has complied with the requirement in 10 CFR 54.21(c)(2) because: (a) the applicant has identified that the regulatory exemptions granted relative to compliance with the applicable requirements of 10 CFR Part 50, Appendix G, for plant P-T limits and LTOP system setpoints are exemptions that are based on a TLAA, (b) the staff has confirmed the acceptability of the applicant's basis, and (c) the staff's review of the CLB did not identify any other regulatory exemptions that were granted in accordance with 10 CFR 50.12 and based on a TLAA.

4.1.2.3 Technical Specifications—Compliance with 10 CFR 54.22

The regulation in 10 CFR 54.22 requires the applicant to identify any additions or changes to the TS that are needed for aging management during the period of extended operation. The staff determined that LRA Appendix D provides the applicant's statement on whether the LRA would need to include any TS additions or changes to comply with the requirement in 10 CFR 54.22. The applicant stated that there are no TS changes or additions that would need to be proposed for aging management of those structures, systems and components (SSCs) that were within the scope of license renewal and subject to an aging management review (AMR).

The staff reviewed the TSs to determine whether the CLB includes any TS requirements that relate to aging management of SSCs that are subject to an AMR. The staff found the following TS Administrative Control requirements may have a relationship to aging management programs (AMPs) or TLAAs that are credited for aging management:

- TS 5.5.2, "Primary Sources Outside Containment," in relation to performing visual examinations of the recirculating loops in the chemical and volume control systems, containment spray systems, residual heat removal (RHR) systems, and safety injection (SI) systems using the applicant's External Surfaces Monitoring of Mechanical Components Program (LRA Section B.2.1.23)

- TS 5.5.5, “Cyclic Component or Transient Limit,” in relation to the applicant’s basis for accepting fatigue-related TLAAs in accordance with 10 CFR 54.21(c)(1)(iii) and for managing cracking due to fatigue using the applicant’s Fatigue Monitoring Program (LRA Section B.3.1.1) for components with fatigue analyses
- TS 5.5.6, “Pre-stressed Concrete Containment Tendon Surveillance Program” in relation to management of loss of material due to corrosion in the containment tendons systems and implementation of the applicant’s Concrete Containment Tendon Prestress Program (LRA Section B.3.1.2)
- TS 5.5.9, “Steam Generator Program,” in relation to management of cracking in the steam generator tubes and implementation of the applicant’s Steam Generators Program (LRA Section B.2.1.10)
- TS 5.5.10, “Secondary Water Chemistry,” in relation to management of corrosion-related aging effects in non-Class 1 components using the applicant’s Water Chemistry Program (LRA Section B.2.1.2)
- TS 5.5.13, “Diesel Fuel Oil Testing Program,” in relation to management of loss of material due to corrosion in the emergency diesel fuel oil storage tanks using the applicant’s Fuel Oil Chemistry Program (LRA B.2.1.18)
- TS 5.5.16, “Containment Leak Rate Testing Program,” in relation to the applicant’s basis for managing loss of material and loss of preload in containment bolting components using the applicant’s 10 CFR Part 50, Appendix J Program (LRA Section B.2.1.32)
- TS 5.6.6, “Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR),” in relation to the applicant’s evaluation of the TLAA on P-T Limits (LRA Section 4.2.5) and the TLAA on LTOP (LRA Section 4.2.6) in accordance with 10 CFR 54.21(c)(1)(iii), and for managing loss of fracture toughness in the reactor vessel components using the applicant’s programmatic process for PTLRs as defined in the TS

With the potential exception of TS 5.6.6, the staff concluded that the applicant would not need to make any amendments to these TS requirements because the staff found that the existing wording in the TSs is sufficient to ensure adequate aging management of the SSCs. The staff also did not identify any aging management criteria in the CLB that would require the applicant to propose new TS requirements for aging management. The staff’s evaluation on whether TS 5.6.6 will need to be modified in accordance the requirement in 10 CFR 54.22 is provided in SER Section 4.2.5.

4.1.3 Conclusion

Based on its review, the staff concludes that, pursuant to the requirements in 10 CFR 54.3(a) and 10 CFR 54.21(c)(1), the applicant identified those analyses in the CLB that conform to the definition of a TLAA in 10 CFR 54.3(a) and are required to be identified as TLAAs for the LRA. The staff also concludes that, pursuant to the requirements in 10 CFR 54.21(c)(2), the applicant has identified those regulatory exemptions in the CLB that were granted by the staff in accordance with the requirements in 10 CFR 50.12 and are based on a TLAA. The staff also concludes that, pursuant to the requirement in 10 CFR 54.22, and with the exception of the staff’s review of the requirements in TS 5.6.6, the applicant does not need to propose any new TS requirements or change the existing TS requirements in order to manage the effects of aging during the period of extended operation.

4.2 Reactor Vessel Neutron Embrittlement Analysis

4.2.1 Neutron Fluence Projections

4.2.1.1 Summary of Technical Information in the Application

LRA Section 4.2.1 describes the applicant's TLAA for neutron fluence projections (energy (e) greater than 1 MeV) for reactor vessel beltline and extended beltline materials. The neutron fluence projections have been used as inputs to the neutron embrittlement analyses that evaluate the loss of fracture toughness resulting from neutron irradiation. Since a request for a Measurement Uncertainty Recapture (MUR) Power Uprate has been submitted to the staff, the MUR neutron flux levels were used to calculate neutron fluence for cycles occurring after the completion of the last full operating cycle prior to November 2012. These neutron fluence analysis methodologies have been approved by the staff as described in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004, and WCAP-16083-NP-A, Revision 0, "Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry," May 2006. These methodologies conform to Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."

The applicant dispositioned the TLAA for the reactor vessel neutron fluence projections in accordance with 10 CFR 54.21(c)(1)(ii) to demonstrate that the analysis has been projected to the end of the period of extended operation.

4.2.1.2 Staff Evaluation

The staff reviewed the applicant's neutron fluence analysis for the reactor vessels, consistent with the review procedures in SRP-LR Section 4.2, which state that the applicant should identify (a) the neutron fluence for the reactor vessel at the end of the period of extended operation, (b) the staff-approved methodology used to determine the neutron fluence (or should submit the methodology for staff review), and (c) whether the methodology follows the guidance in NRC RG 1.190.

The staff noticed that LRA Table 4.2.1-1 describes the applicant's neutron fluence values (energy greater than 1 MeV) for the reactor vessel beltline and extended beltline materials at the end of the period of extended operation, consistent with the review procedures in SRP-LR Section 4.2. The applicant stated that the peak reactor vessel wall neutron fluence values for Byron Units 1 and 2 are 3.21×10^{19} n/cm^2 and 3.19×10^{19} n/cm^2, respectively, at the end of the period of extended operation, which is conservatively estimated as 57 EFPY. The applicant also stated that the peak reactor vessel wall neutron fluence values for Braidwood Station, Units 1 and 2 (Braidwood), are 3.19×10^{19} n/cm^2 and 3.16×10^{19} n/cm^2, respectively, at the end of the period of extended operation (57 EFPY).

During the AMP audit, the staff noticed that the applicant updated the maximum fluence values of Braidwood Unit 1 reactor vessel circumferential welds projected for 32 EFPY. The staff also noticed that these fluence updates were made as part of the applicant's neutron fluence TLAA for license renewal as described in WCAP-17607-NP, Revision 0, "Braidwood Station Units 1 and 2 Reactor Vessel Integrity Evaluation to Support License Renewal Time-Limited Aging Analysis," December 2012. In addition, the staff noticed that the 32-EFPY maximum fluence values of Braidwood Unit 1 reactor vessel welds, which were previously submitted to the staff as docketed information, were described in WCAP-15316, Revision 1, "Analysis of Capsule W from Commonwealth Edison Company Braidwood Unit 1 Reactor Vessel Radiation Surveillance Program," December 1999 (ADAMS Accession No. ML003713874).

The staff noticed that the updated 32-EFPY maximum fluence values of Braidwood Unit 1 reactor vessel welds are different from those described in the docketed reactor vessel surveillance report (e.g., updated fluence of 1.69×10^{19} n/cm^2 versus the previous fluence of 1.92×10^{19} n/cm^2 for weld WR-18). In addition, the staff noticed that clarification is necessary on whether the updated fluence calculations changed the axial flux profile in a manner to reduce the axial flux peaking in the mid-core region.

The staff review considered the 32-EFPY fluence values, although they do not extend to the end of the period of extended operation, because they provide a valid comparison of the flux associated with the previous and updated calculations at the same level of exposure. Since future operation is based on a projected flux value, the fluence associated with either calculation would increase linearly beyond the current cycle. Thus any conclusions drawn from a comparison of the 32-EFPY fluence values are reasonably applicable to fluence values that cover the period of extended operation.

By letter dated February 18, 2014, the staff issued RAI 4.2.1-1 requesting that the applicant justify why the updated 32-EFPY maximum fluence values of the Braidwood Unit 1 reactor vessel welds are different from those described in the docketed reactor vessel surveillance report (i.e., WCAP-15316, Revision 1). The staff also requested that, as part of the response, the applicant clarify whether the updated fluence calculations changed the axial flux profile in a manner to reduce the axial flux peaking in the mid-core region.

In addition, the staff requested that the applicant clarify whether the updated 32-EFPY fluence values for the reactor vessel welds of Byron Units 1 and 2 and Braidwood Unit 2 are different from those reported in docketed documents similar to the Braidwood Unit 1 data. The staff further requested that, if so, the applicant justify why the updated 32-EFPY maximum fluence values are different from those reported in the docketed documents and clarify whether the updated fluence calculations reduced the axial flux peaking in the mid-core region.

In its response dated March 4, 2014, the applicant stated that the methodology used for the WCAP-15316, Revision 1, calculations followed the guidance which was documented in Draft RG DG-1053 (later issued in March 2001 as RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence") and was consistent with the NRC-approved methods in WCAP-14040-NP-A, Revision 2.

In addition, the applicant indicated that the following conservatisms were involved in the adjoint transport methodology which the applicant used in its previous fluence analysis described in WCAP-15316, Revision 1. The applicant indicated that the previous methodology does not allow cycle-to-cycle water density variations in the peripheral fuel assemblies, bypass region, or downcomer region such that water densities were chosen in the analysis to conservatively envelope actual plant operational conditions. The applicant also indicated that the methodology does not account for the flattening of the axial flux distribution that naturally occurs as a function of increasing distance from the reactor core, which results in an overestimate in the high fluence areas of the reactor vessel. The applicant further indicated that the methodology does not account for the shielding effect introduced by the former plates located at several axial elevations between the core baffle plates and the core barrel.

The applicant also stated that the methodology used in the updated neutron fluence calculations for license renewal follows the guidance of RG 1.190 and has been reviewed and approved by the staff. The applicant stated that the methodology used is consistent with WCAP-14040-A, Revision 4, and this updated methodology used a forward neutron transport approach. The applicant further indicated that the fluence analysis methodology for license renewal allows water density to be varied on a cycle-specific basis and accounts for flattening of the axial flux distribution as it propagates from the core to the reactor vessel, as well as the shielding effect of the former plates. The applicant stated that this is more representative of the actual axial neutron flux distribution and reduces the overestimation of the fluence values of the high fluence areas (i.e., mid-core region).

The applicant stated that the prior, adjoint, calculations were based on a less exact representation of the axial variations in the flux levels in the core. The calculations supporting license renewal were performed by synthesizing the three-dimensional flux from lower-dimension calculations. Although the newer calculations employ a more exact approach, RG 1.190 recommends either approach, and thus the staff determined that the flux synthesis method employed in the more recent calculations was acceptable.

The applicant's response also clarified that the updated neutron fluence calculations for license renewal accounted for several cycles of actual plant operation which were treated as projections in the previous neutron fluence calculations described in WCAP-15316, Revision 1. The applicant stated that the methodology differences and updated cycle-specific calculations for license renewal result in an axial flux profile at the pressure vessel with reduced peaking in the mid-core region compared to that in WCAP-15316, Revision 1. The applicant also stated that the reduced peaking in the mid-core region is due to the more refined analysis methodology and is more representative of actual plant operation. The applicant further stated that this refined analysis approach removed some previous dependency on over-conservatisms, and utilized more data based on actual plant operating history.

The applicant stated that there are also differences in the reported 32 EFPY fluence values for the reactor vessel welds of Byron Units 1 and 2, and Braidwood Unit 2 in the updated TLAAs for license renewal compared with those reported in docketed documents. The applicant also stated that the reasons for the differences are the same as those provided in the answer to the request above for Braidwood Unit 1.

The staff found that the applicant's response acceptable because the applicant used the staff-approved fluence analysis methodologies to calculate the neutron fluence for license renewal in accordance with RG 1.190 as described in WCAP-14040, Revision 4 and WCAP-16083-NP-A, Revision 0. The staff also confirmed that differences between previous calculations, performed in 1999, and those supporting the LRA are attributable to the following: (1) the use of cycle-specific water density data, (2) the incorporation of additional recent cycles of actual plant operation, and (3) the rendering of a more exact representation of the axial flux profile in the core, such that over-conservatism was removed from the previous neutron fluence calculation. As discussed above, both calculations were performed using methodologies that adhere to the staff's regulatory guidance; thus, the staff determined that the newer calculations are acceptable despite the differences in the specific results.

Additionally, the updated fluence calculations meet the acceptance criteria in SRP-LR Section 4.2 because the applicant projected the neutron fluence for the reactor vessels to the end of the period of extended operation using staff-approved methodologies in accordance with RG 1.190; therefore, the applicant's TLAA for reactor vessel neutron fluence projections is acceptable.

4.2.1.3 UFSAR Supplement

LRA Section A.4.2.1 provides the UFSAR supplement summarizing the neutron fluence TLAA for the reactor vessels. The staff reviewed LRA Section A.4.2.1, consistent with the review procedures in SRP-LR Section 4.2, which state that the applicant should provide a summary description of the evaluation of the reactor vessel neutron embrittlement. Based on its review of the UFSAR supplement, the staff determines that the applicant provided an adequate summary description of its actions to address the neutron fluence analysis, as required by 10 CFR 54.21(d).

4.2.1.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that the neutron fluence analysis for the reactor vessels 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).

4.2.2 Upper-Shelf Energy

Section IV.A.1 to 10 CFR Part 50, Appendix G, provides the staff's requirements for demonstrating that reactor vessels in U.S. PWRs will have adequate ductility throughout their operating periods. This rule requires that reactor vessel beltline components made from ferritic materials must have a Charpy upper-shelf energy (USE) value equal to or above 75 foot-pounds (ft-lb) initially and must maintain a Charpy USE value of no less than 50 ft-lb throughout the operating period of the reactor vessel. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," provides an expanded discussion regarding the calculations of USE values and describes two methods for determining USE values for reactor vessel beltline materials, depending on whether or not a given reactor vessel beltline material is represented in the plant's Reactor Vessel Surveillance program that is mandated by the requirements in Appendix H to 10 CFR Part 50. Applicants that cannot demonstrate compliance with these requirements are required to demonstrate that lower values of USE will provide adequate margins of safety from fracture equivalent to those required by Appendix G of the ASME Code Section XI.

4.2.2.1 Summary of Technical Information in the Application

LRA Section 4.2.2 describes the applicant's TLAA for the calculation of Charpy USE values for reactor vessel beltline and extended beltline components for the period of extended operation. The applicant projected the USE values using the copper contents of the materials used to fabricate the reactor vessel beltline and extended beltline components, as determined from certified material test reports (CMTRs), and the 57-EFPY fluence values for the components, as determined from the fluence values in LRA Section 4.2.1 and attenuated in accordance with Equation 3 of RG 1.99, Revision 2, the 1/4T location of the reactor vessel wall. The applicant stated that the USE values for the BBS reactor vessel beltline and extended beltline components were determined without the use of surveillance data in accordance with Regulatory Position 1.2 of NRC RG 1.99, Revision 2. In addition, the applicant stated that, where credible surveillance data was available from the Reactor Vessel Surveillance program, the projected USE values were determined by using credible USE data, as established in accordance with Regulatory Position 2.2 of RG 1.99, Revision 2. The applicant stated that USE projections without using the surveillance data resulted in lower (more conservative) USE values and that all of the projected USE values for the BBS reactor vessel beltline and extended beltline materials will remain above the 50 ft-lb requirement through the period of extended operation, as demonstrated in LRA Tables 4.2.2-1 to 4.2.2-4 for Byron Units 1 and 2 and Braidwood Units 1 and 2, respectively.

The applicant dispositioned the USE TLAA for the reactor vessel materials in accordance with 10 CFR 54.21(c)(1)(ii) to demonstrate that the analysis has been projected to the end of the period of extended operation.

4.2.2.2 Staff Evaluation

The staff reviewed the applicant's USE TLAA (LRA Section 4.2.2) and the applicant's basis for dispositioning the TLAA in accordance with 10 CFR 54.21(c)(1)(ii), consistent with the acceptance criteria in SRP-LR Section 4.2.2.1.1.2 and the review procedures in SRP-LR Section 4.2.3.1.1.2. SRP-LR Section 4.2.3.1.1.2 states that the review of the documented revised USE analysis results should be based on the review of the projected 1/4T neutron fluence projections for the reactor vessel beltline components at the end of the period of extended operation and the impacts that those fluence values will have on the USE values for the beltline components at the end of the period of extended operation. The SRP-LR section states that the staff should confirm whether the results of the USE TLAA are in compliance with USE requirements or equivalent margins analysis requirements for reactor vessel beltline components, as defined in 10 CFR Part 50, Appendix G.

RG 1.99, Revision 2 states that the Charpy USE of reactor vessel materials decreases as a function of neutron fluence and copper content. As discussed above, RG 1.99, Revision 2, also describes two methods for determining USE values for reactor vessel materials, depending on whether or not two or more credible surveillance data sets become available from the reactor in question. Regulatory Position 1.2 of RG 1.99, Revision 2, uses Figure 2 of the RG when surveillance data sets are not available. When surveillance data are available, Regulatory Position 2.2 of RG 1.99, Revision 2, is used to determine the decreases in USE by plotting the reduced plant surveillance data on Figure 2 of the RG and fitting the data with a line drawn parallel to the existing lines as the upper bound of all the data.

The applicant stated that it used Regulatory Position 1.2 of RG 1.99, Revision 2, to project the USE values to the 60-year period of extended operation for the reactor vessel beltline and extended beltline materials. The applicant also stated that, when surveillance data was available to determine the USE projections for reactor vessel materials, it listed the projected USE values determined by using Regulatory Position 2.2 of RG 1.99, Revision 2. In addition, LRA Tables 4.2.2-1 to 4.2.2-4 indicate that the applicant's projections without using the surveillance data resulted in lower (more conservative) USE values. The LRA further states that the copper content and initial USE values, which are used in the USE projections, are the data contained in the CMTRs for the reactor vessel beltline and extended beltline materials.

The staff used Position 1.2 of RG 1.99, Revision 2, to confirm the adequacy of the USE values projected at the end of the period of extended operation. Based on the analysis for all beltline and extended beltline materials, the staff confirmed that the applicant's projected USE values were determined conservatively and resulted in 60 ft-lb for the limiting material of Byron Unit 1 (intermediate shell forging-to-lower shell forging circumferential weld), 62 ft-lb for the limiting material of Byron Unit 2 (intermediate shell forging-to-lower shell forging circumferential weld, inlet nozzle-to-nozzle shell forging weld, and outlet nozzle-to-nozzle shell forging weld heat #41403), 59 ft-lb for the limiting material of Braidwood Unit 1 (inlet nozzle-to-nozzle shell forging weld WF-598 and outlet nozzle-to-nozzle shell forging weld WF-598), and 62 ft-lb for the limiting material of Braidwood Unit 2 (intermediate shell forging-to-lower shell forging circumferential weld). Thus, the staff finds that the BBS beltline and extended beltline materials have projected USE values at 1/4T greater than 50 ft-lb in compliance with Appendix G to 10 CFR Part 50.

On the basis of its review, the staff finds the applicant demonstrated pursuant to 10 CFR 54.21(c)(1)(ii), that the USE analysis for the reactor vessels has been projected to the end of the period of extended operation. Additionally, the staff finds that the applicant's USE analysis meets the acceptance criteria in SRP-LR Section 4.2.2.1.1.2 because the applicant's analysis adequately demonstrates that the projected USE values for the reactor vessel beltline and extended beltline material at the end of the period of extended operation are not less than 50 ft-lb in compliance with the requirements of 10 CFR Part 50, Appendix G; therefore, the applicant's USE TLAA is acceptable.

4.2.2.3 UFSAR Supplement

LRA Section A.4.2.2 provides the UFSAR supplement summarizing the USE TLAA for the reactor vessels. The staff reviewed LRA Section A.4.2.2, consistent with the review procedures in SRP-LR Section 4.2.3.2, which states that the applicant should provide a summary description of the evaluation of the reactor vessel neutron embrittlement TLAA and provide information equivalent to SRP-LR Table 4.2-1. Based on its review of the UFSAR supplement, the staff determines that the applicant provided an adequate summary description of its actions to address USE, as required by 10 CFR 54.21(d).

4.2.2.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that the USE for the reactor vessels has been projected to the end of the period of extended operation and meets the acceptance criteria of Appendix G to 10 CFR Part 50. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the USE TLAA evaluation, as required by 10 CFR 54.21(d).

4.2.3 Pressurized Thermal Shock

Section 50.61 of 10 CFR establishes fracture toughness requirements for the protection of PWRs against postulated pressurized thermal shock (PTS) events. Such events, which are caused by severe overcooling concurrent with or followed by significant pressure, can lead to brittle fracture of the reactor pressure vessel (RPV). As such, the PTS requirements are part of the staff's regulatory framework for assuring that the structural integrity of the RPV is adequately maintained. To demonstrate adequate protection against PTS events, 10 CFR 50.61 requires an assessment of the reference temperature for each RPV beltline material. This reference temperature is a measure of the brittleness of the material, and it must be based on the projected effects of neutron irradiation over the period of plant operation. In accordance with 10 CFR 50.61(b), the PTS assessment must be updated upon a request for a change in the expiration date for operation of the facility. Therefore, the PTS assessment must be updated for license renewal.

The requirements of 10 CFR 50.61 prescribe an equation that must be used to calculate the reference temperature for the PTS assessment (RT_{PTS}). This equation is:

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

The $RT_{NDT(U)}$ term is the reference temperature for the RPV material in the preservice or unirradiated condition, as determined in accordance with the procedures of ASME Code, Section III, Paragraph NB-2331, or other NRC-approved methods. The ΔRT_{NDT} term is the mean value of the transition temperature shift for the material due to irradiation. This term is a function of the chemistry factor, which is based on the copper and nickel content of the material, and the best estimate neutron fluence at the clad-base-metal interface on the inside surface of the RPV at the location where the material in question receives the highest fluence for the period of service in question. The term M is a margin to account for uncertainties in the values of $RT_{NDT(U)}$, the copper and nickel content of the material, and the fluence and calculation procedures. The methods for determining the ΔRT_{NDT} and margin term values are described in 10 CFR 50.61(c)(1). Provisions for incorporating credible surveillance test data into the ΔRT_{NDT} estimate are described in 10 CFR 50.61(c)(2).

The results of the RT_{PTS} assessment, as calculated per 10 CFR 50.61(c), must be less than or equal to the PTS screening criteria specified in 10 CFR 50.61(b)(2). The screening criterion for RPV plates, forgings, and axial weld materials is 270 °F (132 °C), and the criterion for circumferential weld materials is 300 °F (149 °C). If the results of the RT_{PTS} assessment show that the PTS screening criteria cannot be met, then 10 CFR 50.61 provides licensees with certain actions that can be taken to permit continued plant operation, including (a) implementation of a flux reduction program to avoid exceeding the screening criteria, (b) submission of a safety analysis to determine what, if any, modifications to equipment, systems, and operation are necessary to prevent potential failure of the RPV as a result of postulated PTS events, or (c) implementation of a thermal annealing treatment of the RPV beltline materials to recover their fracture toughness.

4.2.3.1 Summary of Technical Information in the Application

LRA Section 4.2.3 describes the applicant's evaluation of the PTS TLAA. The LRA states that the applicant used the guidance in RG 1.99, Revision 2, to calculate the RT_{PTS} values for the RPV beltline materials. The LRA states that the applicant used RG 1.99, Revision 2, Regulatory Position C.1 for the calculations involving materials that did not have credible surveillance data, whereas Regulatory Position C.2 was used for the calculations involving materials that had two or more sets of credible surveillance data. Inputs to the calculations included the copper and nickel contents of the beltline materials and the 57 EFPY fluence values projected through the end of the period of extended operation. LRA Tables 4.2.3-1 to 4.2.3-4 provide the results of the applicant's updated RT_{PTS} calculations. Among all four of the BBS units, the applicant projects the limiting (or highest) RT_{PTS} value for the forging materials to be 114 °F (46 °C) at 57 EFPY. According to the LRA, this value corresponds to the Byron Unit 1 intermediate shell forging based on the use of noncredible surveillance data. The applicant projects the limiting RT_{PTS} value for the circumferential weld materials to be 124 °F (51 °C) at 57 EFPY. According to the LRA, this value corresponds to the Byron Unit 2 intermediate shell forging-to-lower shell forging circumferential weld (Heat No. 442002) based on the use of credible surveillance data. The applicant dispositioned the TLAA for the PTS assessments in accordance with

10 CFR 54.21(c)(1)(ii) to demonstrate that the analyses have been projected to the end of the period of extended operation.

4.2.3.2 Staff Evaluation

The staff reviewed the applicant's TLAA for the PTS assessments and the corresponding disposition of 10 CFR 54.21(c)(1)(ii) consistent with the review procedures in SRP-LR Section 4.2.3.1.2.2. Accordingly, the staff reviewed the TLAA for compliance with the requirements of 10 CFR 50.61, which involved an evaluation of the results of the applicant's revised RT_{PTS} calculations based on the projected neutron fluence at the end of the period of extended operation (57 EFPY).

The requirements of 10 CFR 50.61 apply to the RPV beltline materials. The staff considers the beltline to include any RPV material projected to receive a fluence of at least 1×10^{17} n/cm^2 (E greater than 1 MeV). In the past, this definition has limited the beltline to components in the shell course region directly surrounding the effective height of the active reactor core. However, with extended operation, some RPV components outside this region may also experience fluence levels of at least 1×10^{17} n/cm^2 (E greater than 1 MeV) and, therefore, are also evaluated as part of the beltline. The term "beltline materials" is used to refer to the group of materials that surround the effective height of the active reactor core, and the term "extended beltline materials" is used to refer to the group of remaining materials that receive a fluence of at least 1×10^{17} n/cm^2. The applicant's PTS assessments include both beltline and extended beltline materials due to the projected fluence levels at the end of the period of extended operation. At BBS, the extended beltline includes certain inlet and outlet nozzles and welds.

The staff reviewed the applicant's methodology for calculating the RT_{PTS} values. The staff determined that the methodology was acceptable because it followed the requirements of 10 CFR 50.61(c).

The staff also reviewed the adequacy of the applicant's values for $RT_{NDT(U)}$, which is the first term in the equation for calculating RT_{PTS}. The staff compared the $RT_{NDT(U)}$ values in LRA Tables 4.2.3-1 to 4.2.3-4 against the $RT_{NDT(U)}$ values in two CLB sources. One source was the applicant's revised PTLRs for 32 EFPY, which it reported to the staff in 2007 per the requirements of TSs Section 5.6.6. The other source was the UFSAR. Based on this comparison, the staff determined that the $RT_{NDT(U)}$ values in the LRA are consistent with the $RT_{NDT(U)}$ values from both CLB sources, with the exception of the value for the Braidwood, Unit 2 nozzle shell forging-to-intermediate shell forging circumferential weld seam, which is made from Heat No. H4498. For this material, the staff found that LRA Table 4.2.3-4 identifies $RT_{NDT(U)}$ to be −25 °F (−32 °C); however, UFSAR Table 5.3-10 identifies $RT_{NDT(U)}$ to be −30 °F. By letter dated March 11, 2014, the staff issued RAI 4.2.3-1, requesting the applicant to identify and substantiate the correct $RT_{NDT(U)}$ value for this material and explain the discrepancy between the LRA and the UFSAR.

The applicant responded to RAI 4.2.3-1 by letter dated April 8, 2014. The applicant explained that the discrepancy between the two $RT_{NDT(U)}$ values for this material is a historical issue. According to the applicant, a review of the CMTR indicates that the difference in the $RT_{NDT(U)}$ values is due to different interpretations of the raw Charpy test data by separate vendors. The applicant also stated that both values have been used in past PTS analyses submitted on the docket. For example, the −30 °F $RT_{NDT(U)}$ value was used in analyses submitted by letters dated July 12, 1990, and August 8, 1994, and the −25 °F value was used in analyses submitted by letters dated September 3, 1998, and February 28, 2014. The applicant stated that it considers the −25 °F value of $RT_{NDT(U)}$ to be the CLB for Braidwood, Unit 2.

The staff reviewed the letters referenced in the applicant's response to RAI 4.2.3-1 and confirmed that both $RT_{NDT(U)}$ values have been used in past PTS analyses submitted on the docket. However, the staff finds the applicant's use of the −25 °F value of $RT_{NDT(U)}$ for the PTS assessment in the LRA acceptable because it is based on the CMTR data. In addition, the use of this value is acceptable because it produces a more conservative RT_{PTS} value for the Braidwood, Unit 2 nozzle shell forging-to-intermediate shell forging circumferential weld seam (made from Heat No. H4498), as compared to use of the −30 °F value of $RT_{NDT(U)}$. With resolution of this RAI, the staff determined that the applicant used appropriate $RT_{NDT(U)}$ values for its RT_{PTS} calculations. The staff's concern described in RAI 4.2.3-1 is resolved.

The staff also reviewed the adequacy of the applicant's values for ΔRT_{NDT} and margin, which are the remaining terms in the equation for calculating RT_{PTS}. The ΔRT_{NDT} term is the product of a function involving the best estimate neutron fluence and a chemistry factor. As discussed in SER Section 4.2.1, the staff found that the applicant's neutron fluence projections for the period of extended operation are acceptable. In accordance with the requirements of 10 CFR 50.61, the chemistry factor depends on whether the material for a given RPV component is represented in the surveillance program and, if so, whether the surveillance data for that material is credible. When there is no credible surveillance data, the chemistry factor must be based on the copper and nickel content of the material. In these cases, the staff compared the copper and nickel content values from LRA Tables 4.2.3-1 to 4.2.3-4 against the values in the PTLRs for 32 EFPY and the UFSAR. The staff found that all values were in agreement. The staff also determined that the applicant selected appropriate chemistry factors based on the copper and nickel content of the materials, as required by 10 CFR 50.61(c)(1)(iv)(A). When there is credible surveillance data, the chemistry factor must be based on measured values of ΔRT_{NDT} and fluence, as obtained through implementation of the surveillance program. In these cases, since no additional surveillance tests have been conducted since the PTLRs for 32 EFPY were submitted, the staff compared the chemistry factors in LRA Tables 4.2.3-1 to 4.2.3-4 against the material-specific chemistry factors in the PTLRs. The staff found that the values in the LRA were consistent with the values in the PTLRs and, therefore, acceptable. For the margin terms, the staff determined that the applicant used appropriate inputs based on the type of material (i.e., weld or base metal) and whether the material is represented in the surveillance program and whether the surveillance data for the material is credible. The staff determined that the margin terms were calculated consistent with the requirements of 10 CFR 50.61(c)(1)(iii) for components not represented by credible surveillance data and 10 CFR 50.61(c)(2)(iii) for components represented by credible surveillance data. Based on this review, the staff determined that the applicant used appropriate ΔRT_{NDT} and margin values for its RT_{PTS} calculations.

After confirming that the applicant used appropriate inputs to determine the $RT_{NDT(U)}$, ΔRT_{NDT}, and margin terms, the staff used the applicant's values for these terms to independently calculate RT_{PTS} for each RPV beltline material. In all cases the staff's calculations were in agreement with the results reported by the applicant; therefore, the staff determined that the applicant's RT_{PTS} calculations are acceptable. The staff then compared the results against the PTS screening criteria in 10 CFR 50.61(b)(2). Of the four units, the highest calculated RT_{PTS} value for RPV plates, forgings, and axial weld materials is 114 °F (46 °C), which is well below the 270 °F (132 °C) screening criterion. The highest calculated RT_{PTS} value for the circumferential weld materials is 124 °F (51 °C), which is also well below the 300 °F (149 °C) screening criterion. Based on these results, the staff determined that the applicant demonstrated sufficient margins of protection against postulated PTS events for the period of extended operation.

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the PTS assessments for the RPVs have been projected to the end of the period of extended operation. Additionally, the PTS assessments meet the acceptance criteria in SRP-LR Section 4.2.2.1.2.2 because the applicant appropriately recalculated the assessments to consider the period of extended operation, in accordance with the requirements of 10 CFR 50.61. Based on this evaluation, the staff finds that the results of the updated PTS assessments are less than the screening criteria in 10 CFR 50.61(b)(2) for the period of extended operation.

4.2.3.3 UFSAR Supplement

LRA Section A.4.2.3 provides the UFSAR supplement summarizing the PTS assessments for the RPVs. The staff reviewed LRA Section A.4.2.3 consistent with the review procedures in SRP-LR Section 4.2.3.2, which state that the applicant should provide a summary description of the reactor vessel neutron embrittlement TLAA and provide information equivalent to the examples in SRP-LR Table 4.2-1. Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.2.2.2, and is therefore acceptable. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the PTS TLAA assessments, as required by 10 CFR 54.21(d).

4.2.3.4 Conclusion

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

4.2.4 Adjusted Reference Temperature

The guidance in RG 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” provides the staff’s recommended position for calculating the adjusted reference temperature values (ART or RT_{NDT} values) of those RPV components that are within the scope of the P-T limit evaluations, which are required to be calculated in accordance with the regulation in 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements.” These ART values are based on an evaluation of the neutron fluence values of the RPV components, as attenuated from the RPV inside wetted interface to depths at one-quarter and three-quarters of the RPV wall thickness (i.e., 1/4T and 3/4T locations in the RPVs). The ART values are inputs to the plant P-T limit curves, which are required to be included either in the plant-specific TS limiting conditions for operation (LCO) or in a PTLR that is managed in accordance with specific requirements in the Administrative Controls Section of the TSs.

In accordance with Westinghouse Non-Proprietary Class 3 Report No. WCAP-14040-NP-A, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves” (current NRC-approved version is Revision 4 of the WCAP), and RG 1.99, Revision 2, ART values (RT_{NDT} values) are calculated in accordance with Equation 1 below:

$$\text{ART or } RT_{NDT} = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + M \quad \text{(Equation 1)}$$

In this equation, the initial RT_{NDT} is the unirradiated ART value or unirradiated RT_{NDT} value for the component, as derived in accordance with the requirements in Section III of the ASME Boiler and Pressure Vessel (B&PV) Code (ASME Code Section III), Paragraph NB-2331. ΔRT_{NDT} is the shift in the RT_{NDT} value that is induced by neutron irradiation, and M is a margin term that is added into the calculation to account for uncertainties in the calculation methods.

In accordance with RG 1.99, Revision 2, the ΔRT_{NDT} value is calculated in accordance with Equation 2 below:

$$\Delta RT_{NDT} = CF \text{ x } f^{(0.28 - 0.1 \text{ x } \log f)} \quad \text{(Equation 2)}$$

In this equation, f is the neutron fluence of the component (in units of 10^{19} n/cm^2 [E > 1.0 MeV]) and CF is a chemistry factor. The neutron fluence is evaluated for the specific location of interest (e.g., 1/4T or 3/4T), as described below. The chemistry factor is dependent on the Cu and Ni alloying contents of the component's material and determined from either the CF tables in the RG (i.e., Regulatory Position 1.1 in the RG) or from credible RPV material surveillance test data that are obtained through implementation of the applicant's Reactor Vessel Surveillance Program (i.e., Regulatory Position 2.1 in the RG).

RG 1.99, Revision 2, includes a method to calculate the neutron fluence for any location inside the RPV wall thickness. The RG states that the neutron fluence at any depth in RPV wall is calculated in accordance with the following neutron fluence attenuation equation (i.e., Equation 3 below):

$$f = f_{surf}\, e^{(-0.24 * x)} \quad \text{(Equation 3)}$$

In this equation, f_{surf} (in units of 10^{19} n/cm^2 [E > 1.0 MeV]) is the calculated value of the neutron fluence at the inside wetted surface of the vessel, and x (in inches) is the depth into the vessel wall, as measured from the vessel inner (wetted) surface. Alternatively, the RG establishes that, if displacements per atom (dpa*)* calculations are used for the neutron fluence analysis, the ratio of dpa at the depth in question to dpa at the inner surface may be substituted for the exponential attenuation factor in Equation 3. Since the neutron fluence values for the components increase with time, it is the neutron fluence values that establish the time-dependency of these calculations.

4.2.4.1 Summary of Technical Information in the Application

LRA Section 4.2.4 describes the applicant's TLAA for the ART calculations. The applicant stated that the ART value of the limiting RPV beltline material is used to adjust the beltline P-T limit curves to account for irradiation effects. The applicant stated that 10 CFR 50, Appendix G, defines the fracture toughness requirements for the life of the vessel, and that under this rule, the initial RT_{NDT} is evaluated in accordance with the procedures in ASME Code Section III, Paragraph NB-2331.

The applicant also stated that, because accumulated neutron fluence increases the ART for a given RPV beltline component beyond its initial unirradiated value, the shift in the RT_{NDT} (ΔRT_{NDT}) must be evaluated as part of the ART calculations. The applicant also stated that, since the ΔRT_{NDT} values are a function of the neutron fluence values that were assessed for the initial 40-year licensed operating period, these ART calculations meet the criteria of 10 CFR 54.3(a) and have been identified as TLAAs requiring evaluation for 60 years of extended operation. The applicant stated that, since the calculations for the ART TLAA have been updated to project them to the end of the period of extended operation (i.e., to 57 EFPY), the TLAA is acceptable in accordance with 10 CFR 54.21(c)(1)(ii).

4.2.4.2 Staff Evaluation

The staff reviewed LRA Section 4.2.4 to verify that the ART analyses for Byron Units 1 and 2 and Braidwood Units 1 and 2 have been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii). The staff also reviewed LRA Section 4.2.4, consistent with the review procedures in SRP-LR Section 4.7.3.1.2, which state that the documented results of the revised analyses are reviewed to verify that their periods of evaluation are extended such that they are valid for the period of extended operation. The SRP-LR also states that the applicable analysis technique can be the one that is in effect in the plant's CLB at the time of filing of the renewal application.

The staff determined that the applicant uses the methods of analysis in ASME Section XI, Appendix G to generate the P-T limit curves for its reactor units. The staff also noticed that updates of P-T limit curves for the reactor units are performed in accordance with TS 5.6.6, which governs implementation of the applicant's PTLR process and requires updates of the P-T limits to be performed in accordance with specific NRC-approved methodologies referenced by the TS requirements, including the methodology in Westinghouse Topical Report (TR) WCAP-14040-NP-A. The staff further noticed that the methodology in WCAP-14040, Revision 4, as mandated by TS 5.6.6, requires the ART calculations (i.e., RT_{NDT} calculations) to be performed based on an assessment of both the 1/4T and 3/4T neutron fluence values for the RPV beltline and extended beltline components. However, the staff observed that LRA Section 4.2.4 did not include any ART values that were based on the 3/4T fluence values for RPV beltline and extended beltline components at 57 EFPY.

By letter dated April 8, 2014, the staff issued RAI 4.2.4-1/RAI A.4.2.4-1, requesting resolution of these matters. In RAI 4.2.4-1/RAI A.4.2.4-1, Part 1, the staff asked the applicant to amend LRA Section 4.2.4 to provide the ART tables and values that are based on an assessment of the 3/4T neutron fluence values for the RPV beltline and extended beltline components at 57 EFPY. In RAI 4.2.4-1/RAI A.4.2.4-1, Part 2, the staff asked the applicant to provide a basis for dispositioning the ART TLAA in terms of 10 CFR 54.21(c)(1)(ii), given that these values are used to evaluate the P-T limits for the period of extended operation, which are being accepted in accordance with 10 CFR 54.21(c)(1)(iii) (see SER Section 4.2.5). Otherwise, the staff asked the applicant to revise the LRA to disposition the TLAA for projected ART values in accordance with 10 CFR 54.21(c)(1)(iii).

The applicant responded to RAI 4.2.4-1/A.4.2.4-1, Parts 1 and 2, in a letter dated May 6, 2014. In it response to RAI 4.2.4-1/A.4.2.4-1, Part 1, the applicant stated that it amended its basis for accepting the TLAA to be in accordance with 10 CFR 54.21(c)(1)(iii). The applicant also stated that the "limiting 1/4T and 3/4T ART values will continue to be provided with the PTLR report to maintain the P-T limits in accordance with the TS requirements during the period of extended operation, as presented in LRA Section 4.2.5, Pressure-Temperature Limits." In its response to RAI 4.2.4-1/A.4.2.4-1, Part 2, the applicant further stated that it amended LRA Table 4.1-2, LRA Section 4.2.4, and LRA Section A.4.2.4 (UFSAR supplement) to reflect that the ART TLAA is accepted in accordance with 10 CFR 54.21(c)(1)(iii).

The staff noticed that the applicant's responses to RAI 4.2.4-1/A.4.2.4-1, Parts 1 and 2, included an amendment to the LRA to accept the TLAA on ART in accordance with

10 CFR 54.21(c)(1)(iii). The staff finds that the applicant has provided an acceptable basis for accepting the TLAA on ART in accordance with 10 CFR 54.21(c)(1)(iii) because the applicant will manage the TLAA through implementation of TS 5.6.6 and the applicant's basis is consistent with SRP-LR Section 4.2.2.1.3.3. Therefore, the issues identified in RAI 4.2.4-1/A.4.2.4-1 are resolved.

On the basis of its review, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the ART analyses for the reactor vessels will be adequately managed through implementation of the PTLR requirements in TS 5.6.6 during the period of extended operation.

4.2.4.3 UFSAR Supplement

LRA Section A.4.2.4 provides the UFSAR supplement summarizing the basis for accepting the TLAA on ART in accordance with the criterion in 10 CFR 54.21(c)(1)(ii). SRP-LR Section 4.2.3 does not include any recommended guidelines for reviewing TLAAs on ART that are accepted in accordance with the criterion in 10 CFR 54.21(c)(1)(ii). Therefore, the staff reviewed LRA Section A.4.2.4, consistent with the review procedures in SRP-LR Section 4.7.3.1.2, which state that the staff should review the documented results of the revised analyses to verify that their period of evaluation is extended, such that they are valid for the period of extended operation (e.g., 60 years).

The staff found that LRA Section A.4.2.4 stated that "57-EFPY 1/4T fluence values were used to compute ART values for BBS beltline and extended beltline materials in accordance with RG 1.99, Revision 2 requirements." The staff concluded that this basis is not consistent with the requirements in TS 5.6.6 because: (a) the TS provisions require that the calculations of ART values will need to be based, in part, on the methodology that is given in WCAP-14040-NP-A, and (b) WCAP-14040-NP-A, Revision 4, would require that ART calculations would need to include both 1/4T and 3/4T ART calculations for the RPV beltline and extended beltline components at 57 EFPY.

By letter dated April 8, 2014, as part of RAI 4.2.4-1/A.4.2.4-1, Parts 1 and 2, the staff asked the applicant to amend the UFSAR supplement to indicate that both the 57-EFPY 1/4T and 3/4T fluence values were used to compute ART values for BBS beltline and extended beltline materials in accordance with methodology in WCAP-14040-NP, as required by TS 5.6.6. The applicant responded to RAI 4.2.4-1/A.4.2.4-1, Parts 1 and 2, in a letter dated May 6, 2014. In its responses to these parts of the RAI, the applicant t amended the UFSAR supplement to change the basis for accepting the TLAA on ART from 10 CFR 54.21(c)(1)(ii) to 10 CFR 54.21(c)(1)(iii). The applicant also amended the UFSAR supplement to state that the "limiting 1/4T and 3/4T ART values will continue to be provided with the PTLR report to maintain the P-T limits in accordance with the Technical Specification requirements during the period of extended operation, as presented in LRA Section 4.2.5, Pressure-Temperature Limits."

The staff determined that the applicant's basis is consistent with TS 5.6.6 for implementing the applicant's PTLR process. The staff also confirmed that the applicant made the applicable changes to the UFSAR supplement in LRA Section A.4.2.4 on the basis for accepting this TLAA in accordance with 10 CFR 54.21(c)(1)(iii). The issues raised in RAI 4.2.4-1/A.4.2.4-1 are resolved with respect to the contents of LRA UFSAR Section 4.2.4 and the bases for accepting this TLAA in accordance with 10 CFR 54.21(c)(1)(iii).

The staff also noticed UFSAR supplement A.4.2.4 references a 200 °F (90 °C) value that is discussed in Section C.3 of RG 1.99, Revision 2. The staff noticed that the applicant's basis implies that the 200 °F value was included in the RG section to place a maximum limit on the calculation of 1/4T ART values. However, the staff noticed that Section C.3 of RG 1.99, Revision 2, relates to bases for RPV material selection when choosing the ferritic steel materials that would be used to fabricate the RPVs of newly constructed plants. In addition, the staff noticed that the 200 °F value referenced in Section C.3 of the RG serves only as a recommended ART basis for establishing and limiting the copper (Cu) contents of ferritic steel materials that are procured and used for fabrication of the RPVs in new plants. The staff further noticed that the referenced 200 °F value is not used to place a maximum limit on the calculation of 1/4T ART values once the RPV is fabricated and the plant is operated.

By letter dated April 8, 2014, the staff issued RAI A.4.2.4-2, requesting that the applicant amend LRA Section A.4.2.4 to be consistent with the 200 °F value basis that is referenced in Section C.3 of RG 1.99, Revision 2, or provide a technical basis for the applicant's statement as written. Otherwise, the RAI requested that the applicant amend LRA Section A.4.2.4 to delete that statement from the UFSAR supplement section.

The applicant responded to RAI A.4.2.4-2 in a letter dated May 6, 2014. In it response, the applicant stated that it was amending LRA Section 4.2.4 and A.4.2.4 to delete the 200 °F value basis that is referenced to Section C.3 of RG 1.99, Revision 2, in LRA Section A.4.2.4. The staff reviewed the applicant's letter of May 6, 2014, and found that the applicant made acceptable changes to LRA Section A.4.2.4 because RG 1.99, Revision 2, does not place any upper bound limit on the ART values that are calculated for the RPV beltline and extended beltline components. The issue raised in RAI A.4.2.4-2 is resolved.

Based on its review of the UFSAR supplement, as amended in the applicant's letter of May 6, 2014, the staff finds LRA Section A.4.2.4 meets the acceptance criteria in SRP-LR Section 4.2.2.1.3.2, and is therefore acceptable. Additionally, the staff finds that the applicant provided an adequate summary description of the TLAA on ART and its actions for using this TLAA as part of the bases for managing loss of fracture toughness due to neutron irradiation embrittlement of the ferritic steel components in the RPVs, as required by 10 CFR 54.21(d).

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 ART analyses for the reactor vessels will be adequately managed by the applicant's implementation of TS 5.6.6 and the PTLR process activities 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).

4.2.5 Pressure-Temperature Limits

The regulation in 10 CFR 50.36 requires the P-T limits for a licensed nuclear plant to be established and controlled by the TS for the facility. This is accomplished by either including the P-T limits in the LCOs of the TS, or else, if approved in a previously issued license amendment, in a PTLR that is within the scope of and is administratively controlled by the Administrative Controls Section of the TS. The regulation in 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," establishes the requirements for performing calculations of these P-T limits and requires that the P-T limits for the facility must be at least as conservative as those that would be generated if the methods of analysis in Appendix G of the ASME Code Section XI were used to generate the P-T limits.

For ferritic components in the beltline region of the RPV, the regulation requires the P-T limits to account for the effects of neutron irradiation. Therefore, the P-T limits are based, in part, on a function of a time-dependent neutron fluence parameter and must be updated periodically to remain valid for continued service of the facility. In addition, the regulation in 10 CFR Part 50, Appendix G requires that the generation of P-T limits must take into account the relevant neutron dosimetry data and Charpy-impact data that are generated through implementation of the applicant's 10 CFR Part 50, Appendix H, RPV surveillance program, as described in LRA AMP B.1.35.

4.2.5.1 Summary of Technical Information in the Application

LRA Section 4.2.5 describes the applicant's TLAA on P-T limits. The applicant stated that the P-T limits for Byron Units 1 and 2 and Braidwood Units 1 and 2 are required to be calculated in accordance with the requirements of 10 CFR Part 50, Appendix G. The applicant also stated that the P-T limits identify the maximum allowable operating pressure of the RCS as a function of reactor coolant temperature. The applicant further stated that, as the reactor vessel is exposed to increased neutron irradiation, its fracture toughness is reduced.

In addition, the applicant stated that the regulation in 10 CFR Part 50, Appendix G, therefore, requires impacts of the anticipated reactor vessel fluence to be taken into account for P-T limit assessments. In addition, the applicant stated that the current P-T limits are based upon neutron fluence projections for a 40-year licensed operating period. The applicant further stated that, since the P-T limits were originally based upon the 40-year assumption, the P-T limits for the reactor units satisfy the criteria of 10 CFR 54.3(a) and have been identified as a TLAA.

In LRA Section 4.2.5, the applicant summarized its basis for controlling and updating the plant-specific P-T limits for the Byron and Braidwood units in accordance with the Administrative Control requirements in TS 5.6.6 and the applicant's PTLR process. The applicant also provided its basis for using TS 5.6.6 and the PTLR process to accept the TLAA on P-T limits in accordance with 10 CFR 54.21(c)(1)(iii).

4.2.5.2 Staff Evaluation

The staff reviewed the applicant's TLAA on P-T limits and the proposed disposition of the TLAA in accordance with 10 CFR 54.21(c)(1)(iii) in order to: (a) verify whether the impact of loss of fracture toughness due to neutron irradiation embrittlement on the intended reactor coolant pressure boundary (RCPB) function of the RPVs would be adequately managed during the period of extended operation, and (b) determine whether the applicant's implementation of the requirements in TS 5.6.6 and its PTLR process would provide an acceptable basis for managing this aging effect during the period of extended operation.

The staff determined that SRP-LR Section 4.2.3.1.3.3 describes the staff's acceptance criteria for approving a TLAA on P-T limits in accordance with 10 CFR 54.21(c)(1)(iii). For CLB with approved PTLRs, SRP-LR Section 4.2.2.1.3.3 states that updated P-T limits for the period of extended operation must be available prior to entering the period of extended operation and that the requirements for implementing the PTLR process in the Administrative Controls Section of the TSs can be considered adequate AMPs or activities for the period of extended operation. The staff found that the applicant's basis for the disposition of the TLAA on P-T limits in accordance with 10 CFR 54.21(c)(1)(iii) was consistent with the acceptance criteria in SRP-LR Section 4.2.2.1.3.3, with the exception of the following matters that needed additional clarification by the applicant.

The staff noticed the applicant's basis for performing future updates of the P-T limits for Byron Units 1 and 2 and Braidwood Units 1 and 2 lie in the administrative control requirements in TS 5.6.6 and the applicant's procedures for implementing its PTLR process. The staff also noticed that the specifications in TS 5.6.6 require the applicant to perform updates of the P-T limit curves in accordance with the following NRC-approved methodologies:

- the methodologies referenced in the NRC letter of January 21, 1998, "Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance of Referencing Pressure Temperature Limits Report," which included but are not limited to the methodology in Westinghouse nonproprietary report WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigation System Setpoints and RCS Heatup and Cooldown Limit Curves"
- the methodologies referenced in the NRC letter of August 8, 2001, "Issuance of Exemption from the Requirements of 10 CFR 50.60 and Appendix G for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2"
- the methodology in Westinghouse Proprietary Class 2 Report WCAP-16143, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2," November 2003, which was approved in 2006 as an acceptable exemption from the requirements in 10 CFR Part 50, Appendix G, for performing P-T limit evaluations

In addition, the staff found that the regulation in 10 CFR Part 50 Appendix G, "Fracture Toughness Requirements," establishes the minimum fracture toughness requirements that must be met for RCPB components made from ferritic steels. The staff noticed that the rule requires the calculation of P-T limits to be based on an evaluation of all ferritic steel components that are located in the RPV, including those that are located outside of the beltline region of the RPV. The staff also noticed that, for the ferritic RPV components that are located in the beltline region of the vessel, the rule requires that assessment of P-T limits (and in particular, the RT_{NDT} values that are used in the P-T limit calculations) must account for the effects of neutron irradiation, including the results of the RPV surveillance capsule withdrawal program, as required to be implemented in accordance with 10 CFR Part 50, Appendix H.

The staff found that both LRA Section 4.2.5 and UFSAR supplement in LRA Section A.4.2.5 stated that, in order to meet these Appendix G requirements, the "analysis for the P-T curves will consider locations outside of the beltline such as nozzles, penetrations and other discontinuities (i.e., RPV nonbeltline components) to determine if more restrictive P-T limits are required than would be determined by considering only the reactor vessel beltline materials." However, 10 CFR 54.22 requires the applicant to include in its LRA any TS additions or changes that are necessary to manage the effects of aging during the period of extended operation. Section 54.22 also requires that the justification for such TS changes or additions be included in the application. In addition, Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," establishes the criteria that must be included in the Administrative Controls of the TSs and the applicant's PTLR processes as approved by the staff. The criteria in GL 96-03 are based on the requirement that the applicant's methodologies for generating P-T limits, as invoked by the TS requirements for PTLRs, comply with 10 CFR Part 50, Appendix G, unless applicable exemptions from the Appendix G requirements are requested in accordance with 10 CFR 50.60(b) and approved by the staff in accordance with 10 CFR 50.12.

Based on this review, it was not evident to the staff why the assessment of RPV nonbeltline components had been addressed as an enhancement in the UFSAR supplement (LRA Section A.4.2.5) when, in accordance with GL 96-03, this type of assessment should be included as part of the methodologies that were approved as P-T limit methodologies for TS 5.6.6. The staff also noticed that, in 1991 for Braidwood Unit 2 and in 2010 for Byron Unit 2, the applicant modified the RPV closure flange configurations by either removing or cutting one stud from the RPV closure flange assembly or by leaving one stud untensioned when operating the reactor. However, the staff noticed that the methods of analysis in WCAP-16143, as invoked by TS 5.6.6, were based on the original plant design configuration for the RPV closure flange assemblies, and were not on the modified RPV closure flange assembly designs with one stud not fully tensioned. As a result, the staff determined that the applicant would need to justify why a change to TS 5.6.6, Part b, or to the methodologies invoked by TS 5.6.6, Part b, would not need to be processed as part of the LRA, as mandated in accordance with the 10 CFR 54.22 requirements.

By letter dated April 8, 2014, the staff issued RAI 4.2.5-1/RAI A.4.2.5-1, Parts 1 through 3, requesting resolution of these issues. In RAI 4.2.5-1/RAI A.4.2.5-1, Part 1, the staff asked the applicant to clarify how the applicant will assess RPV nonbeltline structural discontinuities for their impact on the future P-T limits for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii) and how this assessment will be factored into the update of the PTLRs that will be performed in accordance TS 5.6.6, Part c. The staff also asked the applicant to justify why this assessment of the RPV nonbeltline structural discontinuities is proposed as an enhancement in LRA Section A.4.2.5 in contrast with the NRC position established in GL 96-03 that this type of assessment be performed in accordance with the 10 CFR Part 50, Appendix G requirements and be included within the scope of at least one of the P-T limit methodologies that are invoked by TS 5.6.6, Part b.

In RAI 4.2.5-1/RAI A.4.2.5-1, Part 2, the staff asked the applicant to justify why the current methodologies specified in TS 5.6.6 and the plant procedures for implementing the PTLR process would be valid for updating the P-T limits for the period of extended operation, given that the P-T limits minimum temperature requirement methodology in WCAP-16143 is not consistent with the current design configurations of the RPV closure flange assemblies at Byron Unit 2 and at Braidwood Unit 2.

In RAI 4.2.5-1/RAI A.4.2.5-1, Part 3, the staff requested the applicant to consider its responses to Request Parts 1 and 2 of the RAI, and based on these responses, to clarify whether changes to TS 5.6.6 need to be proposed in accordance with the requirements in 10 CFR 54.22 and whether changes to the methodologies invoked by TS 5.6.6 need to be proposed for the LRA. The staff also requested that the applicant amend LRA Sections 4.2.5 and A.4.2.5 if either TS 5.6.6 or the methodologies invoked by TS 5.6.6 would need to be amended in accordance with the 10 CFR 54.22 requirements.

The applicant responded to RAI 4.2.5-1/RAI A.4.2.5-1, Parts 1 – 3, in a letter dated May 6, 2014 (ML14126A338). In its response to RAI 4.2.5-1/RAI A.4.2.5-1, Part 1, the applicant stated that the assessment of RPV nonbeltline structural discontinuities for their impact on future P-T limits will be performed in accordance with 10 CFR 54.21(c)(1)(iii) and will be factored into the update of the PTLRs that will be submitted to the staff in accordance with TS 5.6.6, Part c. The applicant also stated that the revisions to the P-T limits beyond the current P-T limits will continue to consider the positions and criteria discussed in GL 96-03, which would have this type of assessment performed in accordance with 10 CFR Part 50, Appendix G requirements. The applicant further stated that the LRA is amended to provide this further clarification of the consideration of the 10 CFR 50, Appendix G requirements, which recognizes the ASME Section XI, Appendix G limits as an acceptable approach for analyzing and ensuring that a sufficient margin of safety is established in the P-T limits that will be calculated in accordance the applicable TS requirements. In addition, the applicant stated that the compliance with the requirements in ASME Section XI, Appendix G is included within the scope of TS 5.6.6, Part b, since ASME Section XI, Appendix G-based methodologies are used and these methodologies have been approved by the staff. The applicant stated that, since TS 5.6.6, Part b, implements the ASME Section XI, Appendix G requirements, and since the ASME Section XI, Appendix G includes the consideration of the assessment of RPV nonbeltline structural discontinuities, no enhancement was intended by the LRA statement.

The applicant stated that the recent issuance of the draft Regulatory Issue Summary (RIS) 2014-XX, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components" (ADAMS Package Accession No. ML14028A179, Memo Accession No. ML14027A577, and *Federal Register* notice Accession No. ML14027A668), also addresses the need to consider the nonbeltline structural discontinuities. The applicant stated that, with the issuance of the final RIS, further clarification will be provided on the expectations for future PTLR submittals on this subject. The applicant further stated that after the issuance of the final RIS, and with PTLRs that will be required to be updated in accordance with TS 5.6.6 for the period of extended operation, Exelon will, in accordance with TS requirements, provide PTLRs which sufficiently address all ferritic materials of pressure-retaining components of the RCPB, including an assessment of the impacts that structural discontinuities in the RPVs and increased neutron fluence accumulation will have on the P-T limits for the period of extended operation.

The staff noticed that TS 5.6.6 requires, in part, that WCAP-14040-NP-A will be used to generate the P-T limits that will be calculated for the period of extended operation in accordance with the TLAA acceptance requirement in 10 CFR 54.21(c)(1)(iii). The staff also noticed that the methodology for calculating P-T limits in WCAP-14040-NP-A is based on compliance with the requirements in 10 CFR Part 50, Appendix G, and Appendix G of the ASME Code Section XI, and with the recommended criteria in RG 1.99, Revision 2. The staff also confirmed that the applicant amended LRA Section 4.2.5 to state that the "PTLR revision necessary to extend the P-T limits into the period of extended operation will consider all ferritic materials of pressure-retaining components of the RCPB including the impact of structural discontinuities, and address the impact of neutron fluence accumulation in accordance with the requirements of 10 CFR 50, Appendix G." Thus, the staff determined that, based on the amendment of this LRA, the applicant has adequately demonstrated that implementation of the P-T limit methodology requirements in both the TS 5.6.6 and 10 CFR Part 50, Appendix G, requirements will ensure that the process for updating the P-T limits for the facilities will include an assessment of the impacts that RPV structural discontinuities and accumulated neutron fluence will have on the P-T limits that will be generated for the period of extended operation. The staff finds this basis to be acceptable because the applicant demonstrated that its P-T limits will be calculated in accordance with methodologies required by TS 5.6.6 and the TLAA acceptance criterion in 10 CFR 54.21(c)(1)(iii). The issue in RAI 4.2.5-1/A.4.2.5-1, Part 1, is resolved.

In its response to RAI 4.2.5-1/RAI A.4.2.5-1, Part 2, the applicant stated that the current P-T limit methodologies, as required by TS 5.6.6, and the plant procedures for implementing the PTLR process, will be valid for updating the P-T limits that will be generated for the period of extended operation. The applicant stated that, given that the P-T limits minimum temperature requirement methodology in WCAP-16143-P is not based on the configurations of current RPV closure flange assemblies at Byron Unit 2 and Braidwood Unit 2, additional commitments have been made in Exelon's response to Notice of Violation dated December 13, 2013, to take corrective steps for revising the methodology in WCAP-16143-P and to reflect the Braidwood Unit 2 configuration of 53 RPV head bolts. In addition, the applicant stated that the revision of WCAP-16143-P will include the 53 RPV head bolt configuration at Byron Unit 2 and that the revision of WCAP-16143-P will bring the methodology in agreement with the current configuration. The applicant stated that, in regard to the period of extended operation, a commitment was made to restore the configuration for Byron Unit 2 and Braidwood Unit 2 RPV closure flange assemblies to that analyzed in WCAP-16143-P prior to the period of extended operation, and that this commitment was made in Exelon's response to NRC RAI B.2.1.3-2, as provided in letter dated December 19, 2013. The applicant further stated that the implementation of these commitments will maintain the current TS 5.6.6 methodologies and plant procedures for implementing the PTLR process valid for the current operating period and the period of extended operation.

The staff noticed that requirements in TS 5.6.6 reference WCAP-16143-P as one of the required methodologies that will be used for updating the P-T limits for the reactor units and that WCAP-16143-P provides an alternative, NRC-approved method for establishing those minimum temperature requirements that need to be within the scope of the P-T limit calculations. The staff noticed that the methodology in WCAP-16143-P for establishing these minimum temperature requirements was approved as an exemption from the minimum temperature requirements that are stated in 10 CFR Part 50, Appendix G, and that the applicant identified this exemption (see LRA page 4.1-10) as an exemption for the LRA that was granted in accordance with provisions in 10 CFR 50.12 and based on a TLAA.

The staff also confirmed that, in letter dated December 19, 2013, the applicant amended LRA Table A.5 to include Commitment No. 47, in which the applicant committed to the repair of the RPV closure flange assembly at Braidwood Unit 2 at least 6 months prior to entering the period of extended operation for the unit, and Commitment No. 48, in which the applicant committed to the repair of the RPV closure flange assembly at Byron Unit 2 at least 6 months prior to entering the period of extended operation for the unit. The staff also noticed that these activities to repair the RPV closure flange assemblies at Byron Unit 2 and Braidwood Unit 2 will make the flange assembly configurations consistent with those analyzed in WCAP-16143-P. As discussed in SER Section 3.0.3.2.2, the applicant updated the status of Commitment No. 47 by reporting that the Byron Unit 2 partially stuck stud No. 11 was removed, and that an inspection showed no damage on the stud or flange hole threads. In addition, in order to ensure that the Braidwood Unit 2 inoperable stud location (No. 35) is restored so that all 54 reactor head closure studs are tensioned during the period of extended operation, the staff has proposed incorporating applicant's Commitment No. 48 into a license condition. Therefore, based on these considerations, the staff finds the technical bases in WCAP-16143-P will remain valid as an alternative methodology for establishing the minimum temperature requirements that need to be within the scope of the P-T limit calculations for the period of extended operation, which will be calculated in accordance with the TS 5.6.6 requirements and the methodologies invoked by those requirements. The issue in RAI 4.2.5-1/A.4.2.5-1, Part 2, is resolved.

In its response to RAI 4.2.5-1/RAI A.4.2.5-1, Part 3, the applicant stated that, based on the responses to Request Parts 1 and 2 above, there are no changes to TS 5.6.6 or to the methodologies invoked by TS 5.6.6 for the LRA in accordance with the requirement in 10 CFR 54.22. Based on the applicant's amendments of the LRA, as clarified in the applicant's responses to RAI 4.2.5-1/RAI A.4.2.5-1, Parts 1 and 2, the staff finds that the applicant demonstrated that proposed changes to TS 5.6.6 do not need to be addressed pursuant to 10 CFR 54.22 because the applicant has committed to the repair of the Byron Unit 2 and Braidwood Unit 2 RPV closure flange assemblies (as evaluated immediately above) such that design configuration of the RPVs for these units will be consistent with the assumptions and alternative methods of analysis in proprietary report WCAP-16143-P, as invoked for use by PTLR process requirements in TS 5.6.6. The issue in RAI 4.2.5-1/A.4.2.5-1, Part 3, is resolved.

The staff also reviewed the applicant's P-T limit basis against applicable information contained in the UFSAR. UFSAR Section 5.3.2.1 states that the RPV "surveillance program withdrawal schedule is contained in Table 4.1 of the PTLR document for each unit, respectively." UFSAR Section 5.3.2.1 also states that "[c]hanges to the withdrawal schedule may be made as part of an update to the PTLR under the provisions of 10 CFR 50.59."

The staff found that GL 96-03 states that P-T limit changes and the LTOP system setpoint changes could be processed through a licensee's 10 CFR 50.59 and PTLR processes, as long at the PTLR methodologies approved in the Administrative Controls Section of the TSs are used to make the changes to the P-T limits and to the low pressure overpressure protection (LTOP) system setpoint values (see SER Section 4.2.6). Because Appendix H to 10 CFR Part 50 requires that proposed changes to the RPV surveillance program withdrawal schedules for the units be submitted to the staff for review and approval, withdrawal schedule changes are not subject to the provisions of 10 CFR 50.59. As a result, the staff noticed that the position in GL 96-03 does not relieve a licensee from compliance with the requirement in 10 CFR Part 50, Appendix H, to submit applicant's proposed changes to the RPV surveillance program withdrawal schedule for NRC review and approval.

By letter dated April 8, 2014, the staff issued RAI 4.2.5-2, requesting that the applicant provide its justification for stating that future "[c]hanges to the withdrawal schedule may be made as part of an update to the PTLR under the provisions of 10 CFR 50.59." The applicant responded to RAI 4.2.5-2, in a letter dated May 6, 2014 (ML14126A338).

In its response, the applicant stated that it agrees that the subject statement in UFSAR Section 5.3.2.1 needs to be corrected. The applicant also stated that the referenced statement in UFSAR Section 5.3.2.1 is inconsistent with both UFSAR Section 5.3.1.6 and the statement in the NRC SER of January 21, 1998 (ADAMS Legacy Library Accession Number 9802040391), which approved the PTLRs for the reactor units. The applicant also stated it was acceptable to control the RPV surveillance capsule withdrawal schedule in PTLRs because changes to the schedules would need to be subjected to the reporting and review and approval requirements in 10 CFR Part 50, Appendix H. The applicant stated that this issue has been entered into the corrective action program (CAP) for revising the UFSAR to correct the inconsistency between UFSAR Section 5.3.2.1 and 5.3.1.6.

The staff concluded that the applicant's response to RAI 4.5.2-2 demonstrates that the applicant is aware that any future changes to the RPV surveillance capsule withdrawal schedules are required to be submitted to the staff for review and approval in accordance with the 10 CFR Part 50, Appendix H, requirements and that any changes to the capsule withdrawal schedules cannot be implemented without prior NRC approval. The staff also noticed that the applicant will make the appropriate amendments of UFSAR Section 5.3.2.1 through implementation of the applicant's process for amending the UFSAR, which is subject to the requirements in 10 CFR 50.71(e). The issue in RAI 4.2.5-2 is resolved.

On the basis of the staff's review described above, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the applicant's P-T limits will be adequately managed during the period of extended operation. Additionally, it meets the acceptance criteria in SRP-LR Section 4.2.2.1.3.3 because, consistent with the SRP-LR recommendations, the applicant adequately demonstrated that it will use the requirements of TS 5.6.6 and the methodologies invoked by TS 5.6.6 to update the P-T limits for the period of extended operation.

4.2.5.3 UFSAR Supplement

LRA Section A.4.2.5, "Pressure-Temperature Limits," provides the applicant's UFSAR supplement summary description for the TLAA on P-T limits. The staff reviewed LRA Section A.4.2.5 against the UFSAR acceptance criteria in SRP-LR Section 4.2.2.2, which state that the summary description for the TLAA on P-T limits should contain appropriate information that demonstrates why the TLAA may be accepted in accordance with one of the three acceptance criteria for accepting TLAAs in 10 CFR 54.21(c)(1)(i), (ii) or (iii). The staff also performed its review consistent with the review procedures in SRP-LR Section 4.2.3.2, which state that the NRC reviewer should verify that the applicant provided sufficient information in its UFSAR supplement that includes a summary description of the evaluation of the TLAA on P-T limits and why the TLAA on P-T limits is acceptable in accordance with 10 CFR 54.21(c)(1)(i), (ii) or (iii). The SRP-LR also states that SRP-LR Table 4.2-1 contains an example of an acceptable UFSAR supplement for this TLAA and that the NRC reviewer should verify that the applicant's UFSAR supplement provides information that is at least as comprehensive as the UFSAR supplement example that is provided for this type of TLAA in SRP-LR Table 4.2-1.

The staff noticed the UFSAR supplement for the TLAA on P-T limits provided an adequate summary of the basis for accepting the TLAA with the requirement in 10 CFR 54.21(c)(1)(iii) and for accepting the basis that implementation of TS 5.6.6 and methodologies invoked by the TS requirements will serve as an acceptable basis for calculating the P-T limit curves that will be needed for the period of extended operation. However, the staff did request further demonstration that the methodologies used for the generation of the P-T limits would appropriately assess all ferritic components in the RPV and that WCAP-16143-P will remain an acceptable alternative minimum temperature requirement methodology for the P-T limits that will be calculated for the period of extended operation. As discussed in Section 4.2.5.2 of this SER, the staff addressed these issues in RAI 4.2.5-1/RAI A.4.2.5-1, Parts 1 - 3, which were issued to the applicant in a letter dated April 8, 2014.

As discussed in SER Section 4.2.5.2, the staff evaluated the applicant's response letters dated May 6, 2014, and January 23, 2015, which include the applicant's responses to this RAI, an LRA amendment of UFSAR supplement Section A.4.2.5, and inclusion of Commitment Nos. 47 (reported as complete) and 48 in UFSAR supplement Table A.5. The staff concluded that the applicant's responses to this RAI, the amendment of the UFSAR supplement in LRA Section A.4.2.5, the completion of Commitment No. 47, and the inclusion of Commitment No. 48 as a license condition, provide reasonable assurance that the methodologies that are required by TS 5.6.6 and will be used to update the P-T limits for the reactor units will assess the impacts that RPV structural discontinuities and increasing neutron fluence will have on the P-T limits for the period of extended operation. The staff also concluded that the applicant's responses to the RAI and the activities of Commitment Nos. 47 (reported as complete) and 48 (incorporated into a license condition) resolved the issue regarding the acceptability of WCAP-16143-P as a basis for calculating the P-T limits for the period of extended operation. The staff also finds that the applicant's responses to the RAI will ensure that the methodology in WCAP-16143-P will remain as a valid, NRC-approved alternative basis for establishing the minimum temperature requirements that will be factored into the P-T limit calculations because the the applicant will repair the RPV closure flange assemblies at Braidwood Unit 2 and Byron Unit 2 in order to ensure that the design configuration for the assemblies will be in conformance with the assumptions and methodology in WCAP-16143-P. Therefore, based on this review, the staff determined that the applicant's responses to RAI 4.2.5-1/RAI A.4.2.5-1, Parts 1 to 3 in the May 6, 2014, letter, the updated response in the letter dated January 23, 2015, and incorporation of Commitment No. 48 into a license condition, provide an acceptable basis for demonstrating that the implementation of the TS 5.6.6 requirements and the applicant's PTLR process may be used to accept this TLAA in accordance with 10 CFR 54.21(c)(1)(iii) and to generate those P-T limits that will be needed to support plant operations during the period of extended operation. The issues in RAI 4.2.5-1/RAI A.4.2.5-1, Parts 1 to 3, are resolved with respect to the acceptability of LRA Section A.4.2.5.

Based on its review of the UFSAR supplement, as amended in the applicant's letter of May 6, 2014, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.2.2.1.3.3, and, therefore, is acceptable. Additionally, the staff determined that the applicant provided an adequate summary description of its actions to address the TLAA on P-T limits, as required by 10 CFR 54.21(d).

4.2.5.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the TLAA on P-T limits will be adequately managed by the TS 5.6.6 requirements and the PTLR process activities 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).

4.2.6 Low Temperature Overpressure Protection Analyses

4.2.6.1 Summary of Technical Information in the Application

LRA Section 4.2.6 describes the applicant's TLAA for the LTOP system setpoints. The LRA states that the LTOP system is required by TS Section 3.4.12 to prevent the pressure in the RCS from exceeding the maximum pressure established in the P-T limits during certain design basis transients. The LTOP system provides the overpressure protection automatically through the use of either: (a) two power-operated relief valves in the pressurizer, (b) two RHR suction relief valves, or (c) a combination of one power-operated relief valve and one RHR suction relief valve. The LRA states that the LTOP system setpoints will need to be re-evaluated because they are based on the current P-T limits, all of which will need to be updated for the period of extended operation. The LRA states that the applicant will use the Reactor Vessel Surveillance program, described in LRA Section B.2.1.19, to re-evaluate the LTOP system setpoints and submit the results to the staff. The applicant dispositioned the TLAA for the LTOP system setpoints in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of loss of fracture toughness on the intended functions will be adequately managed by the Reactor Vessel Surveillance program for the period of extended operation.

4.2.6.2 Staff Evaluation

The staff reviewed the applicant's TLAA for the LTOP system setpoints and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), consistent with the review procedures in SRP-LR

Section 4.7.3.1.3, which state that the staff is to review the applicant's aging management activities to verify that the effects of aging on the intended functions will be adequately managed consistent with the CLB for the period of extended operation.

TS Section 5.6.6 requires the applicant to report the LTOP system setpoints in a PTLR for each reactor vessel fluence period and for any revision or supplement thereto. By letter dated March 27, 2014, the applicant submitted the current PTLRs for Byron, Units 1 and 2, which are applicable for 32 EFPY and 30.5 EFPY, respectively. By letter dated May 9, 2014, the applicant submitted the current PTLRs for Braidwood, Units 1 and 2, which are applicable for 32 EFPY. Since the fluence periods covered by the current PTLRs do not encompass the entire period of extended operation (projected to be 57 EFPY), the applicant will need to re-evaluate the LTOP system setpoints to account for the loss of fracture toughness of the reactor vessels by neutron embrittlement. LRA Section 4.2.6 states that the applicant re-evaluate the LTOP system setpoints and report the results to the staff as part of the Reactor Vessel Surveillance program.

The staff reviewed the adequacy of the Reactor Vessel Surveillance program for re-evaluating and reporting the LTOP system setpoints for the period of extended operation. Based on the description provided in LRA Section B.2.1.19, the staff determined that the program is primarily for condition monitoring because it generates material and dosimetry data for monitoring irradiation embrittlement of the reactor vessels. While this program will provide certain input data needed to generate the LTOP system setpoints, the staff determined that the program does not include the specific analytical methods and processes that are needed to establish, document, and report the LTOP system setpoints as required by the CLB. SRP-LR Section 4.2.2.1.3.3 states that, for P-T limits that will be maintained through the period of extended operation, the use of a PTLR process included in the administrative controls section of the TSs is considered to be an adequate aging management activity for meeting the requirements of 10 CFR 54.21(c)(1)(iii). TS Section 5.6.6 describes the PTLR process for BBS. In addition to providing the requirements for updating the P-T limits, this section also provides the analytical methods and reporting requirements for updating the LTOP system setpoints. By letter dated April 8, 2014, the staff issued RAI 4.2.6-1 requesting that the applicant explain why it did not plan to use the analytical methods and processes required by TS Section 5.6.6 to establish, document, and report the updated LTOP system setpoints that will be needed for the period of extended operation. The staff also requested that the applicant identify and explain the TS changes or additions needed per the requirements of 10 CFR 54.22 if the applicant planned to use an approach different from the requirements of TS Section 5.6.6.

The applicant responded to RAI 4.2.6-1 by letter dated May 6, 2014. In its response, the applicant acknowledged that TS Section 5.6.6 identifies the analytical methods that must be used to establish the LTOP system setpoints. The applicant indicated that these methods are described in an NRC letter dated January 21, 1998. The applicant also acknowledged that TS Section 5.6.6 requires the LTOP system setpoints to be included in the PTLRs, which must be reported to the staff when updated. The applicant stated that these activities constitute its PTLR process and, instead of using the Reactor Vessel Surveillance program, the applicant stated that it will use the PTLR process to generate and submit the appropriate analyses for the LTOP system setpoints for the period of extended operation.

The staff reviewed the applicant's response to RAI 4.2.6-1 and determined that the applicant will follow its PTLR process in accordance with the requirements of TS Section 5.6.6 to re-evaluate, establish, and report the LTOP system setpoints that will be needed for the period of extended operation. The staff finds this process acceptable because it is comparable to the process for updating the P-T limits, as described in the acceptance criteria in SRP-LR Section 4.2.2.1.3.3.

These requirements will ensure that the applicant appropriately accounts for the effects of loss of fracture toughness of the reactor vessels when the LTOP system setpoints are re-evaluated. The staff's concern described in RAI 4.2.6-1 is resolved.

Pursuant to 10 CFR 54.21(c)(2), the applicant must also provide a list of all plant-specific exemptions granted under 10 CFR 50.12 that are in effect and based on a TLAA. The applicant must justify continuation of these exemptions for the period of extended operation. LRA Section 4.1.5 identifies four exemptions that are in effect and based on the TLAA for the LTOP system setpoints. According to the LRA, the staff granted these exemptions on July 13, 1995; November 29, 1996; December 12, 1997; and January 16, 1998. The LRA states that the exemptions permit the applicant to establish the LTOP system setpoints such that the LTOP systems will limit the maximum pressure in the reactor vessels to 110 percent of the pressure determined to satisfy the requirements of ASME Code, Section XI, Appendix G. As to the justification for continuation of these exemptions, LRA Section 4.1.5 states that they are all associated with the current P-T limits. Based on its neutron fluence projections, the applicant expects that BBS will exceed the terms of applicability of these P-T limits prior to the period of extended operation. As such, new P-T limits will be required for the period of extended operation. However, the LRA states that the exemptions related to the LTOP system setpoints will not be needed because the applicant will develop the new setpoints per the requirements of 10 CFR Part 50, Appendix G. The LRA further states that, if the BBS units do not exceed the terms of applicability of their current P-T limits, then continuation of the exemptions is justified because the prior NRC approvals are not limited to the current license term.

The staff reviewed the applicant's justification for continuation of the exemptions related to the TLAA for the LTOP system setpoints. TS Section 5.6.6 specifies the analytical methods that the applicant must use to establish the LTOP system setpoints. One of the required analytical methods is identified as "NRC letter dated January 21, 1998, 'Byron Station Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report.'" The staff confirmed that this document incorporates the four letters listed in LRA Section 4.1.5 and that these letters approve the exemptions related to the LTOP system setpoints. Collectively, the exemptions permit the applicant to use ASME Code Case N-514, "Low Temperature Overpressure Protection, Section XI, Division 1" as the basis for determining the LTOP system setpoints.

Approved code cases are generally incorporated into later editions and addenda of the ASME Code, the staff reviewed the history of ASME Code Case N-514 to determine if its content is included in the most-recent edition and addenda endorsed by the staff. Per RG 1.147, Revision 16, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," dated October 2010, the staff had unconditionally approved ASME Code Case N-514; however, the ASME annulled the code case after incorporating the requirements into ASME Code, Section XI. The staff reviewed a copy of ASME Code Case N-514 and found that it requires the LTOP system to operate in accordance with these two requirements:

(1) The system must be effective at coolant temperatures less than 200 °F or at coolant temperatures corresponding to a reactor vessel metal temperature that is less than RT_{NDT} plus 50 °F (10 °C), whichever is greater.

(2) The system must limit the maximum pressure in the vessel to 110 percent of the pressure determined to satisfy the requirements of ASME Code, Section XI, Appendix G.

Per 10 CFR 50.55a(b)(2), the staff currently endorses up to the 2007 Edition with the 2008 Addenda of ASME Code, Section XI. The staff reviewed this edition and addenda and determined that ASME Code, Section XI, Paragraph G-2215 incorporates the first requirement of ASME Code Case N-514 related to the temperature at which the LTOP system must be effective. However, the staff found that ASME Code, Section XI, does not incorporate the second requirement from ASME Code Case N-514 related to the maximum pressure limit. Specifically, ASME Code, Section XI, Paragraph G-2215 requires the LTOP system to limit the pressure in the reactor vessel to 100 percent of the pressure determined to satisfy the requirements of ASME Code, Section XI, Appendix G. This limit is more restrictive than the limit allowed by ASME Code Case N-514.

After further research, the staff found that the limit currently specified in ASME Code, Section XI, Paragraph G-2215 is based on the now-annulled ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves, Section XI, Division 1." The staff reviewed a copy of ASME Code Case N-640 and determined that it essentially removes the second requirement of ASME Code Case N-514 related to the maximum pressure limit. In addition, the staff determined that the applicant must follow ASME Code Case N-640 because it is one of the other analytical methods required by TS Section 5.6.6 for establishing the P-T limits and LTOP system setpoints. In particular, TS Section 5.6.6 identifies this method as "NRC letter dated August 8, 2001, 'Issuance of Exemption from the requirements of 10 CFR 50.60 and Appendix G, for Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2.'"

The staff determined that TS Section 5.6.6 requires the applicant to follow both ASME Code Cases N-514 and N-640. Therefore, the analytical methods for determining the LTOP system setpoints are equivalent to the current provisions of ASME Code, Section XI, Appendix G, as required by 10 CFR Part 50, Appendix G. Based on the equivalency with current NRC requirements, the staff determined that continuation of the exemption related to ASME Code Case N514 is acceptable for the period of extended operation. This exemption is justified for both: (a) the future LTOP system setpoints, which the applicant will be required to develop according to the analytical methods in TS 5.6.6, and (b) the current LTOP system setpoints, which were also developed according to the analytical methods in TS 5.6.6, as stated in the methodologies described in the current PTLRs.

The staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of loss of fracture toughness on the intended functions of the reactor vessels will be adequately managed for the period of extended operation. This demonstration also meets the acceptance criteria in SRP-LR Section 4.7.2.1.

4.2.6.3 UFSAR Supplement

LRA Section A.4.2.6 provides the UFSAR supplement summarizing the TLAA for the LTOP system setpoints. The staff reviewed LRA Section A.4.2.6 consistent with the review procedures in SRP-LR Section 4.7.3.2, which state that the applicant is to provide a summary description for its evaluation of each TLAA. The SRP-LR also states that the summary description should contain information on the disposition of the TLAA for the period of extended operation and be appropriate such that later changes can be controlled by 10 CFR 50.59. By letter dated May 6, 2014, the applicant amended LRA Section A.4.2.6 to reflect its response to RAI 4.2.6-1. Accordingly, the applicant revised the summary description to state that it will use its PTLR process to demonstrate compliance with 10 CFR 54.21(c)(1)(iii) for the TLAA. Based on its review of the UFSAR supplement, as amended by letter dated May 6, 2014, the staff finds that it meets the acceptance criteria in SRP-LR Section 4.7.2.2 and is therefore acceptable. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address updates to the LTOP system setpoints, as required by 10 CFR 54.21(d).

4.2.6.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the LTOP system setpoints will be adequately managed for the period of extended operation in accordance with the PTLR process required by TS 5.6.6. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3 Metal Fatigue

LRA Section 4.3 provides the applicant's assessment of metal fatigue as a TLAA for license renewal.

4.3.1 Transient Inputs to Fatigue Analyses

4.3.1.1 Summary of Technical Information

LRA Section 4.3.1 includes Tables 4.3.1-1 through 4.3.1-6, which list the 60-year projections of transients applicable to BBS, Units 1 and 2.

LRA Section 4.3.1 states that ASME Code Section III, Class 1 fatigue analyses are based upon explicit numbers and amplitudes of thermal and pressure transients described in the specifications. The LRA states that each BBS component designed in accordance with ASME Code Section III requiring a fatigue analysis was analyzed and shown to have a cumulative usage factor (CUF) less than the allowable design limit of 1.0. Some ASME Code Section III Class 2 heat exchangers at BBS were evaluated for fatigue similar to Class 1 components using transient inputs from LRA Tables 4.3.1-1, 4.3.1-2, 4.3.1-4, and 4.3.1-5. The LRA states that since the fatigue analyses are based upon a number of cycles postulated to bound 40 years of service, projection of the transients' cycles through the period of extended operation is required as an input to demonstrate that the analyses remain valid.

In order to determine that the analyses remain valid for 60 years of service, the applicant reviewed the fatigued monitoring data to determine the number of cumulative cycles of each transient type that has occurred during past plant operation. The LRA provided details of the projection methodology to determine the 60-year projected number of cycles.

The LRA states that an evaluation was performed to determine if the severity of the actual plant transients that have occurred during past operations remains bounded by the transient severity assumed in each transient definition in the design specification. This was done to determine whether the past cycles were appropriately characterized during the fatigue monitoring activities. The administrative and operating procedures were also reviewed to assess the effectiveness of the design transient cycle counting and to validate the cyclic assumptions.

LRA Section 4.3.1 states that, since the transients have been projected to the end of the period of extended operation, this section has been dispositioned in accordance with

10 CFR 54.21(c)(1)(ii). The LRA states the transients are used as inputs in the metal fatigue TLAA evaluations in the remainder of LRA Section 4.3.

4.3.1.2 Staff Evaluation

The staff reviewed LRA Section 4.3.1 to confirm that the transients that are significant fatigue contributors are monitored to ensure that the applicant's fatigue evaluations remain valid. The staff also reviewed the methodology used by the applicant to obtain the 60-year projections. The staff noticed that the applicant will use its Fatigue Monitoring program to track and monitor the transients included in LRA Section 4.3.1. The staff's evaluation of the applicant's Fatigue Monitoring program is documented in SER Section 3.0.3.2.24.

LRA Tables 4.3.1-2 and 4.3.1-5 provides the baseline cycles, 60-year projected cycles, and CLB cycle limit for Transient 6, "Letdown Flow Shutoff Prompt Return to Service." The LRA states that the baseline cycles for Byron Unit 2 and the projected 60-year projected cycles for all four units exceed the CLB cycle limit for the transient. The LRA states that the transient was redefined as four differential temperature range transients. The LRA further states that the number of baseline and 60-year projected cycles for each of the differential temperature range transients were determined and a reanalysis was performed for the bounding location, which confirmed that the CUF will remain below 1.0. The staff noted that the applicant did not provide a technical basis for redefining the original transient definition to four new transients.

By letter dated February 26, 2014, the staff issued RAI 4.3.1-1 requesting that the applicant provide the four redefined differential temperature range transients and include the transient definitions, baseline cycle counts, 60-year projected cycle counts, and CLB cycle limits for each redefined transient. The applicant was also requested to describe and justify the basis for redefining the original transient definition and confirm that its Fatigue Monitoring program will monitor the redefined transient cycles and severities and will require corrective action prior to exceeding design limits.

By letter dated September 11, 2014, the applicant responded to RAI 4.3.1-1. This letter resubmitted RAI responses originally submitted on March 28, 2014, to clarify and reduce the information previously identified as proprietary by the applicant. The applicant stated that each unit has both a normal and an alternate charging line. The applicant stated that these lines are used alternately each refueling outage to distribute the fatigue effects of system transients between the two lines. The applicant stated that the charging nozzle is the limiting component on each line and was evaluated for environmentally assisted fatigue (EAF). The applicant noted that the baseline cycle count for this transient represents the total cycles from both lines, not cycles per nozzle. The applicant provided the four redefined transients for Transient 6, along with the baseline cycles, the projected cycles, and the number of cycles used in the EAF evaluations.

The applicant stated that actual operating transients were less severe than the design transients based on comparison of actual plant-specific transient data against the original design transients. The applicant stated that its review of the plant data showed that a large number of transient cycles associated with flow isolation design transients with temperature changes that were below the temperature changes associated with the design transient, but the total number of transient cycles exceeded the number assumed in the original design analysis. The applicant stated that the cycles were counted within various bounding temperature difference ranges for each unit. The applicant stated that it established the CLB cycle limits, which included the effects of EAF, based on Byron, Unit 2, data because it had the maximum projected number of cycles in each temperature difference range. The applicant stated that the cycle limit distribution was applied to all four units. The applicant confirmed that the Fatigue Monitoring program will track and monitor these transient cycles and severities and will require corrective action prior to exceeding design limits.

The staff finds the response acceptable because the updated transients were redefined based on actual plant parameters and the applicant provided the baseline cycle counts, projected 60-year counts, and CLB cycle limits for the four redefined transients and updated the LRA to reflect them. The applicant also confirmed that the Fatigue Monitoring program will monitor these transients. The staff determined that the enhanced Fatigue Monitoring program ensures that the number of transients will not be exceeded during the period of extended operation or that corrective actions are taken. The staff's evaluation of the Fatigue Monitoring program is documented in SER Section 3.0.3.2.24. The staff's concern in RAI 4.3.1-1 is resolved.

LRA Tables 4.3.1-1 and 4.3.1-4 state that Transient 16, "Recovery of Main Feedwater Flow After Isolation (Units 1 only)," is applicable to the Unit 1 steam generators for both Byron and Braidwood. The LRA further states that Transient 16 is not evaluated because cycles associated with switching between main and auxiliary feedwater flow are implicit in the cycles counted for other RCS transients. The staff noticed that the LRA does not specify which other RCS transients will be monitored to account for Transient 16. It was also unclear to the staff why Transient 16 was applicable to Byron, Unit 1, and Braidwood, Unit 1, only.

By letter dated February 26, 2014, the staff issued RAI 4.3.1-3 requesting the applicant to identify which RCS transients will be monitored to account for Transient 16 and justify that monitoring these other RCS transients will be adequate so that Transient 16 will not need to be monitored through the period of extended operation. The applicant was also requested to clarify and justify why Transient 16 is applicable to Byron, Unit 1, and Braidwood, Unit 1, only.

By letter dated September 11, 2014, the applicant responded to RAI 4.3.1-3. The applicant provided the set of upset condition transients for Byron, Unit 1 that occur before Transient 16 occurs. The applicant stated these transients are captured in LRA Tables 4.3.1-1 and 4.3.1-4, which are monitored by the Fatigue Monitoring program through the period of extended operation. The applicant further stated that Transient 16 is only included in the design basis and CLB for the replacement steam generators (RSGs) installed at Byron, Unit 1, and Braidwood Unit 1, as indicated in the RSG design transient specifications. The applicant stated that the Byron, Unit 2, and Braidwood, Unit 2, steam generator design analysis or CLB do not include this transient.

The staff finds the applicant's response acceptable because the applicant provided the transients that account for the cycles for Transient 16. The staff confirmed that these transients are included in the LRA tables that will be monitored by the Fatigue Monitoring program. The staff determined that the enhanced Fatigue Monitoring program ensures that the number of transients will not be exceeded during the period of extended operation or that corrective actions are taken. The staff's evaluation of the Fatigue Monitoring program is documented in SER Section 3.0.3.2.24. The applicant further provided an adequate basis why Transient 16 is applicable only to Byron, Unit 1, and Braidwood, Unit 1, steam generators. The staff's concerns in RAI 4.3.1-3 are resolved.

LRA Tables 4.3.1-1 and 4.3.1-4 state that Transient 14, "Sampling Line and Nozzles Transients," will not be monitored. The LRA states that chemistry samples are taken at a much lower frequency than that which was assumed in the design, resulting in fewer cycles. The LRA further states that samples are no longer taken from the RCS as specified in the design, and are taken instead from the letdown system. The LRA states that samples from the letdown system result in lower temperature differences and lower transient severity. It was unclear to the staff how the information provided in the LRA regarding the lower frequency and differences support the basis for not monitoring Transient 14.

By letter dated February 26, 2014, the staff issued RAI 4.3.1-4 requesting the applicant to provide the comparison of frequencies and temperature differences at which chemistry samples are taken from the letdown system instead of the RCS system. The applicant was further requested to explain and justify why the lower frequency and lower temperature differences support the basis for not monitoring Transient 14.

By letter dated March 28, 2014, the applicant responded to RAI 4.3.1-4. The applicant stated that it compared the frequency of chemistry samples and temperature difference between the original sample location and design assumptions with the actual sample location and plant procedures. The applicant stated that the original design transient cycle basis assumed that samples were taken three times a day, over a 40-year plant life, totaling 45,000 cycles. The applicant further stated that, based on actual chemistry procedures as confirmed by operator interviews, RCS samples are taken only once per day during power operations and up to a maximum of once per hour, for a maximum of 3 days, during each heatup and cooldown. The applicant stated that the reactor coolant sampling location was changed in approximately 2002 such that chemistry samples were drawn downstream of the Chemical Volume and Control System letdown heat exchangers. The applicant stated that the maximum number of thermal cycles experienced by the original sample piping from the RCS, based on Byron, Unit 1, which has been in service the longest, is estimated to be approximately 11,000 cycles. The staff noticed that in 2002, Byron, Unit 1, had been in operation for approximately 17 years. The applicant stated that when samples were originally taken from the RCS sampling line piping, the temperature difference in the RCS sample line piping was approximately 480 °F to 580 °F. The applicant stated that the temperature difference when taking samples from the letdown heat exchangers is considerably less at approximately 90 °F to 150 °F.

The staff finds it reasonable that Transient 14 in LRA Tables 4.3.1-2 and 4.3.1-5 does not require monitoring by the Fatigue Monitoring program because there is adequate margin between the actual plant occurrence at 11,000 and the design limit of 45,000 when originally taken from the RCS sample line. Since the current plant procedures, which were applied approximately in 2002, require less frequent sampling at a considerably lower temperature difference, the staff finds it reasonable that an adequate margin will be maintained through the period of extended operation. The staff finds the applicant's response to RAI 4.3.1-4 acceptable. The staff's concern in RAI 4.3.1-4 is resolved.

The staff finds that the applicant has demonstrated that it monitors all transients that cause cyclic strain, which support its fatigue analyses with its enhanced Fatigue Monitoring program, such that corrective actions are taken prior to exceeding design limits, including environmental effects when applicable.

LRA Section 4.3.1 states that, since the transients have been projected to the end of the period of extended operation, this section has been dispositioned in accordance with 10 CFR 54.21(c)(1)(ii). The LRA states the transients are used as inputs in the metal fatigue TLAA evaluations in the remainder of LRA Section 4.3. Because the applicant did not provide any analysis, the staff determined that this section does not include a TLAA in accordance with 10 CFR 54.3(a).

4.3.1.3 UFSAR Supplement

LRA Sections A.4.3.1 and A.3.1.1 provide the UFSAR supplement summarizing the applicant's basis of its fatigue analyses and describing its Fatigue Monitoring program to ensure that the numbers of transients actually experienced remain below the assumed number. The staff reviewed LRA Section A.4.3.1 and A.3.1.1, consistent with the review procedures in SRP-LR 4.3.3.2, which state that the reviewer should confirm that the applicant has provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA.

Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR 4.3.2.2. Additionally, the staff determined that the applicant provided an adequate summary description for its Fatigue Monitoring program to monitor the number of transients actually experienced, as required by 10 CFR 54.21(d).

4.3.1.4 Conclusion

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

4.3.2 ASME Code Section III, Class 1 Fatigue Analyses

4.3.2.1 Summary of Technical Information

LRA Section 4.3.2 states that the BBS reactor vessels and RCPB piping, components, and auxiliary lines were designed in accordance with ASME Code Section III, Class 1 requirements. Fatigue analyses were prepared for these components to determine the effects of cyclic loadings resulting from changes in system temperature, pressure, and seismic loading cycles. The LRA states that some ASME Code Section III, Class 2 heat exchangers also have fatigue analyses, which were performed in a manner similar to that used for Class 1 components. The fatigue analyses were required to demonstrate that the CUF will not exceed the design allowable limit of 1.0 when the equipment is exposed to all of the postulated transients. Since the calculation of fatigue usage factors is part of the CLB and is used to support safety determinations and since the number of occurrence of each transient type was based upon 40-year assumptions, these fatigue analyses have been identified as TLAAs requiring evaluation for the period of extended operation. The LRA states that these fatigue analyses are based on the transient cycles listed in design specifications, as shown in LRA Tables 4.3.1-1, 4.3.1-2, 4.3.1-4, and 4.3.1-5. The applicant stated that, in order to ensure the numbers of transients remain bounding of those used in the fatigue analyses, the Fatigue Monitoring program will be used to monitor transients and ensure corrective action is taken prior to exceeding any design cycle limit.

LRA Section 4.3.2 also states that the design analysis of some BBS ASME Code Section III, Class 1 components also used the fatigue exemption provisions of ASME Code Section III, Subparagraphs NB-3222.4(d) (1) through (6). The applicant further states that some ASME Code Section III, Class 2 and 3 components at BBS were designed to ASME Code Section III, Paragraph NC-3219 requirements and were shown to meet the criteria for a fatigue exemption per ASME Code Section III, Subparagraphs NC-3219.2 and NC-3219.3. Since these fatigue exemptions are based upon the 40-year design transients, they have also been identified as TLAAs that require evaluation for the period of extended operation. The LRA states that, in order to demonstrate acceptability from a fatigue exemption basis for the period of extended operation for Classes 1, 2, and 3 components, the transients considered are based on the transients in LRA Tables 4.3.1-1, 4.3.1-2, 4.3.1-4, and 4.3.1-5. The applicant stated that the Fatigue Monitoring program will be used to ensure these cycles will not be exceeded during the period of extended operation.

The applicant dispositioned the ASME Code Section III, Class 1 fatigue analyses, the ASME Code Section III, Class 2 heat exchangers fatigue analyses, and the ASME Code Section III, Class 1, Class 2, and Class 3 fatigue exemptions in accordance with 10 CFR 54.21(c)(1)(iii) such that the effects of metal fatigue on the intended functions will be adequately managed by the Fatigue Monitoring program 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 ASME Code Section III, Class 1, Class 2, and Class 3 fatigue analyses to verify, in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of fatigue will be adequately managed by the Fatigue Monitoring program for the period of extended operation.

The staff reviewed the applicant's TLAAs for the ASME Code Section III, Class 1, Class 2, and Class 3 fatigue analyses, and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), consistent with the review procedures in SRP-LR Section 4.3.3.1.1.3. These procedures state that the reviewer should verify the appropriateness of the applicant's program for monitoring and tracking the number of critical thermal and pressure transients for the selected RCS components.

LRA Section 4.3.2 states that the fatigue analyses and fatigue exemptions for BBS ASME Code Section III, Class 1 vessel, piping, and components are based on the transients listed in LRA Tables 4.3.1-1, 4.3.1-2, 4.3.1-4, and 4.3.1-5. The LRA also states that several ASME Code Section III, Class 2, heat exchangers have fatigue analyses that were evaluated similar to those used for Class 1 components. The LRA further states that the transients assumed to demonstrate acceptable fatigue exemption bases for ASME Code Section III, Class 1, Class 2, and Class 3 components are included in LRA Tables 4.3.1-1, 4.3.1-2, 4.3.1-4, and 4.3.1-5. The LRA states that the Fatigue Monitoring program is credited to monitor the transient cycles and require corrective action prior to exceeding the design limits.

The staff determined that the enhanced Fatigue Monitoring program ensures that the number of transients will not be exceeded during the period of extended operation or that corrective actions are taken. The staff's evaluation of the Fatigue Monitoring program is documented in SER Section 3.0.3.2.24.

The staff finds that the applicant demonstrated, in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of aging related to metal fatigue of the ASME Code Section III, Class 1 vessels, piping, and components, ASME Code Section III Class 2 heat exchangers, and components associated with ASME Code Section III Class 1, Class 2, and Class 3 fatigue exemptions will be adequately managed through the period of extended operation. Additionally, it meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.3 because the applicant's Fatigue Monitoring program monitors and tracks the transient cycles assumed in the analysis and requires corrective action prior to exceeding the number of transient cycles used in the analysis.

4.3.2.3 UFSAR Supplement

LRA Section A.4.3.2 provides the UFSAR supplement which summarizes the TLAA for ASME Code Section III, Class 1 fatigue analyses, ASME Code Section III, Class 2 heat exchanger fatigue analyses, and ASME Code Section III, Class 1, Class 2, and Class 3 fatigue exemptions. The staff reviewed LRA Section A.4.3.2 consistent with the review procedures in SRP-LR Section 4.3.3.2, which state that the information to be included in the UFSAR supplement should include a summary description of the evaluation of the TLAA.

Based on its review of the UFSAR supplement, the staff finds that LRA Section A.4.3.2 meets the acceptance criteria in SRP-LR 4.3.2.2. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the TLAA for ASME Code Section III, Class 1 fatigue analyses, ASME Code Section III, Class 2 heat exchanger fatigue analyses, and ASME Code Section III, Class 1, Class 2, and Class 3 fatigue exemptions 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 on the intended functions of the ASME Code Section III, Class 1 vessels, piping, and components, ASME Code Section III Class 2 heat exchangers, and components associated with ASME Code Section III Class 1, Class 2, and Class 3 fatigue exemptions will be adequately managed by the Fatigue Monitoring program for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.3 ASME Code Section III, Classes 2 and 3 and ANSI B31.1 Allowable Stress Analyses

4.3.3.1 Summary of Technical Information

LRA Section 4.3.3 describes the applicant's allowable secondary stress range reduction factor TLAAs for ASME Code Section III, Class 2 and 3, and ANSI B31.1 piping. A stress range reduction factor to the allowable stress range is required if the number of equivalent full temperature cycles exceeds 7,000. The applicant stated that these are considered to be implicit fatigue analyses since they are based upon cycles anticipated for the life of the component, and are therefore, TLAAs requiring evaluation for the period of extended operation.

<u>Piping and Components Designed in Accordance with ASME Section III, Class 2 and 3, and ANSI B31.1 Associated with the RCS and Auxiliary Systems Transients</u>. LRA Section 4.3.3 states that ASME Code Section III, Class 2 and 3 piping at BBS was designed to ASME Code Section III, Paragraph NC-3611, ASME Code Section III, Paragraph ND-3611, and ANSI B31.1 requirements. The applicant stated that the cyclic qualification of the piping is based on the number of equivalent full temperature cycles as listed in LRA Table 4.3.3-1. LRA Tables 4.3.1-3 and 4.3.1-6 list the transients and their 60-year projections for the Class 2, Class 3, and ANSI B31.1 piping considered to experience transients associated with the RCS and Auxiliary Systems. This transient set is a subset of the transients found in LRA Tables 4.3.1-1, 4.3.1-2,

4.3.1-4, and 4.3.1-5. The applicant further states that, as demonstrated by LRA Tables 4.3.1-3 and 4.3.1-6, the number of projected cycles is less than 7,000, and therefore, the fatigue analyses for these Class 2, Class 3, and ANSI B31.1 piping will remain valid through the period of extended operation. The applicant dispositioned these TLAAs in accordance with 10 CFR 54.21(c)(1)(iii) such that the Fatigue Monitoring program will monitor the transient cycles and severities and require action prior to exceeding design limits that would invalidate these conclusions.

Auxiliary Feedwater, Emergency Diesel Generator, Fire Protection, Heating Water and Heating Steam, and Service Water System ANSI B31.1 Piping and Components. LRA Section 4.3.3 states that, for the remaining systems that are affected by different thermal and pressure cycles, an operational review was performed that concluded that the total number of cycles projected for 60 years are significantly less than 7,000 cycles. This includes the AFW, emergency diesel generator (EDG), fire protection, heating water and heating steam system, and service water systems. The applicant stated that, since the projected number of transient cycles does not exceed the number of equivalent full temperature cycles assumed in the implicit stress analysis, the stress range reduction factors originally selected for the components in all of these systems remain applicable, and therefore the TLAAs remain valid for the period of extended operation. The applicant dispositioned these TLAAs in accordance with 10 CFR 54.21(c)(1)(i) such that the ASME Code Section III, Class 2 and 3, and ANSI B31.1 allowable stress calculations for the AFW, EDG, fire protection, heating water and heating steam, and service water system remain valid for the period of extended operation.

4.3.3.2 Staff Evaluation

Piping and Components Designed in Accordance with ASME Code Section III, Classes 2 and 3, and ANSI B31.1 Associated with the RCS and Auxiliary Systems Transients. The staff reviewed LRA Section 4.3.3 and the TLAA for ASME Code Section III, Class 2 and 3, and ANSI B31.1 piping and components associated with the RCS and Auxiliary Systems for which the allowable range of secondary stresses depends on the number of assumed thermal cycles to verify, in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of fatigue will be adequately managed by the Fatigue Monitoring program for the period of extended operation.

The staff reviewed the applicant's TLAAs and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), consistent with the review procedures in SRP-LR Section 4.3.3.1.2.3. These procedures state that the reviewer should verify the appropriateness of the applicant's program for monitoring and tracking the number of critical thermal and pressure transients for the selected piping and components.

The staff reviewed the 60-year projected cycle counts for the plant transients provided in LRA Tables 4.3.1-3 and 4.3.1-6 to ensure that the full thermal range transient cycle limit of 7,000 will not be exceeded. The total number of design basis thermal events expected to occur in a 60-year life is approximately 2,900 each for Byron, Units 1 and 2, and approximately 1,800 each for Braidwood, Units 1 and 2. The staff concluded that there is an adequate margin to account for unanticipated transient occurrences such that the full-range thermal cycle limit of 7,000 will not be exceeded during the period of extended operation.

The LRA states that the Fatigue Monitoring program is credited to monitor the transient cycles and require corrective action prior to exceeding the design limits that would invalidate this analysis. The staff determined that the enhanced Fatigue Monitoring program ensures that the number of transients will not be exceeded during the period of extended operation or that corrective actions are taken. The staff's evaluation of the Fatigue Monitoring program is documented in SER Section 3.0.3.2.24.

The staff finds that the applicant demonstrated, in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of aging related to metal fatigue on the intended functions of the ASME Code Section III, Class 2 and 3, and ANSI B31.1 piping and components associated with the RCS and Auxiliary Systems will be adequately managed through the period of extended operation. Additionally, it meets the acceptance criteria in SRP-LR Section 4.3.2.1.2.3 because the applicant's Fatigue Monitoring program monitors and tracks the transient cycles assumed in the analysis and requires corrective action prior to exceeding the number of transient cycles used in the analysis.

Auxiliary Feedwater, Emergency Diesel Generator, Fire Protection, Heating Water and Heating Steam, and Service Water System ANSI B31.1 Piping and Components. The staff reviewed LRA Section 4.3.3 and the TLAA for the ANSI B31.1 piping and components associated with the AFW, EDG, fire protection, heating water and heating steam, and service water systems for which the allowable range of secondary stresses depends on the number of assumed thermal cycles to verify, in accordance with 10 CFR 54.21(c)(1)(i), that the analysis remains valid during the period of extended operation.

The staff reviewed the applicant's TLAA and the corresponding disposition of 10 CFR 54.21(c)(1)(i), consistent with the review procedures in SRP-LR Section 4.3.3.1.2.1. These procedures state that the relevant information in the TLAA, operating plant transient history, design basis, and the CLB are reviewed to confirm that the maximum allowable stress range values for the existing fatigue analysis remain valid for the period of extended operation and that the allowable limit for full thermal range transients will not be exceeded during the period of extended operation.

The LRA states that an operational review was performed that concluded that the total number of cycles projected for 60 years for these systems are significantly less than the full-range thermal cycle limit of 7,000. However, the LRA did not contain enough information regarding the applicable thermal cycles and 60-year projections associated for these systems.

By letter dated February 26, 2014, the staff issued RAI 4.3.3-1 requesting that the applicant provide the transients used in the implicit fatigue for the ANSI B31.1 piping and components associated with the AFW, EDG, fire protection, heating water and heating steam, and service water systems. The applicant was requested to provide the 60-year projected cycle counts and justify that the TLAA remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

By letter dated March 28, 2014, the applicant responded to RAI 4.3.3-1. The applicant stated the transients used in the implicit fatigue analysis of the: (a) AFW system are the number of AFW pump diesel engine starts and stops, (b) EDGs are the number of diesel engine starts and stops, (c) fire protection system are the number of fire pump diesel engine starts and stops, and (d) Byron service water system are the number of essential service water makeup pump diesel engine starts and stops. The applicant stated that a diesel engine startup, run, and shutdown is counted as one temperature transient for these systems. The applicant further stated that each diesel engine is started and shutdown once a month for surveillance to satisfy the TSs. The applicant projected approximately 720 thermal cycles for surveillance testing over a 60-year life. The applicant also stated that it projected an additional 720 thermal cycles to account for when a diesel engine is called upon to run and perform its intended function, run for additional surveillance testing requirements, run during spurious starts, or run as a result of maintenance activities. The applicant stated that the total 60-year projected cycle count for these DGs is approximately 1,440 cycles.

The applicant also stated that transients used in the implicit fatigue analysis for the Heating Water and Heating Steam System are the number of auxiliary steam system startups and shutdowns. The applicant stated that the auxiliary steam system, which is common to both units, is used as a backup when the extraction steam supply from both units is simultaneously lost (i.e., dual unit outage). The applicant stated that the 60-year thermal cycle count is projected to be no more than the maximum projected occurrences of plant cooldowns and heatups, reactor trips, and surveillances for a single auxiliary steam boiler. The applicant clarified that Byron Unit 1 is used because it has the greatest number of projected occurrences, which the staff noticed as conservative. The applicant stated that the Byron Unit 1 projects, for a 60-year period, 180 auxiliary steam boiler surveillances, 117 cycles of reactor cooldowns and heatups, and 71 cycles of reactor trips. The applicant stated that this results in a 60-year projection of 368 cycles.

The staff finds the applicant's response to RAI 4.3.3-1 acceptable because the applicant provided the associated transients and its 60-year projections assumed in the implicit analyses and provided adequate demonstration that the cumulative 60-year projected cycles for transients defined as full thermal range transients will remain less than the 7000 cycle allowable and that the analyses remain valid for the period of extend operation. The staff's concern in RAI 4.3.3-1 is resolved.

Based on its review, the staff finds it reasonable that the full-range thermal cycle limit of 7,000 – used in the applicant design basis fatigue evaluations associated with the ANSI B31.1 and ASME Code Section III, Class 2 and 3, piping and components – will not be exceeded and includes margin to account for unanticipated transient occurrences during the period of extended operation.

The staff finds that the applicant demonstrated, in accordance with 10 CFR 54.21(c)(1)(i), that the TLAAs for the ANSI B31.1 piping and components associated with the AFW, EDG, fire protection, heating water and heating steam, and service water systems for which the allowable range of secondary stresses depends on the number of assumed thermal cycles remain valid for the period of extended operation. Additionally, the applicant's analysis meets the acceptance criteria in SRP-LR 4.3.2.1.2.1 because the applicant demonstrated, for those piping and components subject to thermal fatigue described above, the cycle limit for full thermal range transients established in the design analyses will not be exceeded, and therefore the analysis will remain valid for the period of extended operation.

4.3.3.3 UFSAR Supplement

LRA Section A.4.3.3 provides the UFSAR supplement summarizing the TLAA for ANSI B31.1 or ASME Code Section III, Class 2 and 3, piping and components. The staff reviewed LRA Section A.4.3.3 consistent with the review procedures in SRP-LR Section 4.3.3.2, which state 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 metal fatigue TLAAs.

Based on its review of the UFSAR supplement, the staff finds that it meets the acceptance criteria in SRP-LR 4.3.2.2, and is therefore acceptable. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the TLAA for ANSI B31.1 or ASME Code Section III, Class 2 and 3, piping and components, 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, in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of metal fatigue on the intended functions of the ASME Code Section III, Class 2 and 3, and ANSI B31.1 piping and components associated with the RCS and Auxiliary Systems will be adequately managed by the Fatigue Monitoring program for the period of extended operation. The staff also concludes that the applicant has provided an acceptable demonstration, in accordance with 10 CFR 54.21(c)(1)(i), that the TLAA for ANSI B31.1 piping and components associated with the AFW, EDG, fire protection, heating water and heating steam, and service water systems remains valid for the period of extended operation. The staff further concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluations, as required by 10 CFR 54.21(d).

4.3.4 Class 1 Component Fatigue Analyses Supporting GSI-190 Closure

4.3.4.1 Summary of Technical Information

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

The applicant stated that it performed a systematic review of all Safety Class 1 RCPB components in major equipment and piping systems with a fatigue analysis that are susceptible to EAF to ensure that the limiting plant-specific EAF locations have been identified. The LRA states that the methodology in NUREG/CR-5704 (for austenitic stainless steel (SS) components), NUREG/CR-6583 (for carbon/LAS components), and NUREG/CR-6909 (for nickel alloy components) were used to determine applicable values of environmental fatigue life correction factor (F_{en}) for each material type. These values of F_{en} were then used to evaluate CUFs that include environmental effects (CUF_{en}).

The LRA states that the screening process for identifying the limiting locations for EAF divided the Safety Class 1 components into applicable transient sections. The applicant stated that the maximum/bounding F_{en} factors were applied to each of the components in the transient sections based on the material. The applicant then screened out all components with the resulting CUF_{en} of less than 1.0. The applicant used a stress basis comparison to identify the leading locations in each transient section. The applicant ranked the stress analysis methodology applied to each of the component CUFs based the level of technical rigor. The applicant also compared the remaining components of different transient sections that reside in the same piping system or equipment using the CUF_{en} and stress analysis method comparison to remove any additional components from consideration. The resulting leading locations supplemented the locations identified in NUREG/CR-6260.

LRA Table 4.3.4-1 provides the summary of the CUF_{en} values of the NUREG/CR-6260 locations for BBS. LRA Tables 4.3.4-2 and 4.3.4-3 provide the plant-specific EAF screening leading location results. LRA Section 4.3.4 states that the results of the evaluation of other locations from LRA Tables 4.3.4-2 and 4.3.4-3 determined to be potentially limiting will be incorporated into the Fatigue Monitoring program prior to the period of extended operation.

The applicant dispositioned the evaluations associated with EAF of the NUREG/CR-6260 locations for a newer vintage Westinghouse plant in accordance with 10 CFR 54.21(c)(1)(iii), such that the effects of EAF on the intended functions will be adequately managed by the Fatigue Monitoring program for the period of extended operation.

4.3.4.2 Staff Evaluation

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

The staff reviewed the applicant's EAF evaluations, as presented in the LRA and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.3.3.1.3, which state that the reviewer should confirm that the applicant has addressed the effects of the reactor coolant environment on component fatigue life as AMPs are formulated in support of license renewal.

In its review of LRA Section 4.3.4, the staff noticed that this sample of critical components with high-fatigue usage locations should include the locations identified in NUREG/CR-6260, as a minimum, as well as additional locations based on plant-specific considerations. LRA Section 4.3.4 states that 60-year fatigue calculations were performed for these component locations. The LRA states that the methodology in NUREG/CR-5704 (for austenitic SS components), NUREG/CR-6583 (for carbon/LAS components), and NUREG/CR-6909 (for nickel alloy components) were used to determine applicable F_{en} factors and obtain an environmentally adjusted cumulative fatigue usage (CUF_{en}) which included the effects of the reactor water environment. The LRA further states that applicable 60-year projected numbers of transients, which are included in the tables in LRA Section 4.3.1, were used in the EAF evaluations when necessary.

The LRA states that the applicable transient sets to be analyzed for the EAF evaluations for each component were determined by reviewing the transient definitions in the design specifications and the corresponding fatigue analyses. The applicant stated that plant-specific data, when available, were incorporated into the EAF analysis to reduce conservatism on an as-needed basis for qualification. The staff noticed that the applicant did not identify what plant-specific data were used and which evaluations used the plant-specific data to reduce conservatism. The staff noticed that the applicant also did not identify which analyses used 60-year projected cycles.

By letter dated February 26, 2014, the staff issued RAI 4.3.4-1, requesting the applicant to: (a) identify the EAF evaluations in which plant-specific data were issued, (b) describe the plant-specific data used to reduce conservatism, and (c) justify the use of the plant-specific data in the EAF evaluations. The staff also requested that the applicant identify the EAF evaluations, including specific transients and cycles for each location, in which 60-year projected cycles and/or reduced number of cycles were used.

By letter dated September 11, 2014, the applicant responded to RAI 4.3.4-1. This letter resubmitted RAI responses originally submitted on March 28, 2014, to clarify and reduce the information previously identified as proprietary by the applicant. The applicant stated that plant-specific data were used to reduce conservatism for the EAF evaluations of the charging nozzles and the pressurizer spray nozzle. The applicant stated that the CLB transients and cycle counts were used for all other EAF evaluations.

The applicant provided the following information in its RAI 4.3.4-1 response. For the charging nozzle EAF evaluation, the applicant stated that conservatism was reduced for the original Transient 6, “Letdown Flow Shutoff Prompt Return to Service” design transient, which is included in LRA Tables 4.3.1-2 and 4.3.1-5. The applicant stated that the parameters reviewed for this transient include regenerative and letdown heat exchanger outlet temperatures, charging and letdown flows, and reactor coolant loop (RCL) temperatures. The applicant also stated that actual operating transients were less severe than the design transients based on comparison of these actual plant-specific transient data against the original design transients. The applicant stated that its review of the plant data showed that (1) a large number of transient cycles associated with flow isolation design transients were below the temperature changes associated with the design transient, but (2) the total number of transient cycles exceeded the number of cycles assumed in the original design analysis. The applicant stated that the cycles were counted within various bounding maximum temperature difference ranges for each unit. The applicant stated that these four bounding redefined transients were inputs to the EAF evaluation for the charging nozzle. The staff determined that the applicant provided additional information for these transients in its response to RAI 4.3.1-1, which was found acceptable as described in SER Section 4.3.1.2.

The applicant also provided the following information in its RAI 4.3.4-1 response. For the pressurizer spray nozzle EAF evaluation, the applicant stated that the plant-specific data were used to reduce conservatism for the “Plant Heatup” and “Plant Cooldown” design spray transients. The applicant stated that the parameters reviewed for these transients include pressurizer spray line temperature, pressurizer spray line flow demand, pressurizer steam and water temperatures, and RCL temperatures. The applicant stated that the design heatup and cooldown transients were defined with a conservative number of spray events and spray nozzle change in temperature values. The applicant stated that it reviewed plant-specific data for the period of 1999 to 2012 and determined the cycle counts from the reduced data set and prorated counts over past operation and future operation. The applicant stated that the extrapolation was justified based on operator interviews and reviews of plant operating procedures that affect pressurizer spray operation, which did not change significantly from initial startup through 1999. The applicant stated that the bounding redefined “Plant Heatup” and “Plant Cooldown” transients were inputs to the pressurizer spray EAF evaluations. The staff finds this evaluation acceptable because the applicant reviewed actual plant-specific data on spray event occurrences and spray flow demands to determine the bounding redefined cycle counts that were used as inputs to the EAF evaluations.

In its response to RAI 4.3.4-1, the applicant also provided transients and EAF evaluation cycles which were the 60-year projected cycles and/or reduced number of cycles used in the EAF evaluations. The applicant identified four locations for these EAF evaluations: charging nozzles, accumulator nozzles, safety injection nozzles, and the pressurizer spray nozzle. The applicant stated that the limiting number of cycles was used in the EAF evaluations and the reduced cycles used for the EAF evaluations will become the CLB cycle limits for the period of extended operation. The applicant stated that these transients will be monitored and tracked by the Fatigue Monitoring program and will require corrective action prior to exceeding the cycle limits.

The staff finds the applicant's response acceptable because the applicant identified the EAF locations which used plant-specific data to reduce conservatism. The staff confirmed that the applicant provided an adequate justification for the use of the plant-specific data to provide more-accurate CUF_{en} values for the locations evaluated. The staff also finds the response acceptable because the applicant provided the list of transients and locations where 60-year projected cycles or reduced number of cycles were used in the EAF evaluations. The staff finds acceptable that the most limiting number of cycles will be monitored and tracked by the Fatigue Monitoring program at each BBS unit, since this ensures corrective actions can be taken before exceeding a transient count limit. The staff determined that the enhanced Fatigue Monitoring program ensures that the number of transients will not be exceeded during the period of extended operation or that corrective actions are taken. The staff's evaluation of the Fatigue Monitoring program is documented in SER Section 3.0.3.2.24. The staff's concerns in RAI 4.3.4-1 are resolved.

In LRA Section 4.3.4 the applicant stated that the objective of the EAF evaluation methodology was to reduce conservatism in the stress analyses as needed to accommodate the additional F_{en} on the CUF values. The applicant stated that WESTEMS™ (trademarked software technology developed by the Westinghouse Electric Company) was used to determine detailed stress histories for each applicable transient, which considered all applicable mechanical and thermal transient loads during each transient, and to calculate fatigue usage. The applicant stated that the stress histories were used to determine the stress peaks and valleys for the fatigue evaluations. The LRA states that the WESTEMS™ fatigue calculation methodology uses a conservative algorithm for the selection of the stress peaks and valleys for use in the ASME fatigue evaluations. The applicant stated that the analysis can use the optional program tools to remove conservatism to produce a more accurate final result. The applicant stated that when an analyst utilizes these program tools to remove conservatism, the justification of peak removal is fully documented and included in the supporting calculations. The applicant stated that, otherwise, the ASME fatigue evaluations retained the inherent conservatism in the WESTEMS™ software. The staff noticed that the applicant did not clarify whether these optional program tools were used to remove conservatism for fatigue evaluations.

By letter dated February 26, 2014, the staff issued RAI 4.3.4-2, requesting that the applicant identify all of the fatigue evaluations in which the optional program tools in the WESTEMS™ software was used to remove conservatism. The staff also requested that the applicant provide examples of the program tool use, provide the basis for removing conservatism, and justify that a more accurate final result was produced.

By letter dated September 11, 2014, the applicant responded to RAI 4.3.4-2. This letter resubmitted RAI responses originally submitted on March 28, 2014, to clarify and reduce the information previously identified as proprietary by the applicant. The applicant stated that the optional program tools in the WESTEMS™ software were used for the fatigue evaluations of the RCL safety injection nozzles, RCL accumulator nozzles, and the pressurizer spray nozzle. The applicant stated that the optional program tool in the software was the use of peak editing tools and that the peaks removed in these evaluations were determined to be non-controlling, redundant, and resulted in unnecessary conservatism. The applicant also provided an example where the peak editing tools were applied for each of the three components. For each of the three examples, the applicant provided a figure to show a plot in the WESTEMS™ software of two stress intensities for the transient. The applicant stated the resulting analysis removes unnecessary conservatism and results in more accurate usage factors.

The staff finds the applicant response acceptable because the applicant identified the components in which the WESTEMS™ software operational program tools were used in the fatigue evaluations and provided examples of how the program tools were used. The staff further finds the response acceptable because the applicant's use of the peak removal tools in the WESTEMS™ software is documented and removes redundant stress peaks to provide a more accurate CUF_{en}, consistent with the discussion in RIS 2011-14, "Metal Fatigue Analysis Performed By Computer Software," December 29, 2011. The staff's concerns in RAI 4.3.4-2 are resolved.

To ensure that the limiting plant-specific EAF locations have been identified, in LRA Section 4.3.4, the applicant stated that a systematic step-wise review was performed for all Safety Class 1 RCPB components in major equipment and piping systems with a fatigue analysis and susceptible to EAF. The applicant stated that these components were reviewed and categorized into common groups as part of the EAF screening process. The four steps, which are evaluated sequentially below, are:

(1) grouping into transient sections
(2) use of screening F_{en} values to eliminate locations with CUF_{en} values less than 1.0
(3) stress basis comparison to identify leading locations
(4) comparison of locations from different transient sections

The first step in this screening methodology groups the Class 1 components into transient sections, which are defined as groups of subcomponents/locations that experience the same transients. The applicant further stated that the components residing in the same transient section can easily be compared with each other to determine the most limiting component (or leading location), which is the location with the highest CUF value. The applicant stated that the differences in stresses experienced by each component in a transient section are generally the result of the material and geometry differences. The staff finds this first step in the screening methodology appropriate because the applicant included all Safety Class 1 RCPB components susceptible to the reactor coolant water environment in the scope of its EAF evaluation to identify any plant-specific components.

The second step of the screening methodology is to develop environmental correction factors (F_{en}) for each component so that CUF values including environmental fatigue (CUF_{en}) can be calculated. The applicant stated that those components with a screening CUF_{en} of less than 1.0 were removed from the list because they have been calculated using the design-basis fatigue usage factors with a maximum F_{en} based on material. The staff noticed that the applicant did not clarify whether the "maximum F_{en}" is the maximum calculated from the NUREG reports or whether it is the maximum calculated for a particular transient section. The staff noticed that, if it is the latter, it is important to understand the applicant's assumptions in calculating the maximum F_{en} based on material for a particular transient section.

By letter dated February 26, 2014, the staff issued RAI 4.3.4-7, requesting that the applicant clarify whether the maximum F_{en} based on the material is the calculated maximum F_{en} from the applicable NUREG reports or the calculated maximum from a particular transient section. If the maximum F_{en} was based on the transient section, the staff requested that the applicant identify any assumptions (e.g., temperature, sulfur, dissolved oxygen (DO), strain rate) used in calculating the F_{en} and the basis for these assumptions.

By letter dated September 11, 2014, the applicant responded to RAI 4.3.4-7. This letter resubmitted RAI responses originally submitted on March 28, 2014, to clarify and reduce the information previously identified as proprietary by the applicant. The applicant stated that bounding temperature, sulfur, DO, and strain rate parameters were assumed such that use of the parameters would result in the maximum material F_{en} values from the applicable NUREG report that was applied to the component's material. The applicant stated that the only exception was that a value of 0.005 ppm was used for the DO content and that it reviewed its plant chemistry data to confirm that DO content is 0.005 ppm during normal operation and 0.05 ppm during heatup and cooldown operations. The applicant stated that the elevated DO content usually only occurs when reactor coolant temperature is low and in the fluid temperature range where the transformed metal temperature parameter is zero. The applicant stated that it used these maximum material F_{en} values to screen components within the transient sections with CUF_{en} values below 1.0. The applicant also stated that it further refined the F_{en} values based on the maximum temperature and applied them to the remaining components. The applicant stated that it again removed components with CUF_{en} values below 1.0 from consideration. The staff finds the applicant's response acceptable because the applicant provided adequate justification for its assumptions made in determining F_{en} factors, which the staff concluded were bounding. The staff finds the applicant's use of the lower DO content acceptable because the applicant used plant-specific operating data to determine the value during normal operation and during heatup and cooldown. The staff's concerns in RAI 4.3.4-7 are resolved.

The staff recognized that, in order to determine the most limiting component (or leading location) with the highest CUF value, it is important that the CUFs are assessed using the same fatigue curves in ASME Code Section III, Appendix I. The staff further noticed that the LRA did not clarify whether the applicant had considered any differences in component materials when performing its review to compare component CUF values, since material properties may impact the specific CUF value for a given component. The staff reasoned that, through the course of plant operation, it is possible that CUF values for specific components were re-evaluated as part of power uprates, responses to GLs or bulletins, etc., to different editions of ASME Code Section III and with varying levels of rigor when compared to the fatigue evaluations performed for the plant's original design.

By letter dated February 26, 2014, the staff issued RAI 4.3.4-3, requesting that the applicant confirm that the CUF values that were compared with each other in a transient section to identify the location with the highest CUF value were assessed similarly (e.g., amount of rigor in calculating CUF) and used the same fatigue curves in ASME Code Section III, Appendix I to provide a meaningful comparison. If not, the staff requested that the applicant provide the basis for ranking or comparing the CUF values to one another to provide an appropriate method for screening and determining a leading/limiting location. The staff further requested that the applicant clarify whether CUF values of different material types were compared to one another when determining the leading location(s) within a transient section. If so, the applicant was requested to identify the transient section, locations and materials that have been compared and eliminated for consideration of EAF; otherwise the applicant was requested to justify that the comparison of CUF values between different materials within a transient section for the consideration of EAF is appropriate or valid.

By letter dated September 11, 2014, the applicant responded to RAI 4.3.4-3. This letter resubmitted RAI responses originally submitted on March 28, 2014, to clarify and reduce the information previously identified as proprietary by the applicant. The applicant stated that locations within a transient section were compared similarly in regards to the amount of rigor used in calculating the CUF. The applicant stated that, within a transient section, all locations with materials other than nickel alloy used the same fatigue curves from the ASME Code Section III, Appendix I. The applicant stated that the NUREG/CR-6909 fatigue curves were used to compare nickel alloy component locations with the component locations made from different materials. The applicant stated that the EAF screening evaluation for transient sections associated with equipment locations considered different materials. In its response to RAI 4.3.4-3, the applicant described its EAF screening evaluation of the reactor vessel outlet nozzle region to provide an example of the applicant's review of Class 1 components with different material types. The applicant stated that the reactor vessel outlet nozzle region consists of SS (safe end), LAS (nozzle), and nickel alloy (safe end to nozzle weld). The applicant stated that the F_{en} factors applied to the respective CUFs were calculated using NUREG/CR-5704 for the safe end, NUREG/CR-6583 for the nozzle, and NUREG/CR-6909 for the safe end to nozzle weld. The applicant stated that the leading location for this transient section was the safe end location because it produced the highest screening CUF_{en} greater than 1.0. The staff identified that, within a transient section that contains components of various materials (e.g., LAS, nickel alloy, SS), the applicant did not provide a basis for selecting a leading location based on the highest CUF_{en} value. The staff determined that the CUF_{en} value of different materials may respond differently when the EAF is being refined in the future. In the example of the reactor vessel outlet nozzle region, the applicant did not provide sufficient justification that the SS component would bound the components made from other materials after the EAF has been refined to reduce the CUF_{en} of the SS component. More generally, the applicant did not justify that the refinement of the higher CUF_{en} of one material would ensure the reduction of CUF_{en} values for another material within the same transient section.

By letter dated June 30, 2014, the staff issued followup RAI 4.3.4-3a, requesting the applicant to provide additional information and justification that one material can serve as the leading location for other material locations with CUF_{en} values greater than 1.0 within a transient section. The applicant had initially responded to RAI 4.3.4-3 in a letter dated March 28, 2014, which was withdrawn and resubmitted by letter dated September 11, 2014. The staff reviewed both versions and confirmed that the information provided remained the same; therefore, the context of RAI 4.3.4-3a was not affected.

By letter dated September 11, 2014, the applicant responded to RAI 4.3.4-3a. The applicant provided its principles and bases for choosing a location made from one material to serve as the leading location for components within the same transient section that are made from different materials. The applicant stated that there are four transient sections at BBS that included components of different materials. For each of these transient sections, the applicant first evaluated components of similar materials separately to determine if any components can be screened out. The applicant applied the screening CUF_{en} evaluation, which is described in the second step of the methodology, and the stress basis analysis, which is described in the third step of the methodology. The staff's evaluation of this third step of the methodology, the stress basis analysis, is documented later in SER Section 4.3.4.2. The applicant then compared the remaining components within the transient section. To justify selecting the leading location(s) to bound the other components of differing materials, the applicant stated that it applied bases dependent on the screening CUF_{en} values, the conservatism of the analysis method, and the range of the potential reduction F_{en} of each component and material.

In its evaluation of the reactor vessel transient section, the applicant provided its justification to select the outlet nozzle safe ends as the leading location. The applicant stated that the outlet nozzle safe end weld and nozzle body locations were removed from consideration because refined evaluations using the maximum screening F_{en} values resulted in screening CUF_{en} values below 1.0. The staff finds this evaluation and leading location selection for the reactor vessel transient section acceptable because the applicant applied the maximum screening values to the components and will monitor, per Commitment No. 43, the component with resulting CUF_{en} values above 1.0.

In its evaluation of the Unit 2 original steam generator (OSG) transient section, the applicant first compared components for each material separately to screen out components from consideration. The applicant stated that the remaining components were the primary manway (pad/shell) – drain hole in channel head, which is carbon steel; the primary chamber drain, which is nickel alloy; and the tubesheet and shell junction, which is low-alloy steel. The applicant also stated that the tubesheet and shell junction can be removed because the fatigue curves from NUREG/CR-6583 were used for both this component and the carbon steel primary manway (pad/shell) – drain hole in channel head. Because the tubesheet and shell junction had a lower screening CUF_{en} and was evaluated with a more conservative EAF methodology, the primary manway (pad/shell) – drain hole in channel head can bound this component. The staff finds the removal of the tubesheet and shell junction from consideration acceptable because: (a) the two materials could be compared on the same basis because the same fatigue curves were used, and (b) the applicant retained the component with the higher screening CUF_{en} value and more rigorous EAF evaluation method. The applicant stated that it also will retain the nickel alloy primary chamber drain for this transient section. The staff finds the applicant's selection of leading locations for the Unit 2 OSG transient section acceptable because the applicant justified the bounding locations for each material within the transient section and will monitor, per Commitment No. 43, both resulting components.

In its evaluation of the pressurizer transient section, the applicant provided its justification to: (a) select the surge nozzle structural weld overlay (SWOL) as the leading location and (b) remove the lower head at heater penetration and upper shell locations from consideration. The applicant stated that these eliminated components were analyzed using a more conservative methodology; therefore, more reduction in the CUF_{en} values is expected than for the surge nozzle SWOL. In its evaluation for the Unit 1 RSG transient section, the applicant also applied this same justification to eliminate the inlet & outlet nozzle, weld location. It is unclear to the staff how this justification would ensure that refinement of the CUF_{en} value of one material could bound the locations of different materials. The applicant did not provide sufficient justification that removing conservatism for one material would result in a proportional refinement for another material. The applicant did not demonstrate that these components would not need to be monitored by the Fatigue Monitoring program for EAF.

Also in its evaluation of the Unit 1 RSG transient section, the applicant removed the primary head drain hole from consideration. The leading location for this transient section, the primary head/tubesheet juncture, has a screening CUF_{en} value of 2.16. The screening CUF_{en} value for the primary head drain hole has a higher screening CUF_{en} value of 2.234 but was analyzed with a more conservative methodology. As part of its stress analysis ranking methodology, the applicant stated that it would only eliminate components from consideration if: (a) its screening CUF_{en} value is lower or the same, and (b) its analysis method is more conservative. However, the applicant justified removing the primary head drain hole by stating that the screening CUF_{en} value for the leading location was only slightly less than the eliminated location. The applicant stated that this is not a concern because the primary head drain hole has a different analysis rank; therefore, the potential reduction in the CUF_{en} value is greater. It is unclear to the staff why the analysis rank difference alone justifies removing this component from consideration. It is also unclear to the staff if there are other instances where the applicant removed components from consideration that had a higher screening CUF_{en} than the selected leading location.

By letter dated October 28, 2014, the staff issued followup RAI 4.3.4-3b, requesting that the applicant provide justification that the refinement of the leading component material analysis would result in the leading component location bounding these component materials within the pressurizer transient section and the Unit 1 RSG transient section. The applicant was also requested to provide justification why the primary head drain hole was removed from consideration when the screening CUF_{en} value was higher than the screening CUF_{en} for the retained leading location and to identify and provide the basis for any other instance where the screening CUF_{en} value for a component removed from consideration was higher than the screening CUF_{en} value for the retained leading location. This issue was identified as Open Item (OI) 4.3-1.

By letter dated November 25, 2014, the applicant responded to RAI 4.3.4-3b. The applicant provided its justification for removing the lower head at heater penetration (pressurizer transient section), the upper shell (pressurizer transient section), and the inlet & outlet nozzle, weld (Unit 1 RSG transient section) locations from consideration for EAF. The applicant's response expanded its response to RAI 4.3.4-3a and further explained how each component met the criteria to allow the component to be removed from consideration from EAF. The criteria applied bases dependent on the screening CUF_{en} values, the conservatism of the analysis method, and the range of the potential reduction F_{en} of each component and material.

For these three components, the applicant stated that an equivalent refinement of the stress analysis basis between the removed component and leading component can be achieved. The applicant stated that the equivalent refinement would be achieved for: (a) the lower head at heater penetration with the use of explicit finite element modeling of component discontinuities, (b) the upper shell with the use of explicit finite element modeling of component discontinuities, reduction of conservative transient adjustments, and reduction of transient grouping, and (c) the inlet & outlet nozzle, weld with the reduction of transient grouping. The applicant stated that these methodologies would continue to result in the screening CUF_{en} values for these components to be lower than the retained leading location.

However, the applicant did not provide sufficient justification or information in its response to ensure that the methodologies used will ensure that the refinement of the fatigue analyses for different materials is equivalent for the specific components. Without quantitative results of analyses, evaluations, or methodologies, the applicant does not have sufficient justification that the refinements of the fatigue analyses for the specific component locations are equivalent. The staff does not have reasonable assurance that the specified components will not need to be monitored for the effects of EAF throughout the period of extended operation.

On January 27, 2015, the staff held a teleconference call with the applicant to discuss a draft followup RAI. During this teleconference, the applicant proposed to monitor these three locations using the Fatigue Monitoring program in the period of extended operation. By letter dated February 6, 2015, the applicant submitted an LRA Amendment. In this amendment, the applicant amended LRA Table 4.3.4-2, which contains the locations at Byron and Braidwood, Units 1 and 2, that will be monitored for EAF in the period of extended operation. The applicant updated the table to include the lower head at heater penetration (pressurizer transient section), the upper shell (pressurizer transient section), and the inlet & outlet nozzle, weld (Unit 1 RSG transient section) locations. The staff finds this acceptable because: (1) the applicant will monitor these locations with the Fatigue Monitoring program, and (2) with the inclusion of these three locations, the staff has reasonable assurance that the applicant will monitor the bounding locations at Byron and Braidwood, Units 1 and 2, that are susceptible to EAF.

Also in its response to RAI 4.3.4-3b, the applicant provided its justification to remove the primary head drain hole from consideration for EAF. In its justification, the applicant stated that it performed additional stress basis comparisons because this location had a higher screening CUF_{en} value than the retained leading location within the transient section. The applicant stated that the screening CUF_{en} value for the primary head drain hole was less than 4 percent greater than the screening CUF_{en} value of the retained location. The applicant stated that its stress basis comparison evaluation determined that enough refinement of the primary head drain hole analysis can be obtained to justify its removal from consideration. The primary head drain hole was evaluated with a more conservative analysis and has conservatisms that are not applicable to the leading location. One noted conservatism that the applicant provided was that the primary head drain hole fatigue analysis used a conservative inlet/outlet temperature difference in the stress calculation for controlling transients. The applicant stated that it investigated this conservatism impact and determined that it could refine the screening CUF_{en} value by greater than 17 percent.

The staff finds the applicant's basis to remove the primary head drain hole from consideration of EAF acceptable. The applicant provided examples of specific conservatism that impact only the primary head drain hole that could refine the fatigue analysis and CUF_{en} value below that of the retained location. The applicant provided adequate justification in its stress basis comparison that the leading location within the transient section would bound the primary head drain hole for consideration of EAF.

Also in its response, the applicant provided the additional locations that were removed from consideration of EAF but had a higher screening CUF_{en} value than the retained location within its transient section. For these two locations, the hot leg piping location in the reactor coolant pump (RCP) piping transient section and the 3-in. valve butt weld in the pressurizer safety and relief valve (PSARV) piping transient section, additional stress basis comparison evaluations were performed to justify removal from consideration of EAF. The applicant stated that both of these locations were evaluated with a more conservative analysis than the retained leading location in the respective transient section. The applicant identified conservatisms that were unique to the removed locations and not applicable to the retained locations. For the RCP piping transient section, the hot leg piping location analysis was assigned the most conservative and least rigorous ranking. The applicant stated that the conservatisms that were considered for the hot leg piping location and not the retained location included one-dimensional heat transfer analyses, simplified stress intensity range formulas that produced conservative stresses, and absolute combination of stress intensity ranges due to each loading range in NB-3600 equations. The applicant stated these conservatisms would refine the fatigue analyses and the screening CUF_{en} value below that of the retained location. For the PSARV piping transient section, the applicant stated that the finite element analysis used to reduce the stress intensity range terms was done for thermal stress only. The applicant stated that it evaluated the finite element analysis to reduce the stress intensity range term for pressure and determine that the screening CUF_{en} value for the 3-in. valve butt weld could be refined by 50 percent, which would be below the screening CUF_{en} value of the retained location.

The staff finds the applicant's basis to remove the hot leg piping location and the 3-in. valve butt weld acceptable. For both the RCP piping and PSARV piping transient sections, the applicant provided examples of specific conservatism that impact only the removed component that could refine the fatigue analysis and CUF_{en} value below that of the retained location. The applicant provided adequate justification in its stress basis comparison that the leading locations within the transient sections would bound the hot leg piping location and 3-in. valve butt weld for consideration of EAF. The staff's concerns in RAI 4.3.4-3b are resolved. Open Item 4.3-1 is closed.

LRA Section 4.3.4 states that when performing an EAF evaluation, an applicant can either use guidance from NUREG/CR-5704 for austenitic SSs, NUREG/CR-6583 for carbon and low-alloy steels, and NUREG/CR-6909 for nickel alloy, or it can use guidance from NUREG/CR-6909 for all materials. The staff noticed that if NUREG/CR-6909 is used, the corresponding fatigue curves therein should be considered in calculating the CUF values and that this difference must be addressed as part of the EAF screening process. The applicant also indicated that NUREG/CR-6909 was used for nickel alloy locations only. The staff noticed that the applicant did not clarify how many, or if any, nickel alloy components were eliminated based on the CUF_{en} screening process described in the LRA or how the applicant accounted for the difference in fatigue curves used in the fatigue analyses and NUREG/CR-6909 as part of the EAF screening process.

By letter dated February 26, 2014, the staff issued RAI 4.3.4-6, requesting that the applicant identify the nickel alloy locations, or a representative sample set of locations, that were eliminated by the CUF_{en} screening process, including the CUF and F_{en} values for these components. The staff further requested that the applicant discuss and justify how the difference in fatigue curves used in the fatigue analyses of these components and NUREG/CR-6909 was addressed as part of the EAF screening process.

By letter dated September 11, 2014, the applicant responded to RAI 4.3.4-6. This letter resubmitted RAI responses originally submitted on March 28, 2014, to clarify and reduce the information previously identified as proprietary by the applicant. The applicant provided the component locations that were eliminated by the CUF_{en} screening process and included the values of the associated design-basis CUF, revised NUREG/CR-6909 CUF, F_{en}, and CUF_{en} for the components. The applicant also provided its basis for eliminating each of the locations identified. The applicant stated that the contribution from the NUREG/CR-6909 fatigue curve differences would be negligible in design fatigue evaluations representing low cycle regimes because both methodologies would result in the maximum F_{en} penalty factor. The applicant stated that it performed additional evaluations for design fatigue evaluations representing high cycle regimes to determine the impact of the NUREG/CR-6909 fatigue curves on the ASME Code SS curve used in the CLB analysis. The staff finds the applicant's response acceptable because the applicant confirmed that it performed evaluations to determine the impact of NUREG/CR-6909 fatigue curves on the design-basis CUF values, and provided the resulting CUF_{en} values for the nickel alloy locations that were eliminated. The staff's concerns in RAI 4.3.4-6 are resolved.

The staff finds the applicant's second step in the screening methodology, which developed screening F_{en} factors to calculate CUF and CUF_{en} values, acceptable because the applicant used an appropriate and bounding methodology to determine CUF_{en} values and apply an initial screening criteria to retain components with a CUF_{en} value higher than 1.0 for consideration for EAF. This assured that the piping and components remaining after the initial screening criteria can be compared on an equivalent and conservative level.

The third step in the screening methodology was to perform a stress basis comparison on the remaining components within each transient section to identify the leading locations within the transient sections. The LRA states that Westinghouse has developed an approach that was applied to BBS for performing a stress basis comparison for the components included in the screening process.

The applicant stated that the following stress analysis characteristics were considered in determining the limiting locations within a given transient section:

(1) Qualification Criteria (ASME Code Section III, NB-3200, NB-3600, etc.)
(2) Stress Analysis Technique

Furthermore, the applicant stated that in order to perform these stress basis comparisons, a hierarchy of stress analysis techniques was developed based on fatigue analysis experience, to define the relative complexity of the various techniques.

(1) standard NB-3600 analysis
(2) NB-3600 with nonstandard mechanical stress indices or stress quantities used in stress formulas
(3) NB-3600 with nonstandard thermal stress indices or stress quantities used in stress formulas
(4) combination of (2) and (3)
(5) NB-3200 Fatigue Analysis

The staff concluded that the stress basis comparison described in LRA Section 4.3.4 consists of two aspects: (1) consideration of stress analysis characteristics and (2) a hierarchy of stress analysis techniques. The staff noticed that the LRA indicates that the applicant eliminated certain Safety Class 1 reactor pressure boundary locations susceptible to EAF by performing a "stress basis comparison." The staff also noticed that the LRA did not clarify which locations were eliminated or what the technical basis was for removing these locations from consideration of EAF as a leading location using the stress basis comparison.

By letter dated February 26, 2014, the staff issued RAI 4.3.4-5 requesting the applicant to confirm whether the use of a stress basis comparison and screening CUF_{en} of less than 1.0 were the only methods for eliminating locations for consideration of EAF, or describe and justify any other methods that were used. The applicant was further requested to describe and justify the circumstances and situation when locations were eliminated using a stress basis comparison. The staff also requested that the applicant identify the component locations, or a representative sample set of component locations, that were eliminated as a result of performing this stress basis comparison and to provide its basis for eliminating these locations/components, including any assumptions, factors, or criteria that were used to eliminate these locations for consideration in the EAF calculations.

By letter dated September 11, 2014, the applicant responded to RAI 4.3.4-5. This letter resubmitted RAI responses originally submitted on March 28, 2014, to clarify and reduce the information previously identified as proprietary by the applicant. The applicant stated that the only methods used for eliminating locations for consideration of EAF were the use of a stress basis comparison and screening CUF_{en} of less than 1.0. The applicant stated that the stress basis comparison is for locations with a CUF_{en} greater than 1.0 in a transient section. The applicant stated that it uses the hierarchy of stress analysis techniques provided in the LRA and ranks them, with (1) Standard ASME NB-3600 analysis as least rigorous to (5) ASME NB-3200 Fatigue analysis as most rigorous. Therefore, a lower stress analysis ranking equates to a less rigorous evaluation technique. The staff finds this ranking of stress analysis techniques based on technical rigor reasonable. The applicant stated that it eliminates locations with lower screening CUF_{en} values and lower stress analysis ranking (hence less rigor). The applicant stated that this approach is justified because if locations with lower stress analysis rankings were refined using a more rigorous stress analysis technique, the resulting CUF_{en} values for the components would be lower. The applicant stated that, when the most limiting component was difficult to determine based on the stress analysis ranking, multiple locations were retained as leading locations. The applicant stated that a component within a transient section could be eliminated only if its screening CUF_{en} value and stress analysis method ranking were lower than another component being retained, which the staff finds appropriate. The applicant also provided its evaluation of the charging lines to provide additional details on its stress basis comparison and justification for eliminating components. In the example, the transient section comparison resulted in two components with CUF_{en} values greater than 1.0. Using the stress basis comparison, the applicant removed the location with the lower CUF_{en} because the stress analysis method used for that component was less rigorous. The applicant stated that if this eliminated component had used a more rigorous stress analysis method, the resulting CUF_{en} would be a lower value and remain less than the retained component. The staff finds the basis reasonable that refinement using a more rigorous stress analysis method would result in lower CUF_{en} values. The staff finds the applicant's response acceptable because (1) the applicant used the stress basis comparison to only eliminate locations within a transient section with a lower screening CUF_{en} and lower stress analysis method, and (2) the applicant used the stress basis comparison to verify that the CUF_{en} values that were compared in the screening process were conservative and limiting. The staff also finds that the applicant's example demonstrated that its implementation of the stress basis comparison was reasonable in determining a bounding location for EAF monitoring during the period of extended operation. The staff's concern in RAI 4.3.4-5 is resolved. The staff finds this third step in the applicant's screening methodology, which was the stress basis comparison, appropriate as discussed above.

The fourth and final step in the screening methodology, as further detailed in the response to RAI 4.3.4-5, is to compare components that reside in different transient sections (but are within a common system or piece of major equipment) in order to determine leading locations to represent their respective system/equipment. In addition, the applicant stated that the transients themselves often control which components have the highest usage factors in a given system; so, within a particular system, those transient sections with the most severe system transients will usually have the components with the highest usage factors. However, the applicant stated that the comparison of components in different transient sections must be performed after the appropriate F_{en} correction factor is applied to the component usage factor because the F_{en} correction factor is dependent on temperature and strain rate and, therefore, can vary for each transient section.

The staff noticed that the applicant did not clarify when it compared components that reside in different transient sections, but are within a common system or piece of major equipment, to determine leading locations to represent their respective system/equipment. The staff also noticed that the applicant did not identify the assumptions or factors that were considered by the applicant when making this comparison to determine the leading location that resides in different transient sections and the basis for eliminating a location for consideration of EAF.

By letter dated February 26, 2014, the staff issued RAI 4.3.4-4 requesting the applicant to identify the locations, or a representative sample set of locations, that were compared from different transient sections, but within a common system or piece of major equipment. The staff requested that the applicant identify any components that were eliminated from the list of limiting locations and justify the basis, including any assumptions, factors or criteria that were applicable when implementing this comparison.

By letter dated September 11, 2014, the applicant responded to RAI 4.3.4-4. This letter resubmitted RAI responses originally submitted on March 28, 2014, to clarify and reduce the information previously identified as proprietary by the applicant. The applicant stated that the locations within each piping system or major equipment with a fatigue analysis were evaluated and the results were compared, and the leading locations were identified and retained. In its response, the applicant described its screening evaluation of the cold leg safety injection accumulator piping, which resides in two different transient sections. The applicant stated that for this equipment, it compared the RCL nozzle, transient section 1 check valve, valve butt weld, and transient section 2 check valve. The applicant stated that the RCL nozzle, transient section 1 check valve, and the valve butt weld are in transient section 1, and the other check valve is in transient section 2. The applicant stated that it first evaluated the three components in transient section 1 to identify the most limiting component. The applicant stated that the F_{en} screening evaluation and the stress basis comparison determined the RCL nozzle to be the most limiting component for transient section 1. The applicant then applied the same F_{en} screening evaluation and stress basis comparison to evaluate the RCL nozzle and the transient section 2 check valve. The applicant stated that these evaluations determined the RCL nozzle to be the most limiting component within the cold leg safety injection accumulator piping. The staff finds the applicant response acceptable because the applicant provided adequate detail on how its evaluation determined the limiting location within a common system or major equipment and justified the factors and criteria used in the evaluation. The staff's concerns in RAI 4.3.4-4 are resolved. The staff finds this fourth step in the applicant's screening methodology appropriate, as discussed above.

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

4.3.4.3 UFSAR Supplement

LRA Section A.4.3.4 provides the UFSAR supplement summarizing the effects of the reactor coolant environment on fatigue life of piping and components. The staff reviewed LRA

Section A.4.3.4, consistent with the review procedures in SRP-LR 4.3.3.2, which state that the reviewer confirms that the applicant has provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of the reactor coolant environment on component fatigue life.

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

4.3.4.4 Conclusion

On the basis of its review, the staff concludes that the applicant has acceptably demonstrated, in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of EAF on the intended functions of the affected piping and components will be adequately managed by the Fatigue Monitoring Program for the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the EAF evaluations, as required by 10 CFR 54.21(d).

4.3.5 Reactor Vessel Internals Fatigue Analyses

4.3.5.1 Summary of Technical Information

LRA Section 4.3.5 states that the BBS RVIs were designed and procured prior to the issuance of ASME Code Section III, Subsection NG. However, the applicant stated that the intent of the code is applied at BBS with load combinations and allowable stresses, which is consistent with the requirements of ASME Code Section III, Subsection NG. The LRA states that the RVIs were designed to withstand stress originating from the same operating conditions as the reactor vessel. Using the RVI stress reports, CUFs less than 1.0 were determined for the maximum alternating stresses using the design transient cycles from each transient and the design ASME Code fatigue curve. The applicant further states that the bounding CUFs for the RVIs were evaluated for the BBS MUR power uprate project. The evaluation determined that the MUR power uprate did not affect the bounding CUFs, and therefore, no new CUFs were calculated for the MUR power uprate project. The applicant also states that the analyses performed for the RVI components are based upon a subset of the RCS design transients used in the fatigue analyses for the reactor vessel shown in LRA Tables 4.3.1-1 and 4.3.1-4.

The applicant dispositioned the TLAA for the RVIs in accordance with 10 CFR 54.21(c)(1)(iii) such that the effects of metal fatigue on the intended functions will be adequately managed by the Fatigue Monitoring program for the period of extended operation.

LRA Section 4.3.5 also states that the analyses associated with flow-induced vibrations of the RVIs are not based on time-dependent assumptions to be considered a TLAA in accordance with 10 CFR 54.3(a), Criterion 3. The applicant stated that these analyses concluded that the component stress ranges remained below the endurance limit of 10^{11} cycles on the applicable ASME fatigue curves, and therefore, the number of these stress cycles is not limited over the current operating life.

4.3.5.2 Staff Evaluation

The staff reviewed LRA Section 4.3.5 and the metal fatigue TLAAs for the RVIs to verify, in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of fatigue will be adequately managed by the Fatigue Monitoring program for the period of extended operation.

The staff reviewed the applicant's metal fatigue TLAAs for the RVIs and the corresponding disposition of 10 CFR 54.21(c)(1)(iii) consistent with the review procedures in SRP-LR Section 4.3.3.1.1.3. These procedures state that the reviewer should verify the appropriateness of the applicant's program for monitoring and tracking the number of critical thermal and pressure transients for the selected RCS components.

The staff found that the fatigue analyses were performed using the 40-year design transients in UFSAR Table 3.9-1, which are those transients listed in LRA Tables 4.3.1-1 and 4.3.1-4 and monitored by the Fatigue Monitoring program. The applicant stated that the fatigue analyses performed for the RVIs are based upon a subset of these design transients. However, the LRA does not provide the CUF values for the RVIs. By letter December 12, 2013, the staff issued RAI B.2.1.7-4, requesting that the applicant indicate the RVI components with existing CUF analyses.

In its response dated January 14, 2014, the applicant stated that the following RVI components have existing CUF analyses: upper core plate, upper core plate alignment pins, upper support plate, baffle plate, core barrel nozzle, lower radial restraints, lower core plate, and lower support columns. The applicant also provided the CUF values and material type for each of these components. The applicant stated that the Fatigue Monitoring program will be used to monitor the transients used in the fatigue analyses of these RVI components.

The staff confirmed that the associated CUF values were all below the acceptance criteria of 1.0. The staff determined that the Fatigue Monitoring program is capable of managing metal fatigue during the period of extended operation consistent with GALL Report X.M1. The staff finds the applicant's response to RAI B.2.1.7-4 acceptable. The staff's evaluation of the Fatigue Monitoring program is documented in SER Section 3.0.3.2.24.

The staff reviewed the staff's safety evaluation (ADAMS Accession No. ML13281A000) for the BBS MUR power uprate project. The staff noticed that the SE stated that the maximum calculated stresses and cumulative fatigue usage factor for the most limiting component of the RVIs are unaffected by the MUR and remain bounding.

The staff finds that the applicant demonstrated, in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of aging related to fatigue of the RVIs will be adequately managed for the period of extended operation. Additionally, the applicant's disposition meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.3 because the applicant is crediting its Fatigue Monitoring Program to manage metal fatigue to ensure that the allowable design limits on fatigue usage are not exceeded during the period of extended operation, otherwise, the applicant will take corrective actions in accordance with its program.

The applicant stated that analyses associated with flow-induced vibration in the RVIs are not considered a TLAA in accordance with 10 CFR 54.3(a), Criterion 3. The applicant stated that the analyses concluded that the component stress ranges remained below the endurance limit of 10^{11} cycles on the applicable ASME fatigue curves, therefore, the stress range cycles are not limited over the current operating life. The staff concluded that an analysis is only defined as a

TLAA if all six criteria outlined in 10 CFR 54.3 are satisfied. The staff reviewed the UFSAR and did not identify that the design basis for the RVIs for high-cycle fatigue depended on the licensed life of the plant period. Thus, the staff finds that all six criteria for a TLAA were not met for the applicant's evaluation of flow-induced vibration in the RVIs. Although high-cycle fatigue in the RVIs is not evaluated in the application as a TLAA, the staff found that the applicant's PWR Vessel Internals Program addresses the aging effects in the RVIs, including those that could be induced by a flow-induced vibration mechanism. The staff's evaluation of the PWR Vessel Internals Program is documented in SER Section 3.0.3.2.3. The staff further evaluated the absence of a TLAA basis for the RVI flow induced vibrations, as documented in SER Section 4.1.2.1.2.

4.3.5.3 UFSAR Supplement

LRA Section A.4.3.5 provides the UFSAR supplement summarizing the metal fatigue TLAA for the RVIs. The staff reviewed LRA Section A.4.3.5 consistent with the review procedures in SRP-LR Section 4.3.3.2, which state that the reviewer verifies that the applicant provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA.

Based on its review of the UFSAR supplement, the staff finds that LRA Section A.4.3.5 meets the acceptance criteria in SRP-LR 4.3.2.2, and is, therefore, acceptable. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address metal fatigue TLAA for the RVIs, as required by 10 CFR 54.21(d).

4.3.5.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of metal fatigue on the intended functions of the RVIs will be adequately managed by the Fatigue Monitoring program for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.6 High-Energy Line Break (HELB) Analyses Based on Fatigue

4.3.6.1 Summary of Technical Information

LRA Section 4.3.6 states that locations of postulated HELBs are based on two limiting stress criteria and a CUF criterion, as stated in UFSAR Section 3.6. The applicant identifies the postulations of break locations based on the fatigue criterion at BBS as TLAAs. The LRA states that one of the criteria used to determine whether a HELB must be postulated at a given location is that the fatigue usage calculated for the component is greater than 0.1.

The applicant dispositioned the TLAA in accordance with 10 CFR 54.21(c)(1)(iii) such that the Fatigue Monitoring program will be used to monitor transient cycles as inputs for the determination of postulated break locations and require corrective action prior to exceeding the numbers of analyzed cycles.

4.3.6.2 Staff Evaluation

The staff reviewed LRA Section 4.3.6 and the TLAAs for HELB postulations based on a CUF criterion to verify, in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of aging will be adequately managed during the period of extended operation.

The staff reviewed the applicant's TLAA and the corresponding disposition consistent with the review procedures in SRP-LR Section 4.7.3.1.3, which state that the applicant proposes to manage the aging effects associated with the TLAA by an AMP in the same manner as described in the integrated plant assessment (IPA) in 10 CFR 54.21(a)(3). The SRP-LR also states that the staff is to review the applicant's AMP to verify that the effects of aging on the intended function(s) are adequately managed consistent with the CLB for the period of extended operation. In addition, SRP-LR states that a license renewal applicant should identify the structures and components (SCs) associated with the TLAA.

UFSAR Section 3.6.2 states that high energy lines are those larger than 1 in. diameter for which the service temperature is greater than 200 °F (90 °C) or the design pressure is greater than 275 psig. The staff noticed that a given location is identified as a line break location if it was a high-energy line location that satisfied the criteria in UFSAR Section 3.6.2.1.2.1. One such criterion is that any intermediate location between terminal ends where the CUF from the piping fatigue analysis exceeds 0.1 is identified as a line break location. The staff found that the postulations of break location based on CUFs are TLAAs because they are dependent on an assumed number of cycles expected for the design of the plant. The staff concluded that UFSAR Section 3.6.2.1.1 states that the dynamic effects from postulated breaking of the reactor coolant primary piping, accumulator line piping, and reactor coolant loop bypass piping can be eliminated through the application of approved leak-before-break (LBB) technology.

The applicant credits the Fatigue Monitoring program to manage metal fatigue of these postulated HELB locations through the period of extended operation. The staff noticed that as long as the number of transients that occur at the site remain bounded by the 40-year number of cycles assumed in these analyses, the HELB postulation evaluation remains valid. The staff determined that the enhanced Fatigue Monitoring program ensures that the number of transients will not be exceeded during the period of extended operation or that corrective actions are taken. The staff's evaluation of the Fatigue Monitoring program is documented in SER Section 3.0.3.2.24.

The staff finds that the applicant demonstrated, in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of aging related to metal fatigue of the HELB postulated locations based on CUF will be adequately managed through the period of extended operation. Additionally, LRA Section 4.3.6 meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.3 because the applicant's Fatigue Monitoring program monitors and tracks the transient cycles assumed in the analysis and requires corrective action prior to exceeding the number of transient cycles used in the analysis.

4.3.6.3 UFSAR Supplement

LRA Section A.4.3.6 provides the UFSAR supplement which summarizes the TLAA for HELB postulated locations based on CUF. The staff reviewed LRA Section A.4.3.6 consistent with the review procedures in SRP-LR-Section 4.3.3.2, which state that the information to be included in the UFSAR supplement should include a summary description of the evaluation of the TLAA.

Based on its review of the UFSAR supplement, the staff finds that LRA Section A.4.3.6 meets the acceptance criteria in SRP-LR 4.3.2.2. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the TLAA for HELB postulated locations based on CUF as required by 10 CFR 54.21(d).

4.3.6.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the intended functions of the HELB postulated locations will be adequately managed by the Fatigue Monitoring Program for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.7 NRC Bulletin 88-11 Revised Fatigue Analysis of the Pressurizer Surge Line for Thermal Cycling and Stratification

4.3.7.1 Summary of Technical Information

NRC Bulleting 88-11, issued in December 1988, requested utilities to establish and implement a program to confirm the integrity of the pressurizer surge line. The program required both visual inspection of the surge line and demonstration that the design requirements of the pressurizer surge line are satisfied, including the consideration of stratification effects. LRA Section 4.3.7 states that BBS demonstrated consideration of thermal stratification using an ASME Code Section III fatigue analysis. Since the analysis uses time-limited assumptions, such as thermal and pressure transients, operating cycles, and the licensed life of the plant, the analyses required by NRC Bulletin 88-11 have been identified as TLAAs.

LRA Section 4.3.7 further states that pressurizer surge line stratification subtransients were developed for the original analyses. The applicant stated that the ASME Code stress limits and CUF requirements were shown to be acceptable for the current licensed life of BBS. The original analyses were evaluated for impact due to a power uprate project in 2000 and an MUR uprate in 2010. The applicant stated that these evaluations determined that the original analyses were not impacted by the uprate projects. The applicant further states that the fatigue evaluations for the components affected by this bulletin were revised to consider the baseline and projected transients in Tables 4.3.1-1, 4.3.1-2, 4.3.1-4, and 4.3.1-5. The LRA states that the conclusion from the analyses is that the components will continue to meet the design allowable usage of 1.0 with consideration of stratification effects during the period of extended operation.

The applicant dispositioned this TLAA in accordance with 10 CFR 54.21(c)(1)(iii) such that the Fatigue Monitoring program will monitor the transient cycles and severities which are inputs to these analyses and require corrective action prior to exceeding design limits that would invalidate these conclusions.

4.3.7.2 Staff Evaluation

The staff reviewed LRA Section 4.3.7 and the TLAA associated with the fatigue analysis of the pressurizer surge line for thermal cycling and stratification to verify, in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of aging will be adequately managed by the Fatigue Monitoring program for the period of extended operation. The staff reviewed this TLAA and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), consistent with the review procedures in SRP-LR 4.3.3.1.1.3. These procedures state that the reviewer should verify the appropriateness of the applicant's program for monitoring and tracking the number of critical thermal and pressure transients for the selected RCS components.

LRA Section 4.3.7 states that pressurizer surge line stratification subtransients were developed and used in the original analyses which were performed to demonstrate compliance with design requirements. However, the LRA did not include the pressurizer surge line stratification subtransients that were developed.

By letter dated February 26, 2014, the staff issued RAI 4.3.7-1, Part 1, requesting that the applicant identify the pressurizer surge line stratification subtransients that were developed.

By letter dated September 11, 2014, the applicant responded to RAI 4.3.7-1, Part 1. This letter resubmitted RAI responses originally submitted on March 28, 2014, to clarify and reduce the information previously identified as proprietary by the applicant. The applicant stated that in response to Bulletin 88-11, 11 subtransient cases were developed for the surge line piping and nine subtransient cases were developed for the surge line nozzle. The applicant stated that these subtransients were developed based on a detailed evaluation to characterize the cyclic activity during heatup and cooldown and to define a bounding set of differential temperatures. The applicant stated that the fatigue analysis for the pressurizer surge line was performed to account for the surge line pipe stratification subtransients that occur during the postulated 200 heatup and cooldown cycles. The staff finds the applicant's response to RAI 4.3.7-1, Part 1, acceptable because the applicant provided the subtransients developed in response to Bulletin 88-11 and demonstrated that these subtransients are bounded by heatup and cooldown transients that the staff confirmed will be monitored by the Fatigue Monitoring program. The staff's concern in RAI 4.3.7-1, Part 1, is resolved.

LRA Section 4.3.7 states that the fatigue evaluations for the components associated with NRC Bulletin 88-11 were revised to consider the baseline and 60-year projected transients listed in LRA Tables 4.3.1-1, 4.3.1-2, 4.3.1-4, and 4.3.1-5. The applicant stated that the fatigue usage will continue to meet the design limit of 1.0 through the period of extended operation. However, the applicant did not provide enough information on how the fatigue evaluations were revised by the LRA tables listed. By letter dated February 26, 2014, the staff issued RAI 4.3.7-1, Parts 2 and 3, requesting the applicant to identify which transients were considered when the fatigue evaluations were revised and to confirm that its Fatigue Monitoring program will adequately monitor and track the revised transients and require corrective action prior to exceeding design limits.

In the applicant's response to RAI 4.3.7-1, Parts 2 and 3, by letter dated September 11, 2014, the applicant provided the transients that were considered in the fatigue evaluations of the components affected by Bulletin 88-11. The applicant stated that the fatigue evaluations accounted for stratification effects for both heatup/cooldown transients and non-heatup/cooldown transients. The applicant stated that for the affected non-heatup/cooldown transients, stratification was assumed as the maximum applicable temperature difference for each transient cycle. The staff confirmed that the provided transients are included in the transients monitored by the Fatigue Monitoring program. The staff finds the applicant's response to RAI 4.3.7-1, Parts 2 and 3, acceptable because the applicant identified the transients that were considered in the fatigue evaluations that include the effects of thermal stratification and confirmed that the Fatigue Monitoring program will monitor the transient cycles and severities and will require action prior to exceeding design limits. The staff determined that the enhanced Fatigue Monitoring program ensures that the number of transients will not be exceeded during the period of extended operation or that corrective actions are taken. The staff's evaluation of the Fatigue Monitoring program is documented in SER Section 3.0.3.2.24.

LRA Section 4.3.7 states that the original analyses were evaluated for impact due to a power uprate project in 2000 and an MUR uprate in 2010. The applicant stated that the evaluations determined that the original analyses were not impacted by these uprate projects. The staff reviewed the staff's safety evaluation (ADAMS Accession No. ML033040016) for the BBS power uprate project in 2000. The staff noticed that the SE concluded that the fatigue usage factors for the pressurizer surge line will continue to meet the ASME Code requirements for the power uprate. The staff reviewed the staff's safety evaluation (ADAMS Accession No. ML13281A000) for the BBS MUR uprate project in 2010. The staff noticed that the SE concluded there was no adverse effect on the fatigue evaluation of the pressurizer surge line, including the effects of thermal stratification.

The staff finds that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging related to fatigue analysis of the pressurizer surge line, including thermal stratification, will be adequately managed for the period of extended operation. Additionally, TLAA 4.3.7 meets the acceptance criteria in SRP-LR 4.3.2.1.1.3 because the applicant is crediting its Fatigue Monitoring program to manage metal fatigue to ensure that the allowable design limits on fatigue usage are not exceeded during the period of extended operation; otherwise, the applicant will take corrective action, in accordance with its program.

4.3.7.3 UFSAR Supplement

LRA Section A.4.3.7 provides the UFSAR supplement summarizing the metal fatigue TLAA for the pressurizer surge line, including thermal stratification. The staff reviewed LRA Section A.4.3.7 consistent with the review procedures in SRP-LR Section 4.3.3.2, which state that the reviewer confirms that the applicant has provided information to be included in the UFSAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA.

Based on its review of the UFSAR supplement, the staff finds that LRA Section A.4.3.7 meets the acceptance criteria in SRP-LR Section 4.3.2.2. Additionally, the staff determined that the applicant provided an adequate summary description of its actions to address the pressurizer surge line, including thermal stratification, as required by 10 CFR 54.21(d).

4.3.7.4 Conclusion

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

4.3.8 ASME Code Section III, Subsection NF, Class 1 Component Supports Allowable Stress Analyses

4.3.8.1 Summary of Technical Information in the Application

LRA Section 4.3.8 describes the applicant's TLAA for ASME Code Section III, Subsection NF, Class 1 component supports allowable stress analyses, which include supports for the reactor vessel, steam generator, RCP, and pressurizer. The applicant stated that the BBS Class 1 component supports are inherently designed for 20,000 stress cycles of fatigue loading in accordance with the allowable stresses of ASME Code Section III, Subsection NF, 1974 edition through the 1975 summer addenda. The applicant referenced NRC-approved Westinghouse Owners Group (WOG) TR WCAP-14422, Revision 2-A, "License Renewal Evaluation: Aging Management of Reactor Coolant System (RCS) Supports," that performed a technical evaluation of cumulative fatigue aging effects on Class 1 component supports in Westinghouse reactors for the period of extended operation. The report concluded that the number of actual loading transients that affect Class 1 components is projected to be significantly less than 20,000 loading cycles for 60 years and estimated the corresponding fatigue usage to be less than 0.15, which is less than the allowable limit of 1.0. The applicant concluded that the numbers of cycles for the transients analyzed in TR WCAP-14422 are bounded by the transient limits shown in LRA Section 4.3.1. The applicant stated that the NRC Final Safety Evaluation Report, dated November 7, 2000, for TR WCAP-14422, Revision 2-A, required with regard to fatigue that each license renewal applicant justify the use of installed materials not listed in Table 2-4 of the topical report. The applicant addressed this condition by stating that its review of design documents found that the majority of installed Class 1 component support materials used were listed in WCAP Table 2-4. The applicant also stated that evaluation of several materials used that were not listed in the table showed that their yield strength and fatigue resistance properties are consistent with materials in Table 2-4 of WCAP-14422 or the materials are used in bearing plates which do not experience cyclical tensile stresses.

The applicant dispositioned the TLAA for the Class 1 component supports in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of aging due to fatigue on the intended functions will be adequately managed by the Fatigue Monitoring Program for the period of extended operation. The applicant stated that the Fatigue Monitoring Program in LRA Section B.3.1.1 will monitor transient cycles and require action prior to exceeding the design limits that would invalidate the conclusions of the TLAA.

4.3.8.2 Staff Evaluation

The staff reviewed the applicant's metal fatigue TLAA in LRA Section 4.3.8 related to ASME Code Section III, Subsection NF, Class 1 component supports for the reactor vessel, steam generator, RCP, and pressurizer, and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), that the effects of aging due to fatigue will be adequately managed for the period of extended operation by the Fatigue Monitoring Program, consistent with the review procedures in SRP-LR Section 4.3.3.1.1.3. These procedures state that the reviewer should verify the appropriateness of the applicant's program for monitoring and tracking the number of critical thermal and other transients for the selected RCS components, which in this case are the Class 1 RCS component supports.

The staff determined that LRA Section 4.3.8 references NRC-approved WOG TR WCAP-14422, Revision 2-A, "License Renewal Evaluation: Aging Management of Reactor Coolant System (RCS) Supports," as a bounding evaluation in the fatigue TLAA for the Subsection NF Class 1 component supports for the reactor vessel, steam generator, RCP, and pressurizer. In its SER dated November 17, 2000, the staff found TR WCAP-14422, Revision 2-A, acceptable for member plants to reference in an LRA to the extent specified and under the limitations delineated in the SER, which includes completing the renewal applicant action items described in Section 4.1 of the SER. SER Section 4.1, "Renewal Applicant Action Item 6 Fatigue (Section 3.3.1.7)," states that a license renewal applicant will have to justify differences between the materials used for its RCS supports and the values listed in Table 2-4 of the TR. The staff reviewed TR WCAP-14422, Revision 2-A, and confirmed that BBS was included as member operating plants to which the evaluation applies. The staff noticed that Section 3.2 of TR WCAP-14422, Revision 2-A, estimated the fatigue CUF of less than 0.15, which is less than the code allowable limit of 1.0, for 60 years of operation. This estimate was based on 300 thermal cycles (normal conditions with stress amplitude, S_a, of 30 ksi) from plant heatup and cooldown and 600 cycles (upset conditions with S_a of 50 ksi) from operating basis earthquake (OBE) seismic events for a total of 900 transient cycles, which is significantly smaller than the 20,000 cycles implicitly included in ASME Code Section III, Subsection NF design. The staff also found the applicant's statement in LRA Section 4.3.8 that the number of transients analyzed in the topical report are bounded by the transient limits shown in LRA Section 4.3.1.

From the information provided in LRA Section 4.3.8, it was not clear which specific transient limits in LRA Section 4.3.1 were considered in making the comparison to the number of transients analyzed in the WOG Report WCAP-14422. Further, the information provided in LRA Section 4.3.8 with regard to "License Renewal Applicant Action Item 6" in the NRC SER for TR WCAP-14422, Revision 2-A, was not sufficient for the staff to verify that the fatigue evaluation in the TR remains bounded for materials used in the Byron and Braidwood RCS component supports that are not listed in Table 2-4 of the TR. Therefore, by letter dated April 24, 2014, the staff issued RAI 4.3.8-1, requesting the applicant to: (1) identify the specific transient (cycle) limits in LRA Section 4.3.1 that were used in LRA Section 4.3.8 to make the comparison with the number of transients analyzed in the TR, and (2) provide a list of the materials used in the Class 1 RCS component supports that are not documented in Table 2-4 of Topical Report WCAP-14422, Revision 2-A, including information such as the yield strength, fatigue resistance properties, the component and the function of the component where the material is used, such that the staff can verify that the fatigue evaluation in the topical report remains bounding for the RCS support components using these materials.

The applicant provided its response to RAI 4.3.8-1 by letter dated May 23, 2014. In its response to the first part of the RAI 4.3.8-1, the applicant stated that three specific transients and associated cycle limits in LRA Section 4.3.1, which are same as those analyzed in the WOG Topical Report WCAP-14422, were considered in LRA Section 4.3.8. The applicant further clarified that these transients are identified in LRA Tables 4.3.1-1 and 4.3.1-4 as Transient 1, "Plant Heatup at 100 °F/hr," and Transient 2, "Plant Cooldown at 100 °F/hr," each with CLB allowable cycle limits of 200; and Transient 34 "Operating Basis Earthquake (OBE)," with a CLB allowable cycle limit of 20 OBE seismic events each with 20 subcycles for a total of 400 cycles.

The staff finds the response to the first part of RAI 4.3.8-1 acceptable because (a) the applicant clarified the transients applicable to fatigue of RCS component supports, namely cycles from plant heatup and cooldown and OBE seismic events which are the same as those deemed applicable and evaluated in TR WCAP-14422, and the corresponding allowable CLB transient limits documented in LRA Section 4.3.1, and (b) the allowable transient limits (200 plant heatup and cooldown cycles plus 400 seismic OBE cycles) in LRA Section 4.3.1 for applicable transients are bounded by the corresponding number of transient cycles (300 plant heatup and cooldown cycles plus 600 seismic OBE cycles) evaluated for 60 years of operation in TR WCAP-14422. Therefore, the staff's concern described in the first part of RAI 4.3.8-1 is resolved.

In its response to the second part of RAI 4.3.8-1, the applicant stated that BBS RCS Class 1 component supports were designed, fabricated, and installed in accordance with the requirements of ASME Code Section III, Division 1, Subsection NF, 1974 Edition with Summer 1975 addendum, using materials that conform to ASME/ASTM materials meeting the requirements of ASME Code Case 1644, "Additional Materials for Component Supports and Alternate Design Requirements for Bolted Joints Section III Division 1, Subsection NF Class, 1, 2, 3, and MC construction." The applicant further explained that, at BBS, four subcomponent types associated with RCS Class 1 component supports are constructed of materials that are not specifically documented in Table 2-4 of TR WCAP-14422 (which lists only the most commonly specified materials), and addressed the differences as summarized in SER Table 4.3.8-1 below. The applicant concluded that the information provided in the response demonstrates that the fatigue evaluation in TR WCAP-14422 is bounding for the RCS Class 1 component supports using these materials.

Table 4.3.8-1 Summary of Material Differences Addressed in RAI 4.3.8-1 Response

RCS Support Subcomponent	Material used at BBS different from Table 2-4 of TR WCAP-14422	Corresponding material listed in Table 2-4 of TR WCAP-14422 for same application	Impact of material difference on fatigue resistance properties at stress amplitude (Sa) values evaluated in TR WCAP-14422
Steam Generator Lower Lateral Support Inner Frame Structural Plates, 6.5 in. thick (UFSAR Figure 3.9-7a)	ASME SA533 Class 2 alloy steel, minimum yield strength (f_y) = 70 ksi, tensile strength (f_t) = 90-115 ksi	ASTM A588 Grade A or B LAS (f_y = 42 ksi, f_t = 63 ksi)	No impact because fatigue resistance properties (allowable loading cycles) are essentially the same based on Figure I-9.1 in Appendix I of ASME Code Section III, Division 1
Steam Generator Upper Lateral Support Snubber End-Blocks (UFSAR Figures 3.9-6 and 3.9-8)	ASME SA533 Grade B, Class 1 alloy steel (f_y = 50 ksi, f_t = 80-100 ksi), ASME SA516 Grade 70 carbon steel (f_y = 38 ksi, f_t = 70-90 ksi), or ASTM A36 carbon steel (f_y = 36 ksi, f_t = 58-80 ksi)	ASTM A572 Grade 42 LAS (f_y = 42 ksi, f_t = 60 ksi)	No impact because fatigue resistance properties (allowable loading cycles) are essentially the same based on Figure I-9.1 in Appendix I of ASME Code Section III, Division 1
Bolting	ASME SA193 Grade B7 alloy steel (f_y = 75-105 ksi, f_t = 100-125 ksi, depending on bolt size)	ASTM A354 Grade BC alloy steel (f_y = 99-109 ksi, f_t = 115-125 ksi), depending on bolt size, and ASTM A540 Grade B-23 Class 4 alloy steel (f_y = 120 ksi, f_t = 135 ksi)	No impact because fatigue resistance properties (allowable loading cycles) are essentially the same based on Figures I-9.1 and I-9.4 in Appendix I of ASME Code Section III, Division 1
Shim Plate and Spacer Materials	ASTM/ASME A36, A53, A366, A414, A569, A570, A606 Type 4, A607, and A1008 CS Type B	None listed	Fatigue aging effect not applicable since shim plates and spacers are designed for compression loads and not subject to cyclic tensile stresses and fatigue

The staff reviewed ASME Code Section III, Division 1, Appendix I, Figure I-9.1, "Design Fatigue Curves for Carbon, Low Alloy, and High Tensile Steels for Metal Temperatures Not Exceeding 700 °F," and Figure I-9.4, "Design Fatigue Curves for High Strength Steel Bolting for Temperatures Not Exceeding 700 °F." From these figures, for the stress amplitudes (S_a of 30 ksi and 50 ksi) evaluated in the TR, the staff confirmed that the number of allowable fatigue cycles for the materials used at BBS, as described in the table above as different from those listed in Table 2-4 of TR WCAP-14422 are essentially consistent with that for the corresponding materials listed in the Table 2-4. The staff concluded that the fatigue aging effect does not apply to the shim and spacer plates because they are subject to compressive stresses only. Therefore, the staff determines that the fatigue evaluation in TR WCAP-14422 remains bounding for the RCS support components at BBS considering the limited use of these different materials.

The staff finds the applicant's response to the second part of RAI 4.3.8-1 acceptable because (a) the applicant provided the list and fatigue properties of the materials that are also used in some RCS support components at BBS and are different from those listed for the corresponding application in Table 2-4 of TR WCAP-14422, and (b) the information provided enabled the staff to verify that the fatigue evaluation in TR WCAP-14422 remains bounding for the RCS support components at BBS considering the limited use of these different materials. Therefore, the staff's concerns described in the second part of RAI 4.3.8-1 is resolved.

The staff finds that, although Section 3.2.6 of TR WCAP-14422 does not require identification of an AMP for fatigue, the applicant uses the Fatigue Monitoring Program to monitor RCS component supports for thermal and seismic OBE transient cycles against CLB allowable cycle limits of 200 and 400, respectively. The program requires corrective action prior to exceeding these cycle limits through the period of extended operation. The staff noticed that the enhanced Fatigue Monitoring Program is described in LRA Section B.3.1.1 as consistent with the 10 elements of X.M1 "Fatigue Monitoring" AMP in the GALL Report. The staff confirmed that supports and thermal transients were included in the program description. Also, due to the applicant's response to RAI 4.3.9-1, as discussed in SER Section 4.3.9, the staff confirmed that seismic OBE transients are also monitored by the program. The staff determined that the enhanced Fatigue Monitoring Program ensures that the number of transients will not be exceeded during the period of extended operation or that corrective actions are taken. The staff's evaluation of the Fatigue Monitoring Program is documented in SER Section 3.0.3.2.24.

The staff finds the applicant demonstrated pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging due to metal fatigue on the intended functions of the ASME Code Section III, Subsection NF, Class 1 component supports for the reactor vessel, steam generator, RCP, and pressurizer at BBS will be adequately managed for the period of extended operation.

Additionally, the TLAA meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.3 because the applicant's Fatigue Monitoring Program monitors and tracks applicable transient cycles, namely plant heatup and cooldown transients and OBE seismic events, through the period of extended operation and requires corrective action prior to exceeding the CLB allowable transient limits, which bound the number of transient cycles used in the analysis, to ensure that the design CUF limit of 1.0 is not exceeded during the period of extended operation.

4.3.8.3 UFSAR Supplement

LRA Section A.4.3.8 provides the UFSAR supplement summarizing the ASME Code Section III, Subsection NF, Class 1 component supports allowable stress analyses TLAA for supports of the reactor vessel, steam generator, RCP, and pressurizer. The staff reviewed LRA Section A.4.3.8 consistent with the review procedures in SRP-LR Section 4.3.3.2, which state that the staff verifies that the applicant has provided a UFSAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA with information equivalent to that in SRP-LR Table 4.3-2.

Based on its review of the UFSAR supplement, the staff finds LRA Section A.4.3.8 meets the acceptance criteria in SRP-LR Section 4.3.2.2, and is therefore acceptable. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address TLAA evaluations involving ASME Code Section III, Subsection NF, Class 1 component supports allowable stress analyses, as required by 10 CFR 54.21(d).

4.3.8.4 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 due to fatigue on the intended functions of the RCS Class 1 component supports for the reactor vessel, steam generator, RCP, and pressurizer will be adequately managed by the Fatigue Monitoring Program for the period of extended operation. The staff also concludes that the UFSAR supplement in LRA Section A.4.3.8 contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.9 Fatigue Design of Spent Fuel Pool Liner and Spent Fuel Storage Racks for Seismic Events

4.3.9.1 Summary of Technical Information in the Application

LRA Section 4.3.9 describes the applicant's TLAA related to fatigue design of the spent fuel pool (SFP) liner and the spent fuel storage racks for seismic events. The applicant stated that the TLAA includes a fatigue evaluation of the spent fuel storage racks (which were replaced in 2000-2001 and designed in accordance with ASME Code Section III, Subsection NF) and the SFP liner for the cyclic loads imposed by twenty (20) OBE events plus one (1) safe shutdown earthquake (SSE) event using methods similar to those for Class 1 components, in accordance with ASME Code Section III, Subsection NB. The analyses also include a fatigue evaluation of the SFP liner for the loads imposed by the new racks using the same input for seismic events. These analyses calculated a CUF of 0.95 for the spent fuel storage racks and a CUF of 0.00052 for the SFP liner, both of which are less than the allowable CUF of 1.0.

The applicant further stated that OBE events are monitored by the Fatigue Monitoring Program described in LRA Section B.3.1.1 and that no OBE or SSE events have occurred to date. The applicant also stated that the Fatigue Monitoring Program will continue to monitor OBE and SSE transient cycles, to manage fatigue of these components through the period of extended operation.

The applicant dispositioned the TLAA for Fatigue Design of SFP liner and spent fuel storage racks for seismic events in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of aging due to fatigue on the intended functions of these components will be adequately managed by the Fatigue Monitoring Program for the period of extended operation.

4.3.9.2 Staff Evaluation

The staff reviewed the applicant's TLAA in LRA Section 4.3.9 related to the fatigue design of SFP liner and spent fuel storage racks for seismic events and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), that the effects of aging due to fatigue from seismic events will be adequately managed for the period of extended operation by the Fatigue Monitoring Program, consistent with the review procedures in SRP-LR Section 4.3.3.1.1.3. These procedures state that the reviewer should verify the appropriateness of the applicant's program for monitoring and tracking the number of critical transients for the selected components, which in this case are the SFP liner and the spent fuel storage racks.

The staff also confirmed that the applicant's calculated seismic fatigue CUF values of 0.95 and 0.0052 for the spent fuel storage racks and the SFP liner, respectively, based on twenty (20) OBE events and one (1) SSE event, is less than the design code allowable of 1.0. The staff noticed that no seismic events have occurred to date at BBS, and the number of seismic events considered in the fatigue evaluation is conservative and not expected to be exceeded during the period of extended operation. Nevertheless, the staff concluded that the applicant will use its enhanced Fatigue Monitoring Program, described in LRA Section B.3.1.1, to manage aging effects due to seismic fatigue by monitoring number of occurrences of seismic transients through the period of extended operation, and the program requires corrective action prior to exceeding the number of seismic transient cycles assumed in the fatigue evaluation.

The staff also reviewed the applicant's descriptions of the enhanced Fatigue Monitoring Program, in LRA Section B.3.1.1 and LRA Section A.3.1.1, credited as the AMP in the disposition of this TLAA. From the information provided in the descriptions, it was not clear whether the scope of the applicant's Fatigue Monitoring Program included the SFP liner and spent fuel storage racks as components monitored, and load cycles from seismic events as transients monitored. Further, the LRA did not provide information with regard to the number of load cycles considered in the fatigue evaluation for each OBE and SSE event that would define the total bounding limit of seismic transients that would be monitored against by the Fatigue Monitoring Program, such that appropriate corrective action is taken before the number of seismic transient cycles assumed in the fatigue evaluation is exceeded during the period of extended operation. Therefore, by letter dated April 24, 2014, the staff issued RAI 4.3.9-1 requesting the applicant to: (1) clarify whether the LRA Section B.3.1.1, "Fatigue Monitoring" program includes under its scope (a) the SFP liner and spent fuel storage racks as components, and (b) load cycles from OBE and SSE events as parameters monitored and tracked; (2) identify the number of specific load cycles considered, in the fatigue evaluation of the spent fuel storage racks and SFP liner in LRA Section 4.3.9, for each OBE event and the SSE event. The staff also requested the applicant to update the LRA, as necessary, based on the response to RAI 4.3.9-1.

The applicant provided its response to RAI 4.3.9-1 by letter dated May 23, 2014. In its response to the first part of RAI 4.3.9-1, the applicant stated that (a) the SFP liner and the replacement spent fuel storage racks are included as "other components" within the scope of the Fatigue Monitoring program described in LRA Section B.3.1.1, and (b) the occurrence of OBE and SSE seismic transient events are monitored and tracked as parameters in the Fatigue Monitoring program. The applicant further clarified that the number of specific load cycles occurring in each seismic event are evaluated as part of the event analysis using the parameters of duration, magnitude, and cycles of the event. The applicant revised LRA Sections 4.3.9 and A.4.3.9 to clarify OBE and SSE events are monitored by the Fatigue Monitoring Program. The applicant also revised LRA Sections B.3.1.1 and A.3.1.1 to explicitly identify "other components" and seismic transients in the program description of the Fatigue Monitoring Program.

In its response to the second part of RAI 4.3.9-1, the applicant stated that the number of specific load cycles utilized in the fatigue evaluation of the replacement spent fuel storage racks and the SFP are 25 cycles for each of the 20 OBE events, and 20 cycles for the single SSE event. The applicant also revised LRA Section 4.3.9 to include this information.

The staff finds the applicant's response to RAI 4.3.9-1 acceptable because the applicant (a) clarified that the SFP liner and spent fuel storage racks are included as components, and the OBE and SSE seismic event transients are monitored under the scope of the Fatigue Monitoring program, (b) provided the total number of seismic load cycles (20 times 25 OBE load cycles plus 20 SSE cycles for a total of 520 cycles) considered in the fatigue evaluation and monitored against by the Fatigue Monitoring Program credited in the TLAA to manage the effects aging due to seismic fatigue, and (c) updated applicable LRA sections to reflect clarifying information provided in the response. Therefore, the staff's concerns described in RAI 4.3.9-1 are resolved. The staff thus determined that the enhanced Fatigue Monitoring program ensures that the number of seismic transients will not be exceeded during the period of extended operation or that corrective actions are taken. The staff's evaluation of the Fatigue Monitoring program is documented in SER Section 3.0.3.2.24.

The staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging due to fatigue from seismic events on the intended functions of the SFP liner and spent fuel storage racks at BBS will be adequately managed for the period of extended operation.

Additionally, the TLAA meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.3 because the applicant's enhanced Fatigue Monitoring Program monitors OBE and SSE seismic transient cycles and requires corrective action prior to exceeding the conservative allowable seismic transient limits used in the fatigue evaluation to ensure that the design CUF limit of 1.0 is not exceeded during the period of extended operation.

4.3.9.3 UFSAR Supplement

LRA Section A.4.3.9, as amended by letter dated May 23, 2014, provides the UFSAR supplement summarizing the TLAA for fatigue design of SFP liner and spent fuel storage racks for seismic events. The staff reviewed LRA Section A.4.3.9 consistent with the review procedures in SRP-LR Section 4.3.3.2, which state that the staff verifies that the applicant has provided a UFSAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA with information equivalent to that in SRP-LR Table 4.6-1.

Based on its review of the UFSAR supplement, as amended by letter dated May 23, 2014, the staff finds it meets the acceptance criteria in SRP-LR Section 4.3.2.2, and is therefore acceptable. Additionally, the staff determines that the applicant provided an adequate summary description as required by 10 CFR 54.21(d), of its actions to address TLAA evaluations involving fatigue design of SFP liner and spent fuel storage racks for seismic events.

4.3.9.4 Conclusion

On the basis of its review, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging due to fatigue from seismic events on the intended functions of the SFP liner and spent fuel storage racks will be adequately managed by the Fatigue Monitoring Program for the period of extended operation. The staff also concludes that the UFSAR supplement in LRA Section A.4.3.9 contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.10 Pressurizer Heater Sleeve Structural Assessment

4.3.10.1 Summary of Technical Information

LRA Section 4.3.10 describes plant OE regarding a leak that was discovered in the pressurizer heating element penetration sleeve number 52 during the Braidwood Unit 1 refueling outage in May 2006. The applicant stated that the failed sleeve was repaired by cutting the leaking sleeve out and installing a permanent plug in its place. The applicant stated that the design analysis for the sleeve repair plug evaluated fatigue in accordance with ASME Code Section III, Subparagraph NB-3222.4. The fatigue evaluation assumed 200 RCS heatup and cooldown transients and is, therefore, a TLAA requiring evaluation for the period of extended operation. The LRA states that based on the transient design of 200 RCS heatups and cooldowns, a CUF of a maximum 0.003 was calculated, which is below the allowable CUF value of 1.0.

The applicant dispositioned this TLAA in accordance with 10 CFR 54.21(c)(1)(iii) such that the Fatigue Monitoring program will be used to monitor the transient cycles and require corrective action prior to exceeding design limits that would invalidate this analysis.

4.3.10.2 Staff Evaluation

The staff reviewed LRA Section 4.3.10 and the TLAA for the pressurizer heater sleeve structural assessment to verify, in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of aging will be adequately managed during the period of extended operation.

The staff reviewed the applicant's TLAA and the corresponding disposition consistent with the review procedures in SRP-LR Section 4.7.3.1.3, which state that the applicant proposes to manage the aging effects associated with the TLAA by an AMP in the same manner as described in the IPA in 10 CFR 54.21(a)(3). The SRP-LR also states that the staff reviewed the applicant's AMP to verify that the effects of aging on the intended function(s) are adequately managed consistent with the CLB for the period of extended operation. In addition, the SRP-LR requires that a license renewal applicant must identify the SCs associated with the TLAA.

The applicant credits the Fatigue Monitoring program to monitor the transient cycles and requires corrective action prior to exceeding the design limits that would invalidate this analysis. The staff reviewed LRA Tables 4.3.1-1 and 4.3.1-4, which provide the baseline and 60-year cycle projections for RCS transients and noticed that the CLB cycle limit for RCS heatup and cooldown transients are consistent with the limit assumed in the fatigue evaluation for the repair plug. Therefore, the staff determined that the transients assumed in the design analysis for the sleeve repair plug are bounded by their 60-year projection. The staff determined that the enhanced Fatigue Monitoring program ensures that the number of transients will not be exceeded during the period of extended operation or that corrective actions are taken. The staff's evaluation of the Fatigue Monitoring program is documented in SER Section 3.0.3.2.24.

The staff finds that the applicant demonstrated, in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of aging related to fatigue on the intended functions of the Braidwood Unit 1 reactor coolant pressurizer heating element penetration sleeve repair will be adequately managed consistent with the CLB for the period of extended operation. Additionally, LRA Section 4.3.10 meets the acceptance criteria in SRP-LR Section 4.7.2.1 because the applicant's Fatigue Monitoring program monitors and tracks the transient cycles assumed in the analysis and requires corrective action prior to exceeding the number of transient cycles used in the analysis.

4.3.10.3 UFSAR Supplement

LRA Section A.4.3.10 provides the UFSAR supplement which summarizes the Braidwood Unit 1 reactor coolant pressurizer heating element penetration sleeve repair TLAA. The staff reviewed LRA Section A.4.3.10 consistent with the review procedures in SRP-LR-Section 4.7.3.2, which state that the information to be included in the UFSAR supplement should include a summary description of the evaluation of the TLAA.

Based on its review of the UFSAR supplement, the staff finds that LRA Section A.4.3.10 meets the acceptance criteria in SRP-LR 4.7.2.2. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the Braidwood Unit 1 reactor coolant pressurizer heating element penetration sleeve repair TLAA as required by 10 CFR 54.21(d).

4.3.10.4 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 on the intended functions of the Braidwood Unit 1 reactor coolant pressurizer heating element penetration sleeve repair will be adequately managed by the Fatigue Monitoring program for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.4 Environmental Qualification (EQ) of Electric Components

4.4.1 Summary of Technical Information in the Application

LRA Section 4.4 describes the applicant's TLAA for the evaluation of electrical equipment EQ for the period of extended operation. The applicant stated that aging evaluations for electrical components in the Byron and Braidwood EQ program that specify a qualification of at least 40 years have been identified as TLAAs for license renewal because the criteria contained in 10 CFR 54.3 are met. The applicant also stated that the Byron and Braidwood program meets the requirements of 10 CFR 50.49 for the applicable electrical components important to safety. The Byron and Braidwood EQ program manages applicable component thermal, radiation, and cyclic aging effects through aging evaluations for the current operating license using methods for qualification for aging and accident conditions established by 10 CFR 50.49(f). In addition, the applicant stated that 10 CFR 50.49(e)(5) requires replacement or refurbishment of components not qualified for the license term prior to the end of designated life, unless additional life is established through ongoing qualification. Further, the applicant stated that the Byron and Braidwood EQ program implemented under the requirements of 10 CFR 50.49 and the guidance of NUREG-0588 and RG 1.89 is viewed as an AMP under 10 CFR 54.21(c)(1)(iii). The applicant stated that reanalysis of an aging evaluation to extend the qualification of components is performed on a routine basis as part of the EQ program. The applicant further stated that important attributes of reanalysis include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met).

The applicant stated that it dispositioned the Environmental Qualification (EQ) of Electric Components TLAA in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of thermal, radiation, and cyclical aging on the intended functions will be adequately managed by the Environmental Qualification (EQ) of Electric Components program for components associated with the TLAA for the period of extended operation.

4.4.2 Staff Evaluation

The staff reviewed the applicant's TLAA for the electric components and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), consistent with the review procedures in SRP-LR Section 4.4.2.1, which state that pursuant to 10 CFR 54.21(c)(1)(iii), an applicant must demonstrate the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

The EQ requirements established by 10 CFR Part 50, Appendix A, Criterion 4, and 10 CFR 50.49 specifically require each licensee to establish a program to qualify electrical equipment so that such equipment, in its end of life condition, can perform its intended function during accident conditions after experiencing the effects of inservice aging. Title 10 of the CFR

Part 49(e)(5) also requires replacement or refurbishment of components prior to the end of installed life condition (i.e., designated life) unless additional life is established through ongoing qualification. The 10 CFR 50.49 EQ program is considered an AMP for purposes of license renewal. Electric components in the applicant's EQ program with a qualification equal to or greater than the current operating term are considered a TLAA for license renewal. The Environmental Qualification (EQ) of Electric Components TLAA includes long-lived passive and active electrical and instrumentation and control (I&C) components that are important to safety and are located in a harsh environment. Harsh environments are those areas of the plant subject to the environmental effects of a LOCA, a HELB, or post-LOCA environment. EQ equipment comprises safety-related and nonsafety-related equipment, the failure of which could prevent satisfactory accomplishment of any safety-related function, and necessary operation of post-accident monitoring equipment.

As required by 10 CFR 54.21(c)(1), the applicant must provide a list of EQ equipment. The applicant shall demonstrate one of the following for each type of EQ equipment: (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

The staff reviewed LRA Sections 4.4 and B.3.1.3, plant basis documents, additional information provided to the staff, and interviewed plant personnel to verify whether the applicant provided adequate information to meet the requirement of 10 CFR 54.21(c)(1). For electrical equipment, BBS uses 10 CFR 54.21(c)(1)(iii) in its TLAA evaluation to demonstrate that EQ equipment aging mechanisms and effects will be adequately managed during the period of extended operation. Per the GALL Report, plant EQ programs that implement the requirements of 10 CFR 50.49 are considered acceptable AMPs under license renewal 10 CFR 54.21(c)(1)(iii). GALL Report AMP X.E1, "Environmental Qualification (EQ) of Electric Components," provides an acceptable means to meet the requirements of 10 CFR 54.21(c)(1)(iii). The staff reviewed the applicant's Environmental Qualification (EQ) of Electric Components program to determine whether 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 focused on how the EQ program manages the aging effects to meet the requirements pursuant to 10 CFR 50.49. The staff conducted an audit of the information provided in LRA Sections 4.4, A.4.4, B.3.1.3, and A.3.1.3 and the program basis documents. LRA Section 4.4 discusses the component reanalysis attributes, including analytical models, data collection and reduction methods, underlying assumptions, acceptance criteria and corrective actions. On the basis of its audit and as described in SER Section 3.0.3.1.20, the staff finds that the EQ program, which the applicant claimed to be consistent with GALL Report AMP X.E1, "Environment Qualification (EQ) of Electric Components," is consistent with the GALL Report; therefore, the staff concludes that the applicant's Environmental Qualification (EQ) of Electric Components TLAA will be managed consistent with 10 CFR 54.21(c)(1)(iii).

Additionally, the applicant's EQ program meets the acceptance criteria in SRP-LR Section 4.4.2.1 because the applicant's EQ program is capable of programmatically managing the qualified life of components within the scope of the program for license renewal. The continued implementation of the EQ program provides assurance that the aging effects will be managed and that components within the scope of the EQ program will continue to perform their intended functions for the period of extended operation.

4.4.3 UFSAR Supplement

LRA Section A.4.4 provides the UFSAR supplement summarizing the Environmental Qualification (EQ) of Electric Components TLAA. The staff reviewed LRA Section A.4.4 and found it consistent with the review procedures in SRP-LR Section 4.4.1.3, which state that the detailed information on the evaluation of TLAAs is contained in the renewal application. A summary description of the evaluation of TLAAs for the period of extended operation is contained in the applicant's UFSAR supplement.

Based on its review of the Environmental Qualification (EQ) of Electric Components UFSAR supplement, the staff finds LRA Section A.4.4 meets the acceptance criteria in SRP-LR Section 4.4.1.3. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the Environmental Qualification (EQ) of Electric Components TLAA for the period of extended operation, as required by 10 CFR 54.21(d).

4.4.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of thermal, radiation, and cyclical aging on the intended functions of the electric equipment will be adequately managed by the Environmental Qualification (EQ) of Electric Components AMP 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 Concrete Containment Tendon Prestress Analysis

4.5.1 Summary of Technical Information in the Application

LRA Section 4.5 describes the applicant's TLAA for its prestressed concrete containment shell structures, each of which is made up of a cylindrical wall, a shallow dome roof, and a flat foundation slab. The LRA states that the cylindrical portion of BBS's concrete containment is prestressed by a post-tensioning system consisting of 162 vertical and 201 horizontal or hoop ungrouted tendons. The dome has three groups of ungrouted tendons with a total number of 120 (119 for Braidwood Unit 1), oriented 120° to each other and anchored at the vertical face of the dome ring. The hoop tendons are anchored at three equally spaced buttresses 240° apart, bypassing the intermediate buttress. The base foundation slab is a conventional reinforced concrete. The LRA also states that the tendons are enclosed in galvanized steel conduits filled with a corrosion protection medium. Each tendon consists of 170 high strength steel wires, each 6.35 mm (1/4 in.) in diameter.

The LRA states that the containment tendon prestressing forces are time-dependent with losses occurring due to relaxation of the steel tendons and creep and shrinkage of the concrete, which were considered in the design of the plant. The LRA also states:

> The ASME Section XI, Subsection IWL (B.2.1.30) program performs periodic surveillances of individual tendon prestressing values. Predicted lower limit (PLL) force values are calculated for each tendon prior to the surveillances to estimate the magnitude of the tendon relaxation and concrete creep and shrinkage for the given surveillance year. The prestressing forces are measured and plotted, and trend lines are developed, to ensure the average tendon group prestressing values remain above the respective minimum required values

(MRVs) until the next scheduled surveillance, and potentially for the 40-year period. The predicted lower limit force values and regression analyses, utilizing actual measured tendon forces, are used to evaluate the acceptability of the containment structure to perform its intended function over the current 40-year life of the plant, and therefore, are TLAAs requiring evaluation for the period of extended operation.

4.5.1.1 Predicted Lower Limit (PLL)

The LRA states that the initial tendon prestressing force was calculated to accommodate losses for steel tendon relaxation and concrete creep and shrinkage so that the estimated final effective tendon prestressing force at the end of 40 years would be higher than the MRVs. The LRA also states that, as part of the ASME Section XI, Subsection IWL inspections, PLL force values are calculated consistent with the guidance in RG 1.35.1, "Determining Prestressing Forces for Inspection of Prestress Concrete Containments," for each individual tendon scheduled for examination. The LRA further states that the "actual measured values for each tendon are compared to their respective PLL values, with acceptance criteria consistent with ASME Section XI, Subsection IWL requirements."

4.5.1.2 Regression Analysis

The LRA states that a regression analysis is developed for each of the tendon groups (hoop, dome, and vertical) to determine the trend of prestressing values of individual tendons over time. The LRA also states that the regression analysis consists of a trend line utilizing actual individual tendon prestressing forces measured during successive ASME Section XI, Subsection IWL surveillances, consistent with NRC Information Notice (IN) 99-10, "Degradation of Prestressing Tendon Systems in Prestressed Concrete Containments," Attachment 3, "Comparison and Trending of Prestressing Forces." The LRA further states that trend lines are used to demonstrate that the prestressing forces will remain above the MRV until the next scheduled surveillance.

The LRA states that the regression analyses have been reanalyzed to extend the trend lines from 40 to 60 years by using individual tendon prestressing force values based on data incorporating the 20th and 25th year surveillances for each BBS unit. The LRA also states that the extended trend lines predict that the prestressing forces will remain above the MRVs through the period of extended operation. LRA Figures 4.5-1 through 4.5-12 contain the reanalyzed regression analyses for each tendon group. The LRA further states "[t]he Concrete Containment Tendon Prestress (B.3.1.2) program will monitor and manage the TLAAs and the associated loss of tendon prestressing forces during the period of extended operation."

The applicant dispositioned the TLAAs for the Concrete Containment Tendon Prestress Analysis in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of tendon prestress relaxation and the associated effects of loss of prestress forces on the concrete containment prestressing system will be adequately managed by the Concrete Containment Tendon Prestress Program for the period of extended operation.

4.5.2 Staff Evaluation

The staff reviewed the applicant's TLAA for the Concrete Containment Tendon Prestress Analysis and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), consistent with the review procedures in SRP-LR Section 4.5.3.1.3, which state that the reviewer verifies that the applicant has identified the appropriate program (i.e., GALL Report AMP X.S1, "Concrete Containment Tendon Prestress") as described and evaluated in the GALL Report. The SRP-LR also states that the staff is to verify that the applicant has stated that its program contains the same program elements that the staff evaluated and relied upon in approving the corresponding generic program in the GALL Report.

The staff noticed that the SRP-LR states that the evaluation process determines that the applicant's AMP includes plant-specific OE and other relevant OE that occurred at the applicant's plant as well as at other plants. The SRP-LR also states that the applicant should consider in its AMP applicable portions of the OE with prestressing systems described in IN 99-10. The staff further noticed that the GALL Report AMP X.S1 recommends additional evaluation of the applicant's OE, which includes lift off tendon force measurements, calculations, and documentation.

The staff reviewed the UFSAR Sections 3.8.1.1.1, "Description of the Containment," 3.8.1.4.8, "Effects of Losses of Prestress," 3.8.1.7.3.2, "Inservice Tendon Surveillance Program," and Section B.3, "Post-Tensioning Tendons." The staff also reviewed statements contained in the UFSAR regarding RG 1.35, "Inservice Inspection of Ungrouted Tendons in Prestress Concrete Containments," and RG 1.35.1, "Determining Prestressing Forces for Inspection of Prestress Concrete Containments," and confirmed that the applicant has in place a Concrete Containment Tendon Prestress Program. The staff also reviewed LRA Section B.3.1.2, "Concrete Containment Tendon Prestress," and noticed that the existing program will be enhanced for the period of extended operation. The staff's review and evaluation of LRA Section B.3.1.2, "Concrete Containment Tendon Prestress," AMP is documented in SER Section 3.0.3.2.25. During the onsite AMP audit, the staff also reviewed and evaluated the applicant's documentation regarding prestressed tendon regression analyses, surveillances, and inservice inspection (ISI) calculations, which are further discussed below.

4.5.2.1 Predicted Lower Limit (PLL)

The staff confirmed that the onsite available TLAA documentation for LRA Section 4.5, "Concrete Containment Tendon Prestress Analysis," contained calculated and actual initial seating forces (stresses), MRVs, periodic actual lift off forces, as well as predicted lift off forces (minimum design forces with upper and lower limits) for the randomly selected group tendons. The staff also confirmed that the applicant's ISI PLL calculations were developed from the loss of prestress model as discussed in RG 1.35.1. The staff further confirmed that there was an increase in the number of randomly selected tendons in the sampled groups consistent with ASME Code Section XI, Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Plants."

During its review and evaluation of LRA Section B.3.1.2, "Concrete Containment Tendon Prestress," the staff also confirmed that the applicant plans to enhance the AMP prior to the period of extended operation, so that "[f]or each surveillance interval, the predicted lower-limit, minimum required value, and trending lines will be developed for the period of extended operation as part of the regression analysis for each tendon group."

4.5.2.2 Regression Analysis

For the evaluation of the applicant's regression analysis, the staff reviewed LRA Figures 4.5-1 through 4.5-12, which contain the results of the analysis, (i.e., the trend lines, for the vertical, horizontal, and dome tendon lift off force data) and noticed that (1) some of the reported tendon groups exhibited greater control (common) tendon lift off forces in later years than those recorded at earlier periodic surveillances; (2) it was not clear whether the applicant followed the required frequency of tendon lift-off measurements of ASME Code Section XI, Subsection IWL, in the construction of the group trend lines; and (3) although LRA Section A.4.5, "Concrete Containment Tendon Prestress Analyses," (UFSAR supplement) states that trend lines, extended from 40 to 60 years, were calculated based on the most recent tendon surveillances for all three tendons groups, it was not clear to what extent this was followed. Therefore, by letter dated April 7, 2014, the staff issued RAI 4.5-1, requesting that the applicant state the cause for the recorded upward trending lift off force measurement shown in LRA Figures 4.5-1 through 4.5-12 and discuss if and how the higher values were considered and implemented when constructing the extended trend lines to 60 years of operation. In addition, the staff requested that the applicant discuss what were the selected years of measurements for the construction of the regression trend lines shown in LRA Figures 4.5-1 through 4.5-12.

In its response dated May 6, 2014, the applicant stated that "the cause of control tendons exhibiting greater lift-off forces in later years of periodic surveillances […] is consistent with factors associated with the decrease in the rate of lift-off force loss with respect to time and equipment calibration accuracy during containment tendon lift-off force measurements." The applicant referenced ASME Code Section XI, Subsection IWL-2522, "Tendon Force Elongation Measurements," which specifies that the accuracy of the equipment calibration during tendon lift-off forces examination must be within 1.5 percent of the tendon minimum ultimate strength. The applicant stated that allowable calibration variances of measuring equipment can sometimes exceed the actual amount of prestress loss predicted from one surveillance to the next for a specific tendon when there is an extended period (up to 10 years) between surveillances and the measuring equipment used has changed (e.g., different jacks and pressure gauges). The applicant further stated that, based on its review of individual control tendon data used to develop LRA Figures 4.5-1 through 4.5 12, "for all instances where control tendon reported lift-off force values were found greater in later surveillances than earlier surveillances, the difference in the reported values were within the IWL Code allowable variance associated with equipment calibration accuracy."

In regards to whether higher control tendon lift-off force data values were considered and implemented when constructing the regression trend line for each group's overall tendon prestress force losses as shown in LRA Figures 4.5-1 through 4.5-12, the applicant stated that "all control tendon lift off data values were considered and implemented," in the construction of the dome, hoop, and vertical tendon regression analysis trend lines. The applicant stated it reviewed the LRA Section 4.5 figures that reported values of increased control (common) tendons lift-off forces in later years than those recorded at earlier periodic surveillances and concluded that the increased lift-off force values had no significant effect on the affected groups' trend lines extended to 60 years of operation. The applicant also stated that the majority of the tendon lift off force losses occurred during the years 0.1 to 5 of operation (from all charts of data) and the influence of upward trending control tendon lift-off forces on tendon group trend lines becomes less significant as more data is included following future surveillances. The applicant further stated that, given the scatter in the data, none of the tendon group trend lines are dominated by any upward lift-off force measured values of the control tendons.

For the second part of RAI 4.5-1, regarding the actual surveillance years used in the construction of LRA Figures 4.5-1 through 4.5-12, the applicant stated that it included all of the individual tendon lift-off force data obtained, consistent with schedules for multi-unit sites prior to the 15th year of examination as articulated in Regulatory Position 1.5 of RG 1.35, and beginning with the 15th year examinations with those required by IWL-2421. Specifically, the applicant stated that for Byron Unit 1 and Braidwood Unit 1 the scheduled examinations were at 1, 5, 10, and 20 years, while for Byron Unit 2 and Braidwood Unit 2 they were at 1, 5, 15, and 25 years respectively, after the initial Structural Integrity Test (SIT). The applicant stated that for Braidwood additional vertical and horizontal tendon force measurements were documented for Unit 1 during the 15th year surveillance and for Unit 2 during the 3rd and 10th year surveillances. The applicant stated that these additional measurements were included in Braidwood trend line calculations for LRA Figures 4.5-7 through 4.5-10. The applicant also stated that the additional measurements at Braidwood were beyond those required by RG 1.35 and IWL-2421, and were associated with "augmented and followup examinations of tendons, (i.e., tendons affected by steam generator replacement related activities, tendons where free water inspection results did not meet acceptance criteria, tendons where the number of ineffective wires exceeded the original specified limit during construction, and tendons with excessive gaps in shim stacks)."

The staff reviewed the response to RAI 4.5-1 and noticed that the applicant attributed the increase in control tendon trend line force measurements over past successive surveillances to changes in equipment used and their calibration. The staff confirmed during the onsite audit that the applicant routinely performed equipment calibrations to ensure data consistency with measurements obtained during surveillances. Given the reported ultimate strength of the tendons provided in the applicant's response, the staff also confirmed that the recorded upward trending control tendon lift-off force measurement variance observed in LRA Section 4.5 figures is less than the ASME Code Section XI, Subsection IWL overall allowable calibration tolerance. The staff also found that, in the construction of the regression line, the influence of an upward trending control tendon lift-off force measurement is minimal because the measured value falls within the applicable group's closely clustered tendon lift-off force measurements of noncontrol tendons at each surveillance and its influence on the regression line becomes less significant over time as more data is accumulated during future surveillances. The staff also reviewed (1) a letter from NRC to Commonwealth Edison Company dated May 6, 1997 (ADAMS Accession No. ML020870515); (2) a Notice of Consideration dated December 12, 1997 (ADAMS Accession No. ML020870622); and (3) BBS TSs. Based on its review of these three documents, the staff confirmed that the applicant initially followed RG 1.35 and subsequently ASME Code Section XI, Subsection IWL, for scheduling surveillance years and that the frequency of these surveillances is consistent with the recommendations of RG 1.35, and ASME Code Section XI, Subsection IWL-2421. The staff also confirmed that LRA Figures 4.5-7 through 4.5-10 associated to Braidwood include additional lift-off force measurement data points obtained at examinations performed on 3rd, 10th, and 15th years to fulfill ASME Code Section XI, Subsection IWL repair/replacement activities, evaluations, and acceptance criteria.

The staff finds the applicant's response to RAI 4.5-1 acceptable because it clarified the cause for upward trending of tendon lift-off force measurements data recorded in LRA Figures 4.5-1 through 4.5-12 to be associated with the equipment used and its calibration and the reported values were within the ASME Code Section XI, Subsection IWL allowable variance. The staff also finds the applicant's response for the upward trending of control tendon lift-off force measurements in the construction of the trend lines acceptable because it considers their influence in the regression analysis when data is collected and plotted. The staff further finds the applicant's response regarding the years in which tendon prestress lift-off measurements were taken and accounted for in the construction of the trend lines also acceptable, because the applicant has been consistent with the applicable guidance in RG 1.35 and regulatory requirements of ASME Code Section XI, Subsection IWL. The staff's concerns described in RAI 4.5-1 are resolved.

4.5.2.3 Regression Analysis (Byron Unit 2 Only)

The staff also reviewed the onsite Byron Unit 2 regression analyses documentation consisting of two reports: (1) a report of the most recent ASME Section XI, Subsection IWL surveillance titled "Final Report for Exelon Byron Station U1 and U2 25th year Containment Building Tendon Surveillance" (IWL report) and (2) a document titled "Regression Analysis to Predict Post-Tensioning Forces for Byron Unit 2 Containment Tendons in Support of License Renewal" (license renewal analysis report). The staff compared these documents to LRA Figures 4.5-2, 4.5-4, and 4.5 6, which show that the first measurements of lift off forces for Byron Unit 2 occurred at year 1. In contrast, the staff noticed that the IWL report that contains the 60-year tendon lift off force predictions states that the evaluations started at year five. In addition, the IWL report and the license renewal analysis report appeared to differ in the number of reported tendon lift off force data points for the examined tendon groups at certain periodic surveillances. The staff, therefore, requested the applicant clarify whether there is a difference between the two reports. It was also not clear which of the two analyses was used to develop the regression analyses trend lines plotted in LRA Figures 4.5-2, 4.5-4, and 4.5-6. Therefore, by letter dated April 7, 2014, the staff issued RAI 4.5-2 requesting that the applicant clarify discrepancies in data, if any, between the IWL report and the license renewal analysis report. For any discrepancies that may exist, the staff requested the applicant provide an explanation for the differences and discuss which report was used to develop the LRA regression analyses trend lines shown in Figures 4.5-2, 4.5-4, and 4.5-6 for the period of extended operation.

In its response dated May 6, 2014, the applicant stated that in 2009, it performed the 25th year IWL examinations at Byron, including tendon lift-off force measurements for Unit 2. The results of these examinations and regression analyses for each tendon group were documented in the vendor-supplied "25th year IWL report." The applicant also stated that the additional license renewal analysis report extended the regression analyses' trend lines out to 60 years for Byron Unit 2. The applicant further stated that a review of the documents revealed differences in the content as well as in the presentation of the tendon lift-off force data; however, these differences were confirmed not to have an impact on the LRA.

The applicant stated that the difference in the surveillance year number (e.g., 1, 5, 10) between the two aforementioned documents is related to different starting dates in reporting lift off force measurements. The applicant also stated that the 25th year IWL report starts with the tendon tensioning during construction while the license renewal analysis report starts its first surveillance date approximately 4 years later (i.e., a year after the completion of the SIT consistent with RG 1.35.1, Section 4). Accordingly, the applicant stated that the 25th year IWL report documents the first surveillance as "Year 5" while the license renewal analysis report has it as "Year 1."

The applicant stated that with respect to the number of tendon lift-off force data points, a discrepancy of seven data points was identified between the two documents regarding the vertical tendon measurements in "years 1 and 5" surveillances after the SIT. The applicant also stated that this condition was entered into the CAP. The applicant further stated that it reviewed an updated 25th year surveillance graph that includes these additional lift-off values for the Unit 2 vertical tendon group and found no appreciable impact to the trend line at later years, especially for years 40 to 60, since the missing data was from "years 1 and 5" and were of approximately of the same force values as other data points in those surveillance years. Furthermore, the applicant stated that the 60-year regression analysis trend line developed for the LRA included these additional lift-off values and has been shown to remain above the MRV.

Regarding which of the two reports' data sets were used to develop LRA Figures 4.5-2, 4.5-4, and 4.5-6 and the resulting tendon prestress trend lines extending to 60-year of operation, the applicant stated that the data set contained in the license renewal analysis report was used. The applicant also stated the license renewal analysis report is more representative of the loss of prestress because it contains additional tendon lift-off force data points. The applicant also stated that use of the license renewal analysis report data set is appropriate because the surveillance year numbers are presented consistent with ASME Code Section XI, Subsection IWL-2400, which prescribes the examination frequency years relative to the completion of the SIT.

The staff reviewed the response to RAI 4.5-2 and noticed the clarification that year 1 in the license renewal analysis report corresponds to year 5 of the 25th year IWL report as the initial surveillance following the SIT. The staff finds that the surveillance frequencies after the completion of the SIT, as used in the license renewal analysis report and per RG 1.35.1, are the appropriate years to be used in the regression analyses and construction of the trend lines. The staff concluded that no further clarification or action is necessary for this apparent discrepancy. The staff also determined that the applicant also identified the variance in the two reports regarding the number of reported vertical tendons lift-off force measurements for the first two surveillances following completion of the SIT. The staff noticed that the applicant addressed the issue within the CAP and concluded that no further action was required since the missing data, once evaluated and included in the regression analysis, was "found to have no appreciable impact on the trend line" slope during the period of extended operation. Moreover, the staff noticed that the applicant also investigated the LRA figures of concern and concluded that the regression trend lines were a function of all available points past the completion of the SIT. The staff finds the applicant's response acceptable because (1) the applicant clarified the reasons for the discrepancies in the listed years of ISI surveillances and the number of lift-off data points between the 25th year IWL report and the license renewal analysis report, and confirmed that there was no impact on the analyses for license renewal; and (2) the applicant confirmed that the appropriate document was used for the construction of LRA Figures 4.5-2, 4.5-4, and 4.5-6. The staff's concerns described in RAI 4.5-2 are resolved.

Following the review and assessment of PLL and regression analyses (trend lines) methodologies for the evaluation of the TLAA components, above, the staff confirmed that these are consistent with the recommendations provided in GALL Report AMP X.S1. The staff also noticed that the applicant plans to use its enhanced Concrete Containment Tendon Prestress AMP to manage the loss of tendon prestressing forces during the period of extended operation; the program is evaluated in SER Section 3.0.3.2.25.

The staff finds that the applicant demonstrated pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of tendon prestress relaxation and the associated effects of loss of prestress forces on the containment structure prestressing system will be adequately managed for the period of extended operation.

Additionally, LRA Section 4.5 meets the acceptance criteria in SRP-LR Section 4.5.2.1.3, because the applicant has in place an AMP proposed to be enhanced prior to the period of extended operation so that it can adequately manage the effects of loss of tendon prestressing forces.

4.5.3 UFSAR Supplement

LRA Section A.4.5 provides the UFSAR supplement summarizing the concrete containment tendon prestress TLAA. The staff reviewed LRA Section A.4.5 consistent with the review procedures in SRP-LR Section 4.5.3.2, which state that the reviewer verifies that the applicant has provided an UFSAR supplement, that includes a summary description of the evaluation of the concrete containment tendon prestress TLAA.

Based on its review of the UFSAR supplement, the staff finds that LRA Section A.4.5 meets the acceptance criteria in SRP-LR Section 4.5.2.2, and is therefore acceptable. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the PLL and regression analyses (trend lines) of the TLAA associated with predictions of containment tendon prestress losses, as required by 10 CFR 54.21(d).

4.5.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of loss of tendon prestressing forces on the intended function of the concrete containment will be adequately managed by the Concrete Containment Tendon Prestress AMP 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 Containment Liner Plate, Metal Containments, and Penetrations Fatigue Analyses

LRA Section 4.6 states that the Byron and Braidwood prestressed concrete containment structures each include leak-tight liners (membranes) made from welded carbon steel plates, attached to the entire inside surface of the concrete structure. The LRA also states that each prestressed concrete containment structure and the portion of the carbon steel liner backed by concrete were designed to conform to the ASME Code Section III, Division 2 requirements to withstand design-basis accident pressures. The LRA further states that the containment structure design includes Class MC components (emergency personnel airlocks, equipment access hatches and integral personnel airlocks, and all associated penetrations and nozzles), which are designed in accordance with ASME Code Section III, Division 1, requirements.

LRA Section 4.6 provides the applicant's analyses of the following:

- containment liner plates fatigue
- containment airlocks and hatches fatigue
- containment electrical penetrations fatigue
- containment piping penetrations fatigue
- fuel transfer tube bellows fatigue
- recirculation sump guard piping bellows fatigue

4.6.1 Containment Liner Plates Fatigue

4.6.1.1 Summary of Technical Information in the Application

LRA Section 4.6.1 describes the applicant's TLAA for containment liner plates fatigue. The LRA states that the portion of the liner that is backed by concrete was designed in accordance with the 1973 Edition of ASME Code Section III, Division 2, Subarticles CC-2500, CC-4500, and CC-5500, and required that the liner be analyzed for the effects of cyclic loading to satisfy the requirements of the 1973 Edition of ASME Code Section III, Division 1, Subsection NE. The LRA also states that the original design analysis, based on 40-year design inputs, justified that the liner meets the six "exemption criteria" specified in ASME Code Section III, Subparagraph NE-3222.4(d) below, and that no fatigue analysis was required:

(1) atmospheric-to-operating pressure cycles
(2) normal operation pressure fluctuations
(3) temperature difference—startup and shutdown
(4) temperature difference—normal operation
(5) temperature difference—dissimilar materials
(6) mechanical loads

The LRA states:

> …a re-evaluation of the design inputs was performed relative to the six criteria, and it determined that the original inputs remain valid. The temperature differences have not changed because the design transients have not been redefined. The 60-year transient projections provided in Section 4.3.1 show that the transient limits will not be exceeded during the period of operation. Therefore, the numbers of temperature and pressure cycles considered in determining the components were exempt from fatigue analysis will not be exceeded. The Fatigue Monitoring (B.3.1.1) Program will be used to monitor the applicable cycles and ensure that the transient limits will not be exceeded during the period of extended operation.

The applicant dispositioned the TLAA for the containment liner plates in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of fatigue on the intended functions will be adequately managed by the Fatigue Monitoring Program for the period of extended operation.

4.6.1.2 Staff Evaluation

The staff reviewed the applicant's TLAA for the containment liner plates and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended function of the liner plates will be adequately managed by the Fatigue Monitoring Program. The staff reviewed the applicant's TLAA and its disposition consistent with the review procedures in SRP-LR Section 4.6.3.1.1.3, which state that the applicant's proposed AMP is reviewed to ensure that the effects of aging on the intended functions are adequately managed for the period of extended operation.

The staff also reviewed ASME Code Section III, Subparagraph NE-3222.4(d), to ensure that the design inputs considered in the original analysis, which met the conditions for components not requiring analysis for cyclic operation, would continue to meet the conditions for a fatigue waiver, based on the 60-year transient projections in LRA Section 4.3.1. The staff concluded that the applicant's Fatigue Monitoring Program, with enhancements, will monitor transient cycles to ensure that transient limits considered in the original design analysis are not exceeded during the period of extended operation. However, based on the staff's review of the 60-year cycle projections for transients in LRA Section 4.3.1 and UFSAR Section 3.9.1.1, "Design Transients," the staff did not have sufficient information to determine which transients were considered in the original design analysis. To verify that the transients considered in the ASME NE-3222.4(d) analyses will be monitored by the Fatigue Monitoring Program and that the applicable cycles are clearly identified so that they will not be exceeded during the period of extended operation, the staff issued RAI 4.6.1-1 by letter dated April 24, 2014. With respect to the six design inputs to the "exemption criteria" meeting the conditions of ASME Code Section III, Division 1, Subparagraph NE-3222.4(d), through RAI 4.6.1-1, the staff requested that the applicant (1) indicate which transients were considered in each of the TLAAs described in LRA Sections 4.6.1, 4.6.2, and 4.6.3; and (2) provide the number of transient cycles that were assumed in the original design analyses, as well as the number of additional cycles anticipated for LRA Sections 4.6.1, 4.6.2, and 4.6.3 during the period of extended operation.

In its response dated May 23, 2014, the applicant stated:

> [t]he verification that the conditions are met is dependent on the specified number and magnitude of pressure and temperature transients and mechanical load cycles. These assumed inputs are then used to assess the potential effect on the component and consideration of other limitations to determine if the fatigue waiver can be applied. The original design specifications provided the expected number and magnitude of pressure and temperature transients and mechanical load cycles to be considered for the fatigue waiver in the original design analysis…

In its response, the applicant also provided a table relating each of the "exemption conditions" considered in the fatigue waiver to the corresponding LRA transients. The applicant clarified that, for condition 2, the normal operation pressure fluctuations condition is not considered in the LRA Section 4.3.1 tables and is not monitored by the Fatigue Monitoring Program because the pressure fluctuations during normal operation are insignificant. The applicant stated that the projected maximum number of cycles for a Type A leak test are 15, and hence are insignificant when compared to the 2,500 cycles assumed in the design analysis of the liner.

Additionally, the applicant stated that "for conditions 4 and 5, it was conservatively interpreted that the temperature differences could be the result of not only heatups and cooldowns, but also upset conditions. Therefore […], a number of transients were associated with these two conditions." The applicant further stated that "there were no additional cycles, above those used as inputs for the exemption, anticipated for component analyses in LRA Section 4.6.1, 4.6.2, and 4.6.3 during the period of extended operation" and that "[t]he current license basis (CLB) cycle limits are equal to or bounded by the fatigue exemption cycles assumed in the original design analysis."

The staff reviewed the applicant's fatigue "exemption" assessment documented in Table 1, "Byron and Braidwood Units 1 and 2 Fatigue Exemption Inputs Assessment," of the response to RAI 4.6.1-1 and noticed that, for conditions (1) and (3)–(5) of ASME Code Section III, Division 1, Subparagraph NE-3222.4(d), considered in the fatigue waiver, the 60-year cycle projections from LRA Tables 4.3.1-1 and 4.3.1-4 based on the corresponding LRA transients are significantly less than the numbers assumed in the original design analysis, and are bound by the CLB cycle limits provided in LRA Tables 4.3.1-1 and 4.3.1-4. The staff also noticed that for condition (2), the normal operating pressure fluctuations are considered insignificant and, for condition (6), there were no significant mechanical load fluctuations considered on the liner in the original design specification.

The staff finds the applicant's response acceptable because the information in the RAI response provided sufficient information to verify that the design transient cycles considered in the fatigue waiver analysis for the containment liner plates are included in LRA Section 4.3 and will be monitored by the Fatigue Monitoring Program, to ensure that the bounding transient limits will not be exceeded during the period of extended operation. The staff's concern described in RAI 4.6.1-1 is resolved.

Further, the staff reviewed LRA Section B.3.1.1 and noticed that the applicant's Fatigue Monitoring Program, with enhancements, monitors and tracks critical thermal and pressure transients and that:

> [t]he fatigue cycle monitoring data was used to project the numbers of cycles that will occur during 60 years. These projections show that the current 40-year allowable cycle limits will not be exceeded in 60 years. Therefore, the current 40-year cycle limits will be maintained for the period of extended operation. The Fatigue Monitoring aging management program will be enhanced to monitor additional plant transients that are significant contributors to cumulative fatigue damage.

The staff's review and evaluation of the applicant's enhanced Fatigue Monitoring Program is documented in SER Section 3.0.3.2.24.

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

Additionally, the Fatigue Monitoring Program meets the acceptance criteria in SRP-LR Section 4.6.2.1.1.3 because the Fatigue Monitoring Program will monitor transient cycles to ensure that, if a transient limit is approached, corrective action will be taken prior to exceeding a transient limit, ensuring that the "exempt conditions" in ASME Code Section III, Subparagraph NE-3222.4(d), continue to be met for components not requiring fatigue analysis for cyclic operation during the period of extended operation.

4.6.1.3 UFSAR Supplement

LRA Section A.4.6.1 provides the UFSAR supplement summarizing the TLAA for the containment liner plates fatigue. The staff reviewed LRA Section A.4.6.1 consistent with the review procedures in SRP-LR Section 4.6.3.2, which state that the information to be included in the UFSAR supplement should include a summary description of the evaluation of the containment liner plates fatigue TLAA.

Based on its review of the UFSAR supplement, the staff finds LRA Section A.4.6.1 meets the acceptance criteria in SRP-LR Section 4.6.2.2 and is therefore acceptable. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the fatigue monitoring of containment liner plates, as required by 10 CFR 54.21(d).

4.6.1.4 Conclusion

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

4.6.2 Containment Airlocks and Hatches Fatigue

4.6.2.1 Summary of Technical Information in the Application

LRA Section 4.6.2 describes the applicant's TLAA for containment airlocks and hatches fatigue. The LRA states that Byron and Braidwood emergency personnel airlock, personnel airlock with equipment hatch, and all penetrations and nozzles associated with personnel airlocks were designed as Class MC components in accordance with the 1971 Edition of ASME Code Section III, Subsection NE, through the Summer 1973 Addenda. The LRA also states that the original design analyses for containment Class MC components, based on 40-year design inputs, justified that these components meet the six "exemption criteria" specified in ASME Code Section III, Subparagraph NE-3222.4(d), below and that no fatigue analysis was required.

(1) atmospheric-to-operating pressure cycles
(2) normal operation pressure fluctuations
(3) temperature difference—startup and shutdown
(4) temperature difference—normal operation
(5) temperature difference—dissimilar materials
(6) mechanical loads

The LRA also states that:

> …a re-evaluation of the design inputs was performed relative to these criteria from ASME Section III, Subparagraph NE-3222.4(d). The results of the re-evaluation determined that the original inputs remain valid. The temperature differences have not changed because the design transients have not been redefined. The 60-year transient projections provided in Section 4.3.1 show that the transient limits will not be exceeded during the period of extended operation. Therefore, the number of temperature and pressure cycles considered in determining the components, which were exempt from fatigue analysis, will not be exceeded. The Fatigue Monitoring (B.3.1.1) Program will be used to monitor the applicable cycles and ensure that the transient limits will not be exceeded during the period of extended operation.

The applicant dispositioned the TLAA for the containment airlocks and hatches in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of fatigue on the intended functions will be adequately managed by the Fatigue Monitoring Program for the period of extended operation.

4.6.2.2 Staff Evaluation

The staff reviewed the applicant's TLAA for the containment airlocks and hatches and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended function of the containment airlocks and hatches will be adequately managed by the Fatigue Monitoring Program. The staff reviewed the applicant's TLAA and disposition consistent with the review procedures in SRP-LR Section 4.6.3.1.1.3, which state that the applicant's proposed AMP is reviewed to ensure that the effects of aging on the intended functions are adequately managed for the period of extended operation.

The staff also reviewed ASME Code Section III, Subparagraph NE-3222.4(d), to ensure that the design inputs considered in the original analysis, which met the conditions for components not requiring analysis for cyclic operation, would continue to meet the conditions for a fatigue waiver, based on the 60-year transient projections in LRA Section 4.3.1. The staff noticed that the applicant's Fatigue Monitoring Program, with enhancements, will monitor transient cycles to ensure that transient limits considered in the original design analyses are not exceeded during the period of extended operation.

The staff requested additional information through RAI 4.6.1-1 regarding the applicable transients and cycle limits considered in the original fatigue waiver analyses for the containment class MC components (i.e., the emergency personnel airlock, the personnel airlock with equipment hatch, and all the penetrations and nozzles associated with personnel airlocks). The staff's discussion and evaluation of the applicant's response to RAI 4.6.1-1 is documented in SER Section 4.6.1.2. The applicant provided a fatigue exemption input assessment table demonstrating that the total 60-year cycle projections from LRA Tables 4.3.1-1 and 4.3.1-4 for the LRA transients corresponding to the ASME Code Section III, Division 1, Subparagraph NE-3222.4(d), conditions would not exceed the number of cycles assumed in the original design analysis for the evaluated containment class MC components and are bound by the CLB cycle limits provided in LRA Tables 4.3.1-1 and 4.3.1-4. The information in the RAI response provided sufficient information for the staff to verify that the design transient cycles considered in the fatigue waiver analysis for the containment airlocks and hatches are included in LRA Section 4.3 and will be monitored by the Fatigue Monitoring Program to ensure that the bounding transient limits will not be exceeded during the period of extended operation.

The staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the intended functions of the containment airlocks and hatches will be adequately managed for the period of extended operation.

Additionally, it meets the acceptance criteria in SRP-LR Section 4.6.2.1.1.3 because the Fatigue Monitoring Program will monitor transient cycles to ensure that, if a transient limit is approached, corrective action will be taken prior to exceeding a transient limit, ensuring that the "exempt conditions" in ASME Code Section III, Subparagraph NE-3222.4(d), continue to be met for components not requiring fatigue analysis for cyclic operation during the period of extended operation.

4.6.2.3 UFSAR Supplement

LRA Section A.4.6.2 provides the UFSAR supplement summarizing the TLAA for containment airlocks and hatches fatigue. The staff reviewed LRA Section A.4.6.1 consistent with the review procedures in SRP-LR Section 4.6.3.2, which state that the information to be included in the UFSAR supplement should include a summary description of the evaluation of the TLAA.

Based on its review of the UFSAR supplement, the staff finds LRA Section A.4.6.2 meets the acceptance criteria in SRP-LR Section 4.6.2.2, and is therefore acceptable. Additionally, the staff concludes that the applicant has provided an adequate summary description of its actions to address the fatigue monitoring of containment airlocks and hatches, as required by 10 CFR 54.21(d).

4.6.2.4 Conclusion

On the basis of its review, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the intended functions of the containment airlocks and hatches will be adequately managed by the Fatigue Monitoring Program for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.6.3 Containment Electrical Penetrations Fatigue

4.6.3.1 Summary of Technical Information in the Application

LRA Section 4.6.3 describes the applicant's TLAA for containment electrical penetrations fatigue. The LRA states that Byron and Braidwood prestressed concrete containment structures include electrical penetrations that were designed in accordance with the ASME Code Section III, Division 1, Subsection NE, 1977 Edition through Summer 1978 Addenda requirements. The LRA also states that the original design analysis for containment electrical penetrations, based on 40-year design inputs, justified that these components meet the six "exemption criteria" specified in ASME Code Section III, Subparagraph NE-3222.4(d), below and that no fatigue analysis was required.

(1) atmospheric-to-operating pressure cycles
(2) normal operation pressure fluctuations
(3) temperature difference—startup and shutdown
(4) temperature difference—normal operation
(5) temperature difference—dissimilar materials
(6) mechanical loads

The LRA further states that:

> …a re-evaluation of the design inputs was performed relative to the six criteria, and it determined that the original inputs remain valid. The temperature differences have not changed because the design transients have not been redefined. The 60-year transient projections provided in Section 4.3.1 show that the transient limits will not be exceeded during the period of extended operation. Therefore, the number of temperature and pressure cycles considered in determining the components, which were exempt from fatigue analysis, will not be exceeded. The Fatigue Monitoring (B.3.1.1) Program will be used to monitor the applicable cycles and ensure that the transient limits will not be exceeded during the period of extended operation.

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

4.6.3.2 Staff Evaluation

The staff reviewed the applicant’s TLAA for the containment electrical penetrations and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended function of the containment electrical penetrations will be adequately managed by the Fatigue Monitoring Program. The staff reviewed the applicant’s TLAA and disposition consistent with the review procedures in SRP-LR Section 4.6.3.1.1.3, which state that the applicant’s proposed AMP is reviewed to ensure that the effects of aging on the intended functions are adequately managed for the period of extended operation.

The staff also reviewed ASME Code Section III, Subparagraph NE-3222.4(d), to ensure that the design inputs considered in the original analysis, which met the conditions for components not requiring analysis for cyclic operation, would continue to meet the conditions for a fatigue waiver, based on the 60-year transient projections in LRA Section 4.3.1. The staff noticed that the applicant’s Fatigue Monitoring Program, with enhancements, will monitor transient cycles to ensure that design limits considered in the original design analyses are not exceeded during the period of extended operation.

The staff requested additional information through RAI 4.6.1-1 regarding the applicable transients and cycle limits considered in the original fatigue waiver analyses for the containment electrical penetrations. The staff’s discussion and evaluation of the applicant’s response to RAI 4.6.1-1 is documented in SER Section 4.6.1.2. The applicant provided a fatigue exemption input assessment table demonstrating that the total 60-year cycle projections from LRA Tables 4.3.1-1 and 4.3.1-4 for the LRA transients corresponding to the ASME Code Section III, Division 1, Subparagraph NE-3222.4(d), conditions would not exceed the number of cycles assumed in the original design analysis for the evaluated containment class MC components and are bound by the CLB cycle limits provided in LRA Tables 4.3.1-1 and 4.3.1-4. The information in the RAI response provided the staff sufficient information to verify that the design transient cycles considered in the fatigue waiver analysis for the containment electrical penetrations are included in LRA Section 4.3 and will be monitored by the Fatigue Monitoring Program to ensure that the bounding transient limits will not be exceeded during the period of extended operation.

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

Additionally, LRA Section 4.6.3 meets the acceptance criteria in SRP-LR Section 4.6.2.1.1.3 because the Fatigue Monitoring Program will monitor transient cycles to ensure that, if a transient limit is approached, corrective action will be taken prior to exceeding a transient limit, ensuring that the “exemption conditions” in ASME Code Section III, Subparagraph NE-3222.4(d), continue to be met for components not requiring fatigue analysis for cyclic operation during the period of extended operation.

4.6.3.3 UFSAR Supplement

LRA Section A.4.6.3 provides the UFSAR supplement summarizing the TLAA for containment electrical penetrations fatigue. The staff reviewed LRA Section A.4.6.1 consistent with the review procedures in SRP-LR Section 4.6.3.2, which state that the information to be included in the UFSAR supplement should include a summary description of the evaluation of the TLAA.

Based on its review of the UFSAR supplement, the staff finds LRA Section A.4.6.3 meets the acceptance criteria in SRP-LR Section 4.6.2.2 and is therefore acceptable. Additionally, the staff concludes that the applicant has provided an adequate summary description of its actions to address the fatigue monitoring of containment electrical penetrations, as required by 10 CFR 54.21(d).

4.6.3.4 Conclusion

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

4.6.4 Containment Piping Penetrations Fatigue

4.6.4.1 Summary of Technical Information in the Application

LRA Section 4.6.4 describes the applicant's TLAA for the containment piping penetrations fatigue. The LRA states the Byron and Braidwood containment structure penetrations conform to the requirements of ASME Code Section III, Subsection NE, 1971 Edition through the Summer 1973 Addenda. The LRA also states that the instrument and process pipe penetrations required fatigue evaluation of each containment structure penetration, in accordance with ASME Code Section III, Subparagraph NB-3222.4(e) or NE-3222.4(e).

The LRA further states that:

> [t]he design specifications for the containment piping penetrations define the transients applicable to penetration stress analysis. These same transients are listed in Section 4.3.1, along with the 60-year projections…. The Fatigue Monitoring (B.3.1.1) Program is used to monitor the applicable transients and ensure that transient limits are not exceeded. The program also ensures that, if a transient limit is approached, corrective action is taken to reanalyze components prior to exceeding a transient limit.

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

4.6.4.2 Staff Evaluation

The staff reviewed the applicant's TLAA for the containment piping penetrations and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended function of the containment piping penetrations will be adequately managed by the Fatigue Monitoring Program. The staff reviewed the applicant's TLAA and disposition consistent with the review procedures in SRP-LR Section 4.6.3.1.1.3, which state that the applicant's proposed

AMP is reviewed to ensure that the effects of aging on the intended functions are adequately managed for the period of extended operation.

The staff also reviewed ASME Code Section III, Subparagraphs NB-3222.4(e) and NE-3222.4(e), by which a fatigue evaluation was performed for the containment piping penetrations. In its review of UFSAR Section 3.8.2, "Steel Containment and ASME Class MC Components," the staff noticed that penetration sleeves are described as being designed as Class MC components in accordance with Subsection NE of the ASME B&PV Code, Section III, which are directly exposed to worst case loading conditions of the process piping. The head fittings are designed in accordance with Subsection NB, NC, or ND of the ASME Code, Section III, as applicable. The staff also noticed that the fatigue loading conditions include thermal and pressure load transients as well as those of OBE and other mechanical loads. However, it was not clear which of the transients listed in LRA Section 4.3.1 were considered in the analyses of penetration sleeves. Therefore, by letter dated April 24, 2014, the staff issued RAI 4.6.4-1, requesting that the applicant identify the applicable transients, including the cycle limit for each transient, assumed in the fatigue analysis for the containment piping penetrations.

In its response dated May 23, 2014, the applicant provided two tables. One table correlated the containment piping penetration analyses assumed transients and limits to the LRA RCS transients and limits contained in LRA Tables 4.3.1-1 and 4.3.1-4. The other table correlated the containment piping penetration analyses assumed transients and limits to the LRA auxiliary system transients and limits contained in LRA Tables 4.3.1-2 and 4.3.1-5. The applicant stated that the tables:

> …document the pressure and temperature transients and the number of cycles that were assumed in the fatigue analyses for the containment piping penetrations, and also document the corresponding transient number and CLB cycle limits for LRA Tables 4.3.1-1, 4.3.1-2, 4.3.1-4, and 4.3.1-5. These fatigue analyses also applied loads associated with operating basis earthquakes (OBE) in combination with loads created by the pressure and temperature transients.

The applicant also stated that the transients and cycle limits documented in the tables, provided in response to RAI 4.6.4-1:

> …reflect what was assumed in the original fatigue analysis for the containment piping penetrations, are currently monitored by the Fatigue Monitoring aging management program implementing procedures, and will continue to be monitored during the period of extended operation.

The applicant further stated the that main steam and feedwater containment piping penetration analyses were inconsistent with the Westinghouse transient design specifications, and sufficient margin in the original analysis existed such that a corrective action is being taken to revise the analyses to increase the number of cycles to the CLB limits of LRA Tables 4.3.1-1 and 4.3.1-4.

The staff reviewed the applicant's Table 1, "Correlation of Containment Piping Penetration Analyses Assumed Transients and Limits to LRA RCS Transients and Limits Contained in LRA Tables 4.3.1-1 and 4.3.1-4." The LRA states that the applicant's Fatigue Monitoring Program will monitor the transients to the CLB cycle limits; however, a reanalysis needs to be made for the main steam and feedwater containment piping penetrations to "increase the number of cycles to the original CLB cycle limits." The staff confirmed that UFSAR Section 3.9.1.1, "Design Transients," does specify 13,200 cycles for the "Unit Loading and Unloading at

5 percent of Full Power per Minute” transient, which would apply to the associated penetration piping. The staff concluded that the corrective action being taken to revise the main steam and feedwater piping penetration analyses to be consistent with the governing Westinghouse transient design specifications, which the applicant determined had sufficient margin to accommodate the change, is appropriate. In addition the staff concluded that the applicant revised the LRA to indicate that faulted and emergency condition transients and limits are monitored for the RCS through its Fatigue Monitoring Program.

The staff also reviewed Table 2, “Correlation of Containment Piping Penetration Analyses Assumed Transients and Limits to LRA Auxiliary System Transients and Limits Contained in LRA Tables 4.3.1-1 and 4.3.1-4,” and noticed that the limits considered in the analyses were either greater than the CLB cycle limits or no longer relevant (e.g., because of changes in the chemistry sampling strategy leading to a reduction in the originally considered number of thermal cycles); and therefore, the staff finds these limits acceptable.

The staff finds the applicant’s response acceptable because the information in the RAI response provided the staff sufficient information to verify that the design transient cycles, considered in the fatigue evaluation for each containment piping penetration, are included in LRA Section 4.3 and will be monitored by the Fatigue Monitoring Program, to ensure that the bounding transient limits will not be exceeded during the period of extended operation. The staff’s concern described in RAI 4.6.4-1 is resolved.

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

Additionally, LRA Section 4.6.4 meets the acceptance criteria in SRP-LR Section 4.6.2.1.1.3 because the Fatigue Monitoring Program will monitor transient cycles to ensure that, if a transient limit is approached, corrective action is taken prior to exceeding a transient limit.

4.6.4.3 UFSAR Supplement

LRA Section A.4.6.4 provides the UFSAR supplement summarizing the TLAA for containment piping penetrations fatigue. The staff reviewed LRA Section A.4.6.4 consistent with the review procedures in SRP-LR Section 4.6.3.2, which state that the information to be included in the UFSAR supplement should include a summary description of the evaluation of the TLAA.

Based on its review of the UFSAR supplement, the staff finds it meets the acceptance criteria in SRP-LR Section 4.6.2.2 and is therefore acceptable. Additionally, the staff concludes that the applicant provided an adequate summary description of its actions to address fatigue of containment piping penetrations, 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 provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the intended functions of the containment piping penetrations will be adequately managed by the Fatigue Monitoring Program for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.6.5 Fuel Transfer Tube Bellows Fatigue

4.6.5.1 Summary of Technical Information in the Application

LRA Section 4.6.5 describes the applicant's TLAA for fuel transfer tube bellows fatigue. The LRA states that the fuel transfer tubes pass through the containment structure, connecting the refueling cavity to the fuel transfer canal inside the fuel handling building and that the guard pipe assemblies for the fuel transfer tubes also function as penetration sleeves. The LRA also states that there are three expansion bellows in the penetration sleeve around each fuel transfer tube, and three sets of expansion joints (bellows) for the 24-in.-diameter penetration sleeves that comprise the guard pipes for the fuel transfer tubes. The LRA states that the design specification considered 100 load cycles, based on ASME Code Section III, Subsection NE and qualified per Subparagraph NE-3365.2(e)(2), 1974 Edition through Summer 1974, "along with the maximum displacements intended to envelope all postulated design-basis conditions, including 1 Safe Shutdown Earthquake (SSE) transient event, for fatigue consideration."

The LRA also states that:

> [t]hese bellows are affected by seismic transients (1 SSE event) that would cause deflection of the bellows. These transients are listed in Section 4.3.1 and have 60-year projections that are less than the numbers of cycles which form the basis for the design requirement of 100 design load cycles and used for the qualification of bellows. Therefore, the qualification of the bellows is acceptable for the period of extended operation. The Fatigue Monitoring (B.3.1.1) Program monitors SSE transient events.

The applicant dispositioned the TLAA for the fuel transfer tube bellows in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of fatigue on the intended functions will be adequately managed by the Fatigue Monitoring Program for the period of extended operation.

4.6.5.2 Staff Evaluation

The staff reviewed the applicant's TLAA for the fuel transfer tube bellows and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended function of the fuel transfer tube bellows will be adequately managed by the Fatigue Monitoring Program. The staff reviewed the applicant's TLAA and disposition consistent with the review procedures in SRP-LR Section 4.6.3.1.1.3, which state that the applicant's proposed AMP is reviewed to ensure that the effects of aging on the intended functions are adequately managed for the period of extended operation.

The staff also reviewed ASME Code Section III, Subparagraph NE-3365.2(e)(2), by which the fuel transfer tube bellows were qualified. The staff noticed that LRA Section 4.6.5 states that these bellows are limited to 100 design load cycles and the maximum displacements, including one SSE event, which would cause deflection of the bellows, and that the transient cycle projections listed in Section 4.3.1 are fewer than the number of cycles which forms the basis for the design requirement. However, the transients listed in LRA Section 4.3.1 do not include the SSE event. It was not clear that the Fatigue Monitoring Program is monitoring the SSE event to support the applicant's claim that the effects of aging on the fuel transfer tube bellows will be adequately managed by the Fatigue Monitoring Program. Therefore, by letter dated April 24, 2014, the staff issued RAI 4.6.5-1, requesting that the applicant identify what transients along with maximum displacements, other than those associated with SSE, have been considered in the fuel transfer tube bellows fatigue analysis, provide the number of cycles assumed in the design, and clarify why the SSE transients are not listed in the tables in LRA Section 4.3.1.

In its response dated May 23, 2014, the applicant stated that:

> [t]he SSE and LOCA events are the only transients considered in the analysis for the fuel transfer bellows. As described in LRA Section 4.6.5, TLAA Description, the 100 design load cycles envelope the postulated design-basis conditions. Therefore, there are no other transients associated with the analysis. The maximum displacements specified were 1.75 inches axially and 0.5 inches laterally. The Fatigue Monitoring Program monitors and tracks SSE and LOCA events. If a seismic event occurs, the program reviews the duration, magnitude, and number of cycles of the event….

The applicant revised LRA Section A.3.1.1 and B.3.1.1 to clarify that the Fatigue Monitoring Program manages the cumulative fatigue damage of “other components” and monitors design-basis events and counts them in the appropriate design transient category. The applicant also clarified that SSE and LOCA are one-time, faulted events monitored by the current Fatigue Monitoring Program, not normal inputs to fatigue monitoring. Therefore, the SSE and LOCA events are not listed in LRA Tables 4.3.1-1 through 4.3.1-6.

The staff finds the applicant’s response acceptable because the Fatigue Monitoring Program, with enhancements, monitors seismic events including duration, magnitude, and number of cycles, and LOCA events to ensure that, if the maximum displacements due to an SSE or LOCA event are exceeded, corrective action is taken. The staff’s evaluation of the Fatigue Monitoring Program and acceptability of the enhancements is documented in SER Section 3.0.3.2.24. The staff’s concern described in RAI 4.6.5-1 is resolved.

The staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the intended functions of the fuel transfer tube bellows will be adequately managed for the period of extended operation.

Additionally, LRA Section 4.6.5 meets the acceptance criteria in SRP-LR Section 4.6.2.1.1.3 because the Fatigue Monitoring Program will monitor transient cycles to ensure that, if a transient limit is approached, corrective action is taken prior to exceeding a transient limit.

4.6.5.3 UFSAR Supplement

LRA Section A.4.6.5 provides the UFSAR supplement summarizing the TLAA for fuel transfer tube bellows fatigue. The staff reviewed LRA Section A.4.6.5 consistent with the review procedures in SRP-LR Section 4.6.3.2, which state that the information to be included in the UFSAR supplement should include a summary description of the evaluation of the TLAA.

Based on its review of the UFSAR supplement, as amended by letter dated May 23, 2014, the staff finds that LRA Section A.4.6.5 meets the acceptance criteria in SRP-LR Section 4.6.2.2 and is therefore acceptable. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address fatigue of fuel transfer tube bellows, 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 provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the intended functions of the fuel transfer tube bellows will be adequately managed by the Fatigue Monitoring Program for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.6.6 Recirculation Sump Guard Piping Bellows Fatigue

4.6.6.1 Summary of Technical Information in the Application

LRA Section 4.6.6 describes the applicant’s TLAA for recirculation sump guard piping bellows fatigue. The LRA states that the guard pipe, which extends from the recirculation sump to the sump suction valve protection chamber inside the auxiliary building, is composed of a 28-in. diameter sleeve that includes two sets of expansion joints (bellows). The LRA also states that the bellows were analyzed for fatigue in accordance with Expansion Joint Manufacturers Association, 4th Edition, 1975, and substantiated per ASME Code Section III, Subparagraph NE-3365.2(e)(1), 1977 Edition through Summer of 1977 Addenda, which required 7,000 design cycles.

The LRA further states that:

> [t]hese bellows are affected by plant heatup and cooldown transients and other transients associated with accident conditions that would fill the containment recirculation sump, including OBE transients. These transients are listed in Section 4.3.1 and have 60-year projections that are less than the numbers of cycles analyzed for the bellows. Therefore, the design analysis of the bellows is acceptable for the period of extended operation. The BBS Fatigue Monitoring (B.3.1.1) Program monitors plant heatup and cooldown transients, as well as upset, emergency, and faulted conditions, including OBE and SSE events.

The applicant dispositioned the TLAA for the recirculation sump guard piping bellows in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of fatigue on the intended functions will be adequately managed by the Fatigue Monitoring Program for the period of extended operation.

4.6.6.2 Staff Evaluation

The staff reviewed the applicant’s TLAA for the recirculation sump guard piping bellows and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended function of the recirculation sump guard piping bellows will be adequately managed by the Fatigue Monitoring Program. The staff reviewed the applicant’s TLAA and disposition consistent with the review procedures in SRP-LR Section 4.6.3.1.1.3, which state that the applicant’s proposed AMP is reviewed to ensure that the effects of aging on the intended functions are adequately managed for the period of extended operation.

In its review, the staff noticed that LRA Section 4.6.6 states that the recirculation sump guard piping bellows are affected by plant heatup and cooldown transients and other transients associated with accident conditions that would fill the containment recirculation sump. However, it was not clear which of the transients listed in LRA Section 4.3.1 were considered to contribute towards the 7,000-cycle limit for which the bellows were designed; therefore, by letter dated April 24, 2014, the staff issued RAI 4.6.6-1, requesting that the applicant identify the applicable transients, including the cycle limit for each transient that was assumed in the design fatigue analysis for the recirculation sump guard piping bellows.

In its response dated May 23, 2014, the applicant revised LRA Sections 4.6.6 and A.4.6.6 to clarify the description of the configuration of the recirculation sump guard piping bellows. The RAI response clarified that the guard pipe actually includes three (3) sets of expansion joints (bellows), with one bellows sealing between the containment sump piping and the guard pipe located inside the containment structure. A second bellows seals between the guard pipe and the sump suction valve protection chamber inside the auxiliary building structure. The third set of bellows inside the auxiliary building which seal between the sump suction valve protection chamber and the recirculation sump effluent piping. The applicant stated that the cycle limit and the transients, which were assumed in the design fatigue analysis for the recirculation sump guard piping bellows, are different for the three sets of bellows, which the applicant described separately for the containment and auxiliary building as follows:

> <u>Containment Structure</u>
>
> The analysis for the recirculation sump guard piping bellows inside containment was performed in accordance with ASME Section III, Subparagraph NE-3365.2(e)(1), 1977 Edition through Summer of 1977 Addenda to determine the appropriate numbers of fatigue test cycles required to support the design requirement of 10 cycles. The applicable transient associated with the analysis performed for the recirculation sump guard piping bellows is only that associated with the LOCA event. The Current License Basis (CLB) LOCA event limit for this transient is one (1)…The Fatigue Monitoring Program currently includes tracking and monitoring of LOCA events….
>
> <u>Auxiliary Building Structure</u>
>
> …The 7,000 cycles are inputs to the analysis to qualify the bellows and are the cycle limits. The 7,000 cycles input to the analysis are similar to those evaluated in LRA Section 4.3.3, since the process pipe which is attached to the bellows is ASME Section III, Class 2. Both of these bellows assemblies in the auxiliary building do not perform a containment pressure boundary function. Similar to the disposition in LRA Section 4.3.3, for Class 2 fatigue analysis, the Fatigue Monitoring Program will monitor the transients provided in Tables 4.3.1-3 (Byron) and 4.3.1-6 (Braidwood), which are the transients that have the potential to impart differential movement intended for the bellows assemblies. OBE and SSE seismic events are other transients associated with the cyclic differential movement associated with seismic events. Monitoring and tracking of OBE and SSE seismic events is currently performed by the Fatigue Monitoring Program….

In its response, the applicant also revised LRA Section A.3.1.1 and B.3.1.1 to confirm monitoring of LOCA and SSE (seismic) events by the Fatigue Monitoring Program.

In its review of the response to RAI 4.6.6-1, the staff found the applicable transient associated with the analysis performed for the recirculation sump guard piping bellows in the containment structure is a LOCA event, and concluded that the applicant has revised the LRA to clearly indicate that the Fatigue Monitoring Program monitors the event. The staff concluded that the bellows in the auxiliary building structure addressed in this TLAA experience the same number of cycles as process pipe to which they are attached and have been dispositioned in SER Section 4.3.3. The staff also noticed that the transients provided in LRA Tables 4.3.1-3 (Byron) and 4.3.1-6 (Braidwood), which have the potential to impart differential movement intended for the bellows assemblies, show that total projected cycles at each site is fewer than the 7,000 allowed.

The staff finds the applicant's response acceptable because the applicant has clarified that the cycle limits and transients were different for the three sets of bellows, and the applicant has provided sufficient information for the staff to verify that the Fatigue Monitoring Program will adequately monitor and manage aging on the intended functions of the bellows. The staff's evaluation of the Fatigue Monitoring Program and acceptability of the enhancements is documented in SER Section 3.0.3.2.24. The staff's concern described in RAI 4.6.6-1 is resolved.

The staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the intended functions of the recirculation sump guard piping bellows will be adequately managed for the period of extended operation.

Additionally, LRA Section 4.6.6 meets the acceptance criteria in SRP-LR Section 4.6.2.1.1.3 because the Fatigue Monitoring Program will monitor transient cycles to ensure that, if a transient limit is approached, corrective action is taken prior to exceeding a transient limit.

4.6.6.3 UFSAR Supplement

LRA Section A.4.6.6 as amended by letter dated May 23, 2014, provides the UFSAR supplement summarizing the TLAA for recirculation sump guard piping bellows fatigue. The staff reviewed LRA Section A.4.6.6 consistent with the review procedures in SRP-LR Section 4.6.3.2, which state that the information to be included in the UFSAR supplement should include a summary description of the evaluation of the TLAA.

Based on its review of the amended UFSAR supplement, the staff finds LRA Section A.4.6.6 meets the acceptance criteria in SRP-LR Section 4.6.2.2 and is therefore acceptable. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address fatigue of recirculation sump guard piping bellows, as required by 10 CFR 54.21(d).

4.6.6.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the intended functions of the recirculation sump guard piping bellows will be adequately managed by the Fatigue Monitoring Program for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7 <u>Other Plant-Specific Time-Limited Aging Analyses</u>

4.7.1 Leak-Before-Break

Criterion 4, "Environmental and Dynamic Effects Design Bases," of 10 CFR Part 50, Appendix A, (General Design Criterion (GDC)-4) requires SSCs important to safety to be appropriately protected against dynamic effects associated with postulated pipe ruptures. However, protection against such dynamic effects is not required when analyses reviewed and approved by the staff demonstrate that the probability of rupture is extremely low under conditions consistent with the design basis for the piping. An approved LBB analysis permits the removal of protective hardware, such as pipe whip restraints and jet impingement barriers, the redesign of pipe connected components, their supports, and their internals, and other related changes, as described in SRP Section 3.6.3, "Leak-Before-Break Evaluation Procedures" (NUREG-0800).

4.7.1.1 Summary of Technical Information in the Application

LRA Section 4.7.1 describes the applicant's TLAA evaluations for the LBB analyses. The LRA identifies three TLAAs based on the existing LBB analyses: (1) a TLAA for the reactor coolant primary loop piping, (2) a TLAA for the safety injection accumulator piping and reactor coolant bypass piping, and (3) a TLAA for the safety injection accumulator piping cold leg nozzles.

The LRA states that the existing LBB analysis for the reactor coolant primary loop piping is documented in a report by Westinghouse, WCAP-14559, Revision 1, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Byron and Braidwood Units 1 and 2 Nuclear Power Plants," dated April 1996. The LRA also indicates that the staff accepted this analysis by letter dated October 25, 1996. The LRA states that the applicant updated this existing LBB analysis to account for the period of extended operation. Inputs for the updated analysis took into consideration the Mechanical Stress Improvement Process (MSIP®) applied to the reactor vessel inlet and outlet nozzles. In addition, the LRA states that the reactor coolant primary loop piping includes cast austenitic stainless steel (CASS) materials. Since CASS materials are subject to the effects of loss of fracture toughness due to thermal aging embrittlement over time, the applicant accounted for these effects in the updated LBB analysis by using the methodology in NUREG/CR-4513, Revision 1, "Estimation of Fracture Toughness of Cast Stainless Steels during Thermal Aging in LWR Systems," dated May 1994. For the updated LBB analysis, the applicant postulated through-wall flaws at governing critical locations that would cause a leak rate 10 times the capability of the plant leakage detection systems. According to the LRA, the results of the updated LBB analysis demonstrate that such flaws have large margins against instability. Additionally, the LRA states that the applicant analyzed fatigue crack growth based on the design transients and cycles listed in LRA Section 4.3.1.

The LRA states that the existing LBB analysis for the safety injection accumulator piping and reactor coolant bypass piping is documented in a report by Sargent & Lundy, SL-4518, "Leak-Before-Break Evaluation for Stainless Steel Piping Byron and Braidwood Nuclear Power Stations Units 1 and 2," dated May 12, 1989. The LRA also indicates that the staff accepted this analysis by letter dated April 19, 1991. The LRA states that the applicant determined, from a review of the current calculation packages, that the loads used in the existing LBB analysis will still govern in the period of extended operation. The SL-4518 report also documents the existing LBB analysis for the safety injection accumulator piping cold leg nozzles. The LRA states that these nozzles are made of CASS materials, which are subject to the effects of loss of fracture toughness due to thermal aging embrittlement. The LRA states that the applicant updated the existing LBB analysis to account for these effects through the period of extended operation. For the updated LBB analysis, the LRA states that the applicant postulated a through-wall flaw size that would cause a leak rate 10 times the capability of the plant leakage detection systems. According to the LRA, the results of the updated LBB analysis demonstrate that such flaws have large margins against instability.

The applicant dispositioned the TLAA for the safety injection accumulator piping and reactor coolant bypass piping LBB analysis in accordance with 10 CFR 54.21(c)(1)(i) to demonstrate that the analysis remains valid for the period of extended operation. The applicant dispositioned the TLAAs for the reactor coolant primary loop piping LBB analysis and the safety injection accumulator piping cold leg nozzles LBB analysis in accordance with 10 CFR 54.21(c)(1)(ii) to demonstrate that the analyses have been projected to the end of the period of extended operation.

4.7.1.2 Staff Evaluation

The staff reviewed the applicant's TLAA evaluations for the LBB analyses and the corresponding dispositions pursuant to 10 CFR 54.21(c)(1), consistent with the review procedures in SRP-LR Section 4.7.3.1. The staff discusses its evaluation of each TLAA in the sections that follow.

<u>Reactor Coolant Primary Loop Piping</u>. The staff reviewed the applicant's TLAA for the reactor coolant primary loop piping LBB analysis and the corresponding disposition of 10 CFR 54.21(c)(1)(ii), consistent with the review procedures in SRP-LR Section 4.7.3.1.2. These procedures state that the staff is to review the results of the applicant's revised analysis to verify that the evaluation period has been extended, such that the analysis is valid for the period of extended operation. The SRP-LR also states that the applicant may extend the period of evaluation by recalculating the analysis using a methodology that is in effect when the LRA is filed. In addition, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 3.6.3, "Leak-Before-Break Evaluation Procedures," dated March 2007, provides acceptance criteria and procedures for the staff's review of LBB analyses used to demonstrate compliance with GDC-4.

WCAP-14559, Revision 1 documents the existing LBB analysis for the reactor coolant primary loop piping. LRA Section 4.7.1 states that the applicant updated this analysis and the results demonstrate that there are large margins against the instability of postulated flaws. The staff reviewed LRA Section 4.7.1; however, it could not identify the methodology that the applicant used for the updated LBB analysis. Therefore, the staff could not determine whether the applicant updated the analysis using a methodology that is currently in effect, as recommended by SRP-LR Section 4.7.3.1.2. The staff also determined that the LRA did not contain a sufficient level of technical information for the staff to confirm that the updated LBB analysis demonstrates compliance with GDC-4 and the criteria in SRP Section 3.6.3. By letter dated April 24, 2014, the staff issued RAI 4.7.1-1, requesting that the applicant provide for staff review and approval the full update to the LBB analysis or the rationale for not providing it.

The applicant responded to RAI 4.7.1-1 by letter dated May 23, 2014. The response summarizes the applicant's update to the LBB analysis for the reactor coolant primary loop piping. The applicant stated that the primary difference between the existing LBB analysis and the updated LBB analysis is the method used to calculate the fracture toughness properties for the CASS materials. Specifically, the existing LBB analysis used fracture toughness properties after 40 years of aging, as calculated according to a report by Westinghouse, WCAP-10931, Revision 1, “Toughness Criteria for Thermally Aged Cast Stainless Steel,” dated July 1986, whereas the updated LBB analysis used fracture toughness properties after 60 years of aging, as calculated according to NUREG/CR-4513. The applicant stated that use of the new methodology changed the values of the elastic-plastic J-integral fracture mechanics inputs, but clarified that the results still satisfy the flaw stability criteria for the calculated fracture toughness and tearing modulus. In addition, the applicant stated that the existing LBB analysis was developed in accordance with the original (1987) version of SRP Section 3.6.3, whereas the updated LBB analysis was developed in accordance with the current (2007) version. The applicant explained that the primary difference between the two versions is that the current version includes a criterion on determining material susceptibility to primary water stress corrosion cracking (PWSCC), which is considered to be an active degradation mechanism for Alloy 82/182 welds in PWRs. The applicant stated that BBS have Alloy 82/182 welds in the reactor vessel hot and cold leg safe ends, but it has applied MSIP at these locations to mitigate the effects of PWSCC. The applicant’s response to RAI 4.7.1-1 also reports the results of the updated LBB analysis. The applicant stated that, after the application of MSIP for the Alloy 82/182 welds, the leakage flaw size is greater than 10 in. and the critical flaw sizes are all greater than 38 in. The applicant also stated that, per the existing LBB analysis, the two critical locations in the reactor coolant primary loop piping are at: (1) the hot leg elbow fitting into the steam generator, and (2) the cold leg discharge piping of the RCP. The applicant stated that the updated LBB analysis does not result in any change from these locations or any changes to the leakage flaw sizes and critical flaw sizes at these locations.

The staff reviewed the methodology for the applicant’s updated LBB analysis, as described in the response to RAI 4.7.1-1. The response indicates that the applicant prepared the updated LBB analysis using the existing methodology with new inputs to account for thermal aging embrittlement of the CASS materials and for the application of MSIP on the reactor vessel hot and cold leg safe ends. The applicant stated that it used NUREG/CR-4513 to determine the fracture toughness properties of the CASS materials at the end of the period of extended operation. Because NUREG/CR-4513 is the latest NRC-endorsed methodology specifically for this purpose, the staff finds it to be an acceptable methodology for generating the new inputs for the updated LBB analysis.

The staff also reviewed the applicant’s actions to address PWSCC because SRP Section 3.6.3 states that the LBB analysis should demonstrate that it is not a potential source of pipe rupture. The applicant’s response to RAI 4.7.1-1 states that it applied MSIP to the Alloy 82/182 welds in the reactor vessel hot and cold leg safe ends as a means to mitigate PWSCC. The staff noticed that RIS 2010-07, “Regulatory Requirements for Application of Weld Overlays and Other Mitigation Techniques in Piping Systems Approved for Leak-Before-Break,” dated June 8, 2010, states that MSIP is considered to be an adequate means to mitigate PWSCC and thus satisfy the related criterion in SRP Section 3.6.3. Based on the applicant’s statement that MSIP has been applied to the reactor vessel hot and cold leg safe ends, the staff finds that the applicant has taken adequate measures to mitigate PWSCC; therefore, the applicant may continue to apply the LBB analysis to the reactor coolant primary loop piping. RIS 2010-07 also indicates that MSIP may be applied without NRC authorization, since it does not affect any design or inspection requirements. Based on this information, the staff determined that the application of MSIP does not result in a change to the existing LBB methodology. In summary, the response to RAI 4.7.1-1 demonstrates that the applicant used its existing, NRC-approved methodology for the updated LBB analysis; therefore, the staff determined that the applicant does not need to provide the full update for review and approval. In addition, since the applicant used an existing methodology that is currently in effect, the staff determined that the applicant’s approach to updating the TLAA is consistent with the guidance in SRP-LR Section 4.7.3.1.2. The staff's concern described in RAI 4.7.1-1 is resolved.

The staff also reviewed the results of the applicant's updated LBB analysis. For dispositions made pursuant to 10 CFR 54.21(c)(1)(ii), SRP-LR Section 4.7.3.1.2 states that the applicant may recalculate the existing analysis using a 60-year period to show that the acceptance criteria continue to be satisfied for the period of extended operation. The response to RAI 4.7.1-1 states that the applicant's updated LBB analysis does not result in any changes to the leakage flaw sizes or critical flaw sizes at the critical locations, and these locations are unchanged from the existing LBB analysis. The staff reviewed and approved the results of the existing LBB analysis as documented by letter dated October 25, 1996. Since there are no changes to these results, the staff finds them to be acceptable for the reasons stated in the original approval letter. From the response to RAI 4.7.1-1, the safety margin on flaw size is 3.8 for the Alloy 82/182 welds to which MSIP was applied. The staff reviewed this result against the criteria in SRP Section 3.6.3, which state that the results of the deterministic fracture mechanics analysis should demonstrate that there is a safety margin of at least 2 between the critical flaw size and the leakage flaw size. The staff finds the result reported by the applicant acceptable because it is greater and thus more conservative than the safety margin recommended in SRP Section 3.6.3.

In addition, the LRA describes the results of the fatigue crack growth analysis. The staff confirmed that the transients used for this analysis are included in LRA Section 4.3 and that none of the transients are projected to exceed the number of cycles identified in the existing LBB analysis. On the basis of these projections, the staff determined that the applicant has adequately addressed fatigue crack growth for the updated LBB analysis.

The staff finds the applicant demonstrated pursuant to 10 CFR 54.21(c)(1)(ii), that the LBB analysis for the reactor coolant primary loop piping has been projected to the end of the period of extended operation. This demonstration also meets the acceptance criteria in SRP-LR Section 4.7.2.1.

Safety Injection Accumulator Piping and Reactor Coolant Bypass Piping. The staff reviewed the applicant's TLAA for the safety injection accumulator piping and reactor coolant bypass piping LBB analysis and the corresponding disposition of 10 CFR 54.21(c)(1)(i), consistent with the review procedures in SRP-LR Section 4.7.3.1.1. These procedures state that the applicant should show that the existing analysis bounds the period of extended operation, so that no reanalysis is necessary.

The SL-4518 report documents the existing LBB analysis for the safety injection accumulator piping and reactor coolant bypass piping. LRA Section 4.7.1 states that the loads used as inputs for this analysis will still govern in the period of extended operation; therefore, the LRA concludes that the existing LBB analysis remains valid. The staff reviewed LRA Section 4.7.1 and found that it does not identify any specific time-dependent loads or other parameters from the existing LBB analysis, nor does it demonstrate how these time-dependent parameters remain valid for the period of extended operation. By letter dated April 24, 2014, the staff issued RAI 4.7.1-2, requesting that the applicant identify all of the time-dependent parameters used in the existing LBB analysis and justify how each parameter will remain valid for the period of extended operation.

The applicant responded to RAI 4.7.1-2 by letter dated May 23, 2014. In its response, the applicant stated that the methodology used in the existing LBB analysis is consistent with the modified limit load approach described in SRP Section 3.6.3. The applicant also stated that the parameters for this analysis include the piping material properties, dimensions, and loadings. However, the applicant stated that none of these parameters are time-dependent; therefore, the applicant concluded that the existing LBB analysis does not meet the definition of a TLAA. The applicant also amended LRA Sections 4.7.1 and A.4.7.1 in its response, to remove all discussion of the existing LBB analysis for the safety injection accumulator piping and reactor coolant bypass piping as a TLAA.

The staff reviewed the applicant's response to RAI 4.7.1-2. Per the definition in 10 CFR 54.3, TLAAs are those calculations and analyses that, in part, involve time-limited assumptions defined by the current operating term. The staff reviewed the existing LBB analysis for the safety injection accumulator piping and reactor coolant bypass piping and confirmed that it uses a limit load analysis as indicated in the applicant's RAI response. In a limit load analysis, the stability of a postulated flaw is assessed in terms of the applied loads, and there is no assessment of flaw growth over time. In addition, the safety injection accumulator piping and reactor coolant bypass piping are not made of CASS, so the piping is not susceptible to the time-based effects of loss of fracture toughness due to thermal aging. For these reasons, the staff determined that the existing LBB analysis does not involve any time-limited assumptions defined by the current operating term. Therefore, the staff determined that the analysis does not meet the definition of a TLAA in 10 CFR 54.3. Accordingly, the existing LBB analysis for the safety injection accumulator piping and reactor coolant bypass piping does not need to be evaluated against the requirements of 10 CFR 54.21(c)(1). The staff's concern described in RAI 4.7.1-2 is resolved.

Safety Injection Accumulator Piping Cold Leg Nozzles. The staff reviewed the applicant's TLAA for the safety injection accumulator piping cold leg nozzles LBB analysis and the corresponding disposition of 10 CFR 54.21(c)(1)(ii), consistent with the review procedures in SRP-LR Section 4.7.3.1.2. These procedures state that the staff is to review the results of the applicant's revised analysis to verify that the evaluation period has been extended, such that the analysis is valid for the period of extended operation. The SRP-LR also states that the applicant may extend the period of evaluation by recalculating the analysis using a methodology that is in effect when the LRA is filed. In addition, SRP Section 3.6.3 provides acceptance criteria and procedures for the staff's review of LBB analyses used to demonstrate compliance with GDC-4.

The SL-4518 report documents the existing LBB analysis for the safety injection accumulator piping cold leg nozzles. The report indicates that these nozzles are made of CASS, which is susceptible to the effects of loss of fracture toughness due to thermal aging embrittlement. LRA Section 4.7.1 states that the applicant determined new fracture toughness properties for the CASS materials based on their aging through the period of extended operation, and then used the new material properties to generate the updated LBB analysis. The staff reviewed LRA Section 4.7.1; however, it could not identify the methodology that the applicant used for the updated LBB analysis. Therefore, the staff could not determine whether the applicant updated the analysis using a methodology that is currently in effect, as recommended by SRP-LR Section 4.7.3.1.2. The staff also determined that the LRA did not contain a sufficient level of technical detail for the staff to confirm that the updated LBB analysis complies with the requirements of GDC-4 and the criteria in SRP Section 3.6.3. By letter dated April 24, 2014, the staff issued RAI 4.7.1-3, requesting that the applicant provide for staff review and approval the full update to the LBB analysis or the rationale for not providing it. The staff also requested that the applicant identify and provide justification for the methodology used to determine the fracture toughness properties for the CASS materials.

The applicant responded to RAI 4.7.1-3 by letter dated May 23, 2014. The response summarizes the update to LBB analysis for the safety injection accumulator piping cold leg nozzles. The applicant stated that the existing LBB analysis was developed in accordance with the original (1987) version of SRP Section 3.6.3, whereas the updated LBB analysis was developed in accordance with the current (2007) version. The applicant stated that the primary difference between the two versions is that the current version includes a criterion on determining material susceptibility to PWSCC, which is considered to be an active degradation mechanism for Alloy 82/182 welds in PWRs. However, the applicant stated that the safety injection accumulator piping cold leg nozzles are not made of this material, so the criterion is not applicable. The applicant also stated that the primary difference between the existing LBB analysis and the updated LBB analysis is the calculation of the fracture toughness properties for the CASS nozzles. The applicant explained that the existing LBB analysis used fracture toughness properties based on generic material data in the staff's Piping Fracture Mechanics Data Base. However, for the updated LBB analysis, the applicant determined the fracture toughness properties using the plant-specific CMTRs and the methodology in NUREG/CR-4513. As to the results of the updated LBB analysis, the applicant stated that the ratio of the critical flaw size to the leakage flaw size is 3.3. In addition, the applicant stated that the updated fracture toughness properties demonstrate significant margin against the instability of flaws.

The staff reviewed the methodology for the applicant's updated LBB analysis, as described in the response to RAI 4.7.1-3. This response indicates that the applicant prepared the updated LBB analysis using the existing methodology with new inputs to account for thermal aging embrittlement of the CASS safety injection accumulator piping cold leg nozzles. The applicant stated that it used NUREG/CR-4513 to determine the fracture toughness properties of the CASS materials at the end of the period of extended operation. Because NUREG/CR-4513 is the latest NRC-endorsed methodology specifically for this purpose, the staff finds it to be an acceptable methodology for generating the new inputs for the updated LBB analysis. In addition, the staff determined that the safety injection accumulator piping cold leg nozzles are not susceptible to PWSCC because they are not made of Alloy 82/182. Therefore, the related criterion from SRP Section 3.6.3 is satisfied. In summary, the response to RAI 4.7.1-3 demonstrates that the applicant used its existing, NRC-approved methodology for the updated LBB analysis; therefore, the staff determined that the applicant does not need to provide the full update for review and approval. In addition, since the applicant used an existing methodology that is currently in effect, the staff determined that the applicant's approach to updating the TLAA is consistent with the guidance in SRP-LR Section 4.7.3.1.2. The staff's concern described in RAI 4.7.1-3 is resolved.

The staff also reviewed the results of the applicant's updated LBB analysis. For dispositions made pursuant to 10 CFR 54.21(c)(1)(ii), SRP-LR Section 4.7.3.1.2 states that the applicant may recalculate the existing analysis using a 60-year period to show that the acceptance criteria continue to be satisfied for the period of extended operation. The response to RAI 4.7.1-3 reports a safety margin of 3.3 based on the updated LBB analysis. The staff reviewed this result against the criteria in SRP Section 3.6.3, which state that the results of the deterministic fracture mechanics analysis should demonstrate that there is a safety margin of at least 2 between the critical flaw size and the leakage flaw size. The staff finds the result reported by the applicant acceptable because it is greater and thus more conservative than the safety margin recommended in SRP Section 3.6.3.

The staff finds the applicant demonstrated pursuant to 10 CFR 54.21(c)(1)(ii), that the LBB analysis for the safety injection accumulator piping cold leg nozzles has been projected to the end of the period of extended operation. This demonstration also meets the acceptance criteria in SRP-LR Section 4.7.2.1.

4.7.1.3 UFSAR Supplement

LRA Section A.4.7.1 provides the UFSAR supplement summarizing the TLAA evaluations for the LBB analyses. The staff reviewed LRA Section A.4.7.1 consistent with the review procedures in SRP-LR Section 4.7.3.2, which state that the applicant is to provide a summary description for its evaluation of each TLAA. The SRP-LR also states that the summary description should contain information on the disposition of the TLAA for the period of extended operation and be appropriate such that later changes can be controlled by 10 CFR 50.59. By letter dated May 23, 2014, the applicant amended LRA Section A.4.7.1 to reflect its response to RAI 4.7.1-2. Accordingly, the applicant deleted the portion of the summary description on the analysis for the safety injection accumulator piping and reactor coolant bypass piping because this analysis is not a TLAA. Based on its review of the UFSAR supplement, as amended by letter dated May 23, 2014, the staff finds that LRA Section A.4.7.1 meets the acceptance criteria in SRP-LR Section 4.7.2.2 and is therefore acceptable. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the updates to the LBB analyses for the reactor coolant primary loop piping and the safety injection accumulator piping cold leg nozzles, 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 acceptable demonstrations, pursuant to 10 CFR 54.21(c)(1)(ii), that the LBB analyses for the reactor coolant primary loop piping and the safety injection accumulator piping cold leg nozzles 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 evaluations, as required by 10 CFR 54.21(d).

4.7.2 Crane Load Cycle Limits

4.7.2.1 Summary of Technical Information in the Application

LRA Section 4.7.2 describes the applicant’s TLAAs for load cycle limits of cranes designed in accordance with the Crane Manufacturers Association of America Specification 70 (CMAA-70). The LRA states that, based upon frequency of operation and expected size of load relative to their maximum load capacity, these cranes are designated a given service classification with an expected maximum number of design cycles over their life, which correlates to a number of cycles on structural members. The service class is used to define the allowable stress range limits for structural members and fasteners, considering the cyclic operation over the life of the crane. Therefore, since the maximum number of design load cycles over the 40-year life of the crane provides the basis for acceptability of the design for cyclic operation, the load cycles experienced over the period of extended operation need to be evaluated. LRA Table 4.7.2-1 summarizes the evaluation of cyclic operation for each crane.

The applicant dispositioned the TLAAs for the containment polar crane, fuel handling building crane, manipulator crane, SFP bridge crane, and turbine building crane in accordance with 10 CFR 54.21(c)(1)(i) to demonstrate that the analyses remain valid for the period of extended operation.

4.7.2.2 Staff Evaluation

The staff reviewed the applicant's TLAAs for the containment polar crane, fuel handling building crane, manipulator crane, SFP bridge crane, and turbine building crane and the corresponding disposition that the projected number of load cycles is less than the design load cycles used in the cyclic analyses, consistent with the review procedures in SRP-LR Section 4.7.3.1.1, which state that the existing analyses should be shown to be bounding even during the period of extended operation. The SRP-LR also states that the applicant should describe the TLAA with respect to the objectives of the analysis, assumptions used in the analysis, conditions, acceptance criteria, relevant aging effects, and intended functions. The applicant should show that conditions and assumptions used in the analysis already address the relevant aging effects for the period of extended operation, and acceptance criteria are maintained to provide reasonable assurance that the intended functions are maintained for the period of extended operation.

Containment Polar Crane. In its review of the cyclic analysis for the containment polar cranes, the staff confirmed that UFSAR Section 9.1.4.2.2 and Table 9.1-7, "Crane Design," states that the cranes were designed for CMAA-70, 1975 Revision, Class A service (100,000 load cycles), based on a design load of 230 tons on the main hook, and 40 tons on the auxiliary hook. They were also designed to withstand the containment pressure test, and OBE and SSE stresses. The applicant estimated 44 load cycles of 5 tons or greater, for each crane per refueling outage, over the 60-year plant life, and considered an additional 100 load cycles for crane use during original construction and Unit 1 steam generator replacement at both BBS. The staff noticed that the applicant assumed the load cycles performed by all four containment polar cranes were similar, with the steam generator replacement being the only significant difference, and therefore, has considered the analysis for BBS, Unit 1, containment polar cranes to be bounding since steam generators have not been replaced at either Byron or Braidwood, Unit 2.

The estimated number of load cycles for each containment polar crane over the course of the 60-year life of the plant, based on 40 refueling outages, is 1,860, or approximately 1,900 load cycles. This is less than the number of load cycles (100,000) considered when determining the allowable stress for which they were designed and, therefore, is acceptable.

Fuel Handling Building Crane. In its review of the cyclic analysis for the fuel handling building crane, the staff reviewed LRA Section 4.7.2 and UFSAR Section 9.1.4.2.2 and Table 9.1-7, "Crane Design," and noticed that the single fuel handling crane at BBS was designed for CMAA-70, 1975 Revision, Class A Service (100,000 load cycles). The staff also noticed that the fuel handling building overhead crane is equipped with a 125-ton main hoist and 15-ton auxiliary hoist, and is used for lifts associated with RCP motor replacement and refurbishment, dry cask storage campaigns, and outage equipment staging. The applicant estimated 1,200 load cycles for activities other than dry cask storage campaigns and that a normal dry cask storage campaign involves an equivalent of six casks every 18 months and 25 load cycles per cask, thereby resulting in a projected 6,000 loads over the course of 60 years. Considering that BBS began dry cask storage campaigns in 2010 and 2011, respectively, the staff agrees that the applicant's estimated number of load cycles is conservative.

The estimated number of load cycles for the fuel handling building crane over the course of the 60-year life of the plant, based on the normal dry cask storage schedule described above and activities other than the dry cask storage campaigns, is 7,200 load cycles. This is less than the number of load cycles (100,000) considered when determining the allowable stress for which it was designed and, therefore, is acceptable.

Manipulator Crane. In its review of the cyclic analysis for the manipulator crane, the staff reviewed LRA Section 4.7.2, UFSAR Section 9.1.4.2.2, and Table 9.1-7, “Crane Design,” and UFSAR Section 9.1.4.2.2, “Component Description,” which states that the crane, referred to as either manipulator crane or refueling machine, was designed in accordance with CMAA-70, for Class C service (500,000 load cycles). CMAA Table 3.3.3.1.3-1 indicates that the allowable stress range for a crane designed for Class C service is between 100,000 and 500,000 load cycles. The applicant estimated 400 load cycles each refueling outage, which includes offload and reload of 193 assemblies, two pull tests at greater than 150 percent of the weight of the assembly, and two source assembly moves.

The estimated number of load cycles for each manipulator crane (refueling machine) over the course of the 60-year life of the plant, based on 40 refueling outages, is approximately 16,000 load cycles. This is less than the number of load cycles (100,000–500,000) considered when determining the allowable stress for which they were designed and, therefore, is acceptable.

Spent Fuel Pool Bridge Crane. In its review of the cyclic analysis for the SFP bridge crane, the staff reviewed LRA Section 4.7.2, UFSAR Section 9.1.4.2.2 and Table 9.1-7, “Crane Design,” and UFSAR Section 9.1.4.2.2, “Component Description,” and noticed that the SFP bridge crane was designed in accordance with CMAA-70 for Class A service (100,000 load cycles). Because the SFP is common between units at BBS, there is a single SFP bridge crane at each station that handles fuel moves associated with both units. For the single SFP bridge crane at each station, the applicant estimated a total of 41,500 load cycles, about 1032, every 18 months to support an assumed 40 operating cycles for each unit, 9,000 load cycles associated with dry cask storage campaigns, and 26,400 load cycles associated with miscellaneous activities, which include SFP rerack projects, fuel assembly moves for checker-boarding, gamma heating, and insert moves, and B.5.b moves, as described in LRA Section 4.7.2.

The total estimated number of load cycles for the SFP bridge crane over the 60-year life of the plant, based on 40 refueling outages for each unit, is approximately 76,900 load cycles. This is less than the number of load cycles (100,000) considered when determining the allowable stress for which it was designed and, therefore, is acceptable.

Turbine Building Crane. In its review of the cyclic analysis for the turbine building crane, the staff reviewed LRA Section 4.7.2 and UFSAR Section 9.1.4.2.2 and Table 9.1-7, “Crane Design,” and noticed that the turbine building crane was designed in accordance with CMAA-70, 1975 Revision, for Class A service (100,000 load cycles) and includes a 150 ton capacity main hoist and 25 ton capacity auxiliary hoist. The staff also noticed that there is one crane for each unit at BBS, and that the applicant assumed the load cycles performed by each crane are similar. The applicant estimated 4,800 cycles over the 60-year life based on review of crane operation, and has added an additional 200 initial load cycles for use during construction and 100 load cycles for future equipment and system upgrades, for a total of 5,100 load cycles.

The estimated number of load cycles for the turbine building crane over the course of the 60-year life of the plant, is 5,100 load cycles. This is less than the number of load cycles (100,000) considered when determining the allowable stress for which it was designed and, therefore, is acceptable.

In summary and based on its review, the staff finds the applicant demonstrated pursuant to 10 CFR 54.21(c)(1)(i), that the analyses for the containment polar crane, fuel handling building crane, manipulator crane, SFP bridge crane, and turbine building crane remain valid for the period of extended operation.

Additionally, LRA Section 4.7.2 meets the acceptance criteria in SRP-LR Section 4.7.2.1 because the applicant demonstrated that the projected load cycles over 60 years of operation will not exceed the design load cycles used in the cyclic analyses.

4.7.2.3 UFSAR Supplement

LRA Section A.4.7.2 provides the UFSAR supplement summarizing the crane load cycle limits. The staff reviewed LRA Section A.4.7.2 consistent with the review procedures in SRP-LR Section 4.7.3.2, which state that the applicant should provide information to be included in the UFSAR supplement that includes a summary description of the evaluation of each TLAA. SRP-LR Section 4.7.3.2 also states that each summary description should be reviewed to verify that it is appropriate, such that later changes can be controlled by 10 CFR 50.59 and that the description should contain information that the TLAAs have been dispositioned for the period of extended operation.

Based on its review of the UFSAR supplement, the staff finds LRA Section A.4.7.2 meets the acceptance criteria in SRP-LR Section 4.7.2.2, and is therefore acceptable. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address crane load cycle limits, 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 provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses for the containment polar crane, fuel handling building crane, manipulator crane, SFP bridge crane, and turbine building crane 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).

4.7.3 Mechanical Environmental Qualification

4.7.3.1 Summary of Technical Information in the Application

LRA Section 4.7.3 describes the applicant's TLAA for its mechanical environmental qualification (MEQ) program. The applicant stated that qualified lives and replacement intervals are established for safety-related mechanical components located in harsh environments based on aging concerns. Replacement intervals are identified either on the basis of aging performed during an IEEE 323-1974 (Institute of Electrical and Electronics Engineers) qualification test program or on the basis of published material aging data. The results of qualification tests or other published material aging data are documented in individual mechanical component EQ binders. Since some of the variables analyzed are based upon 40-year assumptions, these qualifications have been identified as TLAAs that require evaluation for the period of extended operation. The individual mechanical component's EQ documents will be revised to address the 60-year component service requirements in accordance with the BBS Environmental Qualification Program (EQP).

The applicant dispositioned the TLAA for MEQ in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of aging on the intended functions of mechanical equipment located in harsh environments will be adequately managed by the Byron and Braidwood EQP for the period of extended operation.

4.7.3.2 Staff Evaluation

The staff reviewed the applicant's TLAA for the safety-related mechanical components located in harsh environments and the corresponding disposition of 10 CFR 54.21(c)(1)(iii), consistent with the review procedures in SRP-LR Section 4.7.3.1.3, which state that:

(a) The applicant identifies the SCs associated with the TLAA.

(b) The TLAA is described with respect to the objectives of the analysis, conditions, assumptions used, acceptance criteria, relevant aging effects, and intended function(s).

(c) In cases where a mitigation or inspection program is proposed, the reviewer uses the guidance provided in Branch Technical Position RLSB-1 of the SRP-LR to ensure that the effects of aging on the structure and component intended function(s) are adequately managed for the period of extended operation.

The staff's evaluation of the above criteria is as follows:

(a) The TLAA included a generic description (i.e., safety-related mechanical components located in harsh environments) to define the scope of components associated with the TLAA in lieu of a list of SCs associated with the TLAA. During the AMP audit, the staff reviewed all of the mechanical component environmental qualification binders included within the scope of the Byron and Braidwood EQP and determined that the components are consistent with the generic description of the TLAA.

(b) In LRA Section 4.7.3, the applicant stated that, "[t]he design basis conditions during the period of extended operation will remain the same as those in the current license period." The staff noticed that UFSAR Table 3.11-2, "Plant Environmental Conditions," contains a listing of the environmental parameters associated with temperature, relative humidity, pressure, and integrated dose for normal, abnormal, and accident conditions. The applicant also stated that qualified lives are based IEEE 323-1974 qualification tests or published material aging data.

    During the AMP audit, the staff reviewed all of the mechanical component environmental qualification binders and confirmed that each component or subcomponent has a specific replacement frequency (i.e., qualified life). The staff confirmed that approximately two-thirds of the components within the scope of this TLAA have been analyzed for the impact of a 60-year life. The staff noticed that replacement frequencies vary, resulting in qualified lives ranging from less than 40 years to much more than 60 years. The staff also found that the plant-specific program requires that a component or subcomponent be replaced when it has reached the end of its qualified life. During the audit, the staff confirmed that the Byron and Braidwood EQP plant-specific procedure states that changes to the qualified lives of components are evaluated by the station environmental qualification engineer. Therefore the staff concludes that the remaining one-third of components within the program will either be replaced at the end of their qualified lives or the applicant will perform evaluations to determine if the qualified life can be extended.

In addition, during the AMP audit, the staff reviewed several documents that supported the qualified lives for components within the scope of the TLAA. The staff noticed that replacement frequencies are defined for all the components in the program and frequencies are based on IEEE 323-1974 qualification tests or on the basis of published material aging data (i.e., standard material property data sources).

(c) The applicant has not proposed a mitigation or inspection program, but instead will control the replacement of the components or component subparts using its Byron and Braidwood EQP. The use of the individual mechanical component documents is cited in LRA Section A.4.7.3. However, during its review of the mechanical component environmental qualification binders, the staff noticed that several of the components have condition monitoring surveillance requirements, such as (a) the external parts of the containment spray pumps and main steam power operated relief valves are required to be inspected for aging related degradation during each fuel load outage and be replaced immediately if such degradation is detected, (b) the main feed isolation valves should be inspected for packing and gasket leaks, (c) the main steam power-operated relief valve hydraulic operator should be checked for oil leakage, and (d) the main feed isolation valve hydraulic operators have inspection and oil sample requirements. It was not clear to the staff whether the surveillance requirements have been incorporated into AMPs. By letter dated February 18, 2014, the staff issued RAI 4.7.3-1 requesting that the applicant state the basis for why the condition monitoring activities described in the EQ binders are not required to be performed in order to establish reasonable assurance that the affected components and subcomponents will meet their qualified life, or state how the condition monitoring requirements will be incorporated into AMP.

In its response dated March 4, 2014, the applicant stated that the condition monitoring requirements that are required to establish reasonable assurance that the affected mechanical components and subcomponents will meet their qualified lives will be incorporated into the Environmental Qualification (EQ) of Electric Components program.

The applicant revised LRA Sections 2.5.2.1, 2.5.2.2, 2.5.2.4 to identify MEQ components as a commodity group. The applicant also added MEQ components to LRA Table 3.6.2-1, which states that these components are: constructed from various organic elastomers, exposed to an adverse localized environment, and subject to various aging effects which will be managed by the Environmental Qualification (EQ) of Electric Components program. The applicant revised LRA Sections A.1.3, A.3.1.3, B.1.6, and B.3.1.3, as well as Commitment No. 45, to include the MEQ components in the scope of the Environmental Qualification (EQ) of Electric Components program. The applicant identified an enhancement to the program to include the MEQ components.

The staff's evaluation of the Environmental Qualification (EQ) of Electric Components program and the above enhancement is documented in SER Section 3.0.3.1.20. The staff finds the applicant's response acceptable because the conditioning monitoring activities required by the mechanical component environmental qualification binders will be adequately managed by the Environmental Qualification (EQ) of Electric Components program. The program includes maintenance, surveillance, and replacement activities capable of providing reasonable assurance that the MEQ components will meet their CLB function(s) during the period of extended operation. The staff's concern described in RAI 4.7.3-1 is resolved.

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions of safety-related mechanical components located in harsh environments will be adequately managed for the period of extended operation.

Additionally, the TLAA meets the acceptance criteria in SRP-LR Section 4.7.3.1.3 because: (a) the scope of components associated with this TLAA are controlled by a plant-specific procedure; (b) affected components are identified in individual mechanical component EQ documents; (c) appropriate conditions and assumptions for the evaluation of components are included in the UFSAR; (d) the use IEEE 323-1974 qualification tests or published material aging data are standard industry methods for establishing qualified lives of mechanical equipment located in harsh environments; (e) the UFSAR supplement requires the use of mechanical component EQ documents, which specify the replacement frequency; (f) plant-specific procedures ensure that the appropriate plant staff conducts the review of changes to the mechanical component documents; and (g) conditioning monitoring activities for the MEQ components that are required to establish reasonable assurance that the affected MEQ components will meet their qualified lives will be managed by the maintenance, surveillance, and replacement activities in the Environmental Qualification (EQ) of Electric Components program.

4.7.3.3 UFSAR Supplement

LRA Section A.4.7.3 provides the UFSAR supplement summarizing the safety-related mechanical components located in harsh environments TLAA. The staff reviewed LRA Section A.4.7.3 consistent with the review procedures in SRP-LR Section 4.7.3.2, which state that the summary description is reviewed to verify that it is appropriate, such that later changes can be controlled by 10 CFR 50.59 and it contains information that the TLAA has been dispositioned for 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 4.7.3.2, and is therefore acceptable. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the replacement of safety-related mechanical components located in harsh environments, 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 has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions of safety-related mechanical components located in harsh environments will be adequately managed by the BBS EQ of Electric Components Program for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.4 Residual Heat Removal Heat Exchangers Tube Side Inlet and Outlet Nozzles Fracture Mechanics Analysis

4.7.4.1 Summary of Technical Information in the Application

LRA Section 4.7.4 describes the applicant's TLAA for flaws detected in RHR heat exchanger tube side inlet and outlet nozzles made of SS. During ultrasonic testing (UT) examinations in 1991, indications were found in the Braidwood Unit 2 RHR heat exchanger nozzles. Some of the indications exceeded the acceptance standards of ASME Section XI, IWB-3500 (1983 Edition through Summer 1983 Addenda) and were subjected to further evaluation in accordance with ASME Section XI, IWB-3600. Even though this component is an ASME

Class 2 component, a Class 1 fracture-mechanics-based flaw growth analysis was performed and this flaw growth analysis was used to disposition the indications that did not meet the IWB-3500 acceptance standards. This analysis for the Braidwood Unit 2 flaws was submitted to the staff for review and the staff reviewed and approved the analysis.

Subsequently, UT examinations were performed on all the BBS RHR heat exchanger nozzles and any additional indications exceeding the IWB-3500 acceptance standards were dispositioned with the analytical results. The following documents, as submitted to the staff on August 25, 1992, present the methodology for dispositioning the flaws found at BBS: (1) WCAP-13454, "Fracture Mechanics Evaluation, Byron and Braidwood Units 1 and 2, Residual Heat Exchanger Tube Side Inlet and Outlet Nozzles," August 1992 (Proprietary), and (2) WCAP-13455, "Fracture Mechanics Evaluation, Byron and Braidwood Units 1 and 2, Residual Heat Exchanger Tube Side Inlet and Outlet Nozzles," August 1992 (Non-proprietary, ADAMS Accession No. 9208280207). This analysis uses the startup and shutdowns of the RHR system coincident with the number of plant heatup and cooldowns based on the current licensed operation period as inputs.

The applicant dispositioned the TLAA for the flaws of RHR heat exchanger nozzles in accordance with 10 CFR 54.21(c)(1)(iii) to demonstrate that the effects of fatigue flaw growth on the intended functions will be adequately managed by the Fatigue Monitoring program for the period of extended operation. The Fatigue Monitoring program will monitor transient cycles to ensure the transient inputs used in the flaw growth analysis will not be exceeded during the period of extended operation.

4.7.4.2 Staff Evaluation

The staff reviewed the applicant's fracture-mechanics-based flaw growth analysis for the RHR heat exchanger nozzles made of SS, consistent with the review procedures in SRP-LR Section 4.7.3.1.3. The review procedures state that the applicant proposes to manage the aging effects associated with the TLAA by an AMP in the same manner as described in the IPA in 10 CFR 54.21(a)(3). The review procedures also state that the reviewer reviews the applicant's AMP to verify that the effects of aging on the intended function(s) are adequately managed consistent with the CLB for the period of extended operation. The review procedures further state that the TLAA is described with respect to the objectives of the analysis, conditions, assumptions used, acceptance criteria, relevant aging effects, and intended function(s).

During its review of LRA Section 4.7.4 and related information, the staff noticed a relief request by the applicant, which indicated that an ASME Section XI repair by excavation was completed on the unacceptable flaws of the Braidwood Unit 2 RHR heat exchanger nozzle-to-vessel welds ("Relief from Inservice Inspection Requirements for Residual Heat Removal Heat Exchanger Nozzle-to-Vessel Welds," December 12, 1995, ADAMS Accession No. 951219036). This reference regarding the relief request stated that the Braidwood, Unit 2 flaws were fabrication flaws, slag, incomplete fusion and excess porosity. The staff noticed that the above reference did not identify any other unacceptable flaws of the BBS RHR heat exchanger nozzles.

Therefore, it was unclear to the staff whether there are flaws currently in these nozzles which exceed the acceptance standards of ASME Code Section XI IWB-3500. It was also unclear to the staff whether the applicant's fracture mechanics analysis is relied upon to support the continued service of the heat exchangers, or the applicant's relief request for an alternative to the ASME Code ISI method.

By letter dated February 26, 2014, the staff issued RAI 4.7.4-1. In Part 1 of RAI 4.7.4-1, the staff requested that the applicant clarify whether there are flaws currently in the nozzles that exceed the acceptance standards of ASME Code Section XI IWB-3500. The staff also requested that the applicant clarify whether its flaw growth analysis is relied on to support: (a) continued service of the heat exchanger nozzles with existing flaws, or (b) the applicant's relief request for an alternative to the ASME Code ISI method for these nozzles (e.g., performing VT-2 visual examination in place of UT examination).

In Part 2 of the RAI, the staff stated that relief requests for inservice inspections are only valid for the current ISI ten-year interval and are required to be resubmitted for each interval for the period of extended operation, if desired. The staff also requested that, if the flaw growth analysis is relied upon to support the use of an alternate inspection method under a relief request process, the applicant clarify why the relief request process is not identified as part of the 10 CFR 54.21(c)(1)(iii) aging management basis in conjunction with the applicant's analysis.

In Part 3 of the RAI, the staff requested that the applicant provide the following information for the applicant's analysis: (a) current flaw sizes (i.e., length and depth), orientations (i.e., circumferential and axial) and locations based on the most recent inspection results in comparison with nozzle dimensions, and (b) projected flaw sizes at the end of the period of extended operation. The staff also requested that, as an alternative to (a) and (b), if a bounding-case analysis is applicable to each nozzle, the applicant provide the maximum current flaw size and maximum projected flaw size with the associated orientation and location which bound the other flaws for each nozzle. The staff further requested that the applicant describe the acceptance criteria for the flaws and when the most recent volumetric examination was performed on each nozzle. In addition, the staff requested that, as part of this response, the applicant provide the relevant transient names and projected numbers of transient cycles for the applicant's analysis.

By letter dated March 28, 2014, the applicant responded to RAI 4.7.4-1. In its response to Part 1 of RAI 4.7.4-1, the applicant stated that BBS Unit 1 and 2 RHR heat exchanger tube side inlet and outlet nozzle welds currently contain flaws that exceed the acceptance standards of ASME Code, Section XI, IWB-3500, 1983 Edition through Summer 1983 Addenda. The applicant also stated that these flaws, which were found between 1991 and 1994, were determined to be fabrication flaws. The applicant further stated that, even though these heat exchangers are ASME Class 2 components, a Class 1 fracture mechanics analysis, which met the requirements of ASME Code, Section XI, IWB-3600, was performed on these flaws.

In the response to Part 1, the applicant also stated that the flaws, which satisfied ASME Section XI, IWB-3640 requirements, were determined to be acceptable and remain in service today. The applicant stated that only flaws on the Braidwood Unit 2 "B" heat exchanger outlet nozzle did not satisfy ASME Code, Section XI, IWB-3640 requirements and were repaired in 1994. In addition, the applicant stated that the fracture mechanics analysis supporting the flaw evaluations were submitted to the staff, and the staff reviewed and approved the analysis in a letter dated February 3, 1995, "Residual Heat Exchanger Nozzle Welds, Byron Station, Unit 1 and 2, and Braidwood Station, Units 1 and 2 (TAC Numbers M90894, M90895, M91408, and M90840)," ADAMS Accession Number 9502130037.

In the response to Part 2, the applicant stated that the fracture mechanics analysis of the flaws is relied on to support continued service of the heat exchanger nozzles. The applicant also indicated that ASME Code Case N-706 was endorsed as "acceptable" (not "conditional") by the staff in 2007 as documented in RG 1.147, Revision 15, "Inservice Inspection Code Case

Acceptability ASME Section XI, Division 1." The applicant further stated that as such, the submittal for a relief request to use this code case is not required, and the relief request process is not applicable. The staff concluded that ASME Code Case N-706 was subsequently revised to Code Case N-706-1, which was approved in RG 1.147, Revision 16.

In the response to Part 2, the applicant also stated that ASME Code Case N-706-1 provides relief from the requirement to perform a UT examination of the welds on PWR SS regenerative and residual heat exchangers. The applicant stated that the code case allows VT-2 inspections of the welds in lieu of the UT examination, provided the welds have been volumetrically examined at least once. In addition, the applicant stated that since the BBS RHR heat exchanger nozzle welds have all been volumetrically examined with UT and been dispositioned in accordance with the flaw growth analysis, the use of the code case to perform VT-2 examinations of the welds instead of UT examinations is permissible. The applicant stated that, as discussed above, the use of ASME Code Case N-706-1 relies upon the fracture mechanics analysis.

In the response to Part 3, the applicant stated that the flaw growth analysis provides a bounding-case analysis which is applicable to each RHR heat exchanger tube side inlet and outlet nozzle. The applicant also indicated that Flaw Number 3 on the Braidwood Unit 2 "B" RHR heat exchanger inlet nozzle bounds all flaws that were dispositioned as acceptable for continued service. The applicant further stated that this flaw has an "as found" crack depth of 0.300 in. in a portion of the nozzle with a wall thickness of 0.526 in. In addition, the applicant indicated that applying the fatigue flaw depth growth of 0.001 in., based on an additional 200 cycles, results in a projected flaw depth of 0.301 in. at the end of the period of extended operation. In addition, the applicant stated that the fraction of the projected flaw depth with respect to the nozzle wall thickness would be 57.2 percent, which meets the acceptance criterion. The applicant stated that the flaw growth analysis concludes that for each inlet and outlet nozzle, the appropriate acceptance criterion, in accordance with ASME Code, Section XI, IWB-3460, is a maximum allowed flaw depth of 60 percent of the nozzle wall thickness.

In the response to Part 3, the applicant also stated that the most recent volumetric examinations performed on each nozzle are as follows:

- Braidwood Unit 1, 1RH02AA and 1RH02AB inlet and outlet nozzles, 1992
- Braidwood Unit 2, 2RH02AA and 2RH02AB inlet and outlet nozzles, 1994
- Byron Unit 1, 1RH02AA and 1RH02AB inlet and outlet nozzles, 1993
- Byron Unit 2, 2RH02AA and 2RH02AB inlet and outlet nozzles, 1992

The applicant further stated that the flaw evaluation methodology in the analysis includes loading conditions for thermal expansion, internal pressure, deadweight, and operating basis and safe shutdown earthquakes for the RHR heat exchanger inlet and outlet nozzles. The applicant stated that the flaw growth analysis considers fatigue due to applied stresses during transients and residual stresses, and is also based on ASME Code, Section XI, Appendix C (1989 Addenda). In addition, the applicant stated that the RHR heat exchangers are only used when the RCS is cooled down to cold shutdown and refueling, as the RHR system is placed into service, and later during RCS heatup until the RHR system is taken out of service.

The applicant indicated that the flaw growth analysis conservatively assumed 200 cycles corresponding to 200 plant heatups and plant cooldowns (i.e., transients 1 and 2 in LRA Tables 4.3.1-1 and 4.3.1-4 for Byron and Braidwood, respectively) over the 60-year period of operation. The applicant also stated that, based on the assumed cycles, the analysis results in a fatigue flaw depth growth of less than 0.001 in. such that the projected flaw depth after 200 cycles is calculated by adding 0.001 in. to the "as-found" flaw depth. The applicant further stated that the maximum number of plant heatups and plant cooldowns projected for 60 years is 117 on Byron Unit 1, which bounds all four units.

The staff noticed that the applicant confirmed that RHR heat exchanger tube side nozzles currently contain flaws that exceed the acceptance standards of ASME Code, Section XI, IWB-3500. The staff also noticed that the applicant confirmed that the bounding-case analysis (Braidwood Unit 2, "B" RHR heat exchanger inlet nozzle) indicates that the maximum fatigue flaw growth of 0.001 in. results in the bounding-case flaw depth of 0.301 in. as projected at the end of the period of extended operation. In addition, the staff noticed that the applicant confirmed that the projected flaw depth projected is acceptable against the acceptance criteria of ASME Code Section XI, IWB-3640.

In its review, the staff noticed that the applicant also indicated that the most recent volumetric examinations for the RHR heat exchanger nozzles were those performed in 1994 on Braidwood Unit 2 RHR heat exchanger nozzles. The staff identified that an NRC letter dated February 29, 1996 (ADAMS Accession No. 9603060023), enclosed the staff's safety evaluation regarding the BBS request for relief (Nos. NR-18 and NR-23) from the volumetric examinations of the RHR heat exchanger nozzles for the first 10-year ISI interval. The staff further noticed that the staff's safety evaluation also discusses the previous inspection requirements which were specified in the staff's safety evaluation, dated February 3, 1995 (ADAMS Accession No. 9502130021), regarding the flaws detected from these nozzle inspections and the applicant's fracture mechanics analysis for the flaws subject to the evaluation of ASME Code, Section XI, IWB-3600.

In addition, the staff determined that the February 29, 1996, safety evaluation states that instead of the previous requirements specified in the February 3, 1995, safety evaluation, the applicant is required to perform UT examinations on a sample of RHR nozzle-to-vessel welds (one nozzle per unit) during the next inspection interval (i.e., the second interval) to provide additional assurance that these flaws have not grown and that no new service-induced indication has developed.

The staff also identified that the applicant's letter dated July 25, 2007 (ADAMS Accession No. ML072060413), describes a relief request regarding the Braidwood Units 1 and 2 RHR heat exchanger nozzle examinations for the second 10-year inspection interval. The staff further noticed that even though this 2007 relief request was withdrawn by the applicant's letter dated January 23, 2008 (ADAMS Accession No. ML080240324), the July 25, 2007, letter indicates that UT examinations were performed in September 1998 on a nozzle (1RHR-01-1RHXN1, A HX) of Braidwood Unit 1 RHR heat exchangers to fulfill the requirements specified in the February 22, 1996, NRC safety evaluation (i.e., volumetric examination of one nozzle per unit during the second ISI interval). The staff noticed that the applicant's 2007 letter also states that no appreciable flaw growth was noted from the 1998 examinations on the examined nozzle of Braidwood Unit 1.

As discussed above, it was unclear to the staff why the applicant's response does not discuss the UT examination results for the Braidwood Unit 1 RHR heat exchanger nozzle which were obtained in September 1998, as described in the applicant's letter dated July 25, 2007. It was also unclear to the staff why the applicant's response does not address any results of the UT examinations associated with the applicant's fracture mechanics analysis which are required for the RHR heat exchanger nozzles (i.e., a nozzle per unit), as specified in the staff's safety evaluation dated February 29, 1996. In addition, the staff needed clarification on whether the existing flaws are embedded inside the RHR heat exchanger nozzles without exposure to the reactor coolant in order to confirm the absence of environmental effects on flaw growth. The staff also noticed that the applicant's response did not provide the length of the bounding flaw with an as-found depth of 0.300 in. as baseline information. The staff identified that the applicant's analysis described in the LRA may not adequately address the previous volumetric examination results and the 10 CFR 54.21(c)(1)(iii) aging management basis associated with the flaw growth analysis.

By letter dated May 21, 2014, the staff issued RAI 4.7.4-1a. In Part 1 of the RAI, the staff requested that the applicant clarify why its response does not discuss the ultrasonic examination results of the Braidwood Unit 1 RHR heat exchanger nozzle which were obtained in September 1998 as documented in the applicant's relief request letter dated July 25, 2007.

In Part 2 of the RAI, the staff requested that the applicant clarify why its response does not address results of the ultrasonic examinations, which are associated with the applicant's fracture mechanics analysis and are required for the RHR heat exchanger nozzles (i.e., a nozzle per unit) as specified in the staff's February 29, 1996, safety evaluation. The staff also requested that, if all of these ultrasonic examinations have not been completed, the applicant justify why the applicant does not identify the ultrasonic examinations as part of the 10 CFR Part 54.21(c)(1)(iii) aging management basis associated with the applicant's fracture mechanics analysis.

In Part 3 of the RAI, the staff requested that the applicant provide additional information to confirm whether the previous ultrasonic examinations, including those performed in 1998, revealed any flaw growth. The staff also requested that the applicant define "no appreciable flaw growth," which was mentioned in the applicant's letter dated July 25, 2007.

In Part 4 of the RAI, the staff requested that the applicant clarify whether the existing flaws are embedded inside the RHR heat exchanger nozzles without exposure to the reactor coolant in order to confirm the absence of environmental effects on flaw growth. In addition, the staff requested that the applicant describe the length of the bounding flaw in comparison with the inner diameter of the nozzle as baseline information.

By letter dated June 16, 2014, the applicant responded to RAI 4.7.4-1a. In its response the applicant indicated that it interpreted Part 3 of RAI 4.7.4-1 as a request to provide the results of the most recent (i.e., last performed) examinations for the period between 1991 and 1994 which were addressed in the staff's February 3, 1995, safety evaluation regarding the fracture mechanics analysis of the observed flaws. The applicant clarified that the 1998 ultrasonic examination results were not included in the response to RAI 4.7.4-1 based on this interpretation.

In its response to Part 2 of RAI 4.7.4-1a, the applicant stated that additional ultrasonic examinations after 1994 were performed on RHR heat exchanger nozzles in accordance with the staff's safety evaluation dated February 29, 1996. The applicant also clarified that the ultrasonic examinations were performed on one nozzle per unit at Byron Units 1 and 2 and Braidwood Unit 1 in the second ISI interval, as specified in the 1996 safety evaluation. The applicant stated that these examinations confirmed that there was no flaw growth from the first to the second ISI interval.

In addition, the applicant stated that an ultrasonic examination of one Braidwood Unit 2 RHR heat exchanger nozzle was planned for the spring 2008 refueling outage, which was the last refueling outage in the second ISI interval for the unit. The applicant also stated that because an ultrasonic examination of these nozzles requires extensive labor resources, radiation exposure to the examiners, and significant cost without a commensurate increase in quality or public safety, a relief request was submitted to the staff in the letter dated July 25, 2007. The applicant further indicated that the submitted letter asked a relief from the ASME Code, Section XI requirement to perform ultrasonic examinations on the RHR heat exchanger nozzles based on the alternative visual examination requirements in ASME Code Case N-706. In addition, the applicant indicated that since the staff endorsed ASME Code Case N-706 in 2007 after the submittal of the relief request and all Braidwood Unit 2 RHR heat exchanger nozzles had been ultrasonically examined at least once during the period between 1991 and 1994, thereby satisfying ASME Code Case N-706 requirements, an ultrasonic examination of the Braidwood Unit 2 RHR heat exchanger nozzle in the second interval was no longer required in the spring 2008 refueling outage. The applicant also stated that, on January 23, 2008, a followup letter was submitted to the staff withdrawing the July 2007 relief request.

In its response to Part 2, the applicant also confirmed that the second interval examinations described above found no new flaws. The applicant further indicated that these ultrasonic examination results are consistent with the conclusions of the applicant's fracture mechanics analysis and demonstrate that the RHR nozzle weld flaw growth would be inconsequential (a total growth of less than 0.001 in.) for the 60-year period of operation.

In its response to Part 3 of RAI 4.7.4-1a, the applicant stated that the comparisons of the ultrasonic examination results between the first and second ISI intervals concluded that there was no observed flaw growth greater than the repeatability variances of examination equipment and techniques. The applicant also clarified that the term "no appreciable flaw growth" in the July 25, 2007, letter was intended to explain that any dimensional differences between the flaw examination results between the first and second intervals were small and within the repeatability variances of the examinations. The applicant further indicated that the repeatability variances resulted from factors such as slight variations in the transducer placement angles, orientation, and surface contact during the scanning process, and slight differences in the scanners, search units, and gain levels.

In its response to Part 4 of RAI 4.7.4-1a, the applicant stated that the first interval examinations performed from 1991 through 1994 found that all indications were subsurface flaws, not open to the internal and external surface of the nozzle, and, therefore, not exposed to reactor coolant. The applicant also stated that these flaws were determined to be fabrication flaws. The applicant further clarified that the examinations performed in the second ISI interval found that none of the flaws had grown and the flaws remained subsurface. In addition, the applicant stated that the length of the bounding flaw was 0.8 in. and the inside nozzle diameter is 13.075 in., indicating that the flaw length was not significant compared to the nozzle circumference.

The staff finds the applicant's response acceptable because the applicant confirmed that (1) ultrasonic examinations were conducted on RHR heat exchanger nozzles at Byron Units 1 and 2 and Braidwood Unit 1 during the second ISI interval, (2) these examination results demonstrated that no flaw growth occurred from the first to the second ISI interval and no new flaws were detected in the second interval, (3) the applicant relied on ASME Code Case N-706 to justify why the ultrasonic examination planned for the second interval of Braidwood Unit 2 was not performed, (4) "no appreciable flaw growth" addressed in the July 25, 2007, letter was intended to explain that there was no observed flaw growth greater than the repeatability variances of the ultrasonic examinations, and (5) the ultrasonic examinations performed during the first and second intervals confirmed that all indications were subsurface flaws, and not exposed to the reactor coolant environment. Therefore, the staff's concern described in RAIs B.4.7.4-1 and B.4.7.4-1a are resolved.

Thus, the staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue flaw growth on the intended functions of the RHR heat exchanger nozzles will be adequately managed for the period of extended operation.

Additionally, LRA Section 4.7.4 meets the acceptance criteria in SRP-LR Section 4.7.2 because the applicant appropriately evaluated the TLAA for the flaws of RHR heat exchanger nozzles, consistent with the CLB, and fatigue flaw growth in these nozzles will be adequately managed by the Fatigue Monitoring program for the period of extended operation.

4.7.4.3 UFSAR Supplement

LRA Section A.4.7.4 provides the UFSAR supplement summarizing the flaw growth analysis for the RHR heat exchanger tube side nozzles. The staff reviewed LRA Section A.4.7.4 consistent with the review procedures in SRP-LR Section 4.7.3.2. The review procedures state 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 each TLAA. The review procedures also state that each such summary description is reviewed to verify that it is appropriate. The review procedures further state that the description should contain information that the TLAA has been dispositioned for the period of extended operation.

Based on its review of the UFSAR supplement, the staff determines that the applicant met 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 fatigue flaw growth in the RHR heat exchanger tube side inlet and outlet nozzles, as required by 10 CFR 54.21(d).

4.7.4.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue flaw growth on the intended functions of the RHR heat exchanger tube side inlet and outlet nozzles will be adequately managed by the Fatigue Monitoring program for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.7.5 Reactor Coolant Pump Flywheel Fatigue Crack Growth Analysis

4.7.5.1 Summary of Technical Information in the Application

LRA Section 4.7.5 describes the applicant's TLAA for fatigue crack growth in the RCP motor flywheels. The LRA states that fatigue is an aging effect that was analyzed due to the possibility of flywheel failure, which could create missiles and damage the RCP seals or other pressure boundary components. Accordingly, TS 5.5.7 requires the applicant to periodically inspect the integrity of the flywheels using either ultrasonic or surface tests. Two of the RCP motor flywheels must be inspected at 10-year intervals coinciding with the ISI schedule specified by

ASME Code, Section XI, and all of the other flywheels must be inspected at an interval not to exceed 20 years. The LRA states that Westinghouse reports WCAP-14535A, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination," dated November 1996, and WCAP-15666-A, Revision 1, "Extension of Reactor Coolant Pump Motor Flywheel Examination," dated October 2003, establish the bases for the 10- and 20-year inspection intervals, respectively. The LRA states that both inspection intervals are based on fatigue crack growth analyses that assume 6,000 RCP start-stop cycles, which is more than the greatest number of cycles the applicant projects for any RCP motor flywheel to experience in 60 years of operation. The applicant dispositioned the TLAA for RCP motor flywheel fatigue crack growth in accordance with 10 CFR 54.21(c)(1)(i) to demonstrate that the analyses remain valid for the period of extended operation.

4.7.5.2 Staff Evaluation

The staff reviewed the applicant's TLAA for RCP motor flywheel fatigue crack growth and the corresponding disposition of 10 CFR 54.21(c)(1)(i), consistent with the review procedures in SRP-LR Section 4.7.3.1.1. These procedures state that the applicant should show that the existing analysis bounds the period of extended operation, so that no reanalysis is necessary.

The staff found that there are a total of 18 RCP motor flywheels at Byron and Braidwood, two of which are spares. On September 20, 2001, and September 16, 2010, the staff issued license amendments modifying TS 5.5.7. These amendments impose the current 10- and 20-year inspection intervals for the RCP motor flywheels, respectively. The 10-year inspection interval is based on the staff's prior approval of WCAP-14535A; the 20-year inspection interval is based on the staff's prior approval of WCAP-15666-A. The TSs require different inspection frequencies because two of the RCP motor flywheels were originally designed and built for the cancelled Marble Hill Nuclear Generating Station (serial numbers 4S88P961 and 1S88P961), and these flywheels are not interchangeable with the Byron and Braidwood RCP motors evaluated in WCAP-15666-A. The staff reviewed both topical reports and confirmed that the applicable aging effect considered in the analyses is cracking due to fatigue. The staff also confirmed that the only time-limited assumption involved in the analyses is the number of RCP motor start-stop cycles, which are inputs to the fatigue crack growth analyses.

To demonstrate that the WCAP-14535A and WCAP-15666-A analyses bound the period of extended operation, LRA Section 4.7.5 states that the applicant projected the number of RCP motor start-stop cycles through 60 years. LRA Tables 4.3.1-1 and 4.3.1-4 identify these cycles as operational transients. According to the methodology described in LRA Section 4.3.1, the applicant determined the number of 60-year projected cycles by adding the product of the cycle projection rate and the remaining number of years to the number of baseline cycles, where the cycle projection rate is based on past operating data. As a result, the applicant projects 1,755 RCP motor start-stop cycles for Byron Unit 1, 1,545 cycles for Byron Unit 2, 1,125 cycles for Braidwood Unit 1, and 1,035 cycles for Braidwood Unit 2.

By reworking the applicant's formula for calculating the number of 60-year projected cycles, the staff developed a formula for the cycle projection rate. This rate is equal to the difference of the 60-year projected cycles and the baseline cycles divided by the number of remaining years. The staff calculated the number of remaining years for each unit and then input these values, along with the values in LRA Tables 4.3.1-1 and 4.3.1-4 for the baseline cycles and 60-year projected cycles, into the reworked cycle projection rate formula. As a result, the staff determined that the applicant used approximately 21 cycles per year for the Byron Unit 1 projection, 17 cycles per year for the Byron Unit 2 projection, 20 cycles per year for the

Braidwood Unit 1 projection, and 14 cycles per year for the Braidwood Unit 2 projection. The staff compared these values with recent Byron and Braidwood reactor power status reports and determined that the applicant's cycle projection rates are reasonable; therefore, the staff determined that they are acceptable to demonstrate that the existing analyses bound the period of extended operation.

Although the applicant's projections for the number of RCP motor start-stop cycles are adequate, the staff noticed that past RCP motor flywheel inspection results could invalidate the existing analyses if the inspections detected a flaw and there is evidence of an actual crack growth rate that is greater than the rate assumed in WCAP-14535A and WCAP-15666-A. TS 5.5.7 requires the applicant to periodically inspect the RCP motor flywheels; however, the staff noticed that the LRA does not provide any results from these past inspections. By letter dated February 26, 2014, the staff issued RAI 4.7.5-1, requesting that the applicant summarize the results of the past ISIs of the RCP motor flywheel components. If flaws were detected, the staff requested the applicant to quantify any growth and provide a comparison against the crack growth rates used in WCAP-14535A and WCAP-15666-A.

The applicant responded to RAI 4.7.5-1 by letter dated March 28, 2014. The applicant provided the most recent inspection results for all 18 of its RCP motor flywheels. If these results had recordable indications, then the applicant also provided the previous inspection results for comparison. According to this information, rounded and linear indications have been detected in some of the RCP motor flywheels. However, the applicant stated that all of the recorded indications meet the applicable acceptance criteria from WCAP-14535A and WCAP-15666-A, and there have been no indications of fatigue crack growth between inspections. The staff reviewed the inspection results provided by the applicant and determined that they do not demonstrate any fatigue-induced growth of flaws in the RCP motor flywheels. Therefore, the staff finds the applicant's response acceptable because there is no actual crack growth data that would invalidate the fatigue crack growth rates assumed in WCAP-14535A and WCAP-15666-A. The staff's concern described in RAI 4.7.5-1 is resolved.

Based on the adequacy of the applicant's cycle projections and the lack of actual fatigue crack growth, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses for RCP motor flywheel fatigue crack growth remain valid for the period of extended operation. This demonstration also meets the acceptance criteria in SRP-LR Section 4.7.2.1.

4.7.5.3 UFSAR Supplement

LRA Section A.4.7.5 provides the UFSAR supplement summarizing the TLAA for RCP motor flywheel fatigue crack growth. The staff reviewed LRA Section A.4.7.5 consistent with the review procedures in SRP-LR Section 4.7.3.2, which state that the applicant is to provide a summary description for its evaluation of each TLAA. The SRP-LR also states that the summary description should contain information on the disposition of the TLAA for the period of extended operation and be appropriate such that later changes can be controlled by 10 CFR 50.59.

As described in LRA Section 4.7.5, the TLAA applies to two separate fatigue crack growth analyses. One analysis, which is based on WCAP-14535A, establishes a 10-year inspection interval for RCP motor serial numbers 4S88P961 and 1S88P961. The other analysis, which is based on WCAP-15666-A, establishes a 20-year inspection interval for the other flywheels. However, the staff concluded that the summary description in LRA Section A.4.7.5 only addressed the 10-year inspection interval, and it did not address the 20-year inspection interval. As such, the staff determined that the summary description did not cover the full scope of the TLAA or clearly identify which inspection intervals apply to which RCP motor flywheels. For these reasons, the staff determined that the applicant's summary description would not facilitate control of later changes to the TLAA by 10 CFR 50.59. By letter dated February 26, 2014, the staff issued RAI 4.7.5-2 requesting the applicant to revise LRA Section A.4.7.5 to clearly identify each of the RCP motor flywheels and specify which topical report and corresponding inspection frequency apply to each.

By letter dated March 28, 2014, the applicant responded to RAI 4.7.5-2 by amending the summary description in LRA Section A.4.7.5. The applicant subsequently retracted this amendment by letter dated June 30, 2014; however, by letter dated July 15, 2014, the applicant re-submitted an identical amendment to LRA Section A.4.7.5. The amended summary description specifies the serial numbers for the RCP motor flywheels that are subject to the 10-year inspection interval and states that the 20-year inspection interval applies to all of the other flywheels. The revised summary description also states that the fatigue crack growth analyses in WCAP-14535A and WCAP-15666-A provide the bases for the 10- and 20-year inspection intervals, respectively. The staff reviewed these revisions and finds them acceptable because they clearly identify all of the RCP motor flywheels addressed in the TLAA and their respective inspection requirements consistent with TS 5.5.7. The staff confirmed that the revisions also identify that WCAP-14535A and WCAP-15666-A provide the bases for the inspection requirements, which the applicant demonstrated to remain valid for the period of extended operation. Therefore, the staff's concern described in RAI 4.7.5-2 is resolved.

Based on its review of the UFSAR supplement, as amended by letter dated July 15, 2014, the staff finds that LRA Section A.4.7.5 meets the acceptance criteria in SRP-LR Section 4.7.2.2, and is therefore acceptable. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the TLAA for RCP motor flywheel fatigue crack growth, as required by 10 CFR 54.21(d).

4.7.5.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the TLAA for RCP motor flywheel fatigue crack growth 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).

4.7.6 Byron Unit 2 Pressurizer Seismic Restraint Lug Flaw Evaluation

4.7.6.1 Summary of Technical Information in the Application

LRA Section 4.7.6 states that, in September of 2005, an indication exceeding the acceptance standards of ASME Section XI, Subarticle IWB-3500, 1989 Edition was found on a Byron Unit 2 pressurizer seismic lug. The LRA also states that investigation concluded that the indication was not service induced, but rather was due to lack of fusion in the original weld. The LRA further states that a flaw growth analysis was performed in accordance with ASME Section XI, Subarticle IWB-3600, 1989 Edition, which concluded that the indication size will remain within acceptable limits for the current remaining licensed operating period. The LRA also states that this analysis assumed input transients for the current licensed operating period based on

40 years of operation. Therefore, the applicant concluded that the Byron Unit 2 pressurizer seismic restraint lug flaw evaluation is a TLAA.

The applicant dispositioned the Byron Unit 2 pressurizer seismic restraint lug flaw evaluation in accordance with 10 CFR 54.21(c)(1)(iii), to demonstrate that the effects of aging on the intended function(s) will be adequately managed by the Fatigue Monitoring program, for the period of extended operation.

4.7.6.2 Staff Evaluation

The staff reviewed LRA Section 4.7.6 and the TLAA for the Byron Unit 2 pressurizer seismic restraint lug flaw evaluation to confirm, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions will be adequately managed consistent with the CLB for the period of extended operation.

The staff reviewed the applicant's TLAA and the corresponding disposition consistent with the review procedures in SRP-LR Section 4.7.3.1.3, which state that the applicant proposes to manage the aging effects associated with the TLAA by an AMP in a manner as described in the IPA in 10 CFR 54.21(a)(3). The SRP-LR also states that the staff verifies that the effects of aging on the intended function(s) are adequately managed consistent with the applicant's CLB for the period of extended operation.

The staff also reviewed the applicant's flaw growth analysis for Byron Unit 2 pressurizer seismic lug (ML080580263). The staff noticed that the design transient used in the evaluation is the auxiliary spray actuation transient, which is identified in LRA Table 4.3.1-2. The staff also found that the applicant credited its Fatigue Monitoring Program to manage the effects of pressurizer seismic restraint lug flaw growth on the component fatigue life, by monitoring the transient cycles to assure that the assumed number of transient cycles are not exceeded during the period of extended operations. The staff's review of the Fatigue Monitoring program is documented in SER Section 3.0.3.2.24. The staff determined that the cycle counting method will ensure that the fatigue analysis remains valid by ensuring the assumed number of transients used in the analysis is not exceeded.

Based on its review, the staff finds that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that flaw growth of the Byron Unit 2 pressurizer seismic restraint lug flaw will be adequately managed for the period of extended operation. Additionally, the applicant's demonstration meets the acceptance criteria in SRP-LR Section 4.7.3.1.3 because the Fatigue Monitoring Program will monitor transient cycles used in the analysis to ensure that, if a transient limit is approached, corrective action is taken prior to exceeding a transient limit.

4.7.6.3 UFSAR Supplement

LRA Section A.4.7.6 provides the UFSAR supplement summarizing the Byron Unit 2 pressurizer seismic restraint lug flaw evaluation. The staff reviewed LRA Section A.4.7.6 consistent with the review procedures in SRP-LR Section 4.7.3.2, which state that the information to be included in the UFSAR supplement should include a summary description of the evaluation of the metal fatigue TLAA.

Based on its review of the UFSAR supplement, the staff finds that LRA Section A.4.7.6 meets the acceptance criteria in SRP-LR Section 4.7.3.2. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the Byron Unit 2 pressurizer seismic restraint lug flaw evaluation as required by 10 CFR 54.21(d).

4.7.6.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that flaw growth of the Byron Unit 2 pressurizer seismic restraint lug flaw will be adequately managed by the Fatigue Monitoring Program during the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the Byron Unit 2 pressurizer seismic restraint lug flaw evaluation, as required by 10 CFR 54.21(d).

4.7.7 Braidwood Unit 2 Feedwater Pipe Elbow Crack Growth Evaluation

4.7.7.1 Summary of Technical Information in the Application

LRA Section 4.7.7 describes the applicant's TLAA for the Braidwood Unit 2 feedwater pipe elbow crack growth evaluation. The applicant stated that an axial indication was identified on a Braidwood Unit 2, 16-in. main feedwater line elbow downstream of the feedwater regulating valves. The applicant performed a crack growth analysis, in accordance with ASME Section XI, Subarticle IWB-3600, which concluded that the crack size will remain within the acceptable limits during the 40-year life of the plant. The LRA states that the analysis assumed RCS heatup and cooldown, reactor trip transients, and reactor trips with RCS cooldown transients over the 40-year life of the plant and has therefore been identified as a TLAA requiring evaluation for the period of extended operation. The LRA also states that the number of transients assumed in the analysis bounds the number of transients projected to occur through the period of extended operation as discussed in LRA Section 4.3.1.

The applicant dispositioned the Braidwood Unit 2 feedwater pipe elbow crack growth evaluation in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions will be adequately managed by the Fatigue Monitoring program, for the period of extended operation.

4.7.7.2 Staff Evaluation

The staff reviewed LRA Section 4.7.7 and the TLAA for the Braidwood Unit 2 feedwater pipe elbow crack growth evaluation to confirm, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions will be adequately managed consistent with the CLB for the period of extended operation.

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

The staff reviewed LRA Table 4.3.1-4, which provides the baseline and 60-year cycle projections for RCS transients. The staff found that the applicant will use the Fatigue Monitoring program to ensure that the numbers of transients will not be exceeded during the period of extended operation and the numbers of transients assumed in the analysis bounds the number of transients projected to occur through the period of extended operation. The staff also reviewed the applicant's TLAA evaluation basis and compared the number of transients input to the crack growth evaluation with the 60-year projections from the LRA Table 4.3.1-4, and determined that the numbers of transient cycles assumed in the crack growth analysis are bounded by their 60-year projections.

Bases on this review, the staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions of the Braidwood Unit 2 feedwater pipe elbow will be adequately managed consistent with the CLB for the period of extended operation.

In addition, the TLAA meets the acceptance criteria in SRP-LR Section 4.7.2.1 because the ASME Section XI crack growth analyses will be managed by the Fatigue Monitoring program, which will monitor transient cycles and require corrective action prior to exceeding the number of transient cycles used in the evaluations which support these conclusions.

4.7.7.3 UFSAR Supplement

LRA Section A.4.7.7 provides the UFSAR supplement which summarizes the Braidwood Unit 2 feedwater pipe elbow crack growth evaluation. The staff reviewed LRA Section A.4.7.7 consistent with the review procedures in SRP-LR Section 4.7.3.2, which state that the information to be included in the UFSAR supplement should include a summary description of the evaluation of the metal fatigue TLAA.

Based on its review of the UFSAR supplement, the staff finds that LRA Section A.4.7.7 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 Braidwood Unit 2 feedwater pipe elbow crack Growth Evaluation as required by 10 CFR 54.21(d).

4.7.7.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that flaw growth of the Braidwood Unit 2 feedwater pipe elbow 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).

4.7.8 Analyses Supporting Flaw Evaluations of Primary System Components

4.7.8.1 Summary of Technical Information in the Application

<u>Fatigue Crack Growth Analysis</u>. LRA Section 4.7.8 states that BBS have performed preemptive flaw evaluations on primary system components (such as the reactor vessel, pressurizer, primary steam generator subcomponents, and primary coolant components) consistent with ASME Section XI, Subarticle IWB-3600. The LRA further states flaw evaluations were performed consistently with the methodologies in WCAP-11063, "Handbook on Flaw Evaluations for Byron Unit 1 and 2 Steam Generators and Pressurizers," earlier in plant life and are now performed consistent with those in WCAP-12046, "Handbook on Flaw Evaluations for the Byron and Braidwood Units 1 and 2 Reactor Vessels." LRA Section 4.7.8 further states that the handbooks for flaw evaluation methodology are based on crack growth rate analyses using the design-based transients as inputs for each of the evaluated components to provide crack

growth rate reference curves. Since the flaw evaluation handbooks are based on analyses that have time-limited inputs (e.g., number of design transient cycles assumed over 40 years), these analyses supporting flaw evaluations of primary system components have been identified as TLAAs.

The LRA also states that the numbers of transients used to develop the crack growth rate reference curves bound the numbers of transients in the 60-year projections provided in LRA Section 4.3.1.

The applicant dispositioned this TLAA for Byron and Braidwood, Units 1 and 2, in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions will be adequately managed by the Fatigue Monitoring (B.3.1.1) program, for the period of extended operation.

<u>Fracture Toughness Input to Analyses—Irradiation Embrittlement of Reactor Vessel Beltline and Extended Beltline Components</u>. LRA Section 4.7.8 states that Byron and Braidwood performed flaw evaluations on reactor vessels consistent with ASME Code, Section XI, Subarticle IWB-3600. The LRA further states the flaw evaluations were performed consistently with the methodologies in WCAP-11063 earlier in plant life and are now performed consistent with those in WCAP-12046. LRA Section 4.7.8 further states that these methodologies are based on analyses which use fracture toughness as an input. The applicant stated that the loss of fracture toughness occurs in the portions of the reactor vessel exposed to neutron irradiation embrittlement over the life of the reactor vessel, and therefore the analyses that use fracture toughness as an input supporting the flaw evaluations have been identified as TLAAs.

The applicant dispositioned this TLAA for Byron and Braidwood, Units 1 and 2, in accordance with 10 CFR 54.21(c)(1)(i), such that the ASME Section XI analyses supporting the flaw evaluations for the reactor vessel remain valid during the period of extended operation.

4.7.8.2 Staff Evaluation

The LRA provides two separate analyses to support flaw evaluations of primary system components. One analysis uses transient counts as the time-dependent parameter, whereas the second analysis uses neutron fluence as the time-dependent parameter. As described above, the applicant dispositioned these analyses as two separate TLAAs in accordance with 10 CFR 54.21(c).

<u>Fatigue Crack Growth Analysis</u>. The staff reviewed LRA Section 4.7.8 for the applicant's TLAA for fatigue crack growth analyses for Byron and Braidwood, Units 1 and 2, to verify, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

The staff reviewed the applicant's TLAA and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.7.3.1.3, which state that the applicant proposes to manage the aging effects associated with the TLAA by an AMP in the same manner as described in the IPA in 10 CFR 54.21(a)(3). The SRP-LR also states that the staff reviews the applicant's AMP to verify that the effects of aging on the intended function(s) are adequately managed consistent with the CLB for the period of extended operation. In addition, the SRP-LR requires that a license renewal applicant must identify the SCs associated with the TLAA.

The staff concluded that the methodology in WCAP-11063 and WCAP-12046 uses inputs such as: flaw location, initial size, and the final acceptable size; the base material; and the number of design transient cycles to calculate conservative flaw growth on reactor vessel, pressurizer, primary steam generator subcomponents and primary coolant components. These provide crack growth rate reference curves, which are based on the above factors and that the crack growth rate reference curves provide simplified conclusions as to whether the flaw will propagate to an unacceptable size in 10, 20, 30, or 40 years. The staff also noticed that each of the flaw evaluations used one or more of these curves to demonstrate that flaws will not propagate to unacceptable sizes prior to 40 years.

The staff found that the applicant will use the Fatigue Monitoring program to ensure that the numbers of transients used in these curves will not be exceeded during the period of extended operation. However, it was unclear to the staff which transients were used in the flaw evaluation methodology to confirm that the transients are within the scope of the Fatigue Monitoring program. By letter dated February 6, 2014, the staff issued RAI 4.7.8-1, requesting that the applicant provide the transients that support the ASME Section XI crack growth analyses.

By letter dated March 10, 2014, the applicant responded to RAI 4.7.8-1. The applicant provided the thermal and pressure transients and the number of design transient cycles assumed in the flaw evaluations that support the ASME Section XI crack growth analyses. The applicant stated that these transients are monitored by the Fatigue Monitoring program. The applicant further stated that one of the transients assumed in the flaw evaluation analyses uses a design transient cycle value that is more limiting than the CLB Cycle Limit presented in LRA Tables 4.3.1-1 and 4.3.1-4. The CLB Cycle Limit in LRA Tables 4.3.1-1 and 4.3.1-4 for the Excessive Bypass Feedwater Flow transient is 40 cycles. However, the applicant identified that the steam generator flaw evaluation assumes a design transient cycle limit of 30. The applicant stated that LRA Tables 4.3.1-1 and 4.3.1-4 have been updated to reflect the more conservative cycle limit of 30 cycles for the Excessive Bypass Feedwater Flow transient. The staff confirmed that the transients listed in the RAI response are included in LRA Tables 4.3.1-1 and 4.3.1-4. The staff finds the applicant’s response acceptable because the design transients assumed for the flaw evaluations are included in LRA Section 4.3 tables and will be monitored by the Fatigue Monitoring program, and also because the applicant updated LRA Tables 4.3.1-1 and 4.3.1-4 to reflect the conservative cycle limits for the transients. The staff’s concern in RAI 4.7.8-1 is resolved.

The staff also noticed that the LRA Section 4.7.8 states that the fatigue crack growth analyses are pre-emptive. However, it is not evident whether these analyses were performed in evaluation of actual flaws detected in Class 1 components at the plant or in evaluation of flaws that were assumed to occur in the components. Specifically, the applicant did not clearly identify which reactor pressurize vessel (RPV), steam generator, pressurizer, or RCPB piping components had contained flaws and were analyzed in accordance with the generic flaw evaluation methodology in both WCAP-11063 or WCAP-12046. By letter dated August 20, 2014, the staff issued RAI 4.7.8-2, requesting that the applicant describe the RPV, steam generator, pressurizer, and RCPB flaws that were evaluated with the flaw evaluation criteria in WCAP-11063 or in WCAP-12046 and to identify the NRC safety evaluation references for the approval of these flaws.

By letter dated September 5, 2014, the applicant responded to RAI 4.7.8-2. The applicant stated that it reviewed its OE and its regulatory correspondence database to identify flaws at Byron and Braidwood Station that were evaluated with the methodology and acceptance criteria in WCAP-11063 or in WCAP-12046. The applicant provided a table of the results of its review, which included references to the applicable NRC safety evaluation. The staff reviewed the results provided by the applicant and confirmed that the flaws at BBS were evaluated consistent with the methodology and acceptance criteria of the two WCAP reports. The staff finds the applicant's response acceptable because the applicant confirmed that the analyses described in LRA Section 4.7.8 were used to provide a safety determination for actual flaws identified at BBS. Because the analyses were performed on actual flaws, the staff finds acceptable for the applicant to use its Fatigue Monitoring program to monitor the transient cycles assumed in the analyses, such that the TLAA evaluation of the primary system fatigue crack growth analyses is consistent with 10 CFR 54.21(c)(1)(iii). The staff's concern in RAI 4.7.8-2 is resolved.

The staff finds the applicant demonstrated, pursuant to 10 CFR 54.12(c)(1)(iii), that the impacts of fatigue crack growth on the intended RCPB function of the components analyzed for in the flaw evaluations will be adequately managed for the period of extended operation.

Additionally, the TLAA meets the acceptance criteria in SRP-LR Section 4.7.3.1.3 because the ASME Section XI flaw evaluations will be managed by the Fatigue Monitoring program, which will monitor the transient cycles and require corrective action prior to exceeding the numbers of transient cycles used in the evaluations which support these conclusions.

Fracture Toughness Input to Analyses—Irradiation Embrittlement of Reactor Vessel Beltline and Extended Beltline Components. The staff reviewed LRA Section 4.7.8 and the TLAA for the flaw evaluations of the reactor vessel to confirm, pursuant to 10 CFR 54.21(c)(1)(i), that the analyses remain valid for the period of extended operation. For the flaw evaluations that evaluate the impact of increasing neutron fluence on the fracture toughness values used in the analyses, the staff reviewed the TLAA to demonstrate that the treatment of fracture toughness values used in the analyses will remain valid for the period of extended operation.

The staff reviewed the applicant's TLAA and the corresponding disposition consistent with the review procedures in SRP-LR Section 4.7.3.1.1, which state that the applicant justifies that the existing analyses remain valid and bounding for the period of extended operation. The SRP-LR further states that the applicant should show that relevant aging effects for the period of extended operation are already addressed in the conditions and assumptions in the analyses. The SRP-LR also states that the applicant should show that the acceptance criteria of the analyses are maintained to provide reasonable assurance that the intended function(s) is maintained for renewal.

The applicant stated that the methodology in WCAP-12046 shows that the flaw evaluation charts for the active beltline region are valid for RT_{NDT} less than 200 °F. LRA Section 4.2 describes the applicant's TLAA on reactor vessel neutron embrittlement analysis and provides the projected RT_{NDT} for Byron and Braidwood at 57 EFPY. The applicant stated that the projected RT_{NDT} for the RPV beltline components at 57 EFPY are less than 200 °F, and therefore, the flaw evaluation charts in WCAP-12046 are still applicable for the period of extended operation.

The staff noticed that flaw evaluations performed in accordance with ASME Section XI, Subsection IWB-3600, Appendix C, calculate K_{Ic} and K_{Ia} stress intensity factors as a function of a component's ART (end-of-life RT_{NDT} or RT_{PTS} values), which are fluence-dependent. The staff reviewed the information provided in LRA Section 4.2 and confirmed that the projected RT_{NDT} are less than 200 °F at the end of the period of extended operation for all of the beltline region materials of BBS. The staff noticed that the flaw evaluations performed in accordance with WCAP-12046 assumed a 200 °F end-of-life RT_{NDT} value for the components in the evaluations.

The staff's evaluation of the RT_{NDT} calculations and LRA Section 4.2 is documented in SER Section 4.2. The staff finds the applicant's justification for the active beltline acceptable because there is conservative margin between the projected RT_{NDT} values described in Section 4.2 and the limit of 200 °F that would invalidate the flaw evaluation charts in WCAP-12046.

The applicant stated that the next material below the active beltline in the reactor vessel is the lower shell to bottom head ring circumferential weld. The applicant stated that calculated fluence on the weld is approximately 4×10^{15} n/cm^2, which would result in an increase in RT_{PTS} of 2 °F based on conservative calculations. The applicant stated that the effects of fluence will continue to be negligible. Above the active beltline, the applicant stated that the increase in RT_{PTS} for 57 EFPY due to irradiation at the nozzle shell forgings and associated welds is approximately 10 to 20 °F. The applicant also states that the next material above the active beltline and the inlet and outlet nozzles is the vessel flange-to-nozzle shell forging circumferential weld. The applicant stated that the fluence in this weld is projected to be below 1×10^{17} n/cm^2, and therefore, the embrittlement effects are considered negligible. The applicant also states that, as provided in LRA Section 4.2.3, the highest value of RT_{PTS} from all the extended beltline regions is 90 °F. The applicant stated that, based on RT_{PTS} of 90 °F and the limiting temperature from the bounding transient, the calculated K_{Ic} and K_{Ia} (fracture toughness) value is greater than 200 ksi-in$^{1/2}$. The applicant stated that the flaw evaluation charts in WCAP-12046 remain valid for the period of extended operation because the flaw evaluation charts for the extended baseline region are determined based on an upper-shelf limit of 200 ksi-in$^{1/2}$.

The staff also reviewed the RT_{PTS} calculations for the extended beltline regions. The staff finds the applicant's justification acceptable because, for the material below the active beltline, the change in RT_{PTS} would be negligible based on the calculated fluence on the material. For the material above the active beltline, the staff finds the justification acceptable because calculated K_{Ic} and K_{Ia} based on the most bounding conditions and inputs would result in a value greater than the limit of 200 ksi-in$^{1/2}$ that would invalidate the flaw evaluation charts in WCAP-12046.

The staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the existing analyses are valid for the period of extended operation because the existing analyses utilize fracture toughness values that bound those projected for the end of the period of operation. Additionally, the TLAA meets the acceptance criteria in SRP-LR Section 4.7.3.1.1 because the existing analyses will remain valid for the period of extended operation.

4.7.8.3 UFSAR Supplement

LRA Section A.4.7.8 provides the UFSAR supplement summarizing the analyses supporting flaw evaluations of primary system components. The staff reviewed LRA Section A.4.7.8 consistent with the review procedures in SRP-LR Section 4.7.3.2, which state that the information to be included in the UFSAR supplement should include a summary description of the evaluation of the metal fatigue TLAA.

Based on its review of the UFSAR supplement, the staff finds that the applicant meets the acceptance criteria in SRP-LR Section 4.7.2.2 for the TLAA evaluation of the primary system fatigue crack growth analyses. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the TLAA evaluation of the primary system fatigue crack growth analyses, as required by 10 CFR 54.21(d).

Based on its review of the UFSAR supplement, the staff finds that the applicant meets the acceptance criteria in SRP-LR Section 4.7.2.2 for the TLAA evaluation for the loss of fracture toughness input to the flaw evaluations. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the TLAA evaluation for the loss of fracture toughness input to the flaw evaluations, as required by 10 CFR 54.21(d).

4.7.8.4 Conclusion

On the basis of its review, the staff concludes that the applicant acceptably demonstrated, in accordance with 10 CFR 54.21(c)(1)(iii), that the primary system fatigue crack growth analyses will be adequately managed by the Fatigue Monitoring program for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation of the primary system fatigue crack growth analyses, as required by 10 CFR 54.21(d).

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the loss of fracture toughness input to the flaw evaluations 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 for the loss of fracture toughness input to the flaw evaluations, as required by 10 CFR 54.21(d).

4.8 Conclusion

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

- The TLAAs will remain valid for the period of extended operation, as required by 10 CFR 54.21(c)(1)(i).
- The TLAAs have been projected to the end of the period of extended operation, as required by 10 CFR 54.21(c)(1)(ii).
- The effects of aging on the intended functions will be adequately managed for the period of extended operation, as required by 10 CFR 54.21(c)(1)(iii).

The staff also reviewed the UFSAR supplement for the TLAAs and finds that the supplement contains descriptions of the TLAAs sufficient to satisfy the requirements of 10 CFR 54.21(d). In addition, the staff concludes, as required by 10 CFR 54.21(c)(2), that no plant-specific, TLAA-based exemptions are in effect.

With regard to these matters, the staff concludes that there is reasonable assurance that the activities authorized by the renewed licenses will continue to be conducted in accordance with the CLB. Additionally, any changes made to the CLB 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 U.S. Nuclear Regulatory Commission (NRC or the staff) issued its safety evaluation report (SER) with open items related to the renewal of the operating licenses for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 (BBS), on October 30, 2014. On December 3, 2014, the applicant presented its license renewal application (LRA), and the staff presented its review findings to the Advisory Committee on Reactor Safeguards (ACRS) Plant License Renewal Subcommittee. The staff reviewed the applicant's comments on the SER and completed its review of the LRA. The staff's evaluation is documented in an SER that was issued by letter dated July 6, 2015.

During the 627th meeting of the ACRS held September 9-12, 2015, the ACRS completed its review of the BBS LRA and the staff's SER. The ACRS documented its findings in a letter to the Commission dated September 21, 2015. 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

September 21, 2015

The Honorable Stephen G. Burns
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR BYRON STATION UNITS 1 AND 2 AND BRAIDWOOD STATION UNITS 1 AND 2

Dear Chairman Burns:

During the 627th meeting of the Advisory Committee on Reactor Safeguards (ACRS), September 9-12, 2015, we completed our review of the license renewal application for Byron Station Units 1 and 2 and Braidwood Station Units 1 and 2 and the final Safety Evaluation Report (SER) prepared by the NRC staff. Our Subcommittee on Plant License Renewal reviewed this matter during a meeting on December 3, 2014. During these reviews, we had the benefit of discussions with representatives of the NRC staff and Exelon Generation Company, LLC (Exelon, 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 established programs and commitments by Exelon to manage age-related degradation provide reasonable assurance that Byron Units 1 and 2 and Braidwood Units 1 and 2 can be operated in accordance with their current licensing bases for the period of extended operation without undue risk to the health and safety of the public.

2. Exelon's application for renewal of the operating licenses for Byron Units 1 and 2 and Braidwood Units 1 and 2 should be approved.

BACKGROUND

This unique license renewal action consists of a combined review of two dual-unit nuclear power plants at two different locations approximately 100 miles apart. Each unit has similar nuclear steam supply systems and safety systems, but there are some site-specific differences in their balance of plant design features. Byron is located in north central Illinois, near the town of Byron, Illinois, and near the Rock River, approximately 95 miles northwest of Chicago, Illinois. Braidwood is located in northeastern Illinois, near the town of Braidwood, Illinois, and near the Kankakee River, approximately 60 miles southwest of Chicago, Illinois.

The NRC issued the Byron construction permit on December 31, 1975, and operating licenses on February 14, 1985 (Unit 1), and January 30, 1987 (Unit 2). The NRC issued the Braidwood construction permit on December 31, 1975, and operating licenses on July 2, 1987 (Unit 1), and May 20, 1988 (Unit 2). Each unit utilizes a Westinghouse four-loop pressurized water reactor (PWR) with a dry ambient containment. Sargent & Lundy was the architect-engineer for both stations.

Each unit was originally licensed for a power output of 3,600 MWt and has a current safety evaluation for 3,658 MWt. On June 23, 2011, Exelon requested an increase in licensed power for Byron and Braidwood Stations, Units 1 and 2, from 3,587 MWt to 3,645 MWt based on measurement uncertainty recapture. That power level increase was approved in 2013. The current licensed power output for each unit is about 3,645 MWt with a gross electrical output of approximately 1,260 MWe.

In this application, Exelon requests renewal of the operating licenses for Byron Units 1 and 2 and Braidwood Units 1 and 2 for a period of 20 years beyond the current expiration dates of midnight October 31, 2024 (Byron Unit 1), November 6, 2026 (Byron Unit 2), October 17, 2026 (Braidwood Unit 1), and December 18, 2027 (Braidwood Unit 2).

DISCUSSION

In the final SER, dated July 2015, the staff documented its review of the license renewal application and other information submitted by the applicant and obtained through staff audits and inspections at the plant sites. The staff reviewed the completeness of the identification of structures, systems, and components (SSCs) that are within the scope of license renewal; the integrated plant assessment process; the identification of plausible aging mechanisms associated with passive, long-lived components; the adequacy of the Aging Management Programs (AMPs); and identification and assessment of Time-Limited Aging Analyses (TLAAs) requiring review.

Exelon's license renewal application identified the SSCs that fall within the scope of license renewal. The application demonstrates consistency with the Generic Aging Lessons Learned (GALL) Report (NUREG-1801, Revision 2) or documents and justifies deviations to the specified approaches in that report. Exelon will implement 45 AMPs for license renewal for Byron and 44 AMPs for Braidwood. The AMPs consist of 32 existing programs and 13 new programs for Byron and 32 existing programs and 12 new programs for Braidwood.

For the 13 new AMPs at Byron, 11 of the new programs are consistent with the GALL Report. One of these programs is for fuse holders which is not applicable for Braidwood. Byron also has two additional new programs that are consistent with exceptions.

For the 12 new AMPs at Braidwood, 10 of the new programs are consistent with the GALL Report. Two additional new programs at Braidwood are consistent with exceptions.

Each plant has 32 existing programs. At Byron, 5 are consistent, 20 are consistent with enhancements, 1 is consistent with exceptions, and 6 are consistent with enhancements and exceptions. At Braidwood, 4 are consistent, 20 are consistent with enhancements, 1 is consistent with exceptions, and 7 are consistent with enhancements and exceptions. No AMPs are plant specific.

The license renewal application includes ten exceptions to the GALL Report for Braidwood and nine exceptions for Byron. The Flux Thimble Tube Inspection AMP exception does not apply to Byron Station. We reviewed all of the exceptions (Reactor Head Closure Stud Bolting, Water Chemistry, PWR Vessel Internals, Steam Generators, Compressed Air Monitoring, Fire Water System, Flux Thimble Tube Inspection, Aboveground Metallic Tanks, Buried and Underground Piping, and ASME Section XI Subsection IWF component support inspection). We conclude that all of the GALL exceptions are acceptable.

The staff conducted license renewal audits and performed license renewal inspections at both Byron and Braidwood. The audits verified the appropriateness of the scoping and screening methodology for AMPs, the appropriateness of the aging management review, and the acceptability of the TLAAs. The inspections verified that the license renewal requirements will be implemented appropriately. The inspections, and the reports of those inspections, are thorough. Based on the audits, the inspections, and the staff reviews related to this license renewal application, the staff concluded that the proposed activities will manage the effects of aging of the SSCs and that the intended functions of these SSCs will be maintained during the period of extended operation. The staff concluded that Exelon has demonstrated that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation at Byron and Braidwood, as required by 10 CFR 54.21(a)(3). We concur with that conclusion.

Two important remaining open items were resolved between our Subcommittee meeting on December 3, 2014 and our final review. These items are control rod drive mechanism (CRDM) nozzle wear and environmentally assisted fatigue in Class 1 components.

CRDM Nozzle Wear

Exelon submitted an amendment to its license renewal application that identifies an inspection program for aging management of CRDM nozzle wear. The amendment indicated that the inspection program will be used prior to, and during, the period of extended operation to monitor the nozzle wear. By letter dated February 11, 2015, Exelon revised the application as proposed and provided detailed nondestructive examination procedures that it will implement to manage the CRDM nozzle wear. This commitment in the amendment resolves this open item.

Environmentally Assisted Fatigue (EAF) in Class 1 Components

Exelon compared components of various materials in their EAF evaluations. During the review, it was determined that the environmentally adjusted cumulative usage factor (CUF_{en}) value of different materials may respond differently when the EAF analyses are being refined in the future. It was determined that the initial review did not demonstrate that the refinement of the higher CUF_{en} of one material would ensure the reduction of CUF_{en} values for another material within the same transient section such that the selected leading location would remain appropriate and bounding. Exelon subsequently amended its commitments for the locations at Byron and Braidwood Stations, Units 1 and 2, that will be monitored for EAF in the period of extended operation. With the inclusion of three additional locations at each unit, the staff concluded that there is reasonable assurance that the bounding locations susceptible to EAF will be monitored. This commitment resolves this open item.

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 licenses for Byron and Braidwood. The established programs and commitments by Exelon provide reasonable assurance that Byron and Braidwood can be operated in accordance with their current licensing basis for the period of extended operation without undue risk to the health and safety of the public. The Exelon application for renewal of the operating licenses for Byron and Braidwood should be approved.

Sincerely,

/RA/

John W. Stetkar
Chairman

REFERENCES

1. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2," July 2015 (ML15182A051).

2. Exelon Generation Company, LLC, "Byron and Braidwood Stations License Renewal Application," May 29, 2013 (ML13155A387).

3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report with Open Items Related to the License Renewal of Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2," October 2014 (ML14296A176).

4. U.S. Nuclear Regulatory Commission, "Braidwood Station, Units 1 and 2 NRC License Renewal Scoping, Screening, and Aging Management Inspection Report 05000456/2014009; 05000457/2014009," November 7, 2014 (ML14311A893).

5. U.S. Nuclear Regulatory Commission, "Byron Station Units 1 and 2 – NRC License Renewal Scoping, Screening, and Aging Management Inspection Report 05000454/2014008; 05000455/2014008," November 7, 2014 (ML14311871).

6. U.S. Nuclear Regulatory Commission, "Aging Management Programs Audit Report Regarding the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2," March 13, 2014 (ML14071A620).

7. U.S. Nuclear Regulatory Commission, “Scoping and Screening Methodology Audit Report Regarding the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2,” May 14, 2014 (ML14050A304).

8. U.S. Nuclear Regulatory Commission, NUREG 1801, Revision 2, “Generic Aging Lessons Learned (GALL) Report,” December 2010 (ML103409041).

9. U.S. Nuclear Regulatory Commission, NUREG-1800, Revision 2, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” December 2010 (ML103409036).

10. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.188, Revision 1, “Standard Format and Content for Application to Renew Nuclear Power Plant Operating Licenses,” September 2005 (ML082950585).

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SECTION 6

CONCLUSION

The staff of the United States (U.S.) Nuclear Regulatory Commission (NRC) (the staff) reviewed the license renewal application (LRA) for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, in accordance with NRC regulations and NUREG-1800, Revision 2, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," dated December 2010. Title 10 of the *Code of Federal Regulations* (10 CFR) 54.29, "Standards for Issuance of a Renewed License," sets the standards used for issuing 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 any requirements of 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," Subpart A, "National Environmental Policy Act—Regulations Implementing Section 102(2)," will be documented in separate supplements for Byron and Braidwood to NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants (GEIS)."

APPENDIX A

BYRON STATION, UNITS 1 AND 2, AND BRAIDWOOD STATION, UNITS 1 AND 2, LICENSE RENEWAL COMMITMENTS

During the review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, (BBS) license renewal application (LRA) by the staff of the U.S. Nuclear Regulatory Commission (NRC or the staff), Exelon Generation Company, LLC, made commitments related to aging management programs (AMPs) to manage aging effects of structures and components.

LRA Section A.1.0, “Introduction,” states “The application for a renewed operating license is required by 10 CFR 54.21(d) to include a FSAR Supplement. This Appendix [as revised by supplements, amendments, and RAI responses], which includes the following sections, comprises the FSAR supplement….” It also states “…Section A.5 contains the License Renewal Commitment List.” Therefore LRA Appendix A, as revised by supplements, amendments, and RAI responses, is considered to be the updated final safety analysis report (UFSAR) supplement as discussed in two proposed license conditions in SER Section 1.7, “Summary of Proposed License Conditions.”

The following table lists the commitments, as well as the implementation schedules and the sources for each commitment, as agreed to by the applicant and by the staff.

Explanatory notes (e.g., “Note 1”) within this table provide the basis for station-specific differences as follows:

- Note 1 – Enhancement at one Station only; other Station currently performs activity
- Note 2 – Design difference
- Note 3 – Enhancement due to operating experience

Implementation schedules for Byron, Unit 1 and Unit 2, and Braidwood, Unit 1 and Unit 2, differ according to the start of the respective period of extended operation for each unit. The dates for the start of these respective periods of extended operation for the Byron and Braidwood Units are as follows below and apply to the “Implementation Schedule” column in the table:

- Byron Unit 1, October 31, 2024
- Byron Unit 2, November 6, 2026
- Braidwood Unit 1, October 17, 2026
- Braidwood Unit 2, December 18, 2027

The commitment implementation schedules in this table, as discussed in SER Section 1.7, reflect the applicant’s response to RAI A.1-1 by letter RS-14-216 dated December 15, 2014, and allow time for NRC inspection of commitment implementation prior to a unit’s entry into its respective period of extended operation.

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
1	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD is an existing program that will be enhanced to: 1. Conduct a visual inspection of the accessible portions of the ASME Class 2 reactor vessel flange leakage monitoring tube every other refueling outage. 2. Perform nondestructive examination of the five (5) centermost control rod drive mechanism (CRDM) housing penetrations to determine the thermal sleeve centering tab wear depth on the CRDM housing penetration inner diameter wall. On each unit, these CRDM housings will be examined at least once during the 10-year period prior to the period of extended operation, and on a 10-year frequency during the period of extended operation.	A.2.1.1	Program to be enhanced no later than 6 months prior to the period of extended operation. Inspections prior to the period of extended operation specified in Enhancement 2 will be completed either no later than 6 months prior to the period of extended operation, or before the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	LRA Exelon letter RS-15-067 02/11/2015
2	Existing Water Chemistry program is credited.	A.2.1.2	Ongoing	LRA
3	Reactor Head Closure Stud Bolting is an existing program that will be enhanced to: 1. Revise the procurement requirements for reactor head closure stud material to assure that the maximum yield strength of replacement material is limited to a measured yield strength less than 150 ksi.	A.2.1.3	Program to be enhanced no later than six months prior to the period of extended operation.	LRA Exelon letter RS-13-247 11/5/2013 Exelon letter RS-13-285 12/19/2013
4	Existing Boric Acid Corrosion program is credited.	A.2.1.4	Ongoing	LRA
5	Existing Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components program is credited.	A.2.1.5	Ongoing	LRA

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
6	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) is a new program that manages the aging effects of loss of fracture toughness due to thermal aging embrittlement of ASME Code Class 1 CASS components with service conditions above 250 °C (482 °F). The program will include a screening methodology to determine component susceptibility to thermal aging embrittlement based on casting method, molybdenum content, and percent ferrite. For "potentially susceptible" components, thermal aging embrittlement management will be accomplished through either, qualified visual inspections, such as enhanced visual examination, qualified ultrasonic testing methodology, or component-specific flaw tolerance evaluation.	A.2.1.6	Program to be implemented no later than six months prior to the period of extended operation.	LRA
7	The PWR Vessel Internals is a new program that manages the aging effects of various forms of cracking, including stress-corrosion cracking (SCC), primary water stress corrosion cracking (PWSCC), irradiation-assisted stress-corrosion cracking (IASCC), or cracking due to fatigue/cyclical loading; loss of material due to wear; loss of fracture toughness due to neutron irradiation embrittlement; changes in dimension due to void swelling and irradiation growth; and loss of preload due to thermal and irradiation-enhanced stress relaxation or creep. Program examination methods include visual examination, enhanced visual examination, volumetric examination, and direct physical measurements.	A.2.1.7	Program to be implemented no later than the date that the renewed operating licenses are issued.	LRA
8	The Flow-Accelerated Corrosion aging management program is an existing program that will be enhanced to: 1. Revise program procedures to require the documentation of the validation and verification of updated vendor supplied Flow-Accelerated Corrosion Program software which calculates component wear, wear rates, remaining life, and next scheduled inspection. The validation and verification will verify that the updated software performs these calculations consistently with NSAC-202L-R3 guidelines.	A.2.1.8	Program to be enhanced no later than six months prior to the period of extended operation.	LRA Exelon letter RS-14-143 5/15/ 2014

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
9	Bolting Integrity is an existing program that will be enhanced to: 1. Prohibit the use of lubricants containing molybdenum disulfide on pressure retaining bolted joints. 2. Prohibit the use of high strength bolting (actual measured yield strength equal to or greater than 150 ksi) for pressure retaining bolted joints in portions of systems within the scope of the Bolting Integrity program. 3. Perform visual inspection of submerged bolting on fire protection system pumps (Byron only) (Note 1) and well water system deep well pumps (Byron only) (Note 2) when submerged portions of the pumps are overhauled or replaced during maintenance activities.	A.2.1.9	Program to be enhanced no later than six months prior to the period of extended operation.	LRA

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
10	Steam Generators is an existing program that will be enhanced to: 1. Validate that PWSCC of the divider plate welds to the primary head and tubesheet cladding is not occurring. BBS commits to perform one (1) of the following three (3) resolution options for Units 1 and 2: <u>Option 1: Inspection</u> Perform a one-time inspection, under the Steam Generators program, of each steam generator to assess the condition of the divider plate welds and the effectiveness of the Water Chemistry (A.2.1.2) program. For the Byron and Braidwood, Unit 1 steam generators which were replaced in 1998, the inspection will be performed between 2018 and either no later than 6 months prior to the start of the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later, to allow the steam generators to acquire at least 20 years of service. For the Byron and Braidwood, Unit 2 steam generators which currently have at least 20 years of service, the inspection will be performed prior to entering the period of extended operation. The examination technique(s) will be capable of detecting PWSCC in the divider plate assemblies and associated welds. Or <u>Option 2: Analysis</u> Perform an analytical evaluation of the steam generator divider plate welds in order to establish a technical basis which concludes that the steam generator RCPB is adequately maintained with the presence of steam generator divider plate weld cracking. The analytical evaluation will be submitted to the NRC for review and approval two (2) years prior to entering the associated period of extended operation.	A.2.1.10	Program to be enhanced no later than 6 months prior to the period of extended operation. Schedule for inspection and analysis activities identified in Commitment.	LRA Exelon Letter RS-14-052 03/04/2014

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
10 (cont.)	Or <u>Option 3: Industry/NRC Studies</u> If results of industry and NRC studies and operating experience (OE) document that potential failure of the steam generator RCPB due to PWSCC of the steam generator divider plate welds is not a credible concern, this commitment will be revised to reflect that conclusion. 2. Validate that PWSCC of the tube-to-tubesheet welds is not occurring on BBS Unit 1. BBS commit to perform one (1) of the following three (3) resolution options for Unit 1: <u>Option 1: Inspection</u> Perform a one-time inspection, under the Steam Generators (A.2.1.10) program, of a representative number of tube-to-tubesheet welds in each steam generator to determine if PWSCC cracking is present. Since the Byron and Braidwood Unit 1 steam generators were replaced in 1998, the inspection will be performed between 2018 and either no later than 6 months prior to the start of the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later, to allow the steam generators to acquire at least 20 years of service. The examination technique(s) will be capable of detecting PWSCC in the tube-to-tubesheet welds. If cracking is identified, the condition will be resolved through repair or engineering evaluation to justify continued service, as appropriate, and a periodic monitoring program will be established to perform routine tube-to-tubesheet weld inspections for the remaining life of the steam generators.			

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
10 (cont.)	Or <u>Option 2: Analysis - Susceptibility</u> Perform an analytical evaluation of the steam generator tube-to-tubesheet welds to determine that the welds are not susceptible to PWSCC. The evaluation for determining that the tube-to-tubesheet welds are not susceptible to PWSCC will be submitted to the NRC for review and approval two (2) years prior to entering the associated period of extended operation. Or <u>Option 3: Analysis – Pressure Boundary</u> Perform an analytical evaluation of the steam generator tube-to-tubesheet welds redefining the RCPB of the tubes, where the steam generator tube-to-tubesheet welds are not required to perform an RCPB function. The redefinition of the RCPB will be submitted to the NRC for review and approval two (2) years prior to entering the associated period of extended operation.			

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
11	Open-Cycle Cooling Water System is an existing program that will be enhanced to: 1. Perform periodic volumetric inspections for loss of material in the non-essential service water system piping at a minimum of two (2) locations on each unit in both the auxiliary building and the turbine building for a total of four (4) periodic inspections per unit every refueling cycle. 2. Require inspections of internal coatings be performed by coating inspectors certified to ANSI N45.2.6 or ASTM Standards endorsed in Regulatory Guide (RG) 1.54. 3. Specify that signs of peeling, blistering, or delamination of the coating from the base metal, if identified, shall be entered into the corrective action program (CAP). 4. Require physical testing of internal coatings, where physically possible, to ensure that remaining coating is tightly bonded to the base metal when peeling, blistering, or delamination is detected and the coating is not repaired or replaced. The testing will consist of adhesion testing using ASTM international standards endorsed in RG 1.54 (e.g., ASTM D4541-09 or ASTM D6677-07). 5. Require that evaluations utilized to return a coated component exhibiting signs of peeling, blistering, or delamination to service without repairing or replacing the coating shall consider the potential impact on the intended function of the system. This evaluation shall include consideration of the potential for degraded performance of downstream components due to flow blockage and loss of material of the coated component. 6. Require the as-left condition of a coating that exhibited signs of peeling, blistering, or delamination and that is not repaired or replaced is such that the potential for further degradation of the coating is minimized.	A.2.1.11	Program to be enhanced no later than six months prior to the period of extended operation.	LRA Exelon letter RS-14-124 05/05/2014 Exelon letter RS-14-175 06/30/2014

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
12	Closed Treated Water Systems is an existing program that will be enhanced to: 1. Perform condition monitoring, including periodic visual inspections and NDEs, to verify the effectiveness of water chemistry control at mitigating aging effects. A representative sample of piping and components will be selected based on likelihood of corrosion, fouling, or cracking and inspected at an interval not to exceed once in 10 years during the period of extended operation. The selection of components to be inspected will focus on locations which are most susceptible to age-related degradation, where practical. 2. Perform periodic sampling, analysis, and trending of water chemistry for the essential service water makeup pump engine glycol-based jacket water system to verify the effectiveness of water chemistry control at mitigating aging effects (Byron only) (Note 2).	A.2.1.12	Program to be enhanced no later than six months prior to the period of extended operation.	LRA
13	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems is an existing program that will be enhanced to: 1. Consistently include inspections of structural components and bolting for loss of material due to corrosion, rails for loss of material due to wear and corrosion, and bolted connections for evidence of loss of preload. 2. Ensure periodic inspections are performed on all cranes, hoists, monorails, and rigging beams within the scope of license renewal, including those that are infrequently in use.	A.2.1.13	Program to be enhanced no later than six months prior to the period of extended operation.	LRA
14	Compressed Air Monitoring is an existing program that will be enhanced to: 1. Inspect critical component internal surfaces for signs of loss of material due to corrosion and document deficiencies in the CAP.	A.2.1.14	Program to be enhanced no later than six months prior to the period of extended operation.	LRA

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
15	Fire Protection is an existing program that will be enhanced to: 1. Include visual inspections of the earthen berm enclosing the outdoor fuel oil storage tanks for signs of age-related degradation such as loss of material and loss of form that could affect the intended function of the berm. 2. Provide additional inspection guidance to identify age-related degradation of fire barrier walls, ceilings, and floors or aging effects such as cracking, spalling, and loss of material. 3. Include visual inspection of halon and low-pressure carbon dioxide fire suppression system piping and component external surfaces for signs of corrosion or other age-related degradation.	A.2.1.15	Program to be enhanced no later than six months prior to the period of extended operation.	LRA

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
16	Fire Water System is an existing program that will be enhanced to: 1. Replace sprinkler heads or perform 50-year sprinkler head testing using the guidance of National Fire Protection Association (NFPA) 25 “Standard for the Inspection, Testing and Maintenance of Water-Based Fire Protection Systems” (2002 Edition), Section 5.3.1.1.1. This testing will be performed at the 50-year inservice date and every 10 years thereafter. 2. Provide for chemical addition accompanied with system flushing to allow for adequate dispersal of the chemicals throughout the system, to prevent or minimize microbiologically induced corrosion (Byron only) (Note 3). 3. Perform main drain testing annually, in accordance with NFPA 25, “Standard for the Inspection, Testing and Maintenance of Water-Based Fire Protection Systems,” Section 13.2.5. 4. Perform air flow testing of deluge systems that are not subject to periodic full flow testing on a three (3) year frequency to verify that internal flow blockage is not occurring (Byron only) (Note 1). 5. Perform inspections of Fire Protection System strainers when the system is reset after automatic actuation for signs of internal flow blockage (e.g., buildup of corrosion particles) (Braidwood only) (Note 1). 6. Increase the frequency of visual inspections of the internal surface of the foam concentrate tanks to at least once every ten (10) years. At least one (1) inspection will be performed within the ten (10) year period prior to entry into the period of extended operation, with subsequent inspections performed every ten (10) years thereafter.	A.2.1.16	Program to be enhanced no later than six months prior to the period of extended operation. Pre–period of extended operation activities specified in Enhancements 6 and 8 will be completed either no later than 6 months prior to the period of extended operation, or before the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	LRA Exelon letter RS-14-078 03/13/2014 Exelon letter RS-14-169 06/16/2014 Exelon letter RS-14-175 06/30/2014 Exelon letter RS-14-235 08/29/2014

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
16 (cont.)	7. Perform radiographic testing or internal visual inspections every five (5) years at the end of one (1) fire main and the end of one (1) sprinkler system branch line in half of the wet pipe sprinkler system within the scope of license renewal. If internal flow blockage that could result in failure of the system to deliver the required flow is identified, then perform an obstruction investigation. 8. Perform augmented testing beyond that specified in NFPA 25 on those portions of the water-based fire protection system that are: (a) normally dry but periodically subjected to flow and (b) cannot be drained or allow water to collect. The augmented testing will include: (1) periodic full flow tests at the design pressure and flow rate or internal visual inspections and (2) volumetric wall-thickness examinations. Inspections and testing will commence five (5) years prior to the period of extended operation and will be conducted on a five (5)-year frequency thereafter. 9. Perform a minimum of 30 volumetric examinations of Fire Protection System piping, using radiographic testing or UT, during each three year interval. If volumetric examinations over a 10-year interval do not identify three (3) or more areas exhibiting reduction in wall thickness greater than 50 percent, then this minimum sample size is no longer required. (Byron only) (Note 3). 10. Require inspections of internal coatings be performed by coating inspectors certified to ANSI N45.2.6 or ASTM Standards endorsed in RG 1.54. 11. Specify that signs of peeling, blistering, or delamination of the coating from the base metal, if identified, shall be entered into the CAP.			

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
16 (cont.)	12. Require physical testing of internal coatings, where physically possible, to ensure that remaining coating is tightly bonded to the base metal when peeling, blistering, or delamination is detected and the coating is not repaired or replaced. The testing will consist of adhesion testing using ASTM International standards endorsed in RG 1.54 (e.g., ASTM D4541-09 or ASTM D6677-07). 13. Require that evaluations utilized to return a coated component exhibiting signs of peeling, blistering, or delamination to service without repairing or replacing the coating shall consider the potential impact on the intended function of the system. This evaluation shall include consideration of the potential for degraded performance of downstream components due to flow blockage and loss of material of the coated component. 14. Require the as-left condition of a coating that exhibited signs of peeling, blistering, or delamination and that is not repaired or replaced is such that the potential for further degradation of the coating is minimized. 15. Perform a minimum of 25 volumetric examinations of Fire Protection System piping, using radiographic testing or UT, during each 10-year interval.			

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
17	Aboveground Metallic Tanks is a new program that manages aging effects of loss of material and cracking on the external surfaces of aboveground metallic tanks within the scope of license renewal by performing periodic visual inspections once per eighteen (18) month operating cycle for degradation of the external surface of the insulation lagging, flashing, roof, and accessible sealant. The program also requires periodic visual inspections and liquid penetrant examinations of the tank external surfaces at 25 locations for both tanks combined per site and includes, on a sampling basis, removal of selected tank lagging and insulation to permit inspections of the external tank surfaces and exposed sealants. The tank external surface inspections and examinations will be performed each 10-year period starting 10 years prior to the period of extended operation. The sample locations will include at least four locations below penetrations through the insulation and its jacketing (e.g., instrument nozzles, tank heaters, ladder). The remaining sample locations will be distributed such that inspections will occur on the tank dome, sides, and near the bottom. One-time tank bottom ultrasonic inspections (one CST per station) will be performed within the 5-year period prior to the period of extended operation. The cathodic protection availability and effectiveness criteria in LR-ISG-2011-03 Table 4c, notes 3.ii and 3.iii, respectively, will be required to be met commencing 5 years prior to the period of extended operation and during the period of extended operation.	A.2.1.17	Program to be implemented no later than 6 months prior to the period of extended operation. The pre–period of extended operation inspection activities specified in the commitment will be completed either no later than 6 months prior to the period of extended operation, or before the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	LRA Exelon letter RS-14-003 1/13/2014

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
18	Fuel Oil Chemistry is an existing program that will be enhanced to: 1. Provide for the periodic cleaning of the Fire Protection Fuel Oil Storage Tank (Byron only) (Note 1). 2. Provide for periodic draining of water from the auxiliary feedwater (AFW) day tanks, diesel generator (DG) day tanks, essential Service Water make/up pump fuel oil storage tanks (Byron only) (Note 2), and Fire Protection Fuel Oil Storage Tanks. 3. Include analysis for the levels of microbiological organisms in the AFW day tanks and essential Service Water make-up pumps diesel oil storage tanks (Byron only) (Note 2). 4. Include analysis for water and sediment content, particulate concentration, and the levels of microbiological organisms for the DG Day Tanks. 5. Include analysis for water and sediment content and the levels of microbiological organisms for the DG Fuel Oil Storage Tanks. 6. Include analysis for particulate concentration and the levels of microbiological organisms for the Fire Protection Fuel Oil Storage Tanks. 7. Include internal inspections of the Fire Protection Fuel Oil Storage Tanks at least once during the 10-year period prior to the period of extended operation, and at least once every 10 years during the period of extended operation. Each diesel fuel tank will be drained and cleaned, the internal surfaces visually inspected (if physically possible), and, if evidence of degradation is observed during inspections, or if visual inspection is not possible, these diesel fuel tanks will be volumetrically inspected. 8. Include monitoring and trending for the levels of microbiological organisms for the AFW day tanks and essential Service Water make-up pumps diesel oil storage tanks (Byron only) (Note 2).	A.2.1.18	Program to be enhanced no later than 6 months prior to the period of extended operation. Pre–period of extended operation inspections specified in Enhancement 7 will be completed either no later than 6 months prior to the period of extended operation, or before the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	LRA Exelon letter RS-14-124 05/05/2014 Exelon letter RS-14-175 06/30/2014

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
18 (cont.)	9. Include monitoring and trending for water and sediment content, particulate concentration, and the levels of microbiological organisms for the DG Day Tanks. 10. Include monitoring and trending for water and sediment content and the levels of microbiological organisms for the DG Fuel Oil Storage Tanks. 11. Include monitoring and trending for total particulate concentration and the levels of microbiological organisms for the Fire Protection Fuel Oil Storage Tanks. 12. Require inspections of internal coatings be performed by coating inspectors certified to ANSI N45.2.6 or ASTM Standards endorsed in RG 1.54. 13. Specify that signs of peeling, blistering, or delamination of the coating from the base metal, if identified, shall be entered into the CAP. 14. Require physical testing of internal coatings, where physically possible, to ensure that remaining coating is tightly bonded to the base metal when peeling, blistering, or delamination is detected and the coating is not repaired or replaced. The testing will consist of adhesion testing using ASTM International standards endorsed in RG 1.54 (e.g., ASTM D4541-09 or ASTM D6677-07). 15. Require that evaluations utilized to return a coated component exhibiting signs of peeling, blistering, or delamination to service without repairing or replacing the coating shall consider the potential impact on the intended function of the system. This evaluation shall include consideration of the potential for degraded performance of downstream components due to flow blockage and loss of material of the coated component. 16. Require the as-left condition of a coating that exhibited signs of peeling, blistering, or delamination and that is not repaired or replaced is such that the potential for further degradation of the coating is minimized.			

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
19	Reactor Vessel Surveillance is an existing program that will be enhanced to: 1. Establish operating restrictions to ensure that the plant is operated under the conditions to which the surveillance capsules were exposed. The operating restrictions are as follows: Byron Station, Unit 1: - Cold leg operating temperature limitation: 525 degrees Fahrenheit (minimum) to 590 degrees Fahrenheit (maximum). - RPV beltline material fluence: 3.21E+19 n/cm^2 (E >1.0 MeV) (maximum). Byron Station, Unit 2; Braidwood Station Unit 1: - Cold leg operating temperature limitation: 525 degrees Fahrenheit (minimum) to 590 degrees Fahrenheit (maximum). - RPV beltline material fluence: 3.19E+19 n/cm^2 (E>1.0 MeV) (maximum). Braidwood Station, Unit 2: - Cold leg operating temperature limitation: 525 degrees Fahrenheit (minimum) to 590 degrees Fahrenheit (maximum). - RPV beltline material fluence: 3.16E+19 n/cm^2 (E>1.0 MeV) (maximum). If the reactor pressure vessel exposure conditions (neutron fluence, neutron spectrum) or irradiation temperature (cold leg inlet temperature) are altered, then the basis for the projection to the end of the period of extended operation needs to be reviewed and, if deemed appropriate, updates are made to the Reactor Vessel Surveillance program. Any changes to the Reactor Vessel Surveillance program must be submitted for NRC review and approval in accordance with 10 CFR Part 50, Appendix H.	A.2.1.19	Program to be enhanced no later than six months prior to the period of extended operation. Specimen capsule testing to be performed in accordance with the schedule described in Enhancement 2.	LRA Exelon Letter RS-14-002 01/13/2014 Exelon Letter RS-14-149 05/23/2014 Exelon Letter RS-14-225 07/28/2014

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<tr><th>Item Number</th><th>Commitment</th><th>UFSAR Supplement Section or LRA Section</th><th>Implementation Schedule</th><th>Source</th></tr>
<tr><td>19
(cont.)</td><td>2. One (1) specimen capsule per reactor vessel, as designated below, irradiated to a neutron fluence of one (1) to two (2) times the projected peak neutron fluence at the end of the period of extended operation will be withdrawn from the spent fuel pool (SFP), tested, and the summary technical report submitted to the NRC within one (1) year of receipt of the renewed license. Alternatively, if a request for extension of the testing schedule is submitted in accordance with 10 CFR Part 50, Appendix H and granted by the Director, Office of Nuclear Reactor Regulation, specimen testing will be performed in accordance with that approved extension.
<table>
<tr><td>Reactor Vessel
(Station, Unit)</td><td>Capsule ID</td><td>Capsule Fluence
(n/cm^2)
(E>1.0 MeV)</td></tr>
<tr><td>Byron, Unit 1</td><td>Y</td><td>3.97E+19</td></tr>
<tr><td>Byron, Unit 2</td><td>Y</td><td>4.19E+19</td></tr>
<tr><td>Braidwood, Unit 1</td><td>V</td><td>3.71E+19</td></tr>
<tr><td>Braidwood, Unit 2</td><td>V</td><td>3.73E+19</td></tr>
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Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
20	One-Time Inspection is a new program that will be used to verify the system-wide effectiveness of the Water Chemistry, Fuel Oil Chemistry and Lubricating Oil Analysis programs. The One-Time Inspection AMP will also be utilized, in specific cases where existing data is insufficient: a. to validate that a particular aging effect is not occurring, or b. to verify that the aging effect is occurring slowly enough to not affect a components intended function during the period of extended operation. In these cases, the components will not require additional aging management.	A.2.1.20	Program to be implemented no later than 6 months prior to the period of extended operation. One-time inspections will be performed within the 10-year period prior to the period of extended operation, and will be completed either no later than 6 months prior to the period of extended operation, or before the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	LRA Exelon letter RS-14-003 1/13/2014
21	Selective Leaching is a new program that will include one-time inspections of a representative sample of susceptible components to determine if loss of material due to selective leaching is occurring.	A.2.1.21	Program to be implemented no later than 6 months prior to the period of extended operation. One-time inspections will be performed within the five (5)-year period prior to the period of extended operation, and will be completed either no later than 6 months prior to the period of extended operation, or before the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	LRA

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
22	One-Time Inspection of ASME Code Class 1 Small-Bore Piping is a new program that will manage the aging effect of cracking in Class 1 small-bore piping that is less than nominal pipe size (NPS) 4-inches, and greater than or equal to NPS 1-inch. The socket weld sample population for Byron Unit 1 will include the socket weld on the “D” safety injection system cold leg injection line that was replaced in 1998.	A.2.1.22	Program to be implemented no later than 6 months prior to the period of extended operation. One-time Inspections will be performed and evaluated within the six (6)-year period prior to the period of extended operation, and will be completed either no later than 6 months prior to the period of extended operation, or before the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	LRA Exelon Letter RS-14-002 01/13/2014

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
23	External Surfaces Monitoring of Mechanical Components is a new program that manages aging effects of metallic and elastomeric materials through periodic visual inspection of external surfaces for evidence of loss of material and cracking. Visual inspections are augmented by physical manipulation as necessary to detect hardening and loss of strength of elastomers. The periodic system walkdowns include visual inspection of insulation jacketing to ensure the integrity of the jacketing is maintained. External visual inspections of the jacketing ensure that there is no damage to the jacketing that would permit in-leakage of moisture. The procedures for planning insulation repairs will be revised to document that insulation repairs are performed in accordance with specification requirements (e.g., seams on the bottom, overlapping seams) so as to prevent water intrusion into the insulation. Periodic representative inspections to detect corrosion (i.e., loss of material) under insulation will be conducted on in-scope indoor insulated components, where the process fluid temperature is below the dew point for a period of time sufficient to accumulate condensation, and in-scope outdoor insulated components (with the exception of the condensate storage tanks). These periodic inspections will be conducted during each 10-year period of the period of extended operation. Inspections subsequent to the initial inspection will consist of examination of the exterior surface of the insulation for indications of damage to the jacketing or protective outer layer of the insulation if the initial inspection verifies no loss of material due to general, pitting, or crevice corrosion, beyond that which could have been present during initial construction. If the external visual inspections of the insulation reveal damage to the exterior surface of the insulation or if there is evidence of water intrusion through the insulation (e.g., water seepage through insulation seams/joints), then periodic visual inspections under insulation to detect corrosion and cracking under insulation will continue.	A.2.1.23	Program to be implemented no later than six months prior to the period of extended operation.	LRA Exelon letter RS-14-003 1/13/2014 Exelon letter RS-14-051 2/27/2014 Exelon letter RS-14-218 07/18/2014

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
24	Flux Thimble Tube Inspection is an existing program that will be enhanced as follows: 1. For Braidwood Units 1 and 2 (Note 3): Perform corrective actions to re-establish periodic eddy current testing of the flux thimble tubes prior to the period of extended operation to ensure that wall thickness is monitored to detect loss of material from the flux thimble tubes. Once periodic eddy current testing is reestablished, eddy current testing will be performed for each flux thimble tube every refueling outage until sufficient data has been accumulated to establish a plant-specific eddy current testing frequency to ensure that no flux thimble tube is predicted to incur wear that exceeds 80% before the next inspection. Flux thimble tube wall thickness measurements will be trended and wear rates will be calculated based on plant-specific data. Wall thickness will be projected using plant-specific data in accordance with the WCAP-12866, “Bottom Mounted Instrumentation Flux Thimble Wear,” methodology. 2. For Braidwood Unit 1 (Note 3): a. The 17 Braidwood Station, Unit 1 flux thimble tubes that exhibited indications of wear during eddy current testing performed during Refueling Outage A1R15 (Fall 2010), will be replaced or removed from service during Refueling Outage A1R18 (Spring 2015), unless eddy current data is obtained as required by the Flux Thimble Tube Inspection program. (Flux thimble tubes 1 (J-8), 8 (K-6), 9 (H-11), 12 (E-9), 14 (H-4), 18 (L-11), 19 (L-5), 21 (E-11), 23 (D-10), 36 (J-14), 37 (P-9), 41 (N-4), 44 (R-8), 45 (N-13), 48 (P-4), 54 (A-11), 55 (N-14)) b. The remaining Braidwood Station, Unit 1 flux thimble tubes, not replaced during A1R18, will be replaced or removed from service during Refueling Outage A1R19 (Fall 2016), unless eddy current data is obtained as required by the Flux Thimble Tube Inspection program. c. Following A1R19, any Braidwood Station, Unit 1 flux thimble tube will be replaced every two (2) refueling	A.2.1.24	Byron: Ongoing Braidwood: Schedule for flux thimble tube replacement activities identified in commitment. Corrective actions to reestablish periodic eddy current testing at Braidwood will be completed either no later than six months prior to the period of extended operation, or before the end of the last refueling outage prior to the period of extended operation, whichever occurs later. Braidwood Unit 1: Commitment 2.a was completed during refueling outage A1R18 in Spring 2015.	LRA Exelon letter RS-14-336 11/22/2014 Exelon letter RS-15-071 02/23/15 Exelon letter RS-15-107 04/13/2015

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
	outages or removed from service if eddy current data is not obtained in accordance with the Flux Thimble Tube Inspection program. 3. For Braidwood Unit 2 (Note 3): a. The 29 Braidwood Station, Unit 2 flux thimble tubes that exhibited indications of wear during eddy current testing performed during A2R15 Refueling Outage (Spring 2011) and not replaced during A2R17 Refueling Outage (Spring 2014), will be replaced or removed from service during A2R18 Refueling Outage (Fall 2015), unless eddy current data is obtained as required by the Flux Thimble Tube Inspection program. (Flux thimble tubes 1 (J-8), 4 (H-6), 5 (F-8), 6 (J-10), 7 (F-7), 9 (H-11), 10 (L-8), 11 (G-5), 18 (L-11), 22 (K-12), 23 (D-10), 24 (H-13), 25 (N-8), 26 (H-3), 27 (C-8), 29 (N-6), 32 (L-13), 33 (C-5), 34 (H-2), 36 (J-14), 37 (P-9), 40 (F-14), 41 (N-4), 42 (D-3), 45 (N-13), 46 (J-1), 50 (R-6), 52 (L-15), 56 (N-2)) b. The remaining Braidwood Station, Unit 2 flux thimble tubes, not replaced during A2R17 or A2R18, will be replaced or removed from service during A2R19 Refueling Outage (Spring 2017), unless eddy current data is obtained as required by the Flux Thimble Tube Inspection program. c. Following A2R19, any Braidwood Station, Unit 2 flux thimble tube will be replaced every two (2) refueling outages or removed from service if eddy current data is not obtained in accordance with the Flux Thimble Tube Inspection program.			

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
25	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components is a new program that manages aging effects of metallic and elastomeric materials through visual inspections of internal surfaces for evidence of loss of material. Visual inspections are augmented by physical manipulation as necessary to detect hardening and loss of strength of elastomers. This opportunistic approach is supplemented to ensure a representative sample of components within the scope of this program are inspected. At a minimum, in each 10-year period during the period of extended operation, a representative sample of 20 percent of the population (defined as components having the same combination of material, environment, and aging effect) or a maximum of 25 components per population is inspected. Where practical, the inspections focus on the bounding or lead components most susceptible to aging because of time in service, and severity of operating conditions. Opportunistic inspections continue in each 10-year period despite meeting the sampling minimum requirement.	A.2.1.25	Program to be implemented no later than six months prior to the period of extended operation.	LRA Exelon letter RS-14-003 1/13/2014
26	Existing Lubricating Oil Analysis program is credited.	A.2.1.26	Ongoing	LRA
27	Monitoring of Neutron-Absorbing Materials Other than Boraflex is an existing program that will be enhanced to: 1. Maintain the coupon exposure such that it is bounding for the Boral material in all spent fuel racks prior to coupons being examined, by ensuring that the coupons have been surrounded with a greater number of freshly discharged fuel assemblies than that of any other cell location.	A.2.1.27	Program to be enhanced no later than six months prior to the period of extended operation.	LRA Exelon letter RS-14-052 03/04/2014

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
28	Buried and Underground Piping is an existing program that will be enhanced to: 1. Perform manual examinations, in addition to visual inspections, to detect hardening, softening, or other changes in material properties for buried polymeric piping (Braidwood only) (Note 2). 2. Cracking will be managed for stainless steel components, utilizing a method that has been demonstrated to be capable of detecting cracking, whenever coatings are removed and expose the base material (Braidwood only) (Note 2). 3. Ensure all underground carbon steel essential service water system piping within the scope of license renewal is coated in accordance with National Association of Corrosion Engineers (NACE) SP0169-2007 prior to the period of extended operation (Byron only) (Note 1). 4. Direct visual inspections of coated piping and components will be performed by an individual possessing a NACE Coating Inspector Program Level 2 or 3 operator qualification, or by an individual who has attended the EPRI Comprehensive Coatings Course and completed the EPRI Buried Pipe Condition Assessment and Repair Training Computer Based Training Course. 5. Inspection quantities of buried piping within the scope of license renewal will be performed in accordance with LR-ISG-2011-03, Element 4, Table 4a, and based upon the as-found results of cathodic protection system availability and effectiveness during each ten year period, beginning 10 years prior to the period of extended operation.	A.2.1.28	Program to be enhanced no later than 6 months prior to the period of extended operation. Pre–period of extended operation activities specified in Enhancements 3, 5, 6, and 7 will be completed either no later than 6 months prior to the period of extended operation, or before the last refueling outage prior to the period of extended operation, whichever occurs later.	LRA Exelon letter RS-14-003 1/13/2014

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28 (cont.)	6. The buried carbon steel condensate system piping within the scope of license renewal will be addressed, through means of a long term mitigation strategy, prior to entering the period of extended operation. Mitigation may include activities such as fully recoating, complete replacement with like or upgraded material, installation of internal polymeric sleeves, and routing of pipe above ground or in an engineered trench for leak detection. Inspections of the condensate system piping will be performed in accordance with LR-ISG-2011-03, Element 4, Table 4a, and based on the mitigation strategy implemented (Braidwood only) (Note 3). 7. Inspection quantities of underground piping within the scope of license renewal will be performed in accordance with LR-ISG-2011-03, Element 4, Table 4b, during each 10 year period, beginning 10 years prior to the period of extended operation. a. The piping and components inside the Byron 0SX138A and 0SX138B valve vaults will be visually inspected by engineering on a quarterly basis until either measures to prevent immersion of the piping and components inside the vault are implemented, or a coating system is installed that is designed for periodic immersion applications (Byron only) (Note 3). 8. If adverse indications are detected during inspection, inspection sample sizes within the affected piping categories will be doubled. If adverse indications are found in the expanded sample, an analysis will be conducted to determine the extent of condition and extent of cause. The size of the follow-on inspections will be determined based on the analysis. Timing of the additional inspections will be based on the severity of the identified degradation and the consequences of leakage. In all cases, the additional inspections will be performed within the same 10-year inspection interval in which the original adverse indication was identified. Expansion of sample size may be limited by the extent of piping subject to the observed degradation mechanism.			

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
28 (cont.)	9. In performing cathodic protection surveys, only the −850 mV polarized potential criterion specified in NACE SP0169-2007 for steel piping will be used for acceptance criteria and determination of cathodic protection system effectiveness. Alternatively, soil corrosion, or electrical resistance, probes may also be used to demonstrate cathodic protection effectiveness during the annual surveys. An upper limit of −1200 mV for pipe-to-soil potential measurements of coated pipes will also be established, so as to preclude potential damage to coatings. 10. An extent of condition evaluation will be conducted if observed coating damage caused by non-conforming backfill has been evaluated as significant. The extent of condition evaluation will be conducted to ensure that the as-left condition of backfill in the vicinity of the observed damage will not lead to further degradation.			
29	ASME Section XI, Subsection IWE is an existing program that will be enhanced to: 1. Provide guidance for specification of bolting material, lubricant and sealants, and installation torque or tension to prevent or mitigate degradation and failure of structural bolting. 2. Use the condition of the embedded reinforcing steel at the inner surface of the tendon tunnel as a representative indicator for the potential for corrosion at the exterior surface of the containment liner plate. Use the results of Structures Monitoring (B.2.1.34) AMP, Enhancement 16 activities and results from ongoing examinations of the tendon tunnel performed as part of the ASME Section XI, Subsection IWL (B.2.1.30) and Structures Monitoring (B.2.1.34) AMPs to identify changing conditions. Changing conditions consisting of the identification of significant corrosion of embedded steel in the tendon tunnel structure require an evaluation to determine if augmented examinations in accordance with requirements of IWE-1240 “Surface Areas Requiring Augmented Examination” are required due to the potential for accelerated corrosion at the exterior surface of the containment liner plate.	A.2.1.29	Program to be enhanced no later than six months prior to the period of extended operation.	LRA Exelon Letter RS-14-183 7/8/2014

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
30	ASME Section XI, Subsection IWL is an existing program that will be enhanced to: 1. Include additional augmented examination requirements after post-tensioning system repair/replacement activities in accordance with Table IWL-2521-2. 2. A one-time inspection of one (1) vertical and one (1) horizontal tendon on each unit will be performed prior to the period of extended operation. The inspection will consist of visually examining one (1) wire from each of the two (2) types of tendons at a worst-case location based on evidence of free water, grease discoloration, and grease chemistry results. This location will serve as a leading indicator for potential degradation or tendon surface corrosion. The visual inspection of these wires will be performed in accordance with existing station procedures used for inspections consistent with IWL-2523.2. The acceptance criteria will consist of each wire being free of any active corrosion, including general and pitting corrosion. In the event that the acceptance criteria are not met and corrosion is identified, the condition will be entered into the CAP. The condition will be evaluated to characterize the corrosion, determine the cause of the corrosion, the location, depth, extent of the condition, and applicability of the condition to other wires that comprise that tendon. Corrective actions may include activities such as grease analysis, replacement of grease within the tendon duct, additional wire inspections from the same tendon, evaluation of the tendon capacity, potential replacement of the tendon, and augmented inspections and grease sampling of other leading indicator tendons, based, in part, on previous evidence of free water, observed grease leakage, grease discoloration, and grease chemistry results. Specific corrective actions will depend upon the cause, extent of condition, and grease properties. These corrective actions will be consistent with those actions which would be evaluated during periodic required IWL examinations (Braidwood only) (Note 3).	A.2.1.30	Program to be enhanced no later than 6 months prior to the period of extended operation. Pre–period of extended operation inspections specified in Enhancements 2 and 3 will be completed either no later than 6 months prior to the period of extended operation, or before the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	LRA Exelon Letter RS-14-183 7/8/2014 Exelon Letter RS-14-328 11/21/2014 Exelon Letter RS-14-216 12/15/2014

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
30 (cont.)	3. In order to monitor for tendon exposure to free water and moisture and manage any potential adverse effects, a periodic tendon water monitoring and grease sampling program will be implemented (Braidwood only) (Note 3). The program will consist of: a. A baseline inspection of tendon grease caps at the bottom of all vertical and dome tendons, as well as all below-grade horizontal tendons, prior to the period of extended operation. The baseline inspection will check for evidence of free water and grease discoloration, with further actions taken based on the condition of the grease. b. A followup tendon grease cap inspection of all vertical and dome tendons, as well as all below-grade horizontal tendons, will be performed within 10 years of the initial inspection, using the same approach as the baseline inspection.			

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
30 (cont.)	c. For those tendons where free water, moisture, and grease did not meet acceptance criteria during the two (2) previous inspections, periodic monitoring of grease chemistry and moisture, free water, and grease discoloration will be performed on a frequency not to exceed 10 years. Tendons, which exhibit significant quantities of free water (e.g., more than eight ounces) during periodic monitoring, will be inspected more often, with the timing of followup inspections increased until a frequency is achieved that no longer results in significant amounts of free water observed during successive inspections. Tendon water inspection and draining frequencies may vary from annual to every ten (10) years, depending upon grease chemistry and moisture parameters meeting IWL acceptance criteria. The maximum ten (10) year periodic frequency is meant to address any tendons which exhibit evidence of free water but the quantity is observed to be insignificant, with no observable grease discoloration, and given that the tendon was not inspected for at least ten (10) years prior. More frequent followup inspections will be performed for tendons which exhibit insignificant quantities of free water, but were inspected within the ten (10) years prior. In all cases, the frequency of inspections for water in individual tendons will be adjusted to be commensurate with the severity of the conditions found during each examination.			

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
30 (cont.)	d. Braidwood has performed augmented inspections on additional tendons beyond those selected for the ASME Section XI, Subsection IWL program. The Braidwood augmented inspections are performed on a 5 year frequency, in conjunction with the ASME Section XI, Subsection IWL AMP. The current augmented examinations of additional tendons will continue until the periodic tendon water monitoring and grease sampling program described above is implemented. Corrective actions will be taken as necessary to ensure that the tendon grease meets ASME Section XI, Subsection IWL requirements. 4. Explicitly require that areas of concrete deterioration and distress be recorded in accordance with the guidance provided in American Concrete Institute (ACI) 349.3R. The visual resolution capability of direct and remote examination techniques will be sufficient to detect concrete degradation at the levels described in Chapter 5 of ACI 349.3R. The resolution capability of the optical aids used for remote examinations will be demonstrated as equivalent to direct visual examination. 5. Include quantitative acceptance criteria, based on the "Evaluation Criteria" provided in Chapter 5 of ACI 349.3R, that will be used to augment the qualitative assessment of the Responsible Engineer. In addition, the Responsible Engineer will confirm that the visual resolution capability used for the concrete containment structure examinations was sufficient to evaluate the examination results against the quantitative acceptance criteria described in Chapter 5 of ACI 349.3R.			

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
31	ASME Section XI, Subsection IWF is an existing program that will be enhanced to: 1. Add the MC supports for the transfer tube in the refueling cavity in the containment structure and refueling canal in the fuel handling building to the scope of the program. 2. Revise implementing documents to provide guidance for proper specification of bolting material, storage, lubricants and sealants, and installation torque or tension to prevent or mitigate degradation and failure of structural bolting. Bolting material with actual measured yield strength of 150 ksi or greater shall not be used in plant changes without engineering approval, due to consideration of SCC vulnerability. Storage requirements for high strength bolts shall include the recommendations of the Research Council on Structural Connections, “Specification for Structural Joints Using ASTM A325 or A490 Bolts,” Section 2. Lubricants that contain MoS_2 shall not be applied to high strength structural bolts within the scope of license renewal. 3. Provide procedural guidance, regarding the selection of supports to be inspected on subsequent inspections, when a support is repaired in accordance with the CAP. The enhanced guidance will ensure that the supports inspected on subsequent inspections are representative of the general population.	A.2.1.31	Program to be enhanced no later than 6 months prior to the period of extended operation. Pre–period of extended operation examinations specified in Enhancements 4 and 5 will be completed either no later than 6 months prior to the period of extended operation, or before the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	LRA Exelon Letter RS-14-052 03/04/2014 Exelon Letter RS-14-170 06/16/2014 Exelon Letter RS-14-235 08/29/2014

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
31 (cont.)	4. Perform one-time volumetric examinations on a sample of ASTM A490 bolts, greater than 1-in. nominal diameter for the detection of SCC prior to the period of extended operation. Volumetric examinations will be performed in accordance with the requirements of ASME Code Section XI, Appendix VIII, Supplement 8. The sample will consist of bounding and representative A490 bolt sizes, joint configurations, and environmental exposure conditions. The sample will consist of 20% of the ASTM A490 bolts greater than 1-in. nominal diameter or a maximum of 25 ASTM A490 bolts total for both Byron and Braidwood stations. The selection of the samples will consider susceptibility to SCC (e.g., actual measured yield strength) and ALARA principles. Any adverse results of the volumetric examinations will be entered into the CAP and will be evaluated by engineering to determine if additional actions are warranted such as expansion of sample size, scope, and frequency of any additional supplemental visual or volumetric examinations, as well as any code requirements specified by ASME Section XI, Subsection IWF. Specifically, the implementing documents for performing the one-time volumetric examinations will have criteria for extending the ASTM A490 bolt examination scope to other ASTM A490 bolts used in similar joint configurations and environmental exposure conditions if the volumetric examination of a bolt shows adverse results, which is similar to the methodology used by the ASME Code IWF-2430 for IWF component supports. In addition, the program will be revised to include periodic volumetric examinations, of ASTM A490 bolts in sizes greater than 1-in. nominal diameter, if the one-time volumetric examination of an ASTM A490 bolt shows signs of cracking. The periodic examinations of the ASTM A490 bolts are included in the periodic examination of the supports. For the periodic examinations of supports, the population of the supports examined is specified in Table IWF-2500-1. Consistent with the GALL Report, the periodic examinations will include volumetric examinations of high-strength bolts to detect cracking, if required, in addition to			

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
31 (cont.)	4. (continued) the VT-3 examinations of the high-strength bolts. 5. Revise implementing documents to perform periodic visual examinations to detect a corrosive environment that supports SCC potential for all (100%) of high strength bolting greater than 1-in. nominal diameter prior to the period of extended operation, and then each inspection interval of 10 years thereafter. The periodic visual examinations will include criteria to identify if the bolting has been exposed to moisture or other contaminants by evidence of moisture, residue, foreign substance, or corrosion. Adverse conditions identified during the examinations will be evaluated by engineering to determine if the bolt has been exposed to a corrosive environment with the potential to cause SCC. The bolts determined to have been exposed to corrosive environment with the potential to cause SCC will be included in a sample population for each specific bolt material where SCC is a concern. A sample size equal to 20 percent (rounded up to the nearest whole number) of the bolts in the sample population, with a maximum sample size of 25 bolts will be subject to supplemental volumetric examination to determine if SCC is present. The selection of the samples will consider susceptibility to SCC (e.g., actual measured yield strength) and ALARA principles. Volumetric examinations will be performed in accordance with the requirements of ASME Code Section XI, Appendix VIII, Supplement 8. The results of the volumetric examinations will be evaluated by engineering to determine if additional actions are warranted such as expansion of sample size, scope, and frequency of any additional supplemental visual or volumetric examinations, as well as any code requirements specified by ASME Section XI, Subsection IWF. 6. Add the CRDM seismic support assembly to the scope of the program to implement additional examinations.			
32	Existing 10 CFR Part 50, Appendix J program is credited.	A.2.1.32	Ongoing	LRA

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
33	Masonry Walls is an existing program that will be enhanced to: 1. Add masonry walls in the following structures to the program scope: a. Radwaste and Service Building Complex i. Radwaste Building ii. Original Service Building b. Turbine Building Complex c. Switchyard Structures i. Relay House 2. Provide additional guidance for inspection of masonry walls for shrinkage, separation, and for gaps between the supports and the masonry walls that could impact the intended function of the masonry walls. 3. Require that personnel performing inspections and evaluations meet the qualifications described in ACI 349.3R.	A.2.1.33	Program to be enhanced no later than six months prior to the period of extended operation.	LRA
34	Structures Monitoring is an existing program that will be enhanced to: 1. Add the following structures: a. Radwaste and Service Building Complex i. Radwaste Building ii. Original Service Building b. Turbine Building Complex c. Yard Structures i. Transformer foundations ii. Valve and line enclosures d. Fire protection structures-features i. Transformer fire barrier walls ii. Fuel oil storage tank berm e. Containment structure features i. Containment access facility hallway	A.2.1.34	Program to be enhanced no later than 6 months prior to the period of extended operation. Pre–period of extended operation activities specified in Enhancement 16 will be completed either no later than 6 months prior to the period of extended operation, or before the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	LRA Exelon Letter RS-13-274 12/19/2013 Exelon letter RS-14-097 04/17/2014 Exelon letter RS-14-169 06/16/2014 Exelon Letter RS-14-216 12/15/2014

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
34 (cont.)	2. Add the following components and commodities: a. Blowout panels b. Building features – doors and seals, bird screens, louvers, windows c. Compressible joints and seals, gaskets and moisture barriers d. Concrete curbs e. Electrical cable trays, conduits and tube tracks f. Hatches and plugs g. Insulation including jacketing h. Manholes, handholes and duct banks i. Metal components, including metal decking for concrete slabs, miscellaneous steel, sump screens and trench covers, and scuppers around the SFP j. New fuel storage racks k. Offgas stack and flue l. Panels, racks, cabinets, and other enclosures m. Penetration seals and sleeves n. Pipe whip restraints, jet impingement shields, and spray shields o. Pipe, electrical and equipment component support members p. Sliding surfaces q. SFP gates r. Sumps and liners 3. Monitor groundwater chemistry on a frequency not to exceed five (5) years for pH, chlorides, and sulfates and evaluate results exceeding the threshold criteria to assess impact, if any, on below-grade concrete. 4. Based on groundwater chemistry monitoring results, select and inspect every five (5) years a structure that will be used as a leading indicator for the condition of below grade concrete exposed to groundwater.			

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
34 (cont.)	5. Require (a) evaluation of the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas and (b) examination of representative samples of the exposed portions of the below grade concrete, when excavated for any reason. 6. Provide guidance for proper specification of high strength bolting material and lubricant to prevent or mitigate degradation and failure of structural bolting. 7. Revise storage requirements for high strength bolts to include recommendations of Research Council on Structural Connections (RCSC) Specification for Structural Joints Using High Strength Bolts, Section 2.0. 8. Clarify that loose bolts and nuts, and cracked high strength bolts are not acceptable unless accepted by engineering evaluations. 9. Include the potential for reduction in concrete anchor capacity due to local concrete degradation. 10. Require that personnel performing inspections and evaluations meet the qualifications specified within ACI 349.3R with respect to knowledge of inservice inspection of concrete and visual acuity requirements. 11. Require acceptance and evaluation of structural concrete using quantitative criteria based on Chapter 5 of ACI 349.3R. 12. Perform inspection of elastomeric components such as vibration isolation elements and structural seals for cracking, loss of material and hardening. Visual inspections of elastomeric components are to be supplemented by feel or manipulation to detect hardening. 13. Monitor accessible sliding surfaces to detect loss of mechanical function or significant loss of material due to wear, corrosion, debris, dirt, distortion, or overload that could restrict or prevent sliding of surfaces as required by design.			

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
34 (cont.)	14. Formalize requirements for the monitoring of the leak detection sight glasses associated with the refuel cavity, transfer canal, SFP, and refueling water storage tank on a periodic basis. 15. Require visual inspections of submerged concrete structural elements by dewatering a structure or by a diver if the structure is not dewatered at least once every five (5) years (Byron only) (Note 2).			

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
34 (cont.)	16. At each site, perform one-time sampling activities on below grade, reinforced concrete at specific locations in the tendon tunnels. Select the locations exhibiting significant mineral deposits to serve as leading indicators for potential reinforced concrete degradation as a result of exposure to ground water in-leakage and build-up of mineral deposits. Take corrective actions, if necessary, prior to the period of extended operation. Perform the one-time sampling activities as follows: a. Obtain water in-leakage samples, at representative locations with mineral deposits due to water in-leakage, and analyze for pH, chlorides, sulfates, minerals, and iron content. b. Obtain representative mineral deposit samples and analyze for chemical composition. c. Remove three concrete core samples. i. Test two of the concrete core samples for compressive strength and perform petrographic examination of the core samples. Select representative locations for the concrete core samples that include one with significant mineral deposits and another at a location with no mineral deposits for comparative purposes. ii. Drill an additional core at a crack with significant mineral deposits and subject the core to petrographic examination. d. Expose and examine reinforcing steel at two locations, with water in-leakage, cracks, and significant mineral deposits. e. Collectively evaluate the results from the water inleakage analysis, the chemical composition of the mineral deposits, examination of the exposed reinforcing steel, and the core sample testing to confirm there is no significant degradation to the reinforced concrete material properties and to determine if additional corrective actions are necessary. Additional corrective actions may include, but are not limited to, an extent of condition review for other potentially impacted structures, more frequent examinations, and additional sampling and analysis, as appropriate.			

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
34 (cont.)	17. Perform visual inspections of polymeric components, such as blowout panels, for changes in material properties. Observations of material discoloration, cracking, crazing, and loss of material will provide visual indications of changes in material properties prior to a loss of component intended function.			

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
35	RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants is an existing program that will be enhanced to: 1. Provide guidance for specification of structural bolting material and bolting lubricants to prevent or mitigate degradation and failure of structural bolting. 2. Revise storage requirements for structural bolting to include recommendations of RCSC Specification for Structural Joints Using High Strength Bolts, Section 2.0. 3. Include the potential for reduction in concrete anchor capacity due to local concrete degradation. 4. Include all aging affects addressed by ACI 349.3R in procedures and require acceptance and evaluation of structural concrete using quantitative criteria based on Chapter 5 of ACI 349.3R. 5. Clarify that loose bolts and nuts, and cracked bolts are not acceptable unless accepted by engineering evaluations. 6. Require that steel components subject to RG 1.127 are inspected for loss of material. 7. Require that inspectors work under the direction of a qualified engineer for submerged concrete inspections. 8. Require special inspections also be performed in the event of large floods, hurricanes, and intense local rainfalls. 9. Require increased inspection frequency if the extent of the degradation is such that the structure or component may not meet its design basis if allowed to continue uncorrected until the next normally scheduled inspection. 10. Require (a) evaluation of the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas and (b) examination of representative samples of the exposed portions of the below grade concrete, when excavated for any reason.	A.2.1.35	The Byron Essential Service Water Cooling Tower inspection and maintenance plan (Enhancement 16) will be initiated upon receipt of the renewed licenses, and will continue through the period of extended operation to ensure the condition of the SXCT is maintained. The remainder of the enhancements will be implemented no later than six months prior to the period of extended operation.	LRA Exelon Letter RS-14-216 12/15/2014

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
35 (cont.)	11. Monitor raw water and groundwater chemistry at least once every five (5) years for pH, chlorides, and sulfates and verify that it remains non-aggressive, or evaluate results exceeding criteria to assess impact, if any, on submerged concrete. 12. Based on groundwater chemistry monitoring results, select and inspect every five (5) years a structure that will be used as a leading indicator for the condition of below grade concrete exposed to groundwater. 13. Require visual inspections of submerged concrete structural components by dewatering a structure or by a diver if the structure is not dewatered at least once every five (5) years. Maintenance procedures will be enhanced to require opportunistic inspection of submerged concrete structures when they are dewatered and made accessible. 14. Require that degraded conditions be documented and trended until the condition is no longer occurring or until a corrective action is implemented. 15. Clarify parameters to be monitored and inspected at the Essential Service Water Cooling Towers to include visual inspection for loss of material and reduction of heat transfer for the cooling tower fill, and visual inspection with physical manipulation for change in material properties associated with the PVC drift eliminators and fiberglass support beams for the drift eliminators (Byron only) (Note 2).			

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
35 (cont.)	16. Manage the condition of the Byron Essential Service Water Cooling Towers (SXCTs) as follows: a. Monitor and trend inspection activities at the SXCTs on an increased frequency, with inspections of the entire tower on a three (3) year interval, and inspections of the fill support beams and air-inlet framing on a 1.5-year interval. The recommendations in Chapter 5 of ACI 349.3R will be used for quantitative acceptance and evaluation criteria. b. Develop a repair plan to address degradation of the SXCTs with specific emphasis and consideration for the fill support beams. Repairs that are required will be scheduled based on a ranking of the condition observed and the potential for the degradation to progress or propagate.			
36	Protective Coating Monitoring and Maintenance Program is an existing program that will be enhanced to: 1. Add recurring work orders requiring Service Level I coating inspections every refuel outage. 2. Require qualification of coating inspectors to ASTM D 5498. 3. Require qualification of personnel in accordance with ASTM D 7108. 4. Incorporate guidance for inspection and maintenance of Service Level I coatings per RG 1.54 and impose ASTM D 5163-08 requirements for Service Level I coatings condition assessment, reporting, evaluation, and documentation. 5. Require thorough visual inspections of all coatings near sumps or screens associated with the emergency core cooling system (ECCS) by the coatings inspector(s). 6. Specify instruments and equipment that may be needed for Service Level I coatings inspections.	A.2.1.36	Program to be enhanced no later than six months prior to the period of extended operation.	LRA

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
37	Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements is a new program that will be used to manage aging of the insulation material for non-EQ cables and connections. Accessible cables and connections located in adverse localized environments will be visually inspected at least once every 10 years for indications of reduced insulation resistance, such as embrittlement, discoloration, cracking, melting, swelling, or surface contamination.	A.2.1.37	Program to be implemented no later than 6 months prior to the period of extended operation. Initial inspections will be completed either no later than 6 months prior to the period of extended operation, or before the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	LRA
38	Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits is a new program that will be used to manage aging of non-EQ cable and connection insulation of the in-scope portions of the radiation monitoring system (Byron and Braidwood) and the neutron monitoring inputs to the reactor protection system (Braidwood only) (Note 2). Calibration and cable tests (such as insulation resistance tests, time domain reflectometry tests, or other testing judged to be effective in determining cable system insulation condition) will be performed and results will be assessed for reduced insulation resistance prior to the period of extended operation and at least once every 10 years during the period of extended operation.	A.2.1.38	Program to be implemented no later than 6 months prior to the period of extended operation. Initial calibration, cable tests and evaluation of results will be completed either no later than 6 months prior to the period of extended operation, or before the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	LRA Exelon letter RS-14-030 2/4/2014

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
39	Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements is a new program that will be used to manage the aging effects and mechanisms of non-EQ, in scope, inaccessible power cables. Cables will be tested using one or more proven tests for detecting reduced insulation resistance of the cable's insulation system. The cables will be tested at least once every 6 years. More frequent testing may occur based on test results and OE. Periodic actions will be taken to prevent inaccessible cables from being exposed to significant moisture. Manholes associated with the cables included in this program will be inspected for water collection with subsequent corrective actions (e.g., water removal), as necessary. Prior to the period of extended operation, the frequency of inspections for accumulated water will be established and adjusted based on plant-specific OE with cable wetting or submergence, including water accumulation over time and event driven occurrences such as heavy rain or flooding. Operation of dewatering devices, if installed, will be verified prior to any known or predicted heavy rain or flooding event. During the period of extended operation, the inspections will occur at least annually.	A.2.1.39	Program to be implemented no later than 6 months prior to the period of extended operation. First cable tests and manhole inspections will be completed either no later than 6 months prior to the period of extended operation, or before the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	LRA Exelon letter RS-14-041 2/19/2014
40	Metal Enclosed Bus is an existing program that will be enhanced to: 1. Specify that a sample size of 20 percent of the accessible bolted connection population with a maximum sample size of 25 to be inspected for increased resistance of connection by measuring the connection resistance using a micro-ohmmeter. 2. Specify that the external surfaces of metal enclosed bus enclosure assemblies are to be inspected for loss of material due to general, pitting, and crevice corrosion. 3. Specify maximum allowed bus connection resistance values.	A.2.1.40	Program to be enhanced no later than six months prior to the period of extended operation.	LRA

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
41	Fuse Holders (Byron only) (Note 2) AMP is a new program that applies to fuse holders located outside of active devices that have been identified as susceptible to aging effects. Fuse holders subject to increased resistance of connection or fatigue, will be tested, by a proven test methodology, at least once every 10 years for indications of aging degradation. Visual inspection is not part of this program.	A.2.1.41	Program to be implemented no later than 6 months prior to the period of extended operation. Initial resistance tests will be completed either no later than 6 months prior to period of extended operation, or before the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	LRA
42	Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program is a new program that will implement one-time testing of a representative sample (20 percent with a maximum sample size of 25) of non-EQ electrical cable connections to ensure that either aging of metallic cable connections is not occurring or that the existing preventive maintenance program is effective such that a periodic inspection program is not required.	A.2.1.42	Program to be implemented no later than 6 months prior to the period of extended operation. One-time tests will be completed either no later than 6 months prior to the period of extended operation, or before the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	LRA

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
43	Fatigue Monitoring is an existing program that will be enhanced to: 1. Address the cumulative fatigue damage effects of the reactor coolant environment on component life by evaluating the impact of the reactor coolant environment on critical components for the plant identified in NUREG/CR-6260. Additional plant-specific component locations in the RCPB will be evaluated if they are more limiting than those considered in NUREG/CR-6260. 2. Monitor and track additional plant transients that are significant contributors to component fatigue usage. 3. Evaluate the effects of the reactor coolant system water environment on the reactor vessel internal components with existing fatigue CUF analyses to satisfy the evaluation requirements of ASME Code, Section III, Subsection NG-2160 and NG-3121. 4. Increase the scope of the program to include transients used in the analyses for ASME Code Section III fatigue exemptions, the allowable stress analyses associated with ASME Code Section III and ANSI B31.1, and the flaw evaluation analyses performed in accordance with ASME Section XI, IWB-3600.	A.3.1.1	Program to be enhanced no later than 6 months prior to the period of extended operation. Environmental fatigue evaluations will be completed no later than 6 months prior to the period of extended operation.	LRA Exelon letter RS-14-002 01/13/2014
44	Concrete Containment Tendon Prestress is an existing program that will be enhanced to: 1. For each surveillance interval, the predicted lower-limit, minimum required value, and trending lines will be developed for the period of extended operation as part of the regression analysis for each tendon group.	A.3.1.2	Program to be enhanced no later than six months prior to the period of extended operation.	LRA
45	The Environmental Qualification (EQ) of Electric Components AMP will be enhanced: 1. To expand the scope of the program to include mechanical environmental qualification (MEQ) components.	A.3.1.3	Program to be enhanced no later than six months prior to the period of extended operation.	LRA Exelon letter RS-14-079 3/04/2014

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
46	The Operating Experience Program is an existing program that will be enhanced to: 1. Require the review of internal and external OE for aging-related degradation or impacts to aging management activities, to determine if improvements to Byron and Braidwood Units 1 and 2 aging management activities are warranted. NRC and industry guidance documents and standards applicable to aging management are considered part of this information (e.g., License Renewal Interim Staff Guidance (LR-ISG) documents, NUREG-1801 (GALL) revisions, etc.) Ensure there are written expectations for identifying and processing these documents as OE. 2. Establish criteria to define aging-related degradation. In general, the criteria will be used to identify aging that is in excess of what would be expected, relative to design, previous inspection experience and the inspection intervals. 3. Establish identification coding within the CAP for use in identification, trending and communications of aging-related degradation. Provide a definition for the coding. This coding will assist plant personnel in ensuring that, in addition to addressing the specific issue, the adequacy of existing AMPs is assessed. Station personnel are required to periodically assess the performance of the AMPs, including insights obtained through OE. Adverse trends are entered into the CAP for evaluation. This could lead to AMP revisions or the establishment of new AMPs, as appropriate. 4. Require communication of significant internal aging-related degradation, associated with SSCs in the scope of license renewal, to other Exelon plants and to the industry. Criteria will be established for determining when aging-related degradation is significant. 5. Provide training to those responsible for screening, evaluating and communicating OE items related to aging management and aging-related degradation. This training will be commensurate with their role in the process, will be provided periodically and include provisions to accommodate personnel turnover.	A.1.6	Program to be enhanced no later than the date that the renewed operating licenses are issued and conducted on an ongoing basis throughout the terms of the renewed licenses.	LRA

Item Number	Commitment	UFSAR Supplement Section or LRA Section	Implementation Schedule	Source
47	Byron Unit 2 reactor head closure stud location 11 will be repaired so that all 54 reactor head closure studs are tensioned during the period of extended operation – reported complete by letter dated December 15, 2014 (Byron only. Note 3).		No later than six months prior to the period of extended operation.	Exelon letter RS-13-285 12/19/2013 Exelon Letter RS-14-216 12/15/2014
48	Braidwood Unit 2 reactor head closure stud location 35 will be repaired so that all 54 reactor head closure studs are tensioned during the period of extended operation (Braidwood only. Note 3).		No later than six months prior to the period of extended operation.	Exelon letter RS-13-285 12/19/2013 Exelon Letter RS-14-216 12/15/2014

APPENDIX B

CHRONOLOGY

This appendix contains a chronological listing of the routine correspondence between the staff of the U.S. Nuclear Regulatory Commission (the staff) and Exelon Generation Company, LLC (Exelon or the applicant), and other correspondence regarding the staff's reviews of the Byron Station, Units 1 and 2, and the Braidwood Station, Units 1 and 2, Docket Numbers 50-454, 50-455, 50-456, and 50-457, license renewal application (LRA).

Date	Subject
May 29, 2013	Braidwood and Byron, Units 1 and 2 - Application for Renewed Operating Licenses (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13155A387)
May 29, 2013	Braidwood and Byron, Units 1 and 2 - Information to Support NRC Staff Review of the Application for Renewed Operating Licenses (ADAMS Accession No. ML13155A388)
May 29, 2013	Braidwood License Renewal Boundary Drawings (ADAMS Accession No. ML13154A218)
May 29, 2013	Byron Station, Units 1 & 2, License Renewal Boundary Drawing (ADAMS Accession No. ML13154A227)
May 29, 2013	Byron and Braidwood, Units 1 and 2 - License Renewal Application, Volume 1 of 4 (ADAMS Accession No. ML13155A420)
May 29, 2013	Byron and Braidwood, Units 1 and 2 - License Renewal Application, Volume 2 of 4 (ADAMS Accession No. ML13155A421)
June 3, 2013	Byron and Braidwood, Units 1 and 2 - License Renewal Application (ADAMS Accession No. ML13161A223)
June 6, 2013	Letter to Gallagher M. P., Exelon Generation Company, LLC: Letter-Receipt and Availability of the License Renewal Application for the Byron Nuclear Station, Units 1 and 2, and Braidwood Nuclear Station, Units 1 and 2 (ADAMS Accession No. ML13144A099)
July 16, 2013	Letter to Gallagher M. P., Exelon Generation Company, LLC: Determination Of Acceptability And Sufficiency For Docketing, Proposed Review Schedule, And Opportunity For A Hearing Regarding The Application From Exelon Generation Company, LLC, For Renewal Of The Operating Licenses For Byron Nuclear Station, Units 1 and 2, and Braidwood Nuclear Station, Units 1 and 2 (ADAMS Accession No. ML13134A142)
August 12, 2013	Letter to Gallagher M. P., Exelon Generation Company, LLC: Plan for the Aging Management Program Regulatory Audits Regarding the Byron and Braidwood Nuclear Stations License Renewal Application Review (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML13212A367)
October 7, 2013	Letter to Gallagher M. P., Exelon Generation Company, LLC: Requests For Additional Information For The Review Of The Byron Nuclear Station, Units 1 And 2, And Braidwood Nuclear Station, Units 1 And 2, License Renewal Application -Aging Management, Set 1 (TAC Nos. MF1879, MF1880, MF1881, AND MF1882) (ADAMS Accession No. ML13262A035)
October 28, 2013	Summary Of Telephone Conference Call Held On September 25, 2013 Between The U.S. Nuclear Regulatory Commission And Exelon Generation Company, LLC, Concerning RAIs Pertaining To The Byron-Braidwood License Renewal Application (TAC Nos. MF1879, MF1880, MF1881, MF1882) (ADAMS Accession No. ML13281A557)
November 5, 2013	Braidwood Station, Units 1 & 2 and Byron Station, Units 1 & 2 - Response to NRC Requests for Additional Information, Set 1, dated October 7, 2013, re License Renewal Application (ADAMS Accession No. ML13309B590)
November 22, 2013	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 5 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML13310A576)

Date	Subject
November 25, 2013	Letter to Gallagher M. P., Exelon Generation Company, LLC: Requests For Additional Information For The Review Of The Byron Nuclear Station, Units 1 And 2, And Braidwood Nuclear Station, Units 1 And 2, License Renewal Application - Aging Management, Set 3 (TAC Nos. MF1879, MF1880, MF1881, AND MF1882) (ADAMS Accession No. ML13281A574)
November 25, 2013	Summary Of Telephone Conference Call Held On October 22, 2013, Between The U.S. Nuclear Regulatory Commission And Exelon Generation Company, LLC, Concerning RAI Set 3 For The Byron-Braidwood LRA (TAC Nos. MF1879, MF1880, MF1881, MF1882) (ADAMS Accession No. ML13303B463)
December 12, 2013	Letter to Gallagher, M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Nuclear Station Units 1 and 2, and the Braidwood Nuclear Station Units 1 and 2, LRA - Aging Management, Set 4 (TAC MF1879, MF1880, MF1881, MF1882) (ADAMS Accession No. ML13281A569)
December 13, 2013	Letter to Gallagher, M. P., Exelon Generation Company, LLC: Requests For Additional Information For The Review Of The Byron Nuclear Station, Units 1 And 2, And Braidwood Nuclear Station, Units 1 And 2, License Renewal Application - Aging Management, Set 2 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML13282A369)
December 17, 2013	Braidwood, Units 1 and 2, Byron, Units 1 and 2, Responses to NRC Requests for Additional Information, Set 3, Dated November 25, 2013, Related to the License Renewal Application (ADAMS Accession No. ML13354C055)
December 19, 2013	Braidwood, Units 1 & 2 and Byron, Units 1 & 2 - Response to NRC Requests for Additional Information, Set 5, dated November 22, 2013, related to License Renewal Application (ADAMS Accession No. ML13353A627)
December 19, 2013	Braidwood, Units 1 and 2 and Byron, Units 1 and 2, Updated Responses to Two NRC Requests for Additional Information from Set 1, Dated October 7, 2013, Related to the License Renewal Application (ADAMS Accession No. ML13354B749)
January 13, 2014	Letter to Gallagher, M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 6 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML13317A075)
January 13, 2014	Braidwood, Units 1 and 2, and Byron, Units 1 and 2, Response to NRC Requests for Additional Information, Set 4, dated December 12, 2013, Related to the License Renewal Application (ADAMS Accession No. ML14013A148)
January 13, 2014	Braidwood, Units 1 and 2, and Byron, Units 1 and 2, Response to NRC Requests for Additional Information, Set 2, dated December 13, 2013, Related to the License Renewal Application (ADAMS Accession No. ML14013A293)
January 22, 2014	Letter to Gallagher, M. P., Exelon Generation Company, LLC: Requests for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 9 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14007A658)
January 23, 2014	Summary Of Telephone Conference Call Held On October 22, 2013, Between The U.S. Nuclear Regulatory Commission And Exelon Generation Company, LLC, Concerning RAI Set 4 For The Byron-Braidwood LRA (TAC Nos. MF1879, MF1880, MF1881, MF1882) (ADAMS Accession No. ML13330A932)

Date	Subject
January 23, 2014	Summary of Telephone Conference Call Held on October 31, 2013, Between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC, Concerning Draft Requests for Additional Information Pertaining To the Byron Station and Braidwood Station (ADAMS Accession No. ML13309A932)
January 23, 2014	Summary Of Telephone Conference Call Held Between The U.S. Nuclear Regulatory Commission And Exelon Generation Company, LLC, Concerning Draft Requests For Additional Information Pertaining To The Byron Station And Braidwood Station (ADAMS Accession No. ML13318A415)
February 4, 2014	Braidwood, Units 1 and 2 and Byron, Units 1 and 2, Response to NRC Requests for Additional Information, Set 6, Dated January 13, 2014, Related to License Renewal Application (ADAMS Accession No. ML14035A516)
February 6, 2014	Letter to Gallagher, M. P., Exelon Generation Company, LLC: Requests For Additional Information For The Review Of The Byron Station, Units 1 And 2, And Braidwood Station, Units 1 And 2, License Renewal Application- Aging Management, Set 8 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14006A021)
February 7, 2014	Letter to Gallagher, M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 13 (TAC MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14023A564)
February 10, 2014	Letter to Gallagher, M. P., Exelon Generation Company, LLC: Requests for Additional Information for The Review of The Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 12 (TAC MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14030A596)
February 18, 2014	Letter to Gallagher, M. P., Exelon Generation Company, LLC: Request For Additional Information For The Review Of The Byron Station, Units 1 And 2, And Braidwood Station, Units 1 And 2, License Renewal Application, Set 11 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14034A068)
February 18, 2014	Letter to Gallagher, M. P., Exelon Generation Company, LLC: Request For Additional Information For The Review Of The Byron Station, Units 1 And 2, And Braidwood Station, Units 1 And 2, License Renewal Application, Set 17 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14007A603)
February 19, 2014	Braidwood and Byron Stations, Units 1 and 2 - Response to NRC Requests for Additional Information, Set 9, dated January 22, 2014, related to the License Renewal Application (ADAMS Accession No. ML14051A154)
February 20, 2014	Summary Of Telephone Conference Call Held On January 30, 2014, Between The U.S. Nuclear Regulatory Commission And Exelon Generation Company, LLC, Concerning Draft Request For Additional Information Pertaining To The Byron Station And Braidwood Station, License Renewal Application (TAC Nos. MF1879, MF1880, MF1881, MF1882) (ADAMS Accession No. ML14035A189)
February 20, 2014	January 22, 2014, Summary of Telephone Conference Call held between NRC and Exelon Generation Company, LLC., Concerning Draft Requests for Additional Information Pertaining to the Byron Station and Braidwood Station, License Renewal Application (ADAMS Accession No. ML14035A534)

Date	Subject
February 24, 2014	Summary of Telephone Conference Call Held on January 16, 2014, Between The U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC, Concerning RAI Set 7, tor the Byron-Braidwood License Renewal Application (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML10450A122)
February 26, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Nuclear Station, Units 1 and 2, and Braidwood Nuclear Station, Units 1 and 2, License Renewal Application - Aging Management - Set 10 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14038A336)
February 27, 2014	Braidwood, Units 1 and 2, and Byron, Units 1 and 2, Response to NRC Requests for Additional Information, Set 8, dated February 6, 2014 Related to the License Renewal Application (ADAMS Accession No. ML14058A667)
February 28, 2014	Braidwood and Byron, Units 1 & 2 - Responses to NRC Requests for Additional Information, Set 12, dated February 19, 2014 Related to License Renewal Application (ADAMS Accession No. ML14059A215)
March 4, 2014	Summary of Telephone Conference Call on January 23, 2014, Between U.S. Nuclear Regulatory Commission And Exelon Generation Company, LLC, Concerning RAI Set 11, for Byron-Braidwood License Renewal Application (TAC Nos. MF1879, MF1880, MF1881 & MF1882) (ADAMS Accession No. ML14050A167)
March 4, 2014	Summary of Telephone Conference Call Held on January 28, 2014, Between NRC and Exelon, Concerning RAI Set 10, for the Byron and Braidwood Station, LRA (TAC Nos. MF1879, MF1880, MF1881 & MF1882) (ADAMS Accession No. ML14036A310)
March 4, 2014	Summary of Telephone Conference Call Held on January 29, 2014, Between the NRC and Exelon Generation Company, LLC., Concerning RAI Set 10, for the Byron and Braidwood Station License Renewal Application (TAC Nos. MF1879, MF1880, MF1881, AND MF1882) (ADAMS Accession No. ML14051A431)
March 4, 2014	Braidwood Units 1 & 2 & Byron Station, Units 1 & 2, Response to NRC Requests for Additional Information, Set 13, Dated February 7, 2014 re License Renewal Application (ADAMS Accession No. ML14063A495)
March 4, 2014	Braidwood and Byron, Units 1 and 2 - Response to NRC Requests for Additional Information, Set 11, dated February 18, 2014 related to the License Renewal Application (ADAMS Accession No. ML14063A496)
March 6, 2014	Summary of Telephone Conference Call on January 13, 2014, Between U.S. Nuclear Regulatory Commission And Exelon Generation Company, LLC, Concerning RAI Set 8, for Byron-Braidwood License Renewal Application (TAC Nos. MF1879, MF1880, MF1881 & MF1882) (ADAMS Accession No. ML14051A428)
March 7, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Correction to Request for Additional Information B.2.1.10-1, Letter Dated February 7, 2014, for the Review of the Byron Station and Braidwood Station, Units 1 and 2, LRA Set 13 (ADAMS Accession No. ML14050A081)
March 11, 2014	Summary of Telephone Conference Call on February 18, 2014, Between U.S. Nuclear Regulatory Commission & Exelon Generation Company, LLC Concerning Draft Request for Additional Information Regarding Byron & Braidwood Stations, License Renewal Application (ADAMS Accession No. ML14058B180)

Date	Subject
March 11, 2014	Summary of Telephone Conference Call on February 27, 2014, Between U.S. Nuclear Regulatory Commission & Exelon Generation Company, LLC Concerning Draft Request for Additional Information Re Byron Station & Braidwood Station License Renewal Application (ADAMS Accession No. ML14064A403)
March 11, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 15 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14064A391)
March 13, 2014	Braidwood, Units 1 and 2, and Byron, Units 1 and 2, Response to NRC Requests for Additional Information, Set 7, dated February 18, 2014, Related to the License Renewal Application (ADAMS Accession No. ML14073A118)
March 13, 2014	Aging Management Programs Audit Report Regarding The Byron Station, Units 1 And 2, And Braidwood Station, Units 1 And 2 (TAC Nos. MF1879, MF1880, MF1881, And MF1882) (ADAMS Accession No. ML14071A620)
March 14, 2014	Scoping and Screening Methodology Audit Report Regarding the Byron Station, and Braidwood Station, Units 1 and 2, LRA (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14050A304)
March 18, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 14 (TAC Nos. MF1879, MF1880, MF1881, AND MF1882) (ADAMS Accession No. ML14058B182)
March 20, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Nuclear Station, Units 1 and 2, and Braidwood Nuclear Station, Units 1 and 2, License Renewal Application - Set 17 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14051A503)
March 21, 2014	Braidwood Units 1 and 2, and Byron Units 1 ND 2 - Updated Responses to NRC Requests for Additional Information, Set 3, dated November 25, 2013, License Renewal Application (ADAMS Accession No. ML14080A187)
March 28, 2014	Braidwood, Units 1 and 2, and Byron, Units 1 and 2, Responses to NRC Requests for Additional Information, Set 10, dated February 26, 2014 Related to the License Renewal Application (ADAMS Accession No. ML14090A237)
April 3, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 16 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14084A335)
April 3, 2014	Summary of Telephone Conference Call on February 12, 2014, Between U.S. Nuclear Regulatory Commission & Exelon Generation Company, LLC Concerning Draft Request for Additional Information Re Byron Station & Braidwood Station License Renewal Application (ADAMS Accession No. ML14073A705)
April 7, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Project Manager Change for the License Renewal of Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14092A346)

Date	Subject
April 7, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 20 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14092A261)
April 7, 2014	03/04/2014 Summary of Telephone Conference Call Held between NRC and Exelon Generation Company, LLC, Concerning Draft Request for Additional Information Pertaining to the Byron and Braidwood License Renewal Application Set 16 (ADAMS Accession No. ML14084A488)
April 8, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request For Additional Information For The Review Of The Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 19 (TAC Nos. MF1879, MF1880, MF1881, AND MF1882) (ADAMS Accession No. ML14094A366)
April 8, 2014	Responses to NRC Requests for Additional Information, Set 15, dated March 11, 2014, related to the Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, License Renewal Application (ADAMS Accession No. ML14098A230)
April 10, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request For Additional Information For The Review Of The Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 18 (TAC Nos. MF1879, MF1880, MF1881, AND MF1882) (ADAMS Accession No. ML14093B247)
April 14, 2014	Braidwood and Byron Stations, Units 1 and 2 - Response to NRC Request for Additional Information, Set 17, dated March 20, 2014, related to the License Renewal Application (ADAMS Accession No. ML14104A598)
April 17, 2014	Braidwood and Byron, Units 1 & 2 - Responses to NRC Requests for Additional Information, Set 14, dated March 18, 2014, related to License Renewal Application (ADAMS Accession No. ML14107A027)
April 17, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 & 2, and Braidwood Station, Units 1 & 2, License Renewal Application, Set 21 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14099A485)
April 22, 2014	April 03, 2014 Summary of Telephone Conference Call Held between NRC and Exelon Generation Co., LLC, Concerning Draft Request for Additional Information, Set 21, Pertaining to Byron, Units 1 & 2, Braidwood, Units 1 & 2, License Renewal Application (ADAMS Accession No. ML14099A493)
April 24, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request For Additional Information For The Review Of The Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 22 (TAC Nos. MF1879, MF1880, MF1881, AND MF1882) (ADAMS Accession No. ML14111A118)
April 24, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request For Additional Information For The Review Of The Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 23 (TAC Nos. MF1879, MF1880, MF1881, AND MF1882) (ADAMS Accession No. ML14107A193)

Date	Subject
April 25, 2014	Summary of Telephone Conference Call Held on April 9, 2014, between the U.S. Nuclear Regulatory Commission & Exelon Generation Company, LLC concerning draft request for Additional Information, Set 23, Pertaining to the Byron Station & Braidwood Station (ADAMS Accession No. ML14107A077)
May 5, 2014	October 29, 2013, Summary of Telephone Conference Call Held Between the U.S. Nuclear Regulatory Commission and Exelon Concerning RAI Set 2 for the Byron and Braidwood Station LRA (ADAM Accession No. ML13312A021)
May 5, 2014	Response to NRC Request for Additional Information, Set 16, dated April 3, 2014, related to the Braidwood Station, Units 1 and 2, and Byron Station, Units 1, and 2, License Renewal Application (ADAMS Accession No. ML14125A325)
May 5, 2014	10 CFR 54.21(b) Annual Amendment to the Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, License Renewal Application (ADAMS Accession No. ML14125A300)
May 6, 2014	Response to NRC Request for Additional Information, Set 20, dated April 7, 2014, related to the Braidwood Station, Units 1 and 2, and Byron Station, Units 1, and 2, License Renewal Application (ADAMS Accession No. ML14126A339)
May 6, 2014	Responses to NRC Requests for Additional Information, Set 19, dated April 8, 2014, related to the Braidwood Station, Units 1 and 2, and Byron Station, Units 1, and 2, License Renewal Application (ADAMS Accession No. ML14126A338)
May 12, 2014	Responses to NRC Requests for Additional Information, Set 18, dated April 10, 2014, Related to the Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, License Renewal Application (ADAMS Accession No. ML14132A139)
May 14, 2014	Summary Of Telephone Conference Call Held On March 26, 2014, Between The U.S. Nuclear Regulatory Commission And Exelon Generation Company, LLC Concerning Draft Request For Additional Information, Set 21, Pertaining To The Byron Station And Braidwood (ADAMS Accession No. ML14107A226)
May 14, 2014	Summary Of Telephone Conference Call Held On March 19, 2014, Between The U.S. Nuclear Regulatory Commission And Exelon Generation Company, LLC Concerning Draft Request For Additional Information, Set 18, Pertaining To The Byron Station And Braidwood (ADAMS Accession No. ML14092A440)
May 14, 2014	Summary Of Telephone Conference Call Held On April 10, 2014, Between The U.S. Nuclear Regulatory Commission And Exelon Generation Company, LLC Concerning Draft Request For Additional Information, Set 22, Pertaining To The Byron Station And Braid (ADAMS Accession No. ML14112A418)
May 14, 2014	Summary of Telephone Conference Call Held on April 1, 2014, Between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC Concerning Draft Request for Additional Information, Set 19, Pertaining to the Byron Station and Braidwood (ADAMS Accession No. ML14094A425)
May 14, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Withholding Information from Public Disclosure (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14129A339)

Date	Subject
May 15, 2014	Responses to NRC Requests for Additional Information, Set 21, dated April 17, 2014, related to the Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, License Renewal Application (ADAMS Accession No. ML14135A179)
May 19, 2014	Summary of Telephone Conference Call Held on April 22, 2014, Between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC Concerning Draft Request for Additional Information, Set 24, Pertaining to the Byron Station and Braidwood (ADAMS Accession No. ML14126A543)
May 19, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 25 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14126A806)
May 19, 2014	Summary Of Telephone Conference Call Held On March 19, 2014, Between The U.S. Nuclear Regulatory Commission And Exelon Generation Company, LLC Concerning Draft Request For Additional Information, Set 18, Pertaining To The Byron Station And Braidwood (ADAMS Accession No. ML14094A275)
May 19, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 24 (TAC Nos. MF1879, MF1880, MF1881, MF1882) (ADAMS Accession No. ML14126A434)
May 19, 2014	Summary of Telephone Conference Call Held on May 6, 2014, Between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC Concerning Draft Request for Additional Information, Set 25, Pertaining the Byron Station and Braidwood Station (ADAMS Accession No. ML14133A687)
May 21, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request For Additional Information For The Review Of The Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 27 (TAC Nos. MF1879, MF1880, MF1881, AND MF1882) (ADAMS Accession No. ML14135A540)
May 21, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request For Additional Information for the Review Of the Byron Station, Units 1 and 2, And Braidwood Station, Units 1 And 2, License Renewal Application, Set 26 (TAC MF1879, MF1880, MF1881, AND MF1882) (ADAMS Accession No. ML14133A701)
May 22, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 30 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14136A099)
May 23, 2014	Summary of Telephone Conference Call Held on May 12, 2014, Between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC Concerning Draft Request for Additional Information, Set 26, Pertaining to the Byron Station and Braidwood (ADAMS Accession No. ML14133A639)
May 23, 2014	Responses to NRC Requests for Additional Information, Set 23, dated April 24, 2014, related to the Byron Station, Units 1 and 2, and Braidwood Station, Units 1, and 2, License Renewal Application (ADAMS Accession No. ML14143A313)

Date	Subject
May 23, 2014	Responses to NRC Requests for Additional Information, Set 22, dated April 24, 2014, related to the Byron Station, Units 1 and 2, and Braidwood Station, Units 1, and 2, License Renewal Application (ADAMS Accession No. ML14143A312)
May 23, 2014	Braidwood, Units 1 and 2, Byron, Units 1 and 2, Corrections to the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application (ADAMS Accession No. ML14143A118)
May 28, 2014	Summary of Telephone Conference Call Held on May 15, 2014, Between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC Concerning Draft Request for Additional Information, SET 27, Pertaining to the Byron Station and Braidwood (ADAMS Accession No. ML14140A385)
May 29, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 28 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14143A015)
June 4, 2014	Letter to Gallagher, M. P., Exelon Generation Company, LLC: Request For Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 And 2, License Renewal Application, Set 29 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14149A260)
June 5, 2014	Summary of Telephone Conference Call Held on May 21, 2014, Between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC Concerning Draft Request for Additional Information, Set 29, Pertaining to the Byron Station and Braidwood (ADAMS Accession No. ML14149A141)
June 5, 2014	Summary of Telephone Conference Call Held on May 19, 2014, Between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC Concerning Draft Request for Additional Information, Set 28, Pertaining to the Byron Station and Braidwood (ADAMS Accession No. ML14148A388)
June 5, 2014	Response to NRC Request for Additional Information, Set 25 dated May 19, 2014, related to the Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2, License Renewal Application (ADAMS Accession No. ML14157A151)
June 5, 2014	Request for Withdrawal of Documents in accordance with 10 CFR 2.390(c), related to the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application (ADAMS Accession No. ML14157A311)
June, 9, 2014	Responses to NRC Requests for Additional Information, Set 24, dated May 19, 2014, related to the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application (ADAMS Accession No. ML14160A871)
June 16, 2014	Braidwood, Units 1 & 2, Byron Units 1 & 2, Response to NRC Requests for Additional Information, Set 30 dated May 22, 2014 Related to License Renewal Application (ADAMS Accession No. ML14167A297)
June 16, 2014	Responses to NRC Requests for Additional Information, Set 27, dated May 21, 2014, and Correction of Previously Submitted Information related to the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application (ADAMS Accession No. ML14168A084)

Date	Subject
June 17, 2014	Responses to NRC Requests for Additional Information, Set 26, dated May 21, 2014, related to the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application (ADAMS Accession No. ML14168A020)
June 17, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 35 (TAC Nos. MF1879, MF1880, MF1881 and MF1882) (ADAMS Accession No. ML14160A042)
June 18, 2014	Braidwood Units 1 and 2 & Byron, Units 1 and 2, Responses to NRC Requests for Additional Information, Set 29, dated June 4, 2014, Related to License Renewal Application (ADAMS Accession No. ML14169A026)
June 23, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request For Additional Information For The Review Of The Byron Station, Units 1 And 2, And Braidwood Station, Units 1 And 2, License Renewal Application, Set 33 (TACs MF1879, MF1880, MF1881, And MF1882) (ADAMS Accession No. ML14167A547)
June 23, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 34 (TAC MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14167A540)
June 24, 2014	Summary of Telephone Conference Call Held on June 10, 2014, Between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC Concerning Draft Request for Additional Information, Set 33, Pertaining to the Byron and Braidwood Station (TAC MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14167A025)
June 24, 2014	Summary of Telephone Conference Call Held between NRC and Exelon Generation Co., LLC, Concerning Draft Request for Additional Information, Set 32, Pertaining to the Byron, and Braidwood, License Renewal Application (TAC MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14162A369)
June 25, 2014	Summary of Telephone Conference Call Held on June 11, 2014, Between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC Concerning Draft Request for Additional Information, Set 34, Pertaining to the Byron Station and Braidwood Station (TAC Nos. MF1879, MF1880, MF1881 and MF1882) (ADAMS Accession No. ML14164A446)
June 25, 2014	Summary of Teleconference between the U.S. Nuclear Regulatory Commission & Exelon Generation Company, LLC, concerning Draft RAI, set 31, Pertaining to Byron Station & Braidwood, License Renewal Application) (TAC Nos. MF1879, MF1880, MF1881 and MF1882) (ADAMS Accession No. ML14175A398)
June 26, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 36 (TAC Nos. MF1879, MF1880, MF1881 and MF1882) (ADAMS Accession No. ML14176A090)

Date	Subject
June 30, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 And 2, License Renewal Application, Set 31 (TAC MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14169A627)
June 30, 2014	Response to NRC Request for Additional Information, Set 28, dated May 29, 2014, related to the Byron Station, Units 1 and 2, and Braidwood Station, Units 1, and 2, License Renewal Application (ADAMS Accession No. ML14181B145)
June 30, 2014	Request for Extraction of Enclosure D of Exelon Set 10 RAI Response Letter RS-14-084, dated March 28, 2014, related to the Braidwood and Byron, Units 1 and 2, License Renewal Application (ADAMS Accession No. ML14181B207)
July 2, 2014	Summary of Telephone Conference Call between NRC and Exelon Generation Co., LLC, Concerning the Byron and Braidwood, License Renewal Application (TAC Nos. MF1879, MF1880, MF1881 and MF1882) (ADAMS Accession No. ML14177A430)
July 7, 2014	Summary of Telephone Conference Call Held between the U.S. NRC and Exelon Generation Co., LLC, Concerning Draft Request for Additional Information, Set 36, Pertaining to the Byron Station and Braidwood Station, License Renewal Application (TAC Nos. MF1879, MF1880, MF1881 and MF1882) (ADAMS Accession No. ML14183B230)
July 7, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 37 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14183B617)
July 8, 2014	Braidwood, Units 1 and 2, Byron, Units 1 and 2, Updated Responses to NRC Set 14 Requests for Additional Information, Related License Renewal Application (ADAMS Accession No. ML14189A094)
July 15, 2014	Resubmittal of Information Associated with NRC Set 10 RAIs, Related to the Byron Station, Units 1 & 2, and Braidwood Station, Units 1 and 2, License Renewal Application (ADAMS Accession No. ML14196A553)
July 16, 2014	Summary of Telephone Conference Call Held between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC Concerning Request for Additional Information, Set 35, Pertaining to the Byron Station and Braidwood Station, License Renewal Application (TAC Nos. MF1879, MF1880, MF1881 and MF1882) (ADAMS Accession No. ML14191A693)
July 16, 2014	Summary of Telephone Conference Call Held on June 30, 2014, Between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC Discussing Applicant Responses in Staff Requests for Additional Information B.2.1.16-1A, B.21.23-1, and 3.0.3-3A Concerning the Byron Station and Braidwood Station, License Renewal Application (TAC Nos. MF1879, MF1880, MF1881, MF1882) (ADAMS Accession No. ML14190A464)
July 18, 2014	Braidwood, Units 1 and 2, and Byron, Units 1 and 2, Responses to NRC Requests for Additional Information, Sets 33 and 34, Both dated June 23, 2014; and Corrections and Clarifications Related to the License Renewal Application (ADAMS Accession No. ML14199A346)

Date	Subject
July 23, 2014	Summary of Telephone Conference Call Held on June 26, 2014, Between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC Concerning Draft Request for Additional Information, Set 37, Pertaining to Byron Station and Braidwood Station, Units 1 and 2, License Renewal Application (TAC Nos: MF1879, MF1880, MF1881, MF1882) (ADAMS Accession No. ML14183A017)
July 25, 2014	Byron and Braidwood Station, Units 1 and 2 - Response to NRC Request for Additional Information, Set 31, dated June 30, 2014, related to the License Renewal Application (ADAMS Accession No. ML14206A920)
July 25, 2014	Braidwood, Units 1 and 2, and Byron, Units 1 and 2, Response to NRC Request for Additional Information, Set 36, dated June 26, 2014, Related to the License Renewal Application (ADAMS Accession No. ML14206A729)
July 28, 2014	Braidwood and Byron Stations, Units 1 & 2 - Response to NRC Request for Additional Information, Set 37, dated July 7, 2014, License Renewal Application (ADAMS Accession No. ML14209A045)
August 4, 2014	Summary of Telephone Conference Call Held on July 16, 2014, Between Nuclear Regulatory Commission and Exelon Generation Company, LLC Concerning Responses for Request for Additional Information B.2.1.31-1A Pertaining to Byron Station and Braidwood Station, Units 1 and 2, License Renewal Application, Set 38 (TAC Nos: MF1879, MF1880, MF1881, MF1882) (ADAMS Accession No. ML14202A396)
August 4, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 38 (TAC Nos: MF1879, MF1880, MF1881, MF1882) (ADAMS Accession No. ML14205A595)
August 11, 2014	Summary of Telephone Conference Call Held between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC, Concerning Draft Request for Additional Information, Set 38, Pertaining to the Byron Station and Braidwood Station, License Renewal Application (TAC Nos. MF1879, MF1880, MF1881, MF1882) (ADAMS Accession No. ML14205A228
August 11, 2014	Summary of Telephone Conference Call Held On June 10, 2014, Between U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC Concerning Fire Water System Request for Additional Information Responses Pertaining to Byron Station and Braidwood Station, License Renewal Application (TAC Nos. MF1879, MF1880, MF1881, MF1882) (ADAMS Accession No. ML14205A575)
August 20, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request For Additional Information For the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 40 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14232A313)
August 29, 2014	Response to NRC Request for Additional Information, Set 38, dated 08/04/2014; LRA changes from NRR Staff Feedback on 07/30/2014 Telecon; and, LRA changes from NRC Region III IP-71002 Inspection, to Byron, Units 1 & 2, Braidwood, Units 1 & 2, LRA (ADAMS Accession No. ML14241A527)
September 3, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request to Withholding Information from Public Disclosure (TAC Nos. MF1879, MF1880, MF1881, AND MF1882) (ADAMS Accession No. ML14238A691)

Date	Subject
September 4, 2014	Braidwood Units 1 & 2, Byron, Units 1 & 2, Results of Detailed Review Performed in Response to Request 1 of NRC RAI B.2.1.7-7 from Set 17, Related to License Renewal Application (ADAMS Accession No. ML14247A195)
September 4, 2014	Interim Update Related to Earlier Response to Set 29 RAI B.2.1.5-1a, related to the Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, License Renewal Application (ADAMS Accession No. ML14247A210)
Sept 5, 2014	Response to NRC Request for Additional Information, Set 40, dated August 20, 2014, related to the Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, License Renewal Application (ADAMS Accession No. ML14248A322)
September 8, 2014	Summary of Telephone Conference Call Held between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC, Concerning Request for Additional Information B.2.1.30-3, 3.0.3-3A, and 2.3.3.12-4, and Draft Request for Additional Information Set 38 and 39, Pertaining to the Byron Station and Braidwood Station, License Renewal Application (TAC Nos. MF1879, MF1880, MF1881, MF1882) (ADAMS Accession No. ML14238A092)
September 11, 2014	Withdrawal and Resubmittal of Information associated with NRC Set 31 RAIs, related to the Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2, License Renewal Application (ADAMS Accession No. ML14254A143)
September 11, 2014	Braidwood, Units 1 and 2, Byron, Units 1 and 2, Withdrawal and Resubmittal of Information Associated with NRC Set 10 RAIs, Related to License Renewal Application (ADAMS Accession No. ML14254A136)
September 16, 2014	Summary of Telephone Conference Call Held between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC, Concerning Draft Request for Additional Information Set 40, Pertaining to the Byron Station and Braidwood Station, License Renewal Application (TAC Nos. MF1879, MF1880, MF1881, MF1882) (ADAMS Accession No. ML14245A371)
October 7, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Withholding Information from Public Disclosure (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession Number ML14266A653).
October 9, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 41 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14279A449)
October 10, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 42 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14282A276)
October 16, 2014	Braidwood and Byron, Units 1 and 2 - Response to NRC Request for Additional Information, Set 41, dated October 9, 2014, and LRA changes resulting from NRC Region III IP-71002 Braidwood Inspection, both related to License Renewal Application (ADAMS Accession No. ML14289A423)

Date	Subject
October 28, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 43 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14300A269)
October 30, 2014	Safety Evaluation Report With Open Items Related to the License Renewal of Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 (ADAMS Accession No. ML14296A176)
October 31, 2014	Braidwood and Byron, Units 1 & 2 - Response to NRC Request for Additional Information, Set 42, dated October 10, 2014, License Renewal Application (ADAMS Accession No. ML14304A345)
November 6, 2014	Letter to Gallagher M. P., Exelon Generation Company, LLC: Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application, Set 44 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML14302A417)
November 6, 2014	Summary of Telephone Conference Call Held on October 7, 2014, between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC Concerning Draft Request for Additional Information, Set 41, Pertaining to the Byron Station and Braidwood Station, License Renewal Application (TAC Nos. MF1879, MF1880, MF1881, MF1882) (ADAMS Accession No. ML14300A218)
November 19, 2014	Summary of Telephone Conference Call Held on October 27, 2014, between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC Concerning Draft Request for Additional Information, Set 44, Pertaining to the Byron Station and Braidwood Station, License Renewal Application (TAC Nos. MF1879, MF1880, MF1881, MF1882) (ADAMS Accession No. ML14317A783)
November 21, 2014	Braidwood and Byron Stations, Units 1 & 2 - Response to NRC Request for Additional Information, Set 44, dated November 6, 2014, License Renewal Application (ADAMS Accession No. ML14325A744)
November 22, 2014	Braidwood and Byron, Units 1 and 2 - Supplemental Commitment related to the October 31, 2014 Response to NRC Request for Additional Information, Set 42, dated October 10, 2014, related to License Renewal Application (ADAMS Accession No. ML14330A480)
November 24, 2014	Braidwood, Units 1 and 2, Byron, Units 1 and 2, Update Associated with Earlier Responses to Set 29 RAI B.2.1.5-1a, Related to License Renewal Application (ADAMS Accession No. ML14335A323)
November 25, 2014	Braidwood, Units 1 and 2, Byron, Units 1 and 2, Response to NRC Request for Additional Information, Set 43, dated October 28, 2014, Related License Renewal Application (ADAMS Accession No. ML14335A391)
December 4, 2014	Summary of Telephone Conference Call Held on October 7, 2014, between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC Concerning Draft Request for Additional Information, Set 42, Pertaining to the Byron Station and Braidwood Station, License Renewal Application (TAC Nos. MF1879, MF1880, MF1881, MF1882) (ADAMS Accession No. ML14323A625)

Date	Subject
December 15, 2014	Braidwood and Byron, Units 1 and 2, Response to NRC Request for Additional Information, Set 35, dated June 17, 2014, and Submittal of an updated License Renewal Commitment List related to License Renewal Application (ADAMS Accession No. ML14349A524)
December 15, 2014	Exelon Generation Company, LLC Comments on the Safety Evaluation Report with Open Items, related to the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 License Renewal Application (ADAMS Accession No. ML14349A509)
December 18, 2014	Summary of Telephone Conference Call Held on October 16, 2014, and October 23, 2014, between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC Concerning Draft Request for Additional Information, Set 43, Pertaining to the Byron Station and Braidwood Station, License Renewal Application (TAC Nos. MF1879, MF1880, MF1881, MF1882) (ADAMS Accession No. ML14343A432)
January 23, 2015	Update to Report the Completion of Commitment 47, Related to the Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, License Renewal Application (ADAMS Accession No. ML15023A382)
January 28, 2015	Braidwood and Byron, Units 1 and 2 - LRA Impact Assessment Associated with Earlier Responses to Set 29 RAI B.2.1.5-1a, related to the License Renewal Application (ADAMS Accession No. ML15028A520)
February 6, 2015	Braidwood and Byron, Units 1 & 2 - Update to LRA Section 4.3.4, Class 1 Component Fatigue Analyses Supporting GSI-190 Closure, Related to License Renewal Application (ADAMS Accession No. ML15040A179)
February 11, 2015	Braidwood and Byron Station, Units 1 & 2 - LRA Amendment Providing Commitment for Control Rod Drive Mechanism Examinations, related to License Renewal Application (ADAMS Accession No. ML15042A133)
February 23, 2015	Braidwood, Units 1 & 2 and Byron, Units 1 & 2, Response to NRC Request for Additional Information, Set 45, dated January 22, 2015, Related to License Renewal Application (ADAMS Accession No. ML15054A030)
February 24, 2015	Summary of Telephone Conference Call Held on January 7, 2015, between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC Concerning Draft Request for Additional Information, Set 45, Pertaining to the Byron Station and Braidwood Station (ADAMS Accession No. ML15029A694)
February 24, 2015	Summary of Telephone Conference Call Held on January 27, 2015, between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC, Concerning Draft Request for Additional Information, Set 47, Pertaining to the Byron Station and Braidwood Station (ADAMS Accession No. ML15029A704)
February 25, 2015	Summary of Telephone Conference Call Held on January 29, 2015, between the U.S. Nuclear Regulatory Commission and Exelon Generation Company, LLC Concerning Draft Request for Additional Information, Set 46, Pertaining to the Byron Station and Braidwood Station (ADAMS Accession No. ML15033A059)
February 27, 2015	Byron, Units 1 & 2 - Response to Request for Additional Information Regarding Pressure and Temperature Limits Reports (ADAMS Accession No. ML15058A068)
March 3, 2015	02/05/2015 Summary of Telephone Conference Call No. 2 Held between the NRC and Exelon Generation Co., LLC Concerning Request for Additional Information, Set 45, Pertaining to the Byron and Braidwood, License Renewal Application (ADAMS Accession No. ML15051A361)

Date	Subject
April 2, 2015	Request for Additional Information for the Review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application – Set 48 (TAC Nos. MF1879, MF1880, MF1881, and MF1882) (ADAMS Accession No. ML15089A110)
April 6, 2015	Second 10 CFR 54.21(b) Annual Amendment to the Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, License Renewal Application (ADAMS Accession No. ML15096A409)
April 13, 2015	Braidwood, Units 1 and 2, Byron, Units 1 and 2 - Response to NRC Request for Additional Information, Set 48, dated April 2, 2015 (ADAMS Accession No. ML15103A687)
September 15, 2015	Advisory Committee on Reactor Safeguards Report on the Safety Aspects of the License Renewal Application for Byron Station Units 1 and 2 and Braidwood Station Units 1 and 2 (ADAMS Accession No. ML15264A955)

APPENDIX C

PRINCIPAL CONTRIBUTORS

This appendix lists the principal contributors for the development of this safety evaluation report (SER) and their areas of responsibility.

Principle Contributors	
Name	**Responsibility**
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Fu, Bart	Reviewer – Mechanical
Gavula, Jim	Reviewer – Mechanical
Gitter, Joseph	Management Oversight
Green, Kim	Reviewer – Mechanical
Hiser, Allen	Management Oversight
Holston, William	Reviewer – Mechanical
Homiack, Matthew	Reviewer – Mechanical
Hunt, Christopher	Reviewer – Mechanical
Iqbal, Naeem	Reviewer – Fire Protection
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APPENDIX D

REFERENCES

This appendix lists the references used throughout this safety evaluation report for review of the license renewal application (LRA) for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2.

U.S. Nuclear Regulatory Commission Documents
Entergy Nuclear Vermont Yankee, LLC (Vermont Yankee Nuclear Power Station), CLI-10-17, 72 NRC 1 (2010) (ADAMS Accession No. ML101890775).
GL 89-13, “Service Water System Problems Affecting Safety-Related Equipment,” July 1989.
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Information Notice (IN) 87-67, “Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11,” December 31, 1987.
IN 94-63, “Boric Acid Corrosion of Charging Pump Casings Caused by Cladding Cracks,” August 1994.
IN 99-10, “Degradation of Prestressing Tendon Systems in Prestressed Concrete Containments,” Revision 1.
IN 2013-06, “Corrosion in Fire Protection Piping Due to Air and Water Interaction,” March 2013.
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LR-ISG 2011-03, “Changes to the Generic Aging Lessons Learned (GALL) Report Revision 2 Aging Management Program (AMP) XI.M41, Buried and Underground Piping and Tanks.”
LR-ISG 2011-04, “Updated Aging Management Criteria for Reactor Vessel Internal Components of Pressurized Water Reactors.”
LR-ISG 2011-05, “Ongoing Review of Operating Experience.”
LR-ISG 2012-01, “Wall Thinning Due to Erosion Mechanisms.”
LR-ISG 2012-02, “Aging Management of Internal Surfaces, Fire Water Systems, Atmospheric Storage Tanks, and Corrosion Under Insulation.”
NRC letter dated May 19, 2000, “License Renewal Issue No. 98-0030, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components.”
NUREG-0313, “Technical Report On Material Selection And Processing Guidelines For BWR Coolant Pressure Boundary Piping,” Revision 2, January 1988.
NUREG-0588, “Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment.”
NUREG-0612, “Control of Heavy Loads at Nuclear Plants,” July 1980.
NUREG-0737, “Clarification of TMI Action Plan Requirements,” November 1980.
NUREG-1339, “Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants,” June 1990.
NUREG-1437, “Generic Environmental Impact Statement for License Renewal of Nuclear Plants (GEIS).”
NUREG-1557, “Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal,” October 1996.
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NUREG-1801, “Generic Aging Lessons Learned (GALL) Report,” Revision 2, December 2010.
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11. ABSTRACT (200 words or less)

This safety evaluation report documents the technical review of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, license renewal application (LRA) by the United States Nuclear Regulatory Commission (NRC) staff (the staff). By letter dated May 29, 2013, Exelon Generation Company, LLC (Exelon), submitted the LRA in accordance with Title 10 of the Code of Federal Regulations (10 CFR) Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." Exelon requested renewal of the Byron Units 1 and 2, and Braidwood Units 1 and 2, operating licenses (Operating License Nos. NPF 37, NPF 66, NPF 72, and NPF 77 respectively) for a period of 20 years beyond the current expirations at midnight October 31, 2024; November 6, 2026; October 17, 2026; and December 18, 2027, respectively.

Byron is located in north central Illinois, near the town of Byron, Illinois, and near the Rock River approximately 95 miles from Chicago, Illinois. The Braidwood Station is located in northeastern Illinois, near the town of Braidwood, Illinois, and near the Kankakee River approximately 60 miles from Chicago, Illinois.

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

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT unlimited
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