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10 CFR 50
10 CFR 51
10 CFR 54

September 14, 2018

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Peach Bottom Atomic Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

Subject: Changes to the Peach Bottom Atomic Power Station, Units 2 and 3, Subsequent License Renewal Application

Reference: Letter from Michael P. Gallagher, Exelon Generation Company, LLC (Exelon) to NRC Document Control Desk, dated July 10, 2018, "Application for Subsequent Renewed Operating Licenses"

In the Reference letter, Exelon submitted the Subsequent License Renewal Application (SLRA) for the Peach Bottom Atomic Power Station, Units 2 and 3 (PBAPS). Exelon has identified three minor changes that need to be made to the SLRA.

The Enclosure to this letter provides a description of each change, and corresponding mark-ups to affected portions of the SLRA, thereby supplementing the SLRA.

This submittal has been discussed with the NRC License Renewal Senior Project Manager for the PBAPS Subsequent License Renewal project.

There are no new or revised regulatory commitments contained in this letter.

If you have any questions, please contact Mr. John Hufnagel, Licensing Lead, Exelon License Renewal Projects, at 610-765-5829.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 09-14-2018

Respectfully,

Michael P. Gallagher

Michael P. Gallagher
Vice President - License Renewal and Decommissioning
Exelon Generation Company, LLC

Enclosure: Changes to the Peach Bottom Atomic Power Station, Units 2 and 3, Subsequent License Renewal Application

cc: Regional Administrator – NRC Region I
NRC Senior Project Manager, NRR-DMLR (Safety Review)
NRC Project Manager, NRR-DMLR (Environmental Review)
NRC Project Manager, NRR-DORL – Peach Bottom Atomic Power Station
NRC Senior Resident Inspector, Peach Bottom Atomic Power Station
R.R. Janati, Pennsylvania Bureau of Radiation Protection
D.A. Tancabel, State of Maryland

Enclosure

Changes to the Peach Bottom Atomic Power Station, Units 2 and 3, Subsequent License Renewal Application

Introduction

This enclosure contains three changes that are being made to the Subsequent License Renewal Application (SLRA) that were identified after submittal of the SLRA. For each item, the change is described and the affected page number(s) and portion(s) of the SLRA are provided. For clarity, entire sentences or paragraphs from the SLRA are provided with deleted text highlighted by ~~strikethroughs~~ and inserted text highlighted by ***bolded italics***. Revisions to SLRA tables are shown by providing excerpts from the affected tables.

Change #1: Plant Equipment and Floor Drain System

Affected SLRA Sections: Table 2.3.3-19 and Table 3.3.2-19

SLRA Page Numbers: 2.3-92, 3.3-306, and 3.3-307

Description of Change: Plant Equipment and Floor Drain System pump casings for the recombiner building equipment sump pumps and recombiner building floor drain sump pumps have been removed from the scope of license renewal.

The recombiner building equipment and floor drain sump pumps were included in the scope of license renewal for potential spatial interaction only. However, the Recombiner Building does not contain safety-related SSCs. This precludes the possibility of leakage or spray that could prevent satisfactory accomplishment of a safety-related intended function under 10 CFR 54.4(a)(1).

Accordingly, SLRA Tables 2.3.3-19 and 3.3.2-19 are revised to delete the following component types from the scope of license renewal: Pump Casing (Recombiner Building Equipment Sump Pump), and Pump Casing (Recombiner Building Floor Drain Sump Pump).

SLRA Table 2.3.3-19, Plant Equipment and Floor Drain System - Components Subject to Aging Management Review, page 2.3-92 is revised as shown below:

Component Type	Intended Function
Pump Casing (Circulating Water Pump Structure Sump Pump)	Pressure Boundary
Pump Casing (Conveyor Floor Drain Sump Pump)	Leakage Boundary
Pump Casing (D/G Building Sump Pump)	Leakage Boundary
Pump Casing (Drywell Equipment Drain Sump Pump)	Leakage Boundary
Pump Casing (Drywell Floor Drains Sump Pump)	Leakage Boundary
Pump Casing (Floor Drain Collector Pump)	Leakage Boundary
Pump Casing (Laundry Drain Tank Pump)	Leakage Boundary
Pump Casing (Off Gas Stack Sump Pump)	Leakage Boundary
Pump Casing (RHR Sump Pump)	Leakage Boundary
Pump Casing (Radwaste Building Equipment Drain Sump Pump)	Leakage Boundary
Pump Casing (Radwaste Floor Drain Sump Pump)	Leakage Boundary
Pump Casing (Reactor Building Equipment Drain Sump Pump)	Leakage Boundary
Pump Casing (Reactor Building Floor Drain Sump Pump)	Leakage Boundary
~~Pump Casing (Recombiner Building Equipment Sump Pump)~~	~~Leakage Boundary~~
~~Pump Casing (Recombiner Building Floor Drain Sump Pump)~~	~~Leakage Boundary~~
Pump Casing (Turbine Building Equipment Drain Sump Pump)	Leakage Boundary
Pump Casing (Turbine Building Floor Drain Sump Pump)	Leakage Boundary

SLRA Table 3.3.2-19, Plant Equipment and Floor Drain System, Summary of Aging Management Evaluation pages 3.3-306 and 3.3-307 are revised as shown below:

Page 3.3-306

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	NUREG-2192 Table 1 Item	Notes
Pump Casing (Reactor Building Floor Drain Sump Pump)	Leakage Boundary	Gray Cast Iron	Air - Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B.2.1.24)	VII.I.A-77	3.3.1-078	A
			Waste Water (Internal)	Long-Term Loss of Material	One-Time Inspection (B.2.1.21)	VII.E5.A-785	3.3.1-193	A
				Loss of Material	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25)	VII.E5.AP-281	3.3.1-091	A
					Selective Leaching (B.2.1.22)	VII.E5.A-547	3.3.1-072	A
~~Pump Casing (Recombiner Building Equipment Sump Pump)~~	~~Leakage Boundary~~	~~Gray Cast Iron~~	~~Air - Indoor Uncontrolled (External)~~	~~Loss of Material~~	~~External Surfaces Monitoring of Mechanical Components (B.2.1.24)~~	~~VII.I.A-77~~	~~3.3.1-078~~	~~A~~
			~~Waste Water (Internal)~~	~~Long-Term Loss of Material~~	~~One-Time Inspection (B.2.1.21)~~	~~VII.E5.A-785~~	~~3.3.1-193~~	~~A~~
				~~Loss of Material~~	~~Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25)~~	~~VII.E5.AP-281~~	~~3.3.1-091~~	~~A~~
					~~Selective Leaching (B.2.1.22)~~	~~VII.E5.A-547~~	~~3.3.1-072~~	~~A~~
~~Pump Casing (Recombiner Building Floor Drain Sump Pump)~~	~~Leakage Boundary~~	~~Gray Cast Iron~~	~~Air - Indoor Uncontrolled (External)~~	~~Loss of Material~~	~~External Surfaces Monitoring of Mechanical Components (B.2.1.24)~~	~~VII.I.A-77~~	~~3.3.1-078~~	~~A~~
			~~Waste Water (Internal)~~	~~Long-Term Loss of Material~~	~~One-Time Inspection (B.2.1.21)~~	~~VII.E5.A-785~~	~~3.3.1-193~~	~~A~~
				~~Loss of Material~~	~~Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25)~~	~~VII.E5.AP-281~~	~~3.3.1-091~~	~~A~~

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	NUREG-2192 Table 1 Item	Notes
~~Pump Casing (Recombiner Building Floor Drain Sump Pump)~~	~~Leakage Boundary~~	~~Gray Cast Iron~~	~~Waste Water (Internal)~~	~~Loss of Material~~	~~Selective Leaching (B.2.1.22)~~	~~VII.E5.A-547~~	~~3.3.1-072~~	~~A~~
Pump Casing (Turbine Building Equipment Drain Sump Pump)	Leakage Boundary	Gray Cast Iron	Air - Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B.2.1.24)	VII.I.A-77	3.3.1-078	A
			Waste Water (Internal)	Long-Term Loss of Material	One-Time Inspection (B.2.1.21)	VII.E5.A-785	3.3.1-193	A
				Loss of Material	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25)	VII.E5.AP-281	3.3.1-091	A
					Selective Leaching (B.2.1.22)	VII.E5.A-547	3.3.1-072	A
Pump Casing (Turbine Building Floor Drain Sump Pump)	Leakage Boundary	Gray Cast Iron	Air - Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B.2.1.24)	VII.I.A-77	3.3.1-078	A
			Waste Water (Internal)	Long-Term Loss of Material	One-Time Inspection (B.2.1.21)	VII.E5.A-785	3.3.1-193	A
				Loss of Material	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25)	VII.E5.AP-281	3.3.1-091	A
					Selective Leaching (B.2.1.22)	VII.E5.A-547	3.3.1-072	A
Tanks (Floor Drain Demin)	Leakage Boundary	Stainless Steel	Air - Indoor Uncontrolled (External)	Cracking	One-Time Inspection (B.2.1.21)	VII.E4.AP-209a	3.3.1-004	C
				Loss of Material	One-Time Inspection (B.2.1.21)	VII.I.A-751b	3.3.1-222	A

Change #2: Selective Leaching Aging Management Program

Affected SLRA Section: Appendix B, Section B.2.1.22

SLRA Page Number: B-129

Description of Change: SLRA Appendix B, Section B.2.1.22 describes the two portions of the Selective Leaching program, the one-time portion and the periodic portion. For both portions, visual and mechanical inspections are performed on components susceptible to selective leaching. In addition, destructive examinations are performed on a minimum number of components in the periodic portion of the program. SLRA Section B.2.1.22 further states that the number of visual and mechanical selective leaching inspections may be reduced by two for each component that is destructively examined beyond the minimum number of destructive examinations recommended for each sample population. It does not specifically address whether this provision is applicable to both the one-time and the periodic portions of the program, or if it is only applicable to the periodic portion of the program.

NUREG-2221, "Technical Bases for Changes in the Subsequent License Renewal Guidance Documents NUREG-2191 and NUREG-2192," in Table 2-29 on page 2-348, provides additional clarification to this provision:

"Subsequent to the issuance of the GALL-SLR Report, the staff noted that the option to reduce the number of visual and mechanical inspections was only incorporated into the text addressing periodic inspections. It is the staff's intent that this same option can be incorporated into one-time inspections. Given that the one-time inspections did not incorporate a recommendation for a minimum number of destructive examinations, the first destructive examination (for example) would reduce the total number of visual and mechanical inspections by two."

Accordingly, Section B.2.1.22 of the SLRA is revised to specifically state that this provision is applicable to both the one-time portion and the periodic portion of the Selective Leaching program.

SLRA Section B.2.1.22, page B-129, third paragraph of the Program Description, is revised as shown below:

For the one-time and periodic/opportunistic portions of the program, visual inspections will be conducted on a representative sample of components of each material and environment combination of components. A representative sample consists of three percent of each material and environment population per unit or a maximum of 10 components per population per unit. Additionally, for the periodic/opportunistic portion of the program, two destructive examinations will be performed per population per unit for sample populations with greater than 35 susceptible components, or one destructive examination will be performed per population per unit for sample populations with less than or equal to 35 susceptible components. The number of visual and mechanical inspections may be reduced by two for each component that is destructively examined beyond the minimum number of destructive examinations recommended for each sample population ***in the periodic portion of the program. In addition, the number of visual and mechanical inspections may be reduced by two for each component that is destructively examined in the one-time portion of the program***. Since Peach Bottom is a multi-unit site, a reduced periodic visual inspection sample size of eight components maximum per population per unit will be adopted for sample populations that are not percentage-based. This sample size reduction is acceptable because, for the components in the scope of the periodic program, environmental conditions between the units are similar enough such that the aging effects are not occurring differently. Changes to water chemistry practices and to plant equipment and operating conditions (including power rerates) have been performed on both units at approximately the same time, or within a year of each other for those activities that required outage conditions for implementation. Water chemistry programs monitor various chemistry parameters, and require out-of-spec conditions to be corrected under the corrective action program in a timely manner. Raw water systems for both units draw from the same source, the Susquehanna River. Therefore, a reduced sample size will provide a representative sample of the condition of the plant equipment and the existence of the aging effects involved.

Change #3: Update to Reactor Pressure Vessel Internals System

Affected SLRA Sections: Table 2.3.1-1, Table 3.1.2-1, Section 3.1.2.2.13, Section 4.3.4, and Appendix A, Section A.4.3.4

SLRA Page Numbers: 2.3-7, 3.1-99, 3.1-100, 3.1-101, 3.1-102, 3.1-25, 4-82, and A-76

Description of Change:

SLRA Table 3.1.2-1, Reactor Pressure Vessel and Internals System, contains a line item that addresses “Reactor Vessel Internals Components (Jet Pump Oversized Wedges).” The line item specifies a “Loss of Preload” aging effect due to thermal or neutron irradiation-enhanced stress relaxation of jet pump repair hardware that is evaluated in the SLRA Section 4.2.11. An oversized wedge is a replacement for the original jet pump wedge, either of which is held in place by its own weight. Therefore, loss of preload is not an appropriate aging effect for oversized wedges. Also, it has been determined that although the plant design allows for their installation, oversized wedges have not been installed at PBAPS. Accordingly, the SLRA is revised to remove the Table 3.1.2-1 line items for oversized wedges. The same change is made to Table 2.3.1-1, Reactor Pressure Vessel and Internals System-Components Subject to Aging Management Review.

The TLAA, addressing loss of preload in Section 4.2.11, applies to jet pump auxiliary spring wedges that are installed in addition to the original jet pump wedges. Therefore, the SLRA Table 3.1.2-1 component type description for Reactor Vessel Internals Components “(Jet Pump Auxiliary Wedges)” is revised to Reactor Vessel Internals Components “(Jet Pump Auxiliary Spring Wedges)” for consistency with the SLRA Section 4 component description. The same change is made to Table 2.3.1-1, Reactor Pressure Vessel and Internals System-Components Subject to Aging Management Review.

SLRA Table 3.1.2-1, Intended Functions for Reactor Vessel Internals Systems, components types: Core Spray Repair Hardware, Jet Pump Auxiliary Spring Wedges, Jet Pump Riser Clamps, and Jet Pump Slip Joint Clamps, are revised from “Structural Integrity (Attached)” to “Structural Support.” These components support equipment that performs the safety-related functions of the core spray system and jet pumps and are safety-related. As defined in the SLRA Table 2.1-1, Passive Structure and Component Intended Function Definitions, the definition of “Structural Integrity (Attached)” pertains to nonsafety-related components and “Structural Support” is appropriate for safety-related components. The same change is made to Table 2.3.1-1, Reactor Pressure Vessel and Internals System-Components Subject to Aging Management Review.

SLRA Section 3.1.2.2.13, Loss of Fracture Toughness Due to Neutron Irradiation or Thermal Aging Embrittlement, addresses loss of fracture toughness due to neutron irradiation or thermal aging embrittlement in nickel alloy (including X-750 alloy) reactor internal components including jet pump auxiliary spring wedges. The component description “jet pump auxiliary wedges” is revised to “jet pump auxiliary spring wedges” for clarity and consistency with other SLRA sections and tables. In addition, a reference to “oversized wedges” is removed.

A review was performed to determine if there are other Section 3.0 Table TLAA line items that incorrectly reference sections in Section 4.0. The review concluded there are no other Section 3.0 TLAA line items that incorrectly reference sections in Section 4.0. However, it was identified that Section 4.3.4 does not specifically document the “Reactor Pressure Vessel and Internals System” among the list of systems which contain components that support the piping system’s

implicit ANSI B31.1 fatigue analyses. Therefore, Section 4.3.4 is revised to add the “Reactor Pressure Vessel and Internals System” and Appendix A, Section A.4.3.4 is revised to add the “Reactor Pressure Vessel and Internals System.”

Accordingly, SLRA Table 2.3.1-1, Table 3.1.2-1, Section 3.1.2.2.13, Section 4.3.4, and Section A.4.3.4 are revised.

SLRA Table 2.3.1-1, Reactor Pressure Vessel and Internals System - Components Subject to Aging Management Review, page 2.3-7 is revised as shown below:

Component Type	Intended Function
Jet Pump Assemblies: Jet pump sensing line	Direct Flow
Jet Pump Assemblies: Thermal sleeve inlet header, Riser brace arm, Hold-down beams, and Wedges	Direct Flow
Piping, piping components: Class 1 piping, fittings and branch connections less than 4" NPS and greater than or equal to 1" NPS	Pressure Boundary
Reactor Vessel (Bottom Head and Welds)	Pressure Boundary
Reactor Vessel (Shell and Welds)	Pressure Boundary
Reactor Vessel (Upper Head)	Pressure Boundary
Reactor Vessel Closure Flange Assembly Components	Mechanical Closure
	Pressure Boundary
Reactor Vessel External Attachments, Support Skirt, and Welds	Structural Support
Reactor Vessel Flange Leak Detection Line	Leakage Boundary
	Pressure Boundary
Reactor Vessel Internal Attachments	Structural Support to maintain core configuration and flow distribution
Reactor Vessel Internals Components (Core Spray Repair Hardware)	Pressure Boundary
	Structural ***Support*** ~~Integrity (Attached)~~
Reactor Vessel Internals Components (Jet Pump Auxiliary ***Spring*** Wedges)	Structural ***Support*** ~~Integrity (Attached)~~
~~Reactor Vessel Internals Components (Jet Pump Oversized Wedges)~~	~~Structural Integrity (Attached)~~
Reactor Vessel Internals Components (Jet Pump Riser Clamps)	Structural ***Support*** ~~Integrity (Attached)~~
Reactor Vessel Internals Components (Jet Pump Slip Joint Clamps)	Structural ***Support*** ~~Integrity (Attached)~~
Reactor Vessel Internals Components: Fuel Supports and Control Rod Drive Assemblies	Structural Support to maintain core configuration and flow distribution
	Throttle
Reactor Vessel Internals Components: Instrumentation	Structural Support to maintain core configuration and flow distribution
Reactor Vessel Internals Components: Steam Dryers	Structural Integrity (Attached)
Reactor Vessel Internals Components: Top Guide	Structural Support to maintain core configuration and flow distribution

SLRA Table 3.1.2-1, Reactor Pressure Vessel and Internals System, Summary of Aging Management Evaluation pages 3.1-99 through 3.1-102 are revised as shown below:

Page 3.1-99

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	NUREG-2192 Table 1 Item	Notes
Reactor Vessel Internal Attachments	Structural Support to maintain core configuration and flow distribution	Stainless Steel	Reactor Coolant	Cracking	BWR Vessel ID Attachment Welds (B.2.1.4)	IV.A1.R-64	3.1.1-094	A
					Water Chemistry (B.2.1.2)	IV.A1.R-64	3.1.1-094	B
				Cumulative Fatigue Damage	TLAA	IV.A1.R-04	3.1.1-007	A, 1
				Loss of Material	One-Time Inspection (B.2.1.21)	IV.A1.RP-157	3.1.1-085	A
					Water Chemistry (B.2.1.2)	IV.A1.RP-157	3.1.1-085	B
			Reactor Coolant and Neutron Flux	Cracking	BWR Vessel ID Attachment Welds (B.2.1.4)	IV.A1.R-64	3.1.1-094	A
					Water Chemistry (B.2.1.2)	IV.A1.R-64	3.1.1-094	B
				Cumulative Fatigue Damage	TLAA	IV.A1.R-04	3.1.1-007	A, 1
				Loss of Material	One-Time Inspection (B.2.1.21)	IV.A1.RP-157	3.1.1-085	A
					Water Chemistry (B.2.1.2)	IV.A1.RP-157	3.1.1-085	B
Reactor Vessel Internals Components (Core Spray Repair Hardware)	Pressure Boundary	Nickel Alloy	Reactor Coolant and Neutron Flux	Cracking	BWR Vessel Internals (B.2.1.7)	IV.B1.RP-381	3.1.1-104	B
					Water Chemistry (B.2.1.2)	IV.B1.RP-381	3.1.1-104	B
				Loss of Material	BWR Vessel Internals (B.2.1.7)	IV.B1.RP-26	3.1.1-043	E, 6
					Water Chemistry (B.2.1.2)	IV.B1.RP-26	3.1.1-043	B
	Structural ***Support*** ~~Integrity (Attached)~~	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	BWR Vessel Internals (B.2.1.7)	IV.B1.R-99	3.1.1-103	D
					Water Chemistry (B.2.1.2)	IV.B1.R-99	3.1.1-103	D

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1 Item	Notes
Reactor Vessel Internals Components (Core Spray Repair Hardware)	Structural ***Support*** ~~Integrity (Attached)~~	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Material	BWR Vessel Internals (B.2.1.7)	IV.B1.RP-26	3.1.1-043	E, 6
					Water Chemistry (B.2.1.2)	IV.B1.RP-26	3.1.1-043	B
				Loss of Preload	TLAA			H, 5
		X-750 alloy	Reactor Coolant and Neutron Flux	Cracking	BWR Vessel Internals (B.2.1.7)	IV.B1.RP-381	3.1.1-104	B
					Water Chemistry (B.2.1.2)	IV.B1.RP-381	3.1.1-104	B
				Loss of Material	BWR Vessel Internals (B.2.1.7)	IV.B1.RP-26	3.1.1-043	E, 6
					Water Chemistry (B.2.1.2)	IV.B1.RP-26	3.1.1-043	B
Reactor Vessel Internals Components (Jet Pump Auxiliary ***Spring*** Wedges)	Structural ***Support*** ~~Integrity (Attached)~~	X-750 alloy	Reactor Coolant and Neutron Flux	Cracking	BWR Vessel Internals (B.2.1.7)	IV.B1.RP-381	3.1.1-104	B
					Water Chemistry (B.2.1.2)	IV.B1.RP-381	3.1.1-104	B
				Loss of Material	BWR Vessel Internals (B.2.1.7)	IV.B1.RP-26	3.1.1-043	E, 6
					Water Chemistry (B.2.1.2)	IV.B1.RP-26	3.1.1-043	B
					BWR Vessel Internals (B.2.1.7)			H, 7
				Loss of Preload	TLAA			H, 4
~~Reactor Vessel Internals Components (Jet Pump Oversized Wedges)~~	~~Structural Integrity (Attached)~~	~~Stainless Steel~~	~~Reactor Coolant and Neutron Flux~~	~~Cracking~~	~~BWR Vessel Internals (B.2.1.7)~~	~~IV.B1.R-100~~	~~3.1.1-103~~	~~B~~
					~~Water Chemistry (B.2.1.2)~~	~~IV.B1.R-100~~	~~3.1.1-103~~	~~B~~
				~~Loss of Material~~	~~BWR Vessel Internals (B.2.1.7)~~	~~IV.B1.RP-26~~	~~3.1.1-043~~	~~E, 6~~

Page 3.1-101

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	NUREG-2192 Table 1 Item	Notes
~~Reactor Vessel Internals Components (Jet Pump Oversized Wedges)~~	~~Structural Integrity (Attached)~~	~~Stainless Steel~~	~~Reactor Coolant and Neutron Flux~~	~~Loss of Material~~	~~Water Chemistry (B.2.1.2)~~	~~IV.B1.RP-26~~	~~3.1.1-043~~	~~B~~
		~~X-750 alloy~~	~~Reactor Coolant and Neutron Flux~~	~~Cracking~~	~~BWR Vessel Internals (B.2.1.7)~~	~~IV.B1.RP-381~~	~~3.1.1-104~~	~~B~~
					~~Water Chemistry (B.2.1.2)~~	~~IV.B1.RP-381~~	~~3.1.1-104~~	~~B~~
				~~Loss of Material~~	~~BWR Vessel Internals (B.2.1.7)~~	~~IV.B1.RP-26~~	~~3.1.1-043~~	~~E, 6~~
					~~Water Chemistry (B.2.1.2)~~	~~IV.B1.RP-26~~	~~3.1.1-043~~	~~B~~
					~~BWR Vessel Internals (B.2.1.7)~~			~~H, 7~~
				~~Loss of Preload~~	~~TLAA~~			~~H, 4~~
Reactor Vessel Internals Components (Jet Pump Riser Clamps)	Structural ***Support*** ~~Integrity (Attached)~~	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	BWR Vessel Internals (B.2.1.7)	IV.B1.R-100	3.1.1-103	D
					Water Chemistry (B.2.1.2)	IV.B1.R-100	3.1.1-103	D
				Loss of Material	BWR Vessel Internals (B.2.1.7)	IV.B1.RP-26	3.1.1-043	E, 6
					Water Chemistry (B.2.1.2)	IV.B1.RP-26	3.1.1-043	B
				Loss of Preload	TLAA			H, 4
Reactor Vessel Internals Components (Jet Pump Slip Joint Clamps)	Structural ***Support*** ~~Integrity (Attached)~~	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	BWR Vessel Internals (B.2.1.7)	IV.B1.R-100	3.1.1-103	D
					Water Chemistry (B.2.1.2)	IV.B1.R-100	3.1.1-103	D
				Loss of Material	BWR Vessel Internals (B.2.1.7)	IV.B1.RP-26	3.1.1-043	E, 6
					Water Chemistry (B.2.1.2)	IV.B1.RP-26	3.1.1-043	B

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	NUREG-2192 Table 1 Item	Notes
Reactor Vessel Internals Components (Jet Pump Slip Joint Clamps)	Structural ***Support*** ~~Integrity (Attached)~~	X-750 alloy	Reactor Coolant and Neutron Flux	Cracking	BWR Vessel Internals (B.2.1.7)	IV.B1.RP-381	3.1.1-104	B
					Water Chemistry (B.2.1.2)	IV.B1.RP-381	3.1.1-104	B
				Loss of Material	BWR Vessel Internals (B.2.1.7)	IV.B1.RP-26	3.1.1-043	E, 6
					Water Chemistry (B.2.1.2)	IV.B1.RP-26	3.1.1-043	B
				Loss of Preload	TLAA			H, 4
Reactor Vessel Internals Components: Fuel Supports and Control Rod Drive Assemblies	Structural Support to maintain core configuration and flow distribution	Cast Austenitic Stainless Steel (CASS)	Reactor Coolant and Neutron Flux	Cracking	BWR Vessel Internals (B.2.1.7)	IV.B1.R-104	3.1.1-102	B
					Water Chemistry (B.2.1.2)	IV.B1.R-104	3.1.1-102	B
				Loss of Fracture Toughness	BWR Vessel Internals (B.2.1.7)	IV.B1.RP-220	3.1.1-099	B
				Loss of Material	BWR Vessel Internals (B.2.1.7)	IV.B1.RP-26	3.1.1-043	E, 6
					Water Chemistry (B.2.1.2)	IV.B1.RP-26	3.1.1-043	B
		Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	BWR Vessel Internals (B.2.1.7)	IV.B1.R-104	3.1.1-102	B
					Water Chemistry (B.2.1.2)	IV.B1.R-104	3.1.1-102	B
				Cumulative Fatigue Damage	TLAA	IV.B1.R-53	3.1.1-003	A, 1
				Loss of Material	BWR Vessel Internals (B.2.1.7)	IV.B1.RP-26	3.1.1-043	E, 6
					Water Chemistry (B.2.1.2)	IV.B1.RP-26	3.1.1-043	B
	Throttle	Cast Austenitic Stainless Steel (CASS)	Reactor Coolant and Neutron Flux	Cracking	BWR Vessel Internals (B.2.1.7)	IV.B1.R-104	3.1.1-102	B

SLRA Section 3.1.2.2.13, Loss of Fracture Toughness Due to Neutron Irradiation or Thermal Aging Embrittlement, page 3.1-25, last paragraph, is revised as shown below:

The nickel alloy reactor vessel internal components consist of the following; core shroud support assembly, core shroud access hole covers and bolting, jet pump holddown beams, jet pump repair hardware (e.g., auxiliary ***spring*** wedges~~, oversized auxiliary wedges,~~ and slip joint clamps) and core spray repair hardware (e.g., clamps). The reactor vessel internal components fabricated from nickel alloy are located in relatively low fluence areas or were installed later in plant life (i.e., repair hardware). The projected fluence of the nickel alloy components at the end of the second period of extended operation is less than $5x10^{20}$ n/cm^2, therefore loss of fracture toughness due to neutron irradiation embrittlement is not considered an applicable aging effect, therefore supplemental inspections or enhancements to the BWRVIP guidance are not necessary.

SLRA Section 4.3.4, ASME Section III, Class 2, Class 3, and ANSI B31.1 Allowable Stress Analyses, page 4-82, first full paragraph, is revised as shown below:

Portions of the following license renewal piping systems were designed in accordance with ANSI B31.1 requirements, but are attached to ASME Section III, Class 1 piping and are only affected by the same pressure and temperature transients as the Reactor Coolant System transients that are listed in Table 4.3.1-1 and Table 4.3.1-2: Control Rod Drive, Core Spray, Feedwater, Main Steam, Offgas and Recombiner, Primary Containment Isolation, ***Reactor Pressure Vessel and Internals,*** Reactor Recirculation, Reactor Vessel Instrumentation, Residual Heat Removal, and Standby Liquid Control Systems. Only a subset of the transients listed in Table 4.3.1-1 and Table 4.3.1-2 apply to the Class 2, Class 3, and ANSI B31.1 piping within each system. The summation of all 80-year transient cycle projections from each table is less than 3,500 cycles. Therefore, even if all operational Reactor Coolant System transients (transients 1 through 33) applied to each of these systems, the total number of projected 80-year cycles is less than 50 percent of 7000. Therefore, the stress range reduction factors originally applied for the components within these piping systems remain applicable and these implicit TLAAs remain valid through the second period of extended operation.

SLRA Section A.4.3.4, ASME Section III, Class 2, Class 3, and ANSI B31.1 Allowable Stress Analyses, page A-76, first paragraph, is revised as shown below:

Portions of the following Class 2 and 3 and ANSI B31.1 piping systems within the scope of license renewal are directly connected to Reactor Coolant System (RCS) and are affected by the same operational transients that result in thermal cycles for the attached Class 1 RCS piping: Control Rod Drive, Core Spray, Feedwater, Main Steam, Offgas and Recombiner, Primary Containment Isolation, ***Reactor Pressure Vessel and Internals,*** Reactor Recirculation, Reactor Vessel Instrumentation, Residual Heat Removal, and Standby Liquid Control Systems. These transient cycles have been projected for 80 years. The projections demonstrate that the total number of thermal cycles for these piping systems will not exceed 50 percent of the 7,000-cycle threshold that would result in a reduction in the stress range reduction factor. Therefore, these TLAAs have been demonstrated to remain valid through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).