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10 CFR 50
10 CFR 51
10 CFR 54

RS-15-223

August 26, 2015

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

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Subject: Response to NRC Requests for Additional Information, Set 7, dated July 27, 2015; and a Correction to Information associated with the Set 2 response to RAI B.2.1.20-2, related to the LaSalle County Station, Units 1 and 2, License Renewal Application (TAC Nos. MF5347 and MF5346)

References:

1. Letter from Michael P. Gallagher, Exelon Generation Company LLC (Exelon), to NRC Document Control Desk, dated December 9, 2014, "Application for Renewed Operating Licenses"

2. Letter from Jeffrey S. Mitchell, US NRC to Michael P. Gallagher, Exelon, dated July 27, 2015, "Requests for Additional Information for the Review of the LaSalle County Station, Units 1 and 2 License Renewal Application – Set 7 (TAC Nos. MF5347 and MF5346)"

3. Letter RS-15-165 from Michael P. Gallagher, Exelon, to NRC Document Control Desk, dated June 25, 2015, "Response to NRC Requests for Additional Information, Set 2, dated May 29, 2015 related to the LaSalle County Station, Units 1 and 2, License Renewal Application (TAC Nos. MF5347 AND MF5346)"

In Reference 1, Exelon Generation Company, LLC (Exelon) submitted the License Renewal Application (LRA) for the LaSalle County Station (LSCS), Units 1 and 2. In Reference 2, the NRC requested additional information to support staff review of the LRA.

Enclosure A contains the responses to this request for additional information.

Enclosure B contains updates to sections of the LRA (except for the License Renewal Commitment List) affected by the responses.

Enclosure C provides an update to the License Renewal Commitment List (LRA Appendix A, Section A.5). There are no other new or revised regulatory commitments contained in this letter.

In addition, Enclosure D of this letter provides a correction to a number that was contained in Enclosure B of Reference 3, related to the response to RAI B.2.1.20-2.

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

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

Executed on 08-26-2015

Respectfully,

Michael P. Gallagher

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

Enclosures: A: Responses to Set 7 Requests for Additional Information
B: LSCS License Renewal Application Updates
C: LSCS License Renewal Commitment List Updates
D: Correction to Enclosure B of Exelon letter RS-15-165 Associated with Response to Set 2 RAI B.2.1.20-2

cc: Regional Administrator – NRC Region III
NRC Project Manager (Safety Review), NRR-DLR
NRC Project Manager (Environmental Review), NRR-DLR
NRC Project Manager, NRR-DORL- LaSalle County Station
NRC Senior Resident Inspector, LaSalle County Station
Illinois Emergency Management Agency - Division of Nuclear Safety

Enclosure A

Responses to Requests for Additional Information related to various sections of the LaSalle County Station (LSCS) License Renewal Application (LRA)

RAI 2.3.3.2-1
RAI 2.3.3.4-1
RAI 2.3.3.11-1
RAI 2.3.3.13-1
RAI 2.3.3.13-2
RAI 2.3.3.16-1
RAI 2.3.3.21-1
RAI 3.2.2.2.5-1
RAI 3.5.1.78-1
RAI 3.5.2.2.1.5-1
RAI 4.2.5-1

RAI 2.3.3.2-1

Background:

License Renewal Application (LRA) Section 2.1 describes the applicant's scoping methodology, which specifies how systems or components were determined to be included within the scope of license renewal, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Paragraph 54.4(a). The staff reviewed section 2.3.3.2, "Combustible Gas Control System," and the associated License Renewal Boundary Drawings, and determined that additional information is needed to confirm that all components within the scope of license renewal were properly identified.

Issue:

One review method used by the staff is to confirm the inclusion of all components subject to aging management review (AMR) by reviewing the results of the screening of components within the license renewal boundary. The staff noted during its review of the drawings and locations indicated in the table below that continuations of piping within the scope of license renewal could not be located; therefore, acceptable scoping of systems, structures and components (SSCs) could not be verified.

License Renewal Boundary Drawing Number and Location	Continuation Issue
LR-LAS-M-130 Sheet 1, location A/B-8	212° and 30° continuations at the containment side of penetrations M-102 and M-95, respectively, were not provided.
LR-LAS-M-130 Sheet 2, location A/B-8	212° and 30° continuations at the containment side of penetrations M-102 and M-95, respectively, were not provided.

Request:

The staff requests that the applicant provide sufficient information to locate the license renewal boundary. If the continuation cannot be shown on license renewal boundary drawings, then provide additional information describing the extent of the scoping boundary and verify whether there are additional component types subject to an AMR between the continuation and the termination of the scoping boundary. If the scoping classification of a section of the piping changes over the continuation, provide additional information to clarify the change in scoping classification.

Exelon Response:

On license renewal boundary drawings LR-LAS-M-130, Sheet 1 (A/B-8) and LR-LAS-M-130 Sheet 2 (A/B-8), the references to M-102 and M-95 do not indicate continuations to other drawings. M-102 and M-95 are primary containment penetrations as indicated by the box with the dotted line going vertical above and below the box as shown on the index for the boundary drawings, LR-LAS-M-54, Sheet 2 (B-1). The 212 degree and 30 degree designations at the penetrations indicate the azimuth locations of the penetrations on the suppression pool shell.

The piping segments at primary containment penetrations M-102 and M-95 that extend inside the primary containment air space terminate at open ended piping, as shown on LR-LAS-M-130,

Sheets 1 and 2, and do not continue to other drawings. Therefore, the scoping boundary is not extended and there are no additional components or component types subject to AMR.

RAI 2.3.3.4-1

Background:

In LRA Section 2.1.5.2, subsection “Connected to and Provide Structural Support for Safety-Related SSCs,” the applicant states in part “[f]or nonsafety-related SSCs directly connected to safety-related SSCs the nonsafety-related piping and supports, up to and including the first seismic or equivalent anchor … beyond the safety/nonsafety interface, are within the scope of license renewal per 10 CFR 54.4(a)(2).” The staff reviewed section 2.3.3.4, “Control Rod Drive System,” and the associated License Renewal Boundary Drawings, and determined that additional information is needed to confirm that all components within the scope of license renewal were properly identified.

Issue:

The staff could not locate seismic or equivalent anchors between the safety/nonsafety interface and the end of the 10 CFR 54.4(a)(2) scoping boundary on the following drawings:

- LR-LAS-M-100-4 (B-2) downstream of safety-related valve 1C11-F381
- LR-LAS-M-146-4 (B-1) downstream of safety-related valve 2C11-F381
- LR-LAS-M-93-1 (B-8) upstream of safety-related valve 1B33-F017A
- LR-LAS-M-139-1 (B-8) upstream of safety-related valve 2B33-F017A
- LR-LAS-M-100-3 and LR-LAS-M-146-3 (C-4, 5, 6 and 7) upstream of safety-related valves 5, 3, 4 and 9
- LR-LAS-M-100-5 and LR-LAS-M-146-6 (E-1, B-4, C-4, E-4, F-4 and E-8)

Request:

The staff requests that the applicant provide additional information to locate the seismic or equivalent anchors between the safety/nonsafety interface and the end(s) of the 10 CFR 54.4(a)(2) scoping boundary.

Exelon Response:

Each of the following safety/nonsafety interfaces transition to nonsafety-related (colored red) piping that connect to a Plant Drainage System drain header that includes other piping connections that are not in scope for license renewal (colored Black):

- LR-LAS-M-100-4 (B-2) downstream of safety-related valve 1C11-F381
- LR-LAS-M-146-4 (B-1) downstream of safety-related valve 2C11-F381
- LR-LAS-M-93-1 (B-8) upstream of safety-related valve 1B33-F017A
- LR-LAS-M-139-1 (B-8) upstream of safety-related valve 2B33-F017

Equivalent anchors between the safety/nonsafety interface and the end(s) of the 10 CFR 54.4(a)(2) scoping boundary are not shown. To correct this condition, equivalent anchor symbols “F.4.f” are added for these safety/nonsafety interfaces as described below:

- Nonsafety-related line 1REG9A-2” (colored red) downstream of safety-related valve 1C11-F381 shown on drawing LR-LAS-M-100, Sheet 4 (B-2), continues to drawing LR-LAS-M-91, Sheet 6 (C-3) where it connects to line 1RE06A-8”. Line 1RE06A-8” has a

connection from line 0TE03A-3” that is not in scope for license renewal (colored black) as shown on drawing LR-LAS-M-91, Sheet 6 (A-4). Also, line 1RE06A-8” transitions to line 1RE01AB-6”, which has a connection from 1RT25B-3/4” that is not in scope for license renewal (colored black) as shown on drawing LR-LAS-M-91, Sheet 6 (E-3). Note that the line labeled 1RE01AB-6” is mis-labeled and should be labelled 1RED1AB-6”. This line will be referred to as 1RED1AB-6” throughout this response. Anchor symbol “F.4.f” is added on line 1RT25B-3/4” at the tee connection to line 1RED1AB-6” as shown on LR-LAS-M-91, Sheet 6 (E-3) and on line 0TE03A-3” at the tee connection to line 1RE06A-8” as shown on LR-LAS-M-91, Sheet 6 (A-3). With these anchor endpoints, line 1REG9A-2” continues as a drain with the entire run of downstream piping in scope, ending at the Reactor Building Equipment Drain Tank as shown on drawing LR-LAS-M-91, Sheet 5 (E-1). With this change, the existing anchor symbol “F.4.c” on line 1REG9A-2” shown on LR-LAS-M-100, Sheet 4 (B-2) is correct.

- Nonsafety-related line 2REG9A-2” (colored red) downstream of safety-related valve 2C11-F381 shown on drawing LR-LAS-M-146, Sheet 4 (B-1), continues to drawing LR-LAS-M-137, Sheet 6 (C-3) where it connects to line 2RE06A-8”. Line 2RE06A-8” has a connection from line 2REL7B-1” that is not in scope for license renewal (colored black) as shown on drawing LR-LAS-M-137, Sheet 6 (C-3). Also, line 2RE06A-8” transitions to line 2RED1AB-6”, which has a connection from 2RT25B-3/4” that is not in scope for license renewal (colored black) as shown on drawing LR-LAS-M-137, Sheet 6 (D-3). Anchor symbol “F.4.f” is added on line 2RT25B-3/4” at the tee connection to line 2RED1AB-6” as shown on LR-LAS-M-137, Sheet 6 (D-3) and on line 2REL7B-1” at the tee connection to line 2RE06A-8” as shown on LR-LAS-M-137, Sheet 6 (C-3). With these anchor endpoints, line 2REG9A-2” continues as a drain with the entire run of downstream piping in scope, ending at the Reactor Building Equipment Drain Tank as shown on drawing LR-LAS-M-137, Sheet 5 (E-1). With this change, the existing anchor symbol “F.4.c” on line 2REG9A-2” shown on LR-LAS-M-146, Sheet 4 (B-1) is correct.
- Nonsafety-related branch line 1RE95A-1½” (colored red) upstream of safety-related valve 1B33-F017A shown on drawing LR-LAS-M-93, Sheet 1 (B-8) continues to drawing LR-LAS-M-91, Sheet 6 (C-2) where it connects to line 1RE06A-8”. As discussed above, lines 1RE06A-8” and 1RED1AB-6” have connections from lines 1RT25B-3/4” and 0TE03A-3” that are not in scope for license renewal (colored black), and anchor symbol “F.4.f” is added on lines 1RT25B-3/4” and 0TE03A-3” at the tee connections to lines 1RED1AB-6” and 1RE06A-8”, respectively. With these anchor endpoints, line 1RE95A-1½” continues as a drain with the entire run of downstream piping in scope, ending at the Reactor Building Equipment Drain Tank as shown on drawing LR-LAS-M-91, Sheet 5 (E-1). With this change, the existing anchor symbol “F.4.c” on line 1RE95A-1½” shown on LR-LAS-M-93, Sheet 1 (A-8) is correct.
    - Note that branch lines 1RD12A-3/4”, 1RR24BB-3/4”, and 1RE95AB-3/4” upstream of valve 1B33-F017A do not need additional anchor symbols added. Branch line 1RD12A-3/4”, continuing from drawing LR-LAS-M-100, Sheet 1 (E-3), has a connection with piping that is not in scope for license renewal (colored black) on line 1RD01A-6” at the Turbine Building to Auxiliary Building wall penetration, which is a seismic anchor as indicated by the anchor symbol “F.4.1” shown on drawing LR-LAS-M-100, Sheet 1 (E-8). Branch lines 1RR24BB-3/4” and 1RE95AB-3/4”, continuing to drawing LR-LAS-M-93, Sheet 2 (A-8), do not connect with piping that is not in scope for license renewal.
- Nonsafety-related branch line 2RE95B-1½” (colored black) upstream of safety-related valve 2B33-F017A shown on drawing LR-LAS-M-139, Sheet 1 (B-8) continues to

drawing LR-LAS-M-137, Sheet 6 (B-3) where it connects to line 2RE06A-8”. As discussed above, lines 2RED1AB-6” and 2RE06A-8” have connections from lines 2RT25B-3/4” and 2REL7B-1” that are not in scope for license renewal (colored black), and anchor symbol “F.4.f” is added on lines 2RT25B-3/4” and 2REL7B-1” at the tee connections to lines 2RED1AB-6” and 2RE06A-8”, respectively. With these anchor endpoints, line 2RE95B-1½” continues as a drain with the entire run of downstream piping in scope, ending at the Reactor Building Equipment Drain Tank as shown on drawing LR-LAS-M-137, Sheet 5 (E-1). With this change, the existing anchor symbol “F.4.c” on line 2RE95B-1½” shown on LR-LAS-M-139, Sheet 1 (A-8) is correct.

- Note that branch lines 2RD12A-3/4”, 2RR24BB-3/4”, and 2RE95AB-3/4” do not need additional anchor symbols added. Branch line 2RD12A-3/4”, continuing from drawing LR-LAS-M-146, Sheet 1 (E-3), has a connection to piping that is not in scope for license renewal (colored black) on line 2RD01A-6” at the Turbine Building to Auxiliary Building wall penetration which is a seismic anchor as indicated by the anchor symbol “F.4.1” shown on drawing LR-LAS-M-146, Sheet 1 (E-8). Branch lines 2RR24BB-3/4” and 2RE95AB-3/4”, continuing to drawing LR-LAS-M-139, Sheet 2 (A-8) do not connect with piping that is not in scope for license renewal.

Anchor symbol “F.4.f” indicates that a smaller branch line that does not impose loads on larger piping is considered decoupled from the larger piping within the piping analysis. The Exelon General Station Piping Analysis utilizes a moment of inertia ratio of 7 or larger as criteria to decouple smaller branch lines from larger diameter piping. In the analysis of the larger line, the small branch line does not affect the response of the larger line. The moment of inertia ratio for lines 1RED1AB-6” to 1RT25B-3/4”, and 2RED1AB-6” to 2RT25B-3/4” is 628, therefore lines 1RT25B-3/4” and 2RT25B-3/4” are decoupled from lines 1RED1AB-6” and 2RED1AB-6”, respectively. The moment of inertia ratio for line 1RE06A-8” to 0TE03A-3” is 24.0; therefore line 0TE03A-3” is decoupled from 1RE06A-8”. The moment of inertia ratio for line 2RE06A-8” to 2REL7B-1” is 686; therefore line 2REL7B-1” is decoupled from 2RE06A-8”.

All of the structural endpoints that are added to drawings LR-LAS-M-91, Sheet 6 and LR-LAS-M-137, Sheet 6, as discussed above, are beyond the safety/nonsafety interface and are within the Plant Drainage System license renewal scoping boundary for spatial interaction per 10 CFR 54.4(a)(2). Therefore, the scoping boundary is not extended and there are no additional components or component types subject to AMR. License Renewal Drawings LR-LAS-M-91, Sheet 6 and LR-LAS-M-137, Sheet 6 are revised as described above to add the anchor symbols “F.4.f”.

<u>LR-LAS-M-100-3 and LR-LAS-M-146-3 (C-4, 5, 6 and 7) upstream of safety-related valves 5, 3, 4 and 9</u>

For Unit 1, (LR-LAS-M-100, Sheet 3):

Note that the piping <u>downstream</u> of safety-related valve 5 is nonsafety-related. Nonsafety-related piping that is upstream of safety-related valves 3, 4, and 9 and downstream of safety-related valve 5, (colored red) as shown on drawing LR-LAS-M-100, Sheet 3 (C-4, C-5, C-6, and C-7) interface with piping that is not in scope for license renewal (colored black) on line 1RD01A-6” at the Turbine Building to Auxiliary Building wall penetration, which is a seismic anchor as indicated by the anchor symbol “F.4.1” shown on drawing LR-LAS-M-100, Sheet 1 (E-8). Therefore, the scoping boundary is not extended and there are no additional components or component types subject to AMR.

For Unit 2, (LR-LAS-M-146, Sheet 3):

Note that the piping downstream of safety-related valve 5 is nonsafety-related. Non-Safety-related piping upstream of safety-related valves 3, 4, and 9 and downstream of safety-related valve 5, (colored red) as shown on LR-LAS-M-146, Sheet 3 (C-4, C-5, C-6, and C-7) interface with piping that is not in scope for license renewal (colored black) on line 2RD01A-6" at the Turbine Building to Auxiliary Building wall penetration, which is a seismic anchor as indicated by the anchor symbol "F.4.1" shown on drawing LR-LAS-M-146, Sheet 1 (E-8). Therefore, the scoping boundary is not extended and there are no additional components or component types subject to AMR.

Note that the anchor symbols "F.4.1" at the Turbine Building to Auxiliary Building wall penetrations described above provide the structural endpoints for all safety/nonsafety interfaces associated with piping connected to the discharge of the Control Rod Drive System pumps.

LR-LAS-M-100-5 and LR-LAS-M-146-6 (E-1, B-4, C-4, E-4, F-4 and E-8)

For Unit 1, (LR-LAS-M-100, Sheet 5):

Nonsafety-related piping upstream of continuation flags shown on LR-LAS-M-100, Sheet 5 (E-1, B-4, C-4, E-4, F-4, and E-8) (colored red) interface with piping that is not in scope for license renewal (colored black) on line 1RD01A-6" at the Turbine Building to Auxiliary Building wall penetration, which is a seismic anchor as indicated by the anchor symbol "F.4.1" shown on drawing LR-LAS-M-100, Sheet 1 (E-8). Therefore, the scoping boundary is not extended and there are no additional components or component types subject to AMR.

For Unit 2, (LR-LAS-M-146, Sheet 6):

Nonsafety-related piping upstream of continuation flags shown on LR-LAS-M-146, Sheet 6 (E-1, B-4, C-4, E-4, F-4, and E-8) (colored red) interface with piping that is not in scope for license renewal (colored black) on line 2RD01A-6" at the Turbine Building to Auxiliary Building wall penetration, which is a seismic anchor as indicated by the anchor symbol "F.4.1" shown on drawing LR-LAS-M-146, Sheet 1 (E-8). Therefore, the scoping boundary is not extended and there are no additional components or component types subject to AMR.

The Response to RAI 2.3.3.16-1 includes other branch lines that are beyond a safety/nonsafety interface, are connected to Unit 1 line 1RE06A-8" shown on drawing LR-LAS-M-91, Sheet 6 or Unit 2 line 2RE06A-8" shown on drawing LR-LAS-M-137, Sheet 6, and do not include proper equivalent anchor symbols. The Response to RAI 2.3.3.16-1 includes a summary of the extent of condition review performed for this condition.

RAI 2.3.3.11-1

Background:

LRA Section 2.1 describes the applicant's scoping methodology, which specifies how systems or components were determined to be included within the scope of license renewal, in accordance with 10 CFR 54.4(a). The staff confirms the inclusion of all components subject to AMR by reviewing the results of the screening of components within the license renewal boundary. The staff reviewed section 2.3.3.11, "Essential Cooling Water System" and the associated License Renewal Boundary Drawings, and determined that additional information is needed to confirm that all components within the scope of license renewal were properly identified.

Issue:

Drawings LR-LAS-M-87-3 (E-6) and LR-LAS-M-134-3 (D-6) show 10 CFR 54.4(a)(1) piping whose scope changes to 10 CFR 54.4(a)(2) without a change in piping classification.

Request:

The staff requests that the applicant provide sufficient information to clarify the change in scoping classification.

Exelon Response:

The 3G designation in the half circle shown just downstream of valve 1DG039 on LR-LAS-M-87, Sheet 3 (E-6) indicates that valve 1DG039 and upstream piping are ASME Code Class 3 components and tested to ASME Code Section XI requirements specified on drawing LR-LAS-M-54, Sheet 6 (D-8). Valve 1DG039 is safety-related with a 10 CFR 54.4(a)(1) pressure boundary intended function. The piping components downstream of valve 1DG039 are not ASME Code components, are not safety-related, and do not have a 10 CFR 54.4(a)(1) intended function. These piping components have 10 CFR 54.4(a)(2) intended functions to provide structural support of attached piping with 10 CFR 54.4(a)(1) intended functions and provide a leakage boundary for nearby components that have 10 CFR 54.4(a)(1) intended functions.

The 3G designation in the half circle shown just downstream of valve 2DG039 on LR-LAS-M-134, Sheet 3 (D-6) indicates that valve 2DG039 and upstream piping are ASME Code Class 3 components and tested to ASME Code Section XI requirements specified on drawing LR-LAS-M-54, Sheet 6 (D-8). Valve 2DG039 is safety-related with an (a)(1) pressure boundary intended function. The piping components downstream of valve 2DG039 are not ASME Code components, are not safety-related, and do not have a 10 CFR 54.4(a)(1) intended function. These piping components have 10 CFR 54.4(a)(2) intended functions to provide structural support of attached piping with 10 CFR 54.4(a)(1) intended functions and provide a leakage boundary for nearby components that have 10 CFR 54.4(a)(1) intended functions.

RAI 2.3.3.13-1

Background:

LRA Section 2.1 describes the applicant's scoping methodology, which specifies how systems or components were determined to be included within the scope of license renewal, in accordance with 10 CFR 54.4(a). The staff confirms the inclusion of all components subject to AMR by reviewing the results of the screening of components within the license renewal boundary. The staff reviewed section 2.3.3.13, "Fuel Pool Cooling and Storage System" and the associated License Renewal Boundary Drawings, and determined that additional information is needed to confirm that all components within the scope of license renewal were properly identified.

Issue:

Unit 1 drawing LR-LAS-M-98-1 shows 10 CFR 54.4(a)(1) piping whose scope changed to 10 CFR 54.4(a)(2) without a change in piping classification at the following locations:

LR-LAS-M-98, Sheet-1, Location	Piping ID
Location C-2	1FC87A upstream of valve 1FC130
Location C-5	1FC19AA downstream of valve 1FC118
Location C-8	1FC110A upstream of valve 1FC141
Location B-8	1FC01DA downstream of valve 1FC139A
Location A-8	1FC01DB downstream of valve 1FC139B

Similarly for Unit 2, drawing LR-LAS-M-144, Sheet-1, shows 10 CFR 54.4(a)(1) piping whose scope changed to CFR 54.4(a)(2) without any change in system scope or piping classification at the following locations:

LR-LAS-M-144, Sheet-1, Location	Piping ID
Location C-1	2FC11DA upstream of valve 2FC141
Location B-1	2FC01DA downstream of valve 2FC139A
Location B-1	2FC01DB downstream of valve 2FC139B
Location C-4	2FC19AA downstream of valve 2FC118
Location C-7	2FC87A upstream of valve 2FC130

Request:

The staff requests that the applicant provide sufficient information to clarify the change in scoping classification.

Exelon Response:

The piping sections listed in the above Issue that are shown on drawings LR-LAS-M-98, Sheet 1 and LR-LAS-M-144, Sheet 1 are all properly scoped as follows:

LR-LAS-M-98, Sheet 1, Location	Piping ID	Scoping Classification
Location C-2	1FC87A upstream of valve 1FC130	10 CFR 54.4(a)(2)
Location C-5	1FC19AA downstream of valve 1FC118	10 CFR 54.4(a)(2)
Location C-8	1FC110A upstream of valve 1FC141	10 CFR 54.4(a)(2)
Location B-8	1FC01DA downstream of valve 1FC139A	10 CFR 54.4(a)(2)
Location A-8	1FC01DB downstream of valve 1FC139B	10 CFR 54.4(a)(2)

LR-LAS-M-144, Sheet 1, Location	Piping ID	Scoping Classification
Location C-1	2FC11DA upstream of valve 2FC141	10 CFR 54.4(a)(2)
Location B-1	2FC01DA downstream of valve 2FC139A	10 CFR 54.4(a)(2)
Location B-1	2FC01DB downstream of valve 2FC139B	10 CFR 54.4(a)(2)
Location C-4	2FC19AA downstream of valve 2FC118	10 CFR 54.4(a)(2)
Location C-7	2FC87A upstream of valve 2FC130	10 CFR 54.4(a)(2)

The LSCS System Safety Boundary Document for the Fuel Pool Cooling Filter and Demineralizer (FC) System, that was prepared by Sargent & Lundy Engineers, states that the normal fuel pool cooling loops are not safety-related, are designed as a nonseismic and electrical non-1E system, but are designed and procured to ASME Section III standards. The residual heat removal (RHR) system pump, heat exchanger, and piping that provides the flow path to and from fuel pool is safety-related and designed to seismic Category 1 and electrical Class 1E requirements. The emergency makeup water supply system to the fuel pool from the core standby cooling system (CSCS), piping components that form primary containment boundaries, or are required to maintain fuel pool integrity are also safety-related and designed to seismic Category 1 requirements. The following safety-related boundary valves maintain the flow path integrity for the safety-related cooling loop: 1(2)FC130, 1(2)FC141, 1(2)FC139A, and 1(2)FC139B. Valves 1(2)FC118 are required to maintain fuel pool integrity and are safety-related.

The piping sections listed in the Issue on the nonsafety-related side of these boundary valves do not have 10 CFR 54.4(a)(1) intended functions. These piping sections have 10 CFR 54.4(a)(2) intended functions to provide structural support for attached piping with 10 CFR 54.4(a)(1) intended functions and provide a leakage boundary for nearby components that have 10 CFR 54.4(a)(1) intended functions.

UFSAR Table 3.2-1 Sheet 8, Section XX, Fuel Pool Cooling and Cleanup System and associated notes 4a, 4b, 5, and 18 provide design bases information that is consistent with the information discussed above from the System Safety Boundary Document for the Fuel Pool Cooling Filter and Demineralizer (FC) System. The only fuel pool cooling and cleanup system

components that are listed as being safety-related (Quality Assurance Requirement I) and Seismic Category I are the pumps, valves, and piping associated with emergency makeup. Except for some nonsafety-related piping and valves and the precoat facility that are designed to Quality Group Classification D (ANSI B31.1), and safety-related piping associated with the primary containment boundary at penetrations M-65 and M-59 that are designed to Quality Group Classification B (ASME Section III Class 2); all other fuel pool cooling and cleanup system piping, including safety-related and nonsafety-related piping, is designed to Quality Group Classification C (ASME Section III Class 3). Therefore, although there is no change in piping classification at the locations discussed in this RAI, the design and licensing basis information described above provides the basis for why the license renewal scoping classification change from 10 CFR 54.4(a)(1) to 10 CFR 54.4(a)(2) is appropriate.

RAI 2.3.3.13-2

Background:

LRA Section 2.1 describes the applicant's scoping methodology, which specifies how systems or components were determined to be included within the scope of license renewal, in accordance with 10 CFR 54.4(a). The staff confirms the inclusion of all components subject to AMR by reviewing the results of the screening of components within the license renewal boundary. The staff reviewed section 2.3.3.13, "Fuel Pool Cooling and Storage System" and the associated License Renewal Boundary Drawings, and determined that additional information is needed to confirm that all components within the scope of license renewal were properly identified.

Issue:

Unit 1 drawing LR-LAS-M-98-1 (C-7) shows a 10 CFR 54.4(a)(2) line 1FC11DC 10 downstream of a 10 CFR 54.4(a)(1) valve 1FC086 whose scope changed to 10 CFR 54.4(a)(2) while the piping classification changed to 'Class C' indicating ASME Section III-Class 3 piping.

Unit 2 drawing LR-LAS-M-144-1 (C-2) shows a 10 CFR 54.4(a)(2) line 2FC11DC 10 downstream of a 10 CFR 54.4(a)(1) valve 2FC086 whose scope changed to 10 CFR 54.4(a)(2) while the piping classification changed to 'Class C' indicating ASME Section III-Class 3 piping.

Request:

The staff requests that the applicant provide sufficient information to clarify the change in scoping classification.

Exelon Response:

The piping sections listed in the above Issue that are shown on drawings LR-LAS-M-98, Sheet 1 and LR-LAS-M-144, Sheet 1 are properly scoped as being in scope for 10 CFR 54.4(a)(2) intended functions. These piping sections are classified within the LSCS current design basis as nonsafety-related and they do not have 10 CFR 54.4(a)(1) intended functions.

The LSCS System Safety Boundary Document for the Fuel Pool Cooling Filter and Demineralizer (FC) System, that was prepared by Sargent & Lundy Engineers, states that the normal fuel pool cooling loops are not safety-related, are designed as a nonseismic and electrical non-1E system, but are designed and procured to ASME Section III standards. The residual heat removal (RHR) system pump, heat exchanger, and piping that provides the flow path to and from fuel pool are safety-related and designed to seismic Category 1 and electrical Class 1E requirements. The emergency makeup water supply system to the fuel pool from the core standby cooling system (CSCS), piping components that form primary containment boundaries, or are required to maintain fuel pool integrity are also safety-related and designed to seismic Category 1 requirements. Piping that is outside of these boundaries is not safety-related and does not have 10 CFR 54.4(a)(1) intended functions. These classifications are consistent with UFSAR Table 3.2-1 Sheet 8, Section XX, Fuel Pool Cooling and Cleanup System and associated notes 4a, 4b, 5, and 18.

Lines 1(2)FC11C-10”, up to and including valves 1(2)FC086, are safety-related and have a 10 CFR 54.4(a)(1) intended function to form the primary containment boundary for primary containment penetration M-65. Lines 1(2)FC11DC-10” that are downstream of valves 1(2)FC086 are not safety-related and do not have a 10 CFR 54.4(a)(1) intended function since they are not within the safety-related flow path for the cooling loop to or from fuel pool to the RHR system; are not part of the emergency water makeup system to the fuel pool; are not part of the primary containment boundary; and are not required to maintain fuel pool integrity. Lines 1(2)FC11DC-10” have 10 CFR 54.4(a)(2) intended functions to provide structural support for attached piping that have 10 CFR 54.4(a)(1) intended functions and to provide a leakage boundary for nearby components that have 10 CFR 54.4(a)(1) intended functions.

RAI 2.3.3.16-1

Background:

In LRA Section 2.1.5.2, subsection “Connected to and Provide Structural Support for Safety-Related SSCs,” the applicant states in part “[f]or nonsafety-related SSCs directly connected to safety-related SSCs the nonsafety-related piping and supports, up to and including the first seismic or equivalent anchor … beyond the safety/nonsafety interface are within the scope of license renewal per 10 CFR 54.4(a)(2).” The staff reviewed section 2.3.3.16, “Plant Drainage System,” and the associated License Renewal Boundary Drawings, and determined that additional information is needed to confirm that all components within the scope of license renewal were properly identified.

Issue:

On drawing LR-LAS-M-142-1 (A-4), the staff could not locate seismic or equivalent anchors between the safety/nonsafety interface at the F.4.c termination symbol (valve 2E12-F070) and the end of the 10 CFR 54.4(a)(2) scoping boundary.

Request:

The staff requests that the applicant provide additional information to locate the seismic or equivalent anchors between the safety/nonsafety interface and the end(s) of the 10 CFR 54.4(a)(2) scoping boundary.

Exelon Response:

The safety/nonsafety interface downstream of safety-related valve 2E12-F070 at the anchor symbol “F.4.c”, shown on drawing LR-LAS-M-142, Sheet 1 (A-4) transitions to nonsafety-related (colored red) piping that connects to a Plant Drainage System drain header that includes other piping connections that are not in scope for license renewal (colored black). To correct this condition, equivalent anchor symbols “F.4.f” are being added for these safety/nonsafety interfaces as described below.

Nonsafety-related line 2RE26A-3” (colored red) downstream of safety-related valve 2E12-F070 shown on drawing LR-LAS-M-142, Sheet 1 (A-4) continues to drawing LR-LAS-M-137, Sheet 6 (B-3) where it connects to line 2RE06A-8”. Line 2RE06A-8” has a connection with line 2REL7B-1” that is not in scope for license renewal (colored black) as shown on drawing LR-LAS-M-137, Sheet 6 (C-3). Also, line 2RE06A-8” transitions to line 2RED1AB-6” that has a connection to 2RT25B-3/4” that is not in scope for license renewal (colored black) as shown on drawing LR-LAS-M-137, Sheet 6 (D-3). Therefore, a seismic or equivalent anchor must be identified between valve 2E12-F070 and the lines that are not in scope for license renewal.

RAI 2.3.3.4-1 identified other branch lines from safety/nonsafety interfaces that connect to line 2RE06A-8” as shown on LR-LAS-M-137, Sheet 6. To correct that condition, the Response to RAI 2.3.3.4-1 adds anchor symbol “F.4.f” on lines 2RT25B-3/4” as shown on drawing LR-LAS-M-137, Sheet 6 (D-3) and 2REL7B-1” as shown on drawing LR-LAS-M-137, Sheet 6 (C-3) at the tee connections to lines 2RED1AB-6” and 2RE06A-8”, respectively. With these additional anchor endpoints, line 2RE26A-3” continues as a drain pipe with the entire run of piping in scope, ending at the Reactor Building Equipment Drain Tank as shown on drawing LR-LAS-

M-137, Sheet 5 (E-1). With this change, the existing anchor symbol “F.4.c” downstream of valve 2E12-F070 on line 2RE26A-3” shown on LR-LAS-M-142, Sheet 1 (A-4) is correct.

The Response to RAI 2.3.3.4-1 also addressed safety/nonsafety interfaces that connect to Unit 1 line 1RE06A-8” and did not include proper anchor endpoints. Lines 1RE06A-8” and 1RED1AB-6” have connections from lines 0TE03A-3” and 1RT25B-3/4”, respectively that are not in scope for license renewal (colored black) as shown on drawing LR-LAS-M-91, Sheet 6. To correct that condition, the Response to RAI 2.3.3.4-1 adds anchor symbol “F.4.f” on lines 1RT25B-3/4” as shown on LR-LAS-M-91, Sheet 6 (E-3) and 0TE03A-3” as shown on LR-LAS-M-91, Sheet 6 (A-3).

An extent of condition review identified the following additional branch lines that are beyond a safety/nonsafety interface and are connected to Unit 2 line 2RE06A-8” shown on drawing LR-LAS-M-137, Sheet 6, or Unit 1 line 1RE06A-8” shown on drawing LR-LAS-M-91, Sheet 6 that do not include proper equivalent anchor symbols.

Drawing Location	**Large Line**	**Smaller Branch Line**
LR-LAS-M-137, Sheet 6 (B-3)	2RE06A-8”	2RE37AA-3”
LR-LAS-M-91, Sheet 6 (A-3)	1RE06A-8”	1RE26A-3”
LR-LAS-M-91, Sheet 6 (A-3)	1RE06A-8”	1RE37AA-3”
LR-LAS-M-91, Sheet 6 (B-3)	1RE06A-8”	1REP4A-1½ ”

For all these conditions, the anchor symbols “F.4.f” that are added by the Response to RAI 2.3.3.4-1 provide the proper endpoints for the 10 CFR 54.4(a)(2) scoping boundary.

RAI 2.3.3.21-1

Background:

LRA Section 2.1 describes the applicant's scoping methodology, which specifies how systems or components were determined to be included within the scope of license renewal, in accordance with 10 CFR Paragraph 54.4(a). The staff confirms the inclusion of all components subject to AMR by reviewing the results of the screening of components within the license renewal boundary. The staff reviewed section 2.3.3.21, "Reactor Water Cleanup System" and the associated License Renewal Boundary Drawings, and determined that additional information is needed to confirm that all components within the scope of license renewal were properly identified.

Issue:

Unit 1 drawing LR-LAS-M-97-1 shows 10 CFR 54.4(a)(1) piping whose scope changes to 10 CFR 54.4(a)(2) while the piping classification changes to 'Class C' indicating ASME Section III-Class 3 piping at the following locations:

LR-LAS-M-97, Sheet-1, Location	Piping ID
Location E-7	Line 1RT01C 4 downstream of valve 1G33-F004
Location F-4	Line 1RT06B 4 downstream of valve 1G33-F040

Similarly for Unit 2, drawing LR-LAS-M-143, Sheet-1, shows 10 CFR 54.4(a)(1) piping whose scope changed to CFR 54.4(a)(2) while the piping classification changes to 'Class C' indicating ASME Section III-Class 3 piping at the following locations:

LR-LAS-M-143, Sheet-1, Location	Piping ID
Location E-7	2RT01C 4 downstream of valve 2G33-F004
Location F-4	2RT06B 4 downstream of valve 2G33-F040

Request:

The staff requests that the applicant provide sufficient information to clarify the change in scoping classification.

Exelon Response:

The piping sections listed in the above Issue that are shown on drawings LR-LAS-M-97 Sheet 1 and LR-LAS-M-143 Sheet 1 are properly scoped as being in scope for 10 CFR 54.4(a)(2) intended functions. These piping sections are classified within the LSCS current design basis as nonsafety-related and they do not have 10 CFR 54.4(a)(1) intended functions.

UFSAR Table 3.2-1 Sheet 8, Section XIX, Reactor Water Cleanup System and associated notes 4a, 4b, 5, 10, 14, and 39 provide design bases information that describe the design bases for the reactor water cleanup system. The only reactor water cleanup system components that are listed as being Quality Assurance Requirement I (safety-related) and Seismic Category I are the piping and valves within the reactor coolant pressure boundary (RCPB) and the primary containment boundary. Piping and valves within the RCPB are Quality Group Classification A

(ASME Section III Class 1), and piping and valves within the primary containment boundary (that are not also within the RCPB) are Quality Group Classification B (ASME Section III Class 2). All other reactor water cleanup system piping is Quality Assurance Requirement II (nonsafety-related), Seismic Category II (nonseismic), and Quality Group Classification C (ASME Section III Class 3). Note that piping components within the plant reactor water cleanup system that are part of the RCPB are scoped within the Reactor Coolant Pressure Boundary license renewal system.

Piping classification designations on the boundary drawings that start with “A” (e.g. A901LS) indicates ASME Section III Class 1; “B” indicates ASME Section III Class 2; and “C” indicates ASME Section III Class 3, as shown on drawing LR-LAS-M-54, Sheet 1 (F-5).

Valves 1G33-F004 and 2G33-F004 are ASME Section III Class 1 safety-related valves that function as outboard RCPB and outboard primary containment boundary valves. Downstream piping 1(2)RT01C-4” (colored red) is outside of the RCPB and primary containment boundary and is ASME Section III Class 3 as shown on the drawings. Lines 1(2)RT01C-4” are therefore nonsafety-related, nonseismic, and do not have 10 CFR 54.4(a)(1) intended functions. Lines 1(2)RT01C-4” have 10 CFR 54.4(a)(2) intended functions to provide structural support for attached piping with 10 CFR 54.4(a)(1) intended functions and to provide a leakage boundary for nearby components that have 10 CFR 54.4(a)(1) intended functions.

Valves 1G33-F040 and 2G33-F040 are ASME Section III Class 1 safety-related valves that function as outboard primary containment boundary valves. Upstream piping 1(2)RT06B-4” (colored red) is outside of the RCPB and primary containment boundary and is ASME Section III Class 3 as shown on the drawings. Lines 1(2)RT06B-4” are therefore nonsafety-related, nonseismic, and do not have 10 CFR 54.4(a)(1) intended functions. Lines 1(2)RT06B-4” have 10 CFR 54.4(a)(2) intended functions to provide structural support for attached piping with 10 CFR 54.4(a)(1) intended functions and to provide a leakage boundary for nearby components that have 10 CFR 54.4(a)(1) intended functions.

RAI 3.2.2.2.5-1

Background:

Standard Review Plan for License Renewal (SPR-LR) Section 3.2.2.2.5 addresses loss of material due to general corrosion and fouling that leads to corrosion for steel drywell and suppression pool spray system nozzles exposed to uncontrolled indoor air. The further evaluation discussed in the section states "[t]his aging mechanism and effect [plugging of the spray nozzles and flow orifices] will apply since the spray nozzles and flow orifices are occasionally wetted, even though the majority of the time the system is on standby. The wetting and drying of these components can accelerate corrosion and fouling." The associated Table 1 item (3.2.1-6) states that a plant-specific aging management program is to be evaluated. LRA Section 3.2.2.2.5 states that this further evaluation item is not applicable because the spray system nozzles are stainless steel, not steel. The discussion for this item in the LRA does not mention that the system is occasionally wetted; however, the staff notes that technical specification surveillance requirement 3.6.2.4.2 for the residual heat removal's suppression pool spray subsystem requires periodic verification of a flow rate greater than 450 gallons per minute through the associated spray sparger (identified as "spray header" on drawing LR-LAS-M-96).

The spray nozzles in LRA Table 3.2.2-4, "Residual Heat Removal System," show the internal environment as "condensation" that will be managed for loss of material by the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program. The same table shows carbon steel piping and piping components with internal environments of both "condensation" and "treated water" that will be managed for loss of material by either the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program or the Water Chemistry with One-Time Inspection program, respectively.

Issue:

1. Plugging of the spray nozzles is not precluded simply because they are constructed of a material that is not susceptible to general corrosion in an uncontrolled indoor air environment. The wetting and drying that "can accelerate corrosion and fouling" also applies to steel piping and components upstream of the spray nozzles. It is unclear to the staff how the difference in material for the spray nozzles provides sufficient justification for non-applicability of the further evaluation item, if the steel piping and components upstream of the nozzles are occasionally wetted.
2. The internal environment for the spray header (i.e., piping upstream of the spray nozzles) is either "condensation" or "treated water," and the associated components will be considered as part of a population for either periodic inspections or one-time inspections, depending on the program. It is not clear to the staff which components are considered to have a "condensation" environment, and whether it is applicable to the spray header piping. In addition, the staff notes that a total flow of 450 gallons per minute with 18 spray nozzles on a 4-inch spray header results in flow velocities between 0.6 and 5.7 feet per second, depending on the spray header segment relative to the supply point. It is not clear whether the internal environment of the spray header will be considered unique due to the flow variations throughout the line or whether the internal environment will be grouped with other system components from a sampling perspective.

Request:

1. For the drywell and suppression pool spray system nozzles, provide the bases for not needing a plant-specific aging management program to manage plugging of spray nozzles caused by accelerated corrosion of steel components upstream of the nozzles due to occasional wetting and drying.
2. For the piping upstream of the suppression pool spray nozzles, clarify which environment (condensation or treated water) is being considered and discuss whether the suppression pool spray header will be considered as a unique environment due to the variations in flow, or provide the bases for considering it as part of a larger population from a sampling perspective.

Exelon Response:

1. For the drywell and suppression pool spray system nozzles, the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25) aging management program will be used to manage plugging of the spray nozzles caused by corrosion of steel components upstream of the nozzles due to occasional wetting and drying.

   - Drywell Spray System

     The drywell spray nozzles are stainless steel and the spray header and piping back to the normally closed spray isolation valves (valves 1E12-F017A/B as shown on LR-LAS-M-96 sheets 1 and 2 (Unit 1); valves 2E12-F017A/B as shown on LR-LAS-M-142 sheets 1 and 2 (Unit 2)) are carbon steel. The drywell spray system is not water tested and the spray nozzles, spray header, and piping are not subject to occasional wetting and drying from periodic system flow testing. Therefore, plugging of the spray nozzles due to corrosion products from periodic system flow testing cannot occur. However, the spray nozzles, spray header, and piping are susceptible to the loss of material due to their internal environment. For the purpose of aging management, the drywell spray nozzles, header, and piping were considered to be exposed to an internal environment of condensation. LRA Table 3.0-1 defines the condensation environment as an air environment containing warm or moist air where condensation may occur and periodically wet the component surface. This environment includes air with enough moisture to facilitate loss of material.

The drywell spray nozzles, header, and piping are included in LRA Table 3.2.2-4 "Residual Heat Removal System Summary of Aging Management Evaluation," as follows:

Component Type	Intended Function	Material	Environment	Aging Effect
Spray Nozzles	Spray	Stainless Steel	Condensation	Loss of Material
Piping, piping components, and piping elements	Pressure Boundary	Carbon Steel	Condensation	Loss of Material

For managing the aging effects associated with these components, NUREG-1801 line items with the internal environment of condensation were used. In accordance with NUREG-1801, and as identified in LRA Table 3.2.2-4, the aging effect of loss of material in these components is managed by the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25) aging management program.

To ensure that the stainless steel nozzles are managed for plugging due to aging effects associated with their exposure to an internal environment of condensation, LRA Table 3.2.2-4 is revised to add the aging effect of "Flow Blockage" for the stainless steel spray nozzles associated with the drywell sprays. This aging effect will be managed by the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25) aging management program. Additionally, plant specific notes 5 and 7 are added to identify that the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25) aging management program is used to manage the aging effect of "Flow Blockage" associated with the drywell spray system nozzles and will verify that plugging of the spray nozzles due to corrosion of the upstream piping does not exist.

- Suppression Pool Spray System

The suppression pool spray nozzles are stainless steel and the spray header and piping back to the normally closed spray isolation valves (valves 1E12-F027A/B as shown on LR-LAS-M-96 sheets 1 and 2 (Unit 1); valves 2E12-F027A/B as shown on LR-LAS-M-142 sheets 1 and 2 (Unit 2)) are carbon steel. The suppression pool spray system is flow tested with water quarterly. Following testing the piping and spray header remain partially water filled. Because the spray nozzle connections tap off of the spray header towards the top of the horizontal header pipe (60° from top dead center of the pipe), the majority of the header will remain water filled following flow testing and therefore not subject to occasional wetting and drying.

These water filled portions are included in LRA Table 3.2.2-4 as follows:

Component Type	Intended Function	Material	Environment	Aging Effect
Piping, piping components, and piping elements	Pressure Boundary	Carbon Steel	Treated Water	Loss of Material

In accordance with NUREG-1801, and as identified in LRA Table 3.2.2-4, the aging effect of loss of material in these components is managed by the Water Chemistry (B.2.1.2) and One-Time Inspection (B.2.1.21) aging management programs.

The non-water filled portions of the piping, header, and spray nozzles are subject to occasional wetting and drying from periodic system flow testing and are considered to have an internal environment of condensation. These items are included in LRA Table 3.2.2-4 as follows:

Component Type	Intended Function	Material	Environment	Aging Effect
Piping, piping components, and piping elements	Pressure Boundary	Carbon Steel	Condensation	Loss of Material
Spray Nozzles	Spray	Stainless Steel	Condensation	Loss of Material

In accordance with NUREG-1801, and as identified in LRA Table 3.2.2-4, the aging effect of loss of material in these components is managed by the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25) aging management program.

To ensure that the stainless steel nozzles are managed for plugging due to aging effects associated with their exposure to an internal environment of condensation; and, from occasional wetting and drying of the upstream carbon steel components from periodic system flow testing, LRA Table 3.2.2-4 is revised to add the aging effect of “Flow Blockage” for the stainless steel spray nozzles associated with the suppression pool sprays. This aging effect will be managed by the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25) aging management program. Additionally, plant specific notes 5, 6, and 7 are added to identify that the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25) aging management program is used to manage the aging effect of “Flow Blockage” associated with the suppression pool spray system nozzles and will verify that plugging of the spray nozzles due to corrosion of the upstream piping does not exist.

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25) aging management program requires that, 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 will be inspected. The stainless steel spray nozzles with an internal environment of condensation and an aging effect of flow blockage is a new population for this program. This population consists of 310 stainless steel spray nozzles (119 nozzles per unit for drywell sprays and 36 per unit for suppression pool sprays). The drywell and suppression pool spray nozzles are ramp type nozzles with an orifice size of 15/16 inches and 9/32 inches, respectively. The inspection sample will be 25 spray nozzles per unit. Inspections will be a combination of drywell spray nozzles and suppression pool spray nozzles. A review of plant operating experience did not identify drywell or suppression pool spray nozzle plugging; therefore this aging management approach is considered appropriate.

2. For the piping upstream of the suppression pool spray nozzles, both the treated water and condensation environments are addressed for aging management. The suppression pool spray system consists of piping and a single spray header with 36 nozzles. The piping upstream of the normally closed spray isolation valves (valves 1E12-F027A/B as shown on LR-LAS-M-96 sheets 1 and 2 (Unit 1); valves 2E12-F027A/B as shown on LR-LAS-M-142 sheets 1 and 2 (Unit 2)) has an internal environment of treated water and the loss of material is managed by the Water Chemistry (B.2.1.2) and One-Time Inspection (B.2.1.21) aging management programs. As discussed in the response to item 1 above, portions of the spray header and piping downstream of the normally closed spray isolation valves are water filled. Non-water filled portions of the spray header and piping, and the spray nozzles, are subject to occasional wetting and drying from periodic system flow testing and have been assigned an internal environment of condensation.

   For the purpose of aging management, the internal environment of the spray header is not considered unique due to the flow variations occurring throughout the line during infrequent (quarterly) flow testing. Instead, as discussed in the response to item 1 above, the spray header is considered exposed to an internal environment of both treated water and condensation. The periodic inspection of the spray nozzles by the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25) aging management program will verify that plugging of the spray nozzles due to corrosion of upstream piping does not exist; and, will verify that a unique environment causing accelerated corrosion does not exist. Although variations in flow are not considered a unique environment for the purpose of establishing a sample population, the aging effect of flow blockage represents a unique population for the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25) aging management program in which the spray nozzles are the only components.

LRA Table 3.2.2-4, Section 3.2.2.1.4, Appendix A, Section A.2.1.25, and Appendix B, Section B.2.1.25 are revised as shown in Enclosure B.

RAI 3.5.1.78-1

Background:

LRA Table 3.5.1, item 3.5.1-78 addresses the spent fuel pool (SFP) liner exposed to treated water, which will be managed for loss of material and cracking due to stress corrosion cracking (SCC). LRA Table 3.5.2-9, associated with item 3.5.1-78, cites generic note B, indicating that the items are consistent with the GALL Report item; however, these items also include components such as stainless steel gates, liner anchors, and integral attachments, in addition to the SFP liner. Based on the discussion in LRA Table 3.5.1, item 3.5.1-78, these additional items will only be managed for loss of material through the Water Chemistry program and by monitoring the SFP water level and leakage from the leak chase channels. LRA Table 3.5.2-9 does not address cracking for stainless steel components exposed to treated water that are associated with item 3.5.1-78.

The staff recognizes that treated water less than 140 °F is not an environment that is conducive to SCC. However, although the Updated Final Safety Analysis Report (UFSAR) Section 9.1.3.2.1.1 states that normal operating and refueling temperatures in the SFP are at or below 140 °F, during an "emergency" full core offload, the UFSAR states that the SFP water temperature remains below boiling (212 °F). In addition, SFP temperature limits maintain bulk water temperature less than 140 °F, but there may be local temperatures that exceed the limitation for susceptibility to cracking.

Issue:

It is unclear to the staff if appropriate activities to verify the effectiveness of the Water Chemistry program are being performed for the components in LRA Table 3.5.2-9 that reference item 3.5.1-78. It is also unclear to the staff why the aging effect of cracking is not being managed for stainless steel components that reference item 3.5.1-78, since temperatures in the SFP can exceed 140 °F.

Request:

1. State the basis for why the aging effect of cracking is not being managed for items that reference 3.5.1-78. Include information regarding past SFP temperatures and discuss whether local temperatures need to be considered.

2. If it is determined that cracking needs to be managed, provide clarification on how the effectiveness of the Water Chemistry program is being verified for items that reference item 3.5.1-78 in LRA Table 3.5.2-9 since some of the components cannot be monitored via leakage through the leak chase channels.

Exelon Response:

1. The environment for items that reference 3.5.1-78 is treated water less than 140 °F, which is not an environment that is conducive to SCC. Therefore, the aging effect of cracking is not being managed for items that reference 3.5.1-78.

   The RAI “Background” discussion above states “that treated water less than 140 °F is not an environment that is conducive to SCC”. The GALL Report Chapter IX Section D pages IX-14 and IX-21 indicate the temperature threshold for requiring consideration of stainless steel SCC is greater than 140 °F. The UFSAR Section 9.1.3.2.1.1 referenced above states that spent fuel pool temperatures are maintained less than or equal to 140 °F during normal operations and normal refueling. Spent fuel pool water temperatures recorded since January 2000 were reviewed and confirm that spent fuel pool water temperatures have been maintained well below 140 °F. The highest recorded temperature was less than 114 °F degrees and the average normal spent fuel pool water temperature was approximately 95 °F. 100 °F is an existing operational limit for spent fuel pool water temperature in LSCS Operating Procedures which was established in order to reduce refuel floor airborne reactivity levels.

   NUREG 1800 Appendix A, Section A1, A.1.2.1.7 states that aging effects from abnormal events need not be postulated for license renewal, unless such an event has occurred at the particular plant. No record of an emergency full core offload event defined in UFSAR Section 9.1.3.2.3.2 has been identified for LSCS. The “emergency” full core offload, mentioned in the UFSAR was a potential worst case design condition that was evaluated when high density racks were installed in the spent fuel pools. An emergency full core offload has not occurred and is not expected to occur.

   As stated in LRA B.2.1.2, the Water Chemistry program includes periodic monitoring of the treated water and control of known detrimental contaminants such as chlorides, dissolved oxygen, and sulfate concentrations below the levels known to result in loss of material or cracking. Spent Fuel Pool water chemistry parameters are also provided in UFSAR Section 9.1.3.1.1.2. The spent fuel pool for each unit by design has its own cooling and cleanup system and each of the two spent fuel pool pumps per unit have a flowrate of 3000 gallons per minute which provides effective cooling and cleanup of spent fuel pool treated water. Water is withdrawn from the top of the spent fuel pool for cooling and cleaning and returned towards the bottom of the pool. As a result of this LSCS design and the previously described operating experience where treated water temperatures in the spent fuel pool have been maintained well below the greater than 140 °F threshold, local temperatures need not be considered during normal operations because the mixing from the spent fuel pool cooling system flow rate would also maintain local temperatures at or below the 140 °F threshold requiring consideration of stainless steel SCC.

   Therefore, stress corrosion cracking is not considered to be an applicable aging mechanism for the stainless steel water retaining components exposed to the treated water in the spent fuel pool.

2. As described above, stress corrosion cracking is not considered to be an applicable aging mechanism for the stainless steel water retaining components exposed to treated water in the spent fuel pool. Therefore, there is no need to manage stress corrosion cracking of the stainless steel components exposed to treated water in the spent fuel pool.

   The spent fuel pool stainless steel water retaining components are the liner, integral attachments, and gates. These stainless steel water retaining components are directly monitored for leakage, and in addition, they are also managed by the Water Chemistry (B.2.1.2) aging management program and spent fuel pool water level monitoring. As shown in LRA Table 3.5.2-9, the carbon steel provided for structural support, which are the liner anchors, are exposed only to the concrete environment on the concrete side of the liner. These carbon steel liner anchors are also not addressed by LRA Table 1 (Table 3.5.1) item 3.5.1-78 but instead are addressed by item 3.5.1-41. The LSCS spent fuel pools are each provided with two separate gates for redundancy which are installed between the spent fuel pool and the reactor cavity during operations. A flow switch monitors the drain line from the fuel pool gates to detect leakage from a spent fuel pool gate or gate seal. An annunciator alarms when excessive flow is detected. Drain lines which collect possible leakage from behind the spent fuel pool liners are also provided with flow switches that trigger annunciators that alarm on excessive flow. The Unit 1 and Unit 2 spent fuel pools are connected by a normally filled transfer canal and cask well, which are also provided with leak detection. UFSAR Section 9.1.2.1.3.3 also states: “a means of leak detection is available at all times during normal and abnormal operations and is used to monitor any flow paths which could reduce the level of the pool from its normal level.”

   In conclusion, the Water Chemistry program with the water level and leakage monitoring provided is adequate and consistent with the GALL Report to manage the loss of material aging effect for the subject stainless steel, spent fuel pool, water retaining components, and is also consistent with the GALL Report recommendation for managing the cracking due to stress corrosion cracking aging effect. As discussed previously in response to request 1 above, stress corrosion cracking is not considered to be an applicable aging mechanism for the stainless steel water retaining components exposed to the treated water in the spent fuel pool.

RAI 3.5.2.2.1.5-1

Background:

Section 54.21(a)(3) of 10 CFR requires applicants to demonstrate that the effects of aging will be adequately managed so that intended functions will be maintained consistent with the current licensing basis for each structure and component subject to aging management review. Section 54.2(c)(1) of 10 CFR requires the evaluation of time-limited aging analyses (TLAA) to demonstrate that: (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 will be adequately managed for the period of extended operation.

LRA Section 3.5.2.2.1.5 "Cumulative Fatigue Damage," corresponding to LRA Table 3.5.1, item 3.5.1-9, states in part that the evaluation of fatigue as a TLAA for the LaSalle County Station, Units 1 and 2, primary containment liner is addressed in LRA Section 4.6. In LRA Section 4.6.1, the applicant provides the TLAA evaluation of the primary containment liner plate and penetrations. The applicant dispositioned this TLAA in accordance with 10 CFR 54.21(c)(1)(iii), and stated that the effects of aging on the intended functions of components analyzed in accordance with ASME Section III, Class 1 requirements will be managed by the Fatigue Monitoring program through the period of extended operation.

LRA Table 3.5.2-7 AMR results line items corresponding to LRA Table 3.5.1, item 3.5.1-9, address electrical penetration assemblies, mechanical penetration assemblies, penetration sleeves, refueling bellows assembly, and downcomers components which will be managed for the cumulative fatigue damage aging effect by the TLAA program. However, LRA Table 3.5.2-7 does not identify any AMR results line items corresponding to Table 3.5.1, item 3.5.1-9, for the primary containment liner.

Issue:

The staff is not clear why the LRA Table 3.5.2-7 AMR results line items corresponding to Table 1, item 3.5.1-9, do not address the containment liner component(s), and how this component is being adequately managed for the cumulative fatigue damage aging effect for the period of extended operation.

Request:

Describe how the primary containment liner is adequately managed for the cumulative fatigue damage aging effect through the period of extended operation, and provide the technical basis for not addressing the containment liner component(s) in LRA Table 3.5.2-7 AMR results line items corresponding to Table 1, item 3.5.1-9.

Exelon Response:

The primary containment components will be adequately managed for the cumulative fatigue damage aging effect through the period of extended operation by the Fatigue Monitoring (B.3.1.1) aging management program. Although no line item was included in LRA Table 3.5.2-7 for the cumulative fatigue damage aging effect for the primary containment liner or associated components, the cumulative fatigue damage aging effect has been addressed as stated in LRA Sections 3.5.2.2.1.5, 4.6, and in Table 3.5.1 by Item 3.5.1-9 for these components with appropriate aging management activities.

Based on the above, LRA Table 3.5.2-7 is revised for consistency with LRA Section 4.6 by adding the cumulative fatigue damage aging effect line items for the primary containment liner and associated components.

As a result of this RAI, LRA Table 2.4-7, Table 3.5.2-7, and Section 3.5.2.2.1.5 are revised as shown in Enclosure B.

RAI 4.2.5-1

Background:

LRA Section 4.2.5 addresses a TLAA on reactor vessel axial weld failure probability assessment. Specifically, LRA Tables 4.2.5-1 and 4.2.5-2 provide comparison of axial weld failure probability parameters between the applicant's assessment and the staff's safety assessment in the safety evaluation (July 28, 1998) of the BWRVIP-05 report. The comparison of parameters includes initial RT_{NDT} (nil-ductility transition reference temperature) and mean RT_{NDT}. The LRA indicates that Unit 1 has a Combustion Engineering (CE) reactor vessel and Unit 2 has a Chicago Bridge and Iron (CB&I) reactor vessel. The applicant dispositioned this TLAA in accordance with 10 CFR 54.21(c)(1)(ii) by projecting the axial weld failure probability assessments through the period of extended operation.

The NRC staff's safety evaluation for the methodology in BWRVIP-05 was supplemented in a safety evaluation to the BWRVIP dated March 7, 2000. In the supplemental evaluation, the staff updated the reactor vessel failure probability and mean RT_{NDT} analyses for axial welds in reactor vessels designed by CE and by CB&I.

Issue:

The staff noted that LRA Section 4.2.5 does not provide a comparison of axial weld failure probability parameters between the applicant's assessment and the staff's March 7, 2000, supplemental safety evaluation regarding the BWRVIP-05 Report. Specifically, Table 3 of the supplemental safety evaluation addresses the axial weld failure frequencies determined in the staff's assessment. The LRA does not discuss these failure frequencies in comparison with the applicant's assessment results.

In addition, the applicant did not provide a probabilistic analysis establishing that the reactor vessel failure frequency for axial welds is less than 5×10^{-6} per reactor year in accordance with the staff's March 7, 2000, supplemental safety evaluation.

Request:

1. Justify why LRA Section 4.2.5 does not need to be revised to provide comparison of axial weld failure probability parameters between the applicant's assessment and the staff's supplemental safety evaluation dated March 7, 2000.
2. If the applicant's failure probability comparison does not indicate that the reactor vessel failure frequency for axial welds is less than 5×10^{-6} per reactor year, provide the following information:
    a. a plant-specific probabilistic analysis for axial weld failure, consistent with SRP-LR Section 4.2.3.1.5, or
    b. information demonstrating that loss of fracture toughness due to neutron irradiation embrittlement of axial welds does not affect the structural integrity of the reactor vessel for the period of extended operation.

Exelon Response:

Subsequent to the original SER for BWRVIP-05 (July 28, 1998), additional failure probability assessments were prepared by the BWRVIP to more accurately determine the conditional failure probability of BWR axial welds. The staff reviewed these BWRVIP-05 assessments and provided separate conditional failure probability assessments in the Supplement to the Final Safety Evaluation for the BWRVIP-05 Report dated March 7, 2000. The staff assessments resulted in higher failure probabilities than the BWRVIP assessments and are therefore limiting. The staff determined that the RPV failure frequency due to failure of the axial welds in the BWR fleet is no greater than 5.02E-06 per reactor year. Since these assessments are applicable to LSCS Units 1 and 2, they are identified as TLAAs requiring evaluation through the period of extended operation.

LRA Section 4.2.5 is revised as shown in Enclosure B to describe these TLAAs and to provide their evaluations. LRA Tables 4.2.5-1 and 4.2.5-2 are revised to show the comparisons between the mean RT_{NDT} values in the staff axial weld assessments (March 7, 2000) and the limiting LSCS axial weld mean RT_{NDT} values for each unit at 54 EFPY. For clarity, the responses to requests 1 and 2 are provided separately below for each unit.

Unit 1

1. LRA Table 4.2.5-1 is revised to provide a comparison of the limiting Unit 1 axial weld mean RT_{NDT} value at 54 EFPY of 139.9ºF (without margin) to the 114ºF mean RT_{NDT} value (without margin) determined for the limiting Combustion Engineering (CE) reactor, Mod 2 Variant, in the Supplement to the Final Safety Evaluation of the BWRVIP-05 Report dated March 7, 2000, as shown in Enclosure B. Since the 114ºF mean RT_{NDT} value from the staff assessment does not bound the limiting Unit 1 mean RT_{NDT} value of 139.9ºF, the staff failure probability of 5.02E-06 does not bound the Unit 1 axial welds at 54 EFPY. The limiting Unit 1 axial welds are projected to reach the 114ºF mean RT_{NDT} value (without margin) at approximately 39.15 EFPY. For reference, the Unit 1 cumulative neutron exposure through July 2015 is equivalent to 24.05 EFPY.

2. Since the staff failure probability assessment for Unit 1 is not projected to remain valid through the period of extended operation, the axial weld failure probability assessment TLAA for Unit 1 is dispositioned in accordance with 10 CFR 54.21(c)(1)(iii): The effects of aging on the Unit 1 reactor pressure vessel axial welds will be managed by:

    (1) The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.1.1) aging management program, which requires periodic examination of the axial welds in accordance with the requirements of ASME Section XI, Table IWB-2500-1, Examination Category B-A, Pressure Retaining Welds in Reactor Vessel, and

    (2) The Reactor Vessel Surveillance (B.2.1.20) aging management program, which manages neutron embrittlement by monitoring neutron fluence and ensuring that neutron embrittlement analyses are updated as necessary to evaluate bounding neutron fluence values. The Reactor Vessel Surveillance program will be enhanced as follows: 1) prior to the period of extended operation, establish a monitoring limit for neutron fluence at the limiting Unit 1 axial weld (currently 39.15 EFPY) that corresponds to the axial weld failure probability of 5.02E-06 per reactor year specified in the Supplement to the Final Safety Evaluation of the BWRVIP-05

Report dated March 7, 2000; and 2) prior to 39.15 EFPY, complete a probabilistic axial weld failure analysis for Unit 1 that demonstrates the 60-year axial weld failure probability is no greater than 5.02E-06 per reactor year.

The combination of periodic axial weld examinations by the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program and management of neutron embrittlement by the enhanced Reactor Vessel Surveillance program ensures that the loss of fracture toughness due to neutron irradiation embrittlement of axial welds will not affect the structural integrity of the Unit 1 reactor vessel through the period of extended operation.

The LRA Table of Contents is revised as shown in Enclosure B to provide an updated title for LRA Table 4.2.5-1. LRA Table 4.1-2, LRA Section 4.2.5, including LRA Table 4.2.5-1, LRA Section 4.8, and LRA Appendix A, Section A.4.2.5 are revised as shown in Enclosure B to evaluate the Unit 1 axial weld failure probability TLAA from the Supplement to the Final Safety Evaluation of the BWRVIP-05 Report dated March 7, 2000. LRA Sections A.1.1, A.2.1.20, B.1.5, and B.2.1.20 are revised as shown in Enclosure B to incorporate the enhancement to the Reactor Vessel Surveillance program. LRA Appendix C, Applicant Action Item 12 for BWRVIP-74-A, LaSalle Response, is revised as shown in Enclosure B.

LRA Appendix A, Section A.5, Commitment 20, is revised to incorporate the enhancement to the Reactor Vessel Surveillance program, as shown in Enclosure C.

Unit 2

1. LRA Table 4.2.5-2 is revised to provide a comparison of the limiting Unit 2 axial weld mean RT_{NDT} value at 54 EFPY of 12.2ºF (without margin) to the 91ºF mean RT_{NDT} value (without margin) determined for the limiting Chicago Bridge and Iron (CB&I) reactor in the Supplement to the Final Safety Evaluation of the BWRVIP-05 Report dated March 7, 2000, as shown in Enclosure B. Since the 91ºF mean RT_{NDT} value from the staff assessment bounds the limiting axial weld mean RT_{NDT} value of 12.2ºF for Unit 2, the staff failure probability of 2.73E-06 is also bounding for Unit 2 for 54 EFPY.

   The Unit 2 TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(ii): The Unit 2 axial weld failure probability TLAA has been satisfactorily projected through the period of extended operation, consistent with SRP-LR 4.2.3.1.5. The Unit 2 axial welds will also be periodically examined in accordance with the requirements of ASME Section XI, Table IWB-2500-1, Examination Category B-A, Pressure Retaining Welds in Reactor Vessel.

2. Request 2 is not applicable to Unit 2.

   The LRA Table of Contents is revised as shown in Enclosure B to provide an updated title for LRA Table 4.2.5-2. LRA Section 4.2.5, including LRA Table 4.2.5-2, is revised as shown in Enclosure B to identify and evaluate the Unit 2 axial weld failure probability assessment TLAA from the Supplement to the Final Safety Evaluation of the BWRVIP-05 Report dated March 7, 2000.

Enclosure B

**LSCS License Renewal Application Updates
Resulting from the Response to the following RAIs:**

RAI 3.2.2.2.5-1
RAI 3.5.2.2.1.5-1
RAI 4.2.5-1

Notes:

- Updated LRA Sections and Tables are provided in the same order as the RAI responses contained in Enclosure A.
- To facilitate understanding, portions of the LRA have been repeated in this Enclosure, with revisions indicated. Previously submitted information is shown in normal font. Changes are highlighted with ***bolded italics*** for inserted text and ~~strikethroughs~~ for deleted text.

As a result of the response to RAI 3.2.2.2.5-1 provided in Enclosure A of this letter, the affected portions of LRA Table 3.2.2-4, beginning on page 3.2-60 of the LRA, are revised as shown below:

Table 3.2.2-4 Residual Heat Removal System (Continued)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Item	Table 1 Item	Notes
Piping, piping components, and piping elements	Pressure Boundary	Carbon Steel	Air - Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B.2.1.24)	V.D2.E-26	3.2.1-40	A
			Condensation (Internal)	Loss of Material	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25)	V.D2.E-27	3.2.1-46	A***, 5***
			Lubricating Oil (Internal)	Loss of Material	Lubricating Oil Analysis (B.2.1.26)	V.D2.EP-77	3.2.1-49	A
					One-Time Inspection (B.2.1.21)	V.D2.EP-77	3.2.1-49	A
			Treated Water (External)	Loss of Material	One-Time Inspection (B.2.1.21)	V.D2.EP-60	3.2.1-16	A, 1
					Water Chemistry (B.2.1.2)	V.D2.EP-60	3.2.1-16	B, 1
			Treated Water (Internal)	Cumulative Fatigue Damage	TLAA	V.D2.E-10	3.2.1-1	A, 4
				Loss of Material	One-Time Inspection (B.2.1.21)	V.D2.EP-60	3.2.1-16	A***, 6***
					Water Chemistry (B.2.1.2)	V.D2.EP-60	3.2.1-16	B***, 6***
				Wall Thinning	Flow-Accelerated Corrosion (B.2.1.10)	V.D2.E-09	3.2.1-11	B

Table 3.2.2-4 Residual Heat Removal System (Continued)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Item	Table 1 Item	Notes
Spray Nozzles	Spray	Stainless Steel	Air - Indoor Uncontrolled (External)	None	None	V.F.EP-18	3.2.1-63	A
			Condensation (Internal)	***Flow Blockage***	***Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25)***			***H, 7***
				Loss of Material	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25)	V.D2.EP-61	3.2.1-48	A

Plant Specific Notes:

5. The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25) aging management program manages the loss of material for this component type, material, and environment combination. In addition, the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25) aging management program is used to manage the aging effect of "Flow Blockage" associated with the drywell and the suppression pool spray system spray nozzles and will verify that plugging of the spray nozzles due to corrosion of the upstream piping does not exist.

6. The Water Chemistry (B.2.1.2) and One-Time Inspection (B.2.1.21) aging management programs manage the loss of material for this component type, material, and environment combination. In addition, the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25) aging management program is used to manage the aging effect of "Flow Blockage" associated with the suppression pool spray system spray nozzles and will verify that plugging of the spray nozzles due to corrosion of the upstream piping does not exist.

7. The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25) program is used to manage the aging effect of "Flow Blockage" associated with this component type, material, and environment combination. The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25) aging management program requires that, 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 will be inspected.

As a result of the response to RAI 3.2.2.2.5-1 provided in Enclosure A of this letter, LRA Section 3.2.2.1.4, on page 3.2-6 of the LRA, is revised as shown below:

Aging Effects Requiring Management

The following aging effects associated with the Residual Heat Removal System components require management:

- Cumulative Fatigue Damage
- ***Flow Blockage***
- Loss of Material
- Loss of Preload
- Reduction of Heat Transfer
- Wall Thinning

As a result of the response to RAI 3.2.2.2.5-1 provided in Enclosure A of this letter, LRA Appendix A, Section A.2.1.25, beginning on page A-28 of the LRA, is revised as shown below:

A.2.1.25 Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components aging management program is a new condition monitoring program that will consist of inspections of the internal surfaces of metallic and elastomeric components such as piping, piping components and piping elements, ducting components, tanks, heat exchanger components, elastomers, and other components that are exposed to environments of condensation, diesel exhaust, and waste water. These internal inspections will be performed during the periodic system and component surveillances or during the performance of maintenance activities when the surfaces are made accessible for visual inspection. 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 will be inspected. Where practical, the inspections will focus on the bounding or lead components most susceptible to aging because of time in service and severity of operating conditions. Opportunistic inspections will continue in each period even after meeting the sampling limit.

The program will manage the aging effects of loss of material, reduction of heat transfer, ***flow blockage,*** and cracking for metallic components. The program will also manage the aging effects of loss of material, hardening and loss of strength, and change in material properties for elastomeric components. The program will include visual inspections to ensure that existing environmental conditions are not causing material degradation ***or flow blockage*** that could result in a loss of the component's intended function. ***For example, the program verifies that plugging of the suppression pool or drywell spray nozzles due to corrosion of upstream piping does not occur.*** For certain materials, such as elastomers, physical manipulation to detect hardening or loss of strength will be used to supplement the visual examinations conducted under this program.

In addition, a review of LSCS operating experience has revealed instances of recurring internal corrosion in plant floor drain piping that is within the scope of the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program. This program will include periodic inspections on this population of carbon steel piping in the floor drain systems to ensure that recurring aging effects are adequately managed.

This new aging management program will be implemented prior to the period of extended operation.

As a result of the response to RAI 3.2.2.2.5-1 provided in Enclosure A of this letter, LRA Appendix B, Section B.2.1.25, “Program Description” paragraph beginning on page B-110 of the LRA is revised as shown below:

B.2.1.25 Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components

Program Description

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components aging management program is a new condition monitoring program that manages the applicable aging effects by directing visual inspections of internal surfaces of components within the scope of license renewal be performed when they are made accessible during periodic system and component surveillances or during the performance of maintenance activities. The program provides assurance that condensation, diesel exhaust, and waste water environments are not causing material degradation that could result in loss of intended function.

The program consists of visual inspections of the internal surfaces of metallic components such as piping, piping components and piping elements, ducting components, tanks, heat exchangers components, and other components that are exposed to condensation, diesel exhaust, and waste water. The program also consists of visual inspections of the internal surfaces of elastomeric components that are exposed to condensation, supplemented by physical manipulation to detect hardening or loss of strength where appropriate. The program will manage the aging effects of loss of material, reduction of heat transfer, ***flow blockage,*** and cracking for metallic components. The program will also manage the aging effects of loss of material, hardening and loss of strength, and change in material properties for elastomeric components. The program includes provisions for visual inspections of the internal surfaces of components not managed under other aging management programs.

Acceptance criteria for each component and aging effect combination are defined to ensure that the need for corrective actions is identified before the loss of intended functions. For metallic surfaces, any abnormal surface condition is evaluated by engineering. ***Flow blockage of the suppression pool or drywell spray nozzles and verification that plugging of the spray nozzles due to corrosion of upstream piping does not occur is evaluated by engineering.*** For stainless steel surfaces, a clean, shiny surface is expected. For flexible polymers, changes in material properties (e.g., hardness, flexibility, physical dimensions, and color) and cracks are evaluated. For rigid polymers, surface changes affecting performance such as erosion and cracking are evaluated.

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 will be inspected. Where practical, the inspections will focus on the bounding or lead components most susceptible to aging because of time in service and severity of operating conditions. Opportunistic inspections will continue in each period even after meeting the sampling limit.

A review of LSCS operating experience has revealed instances of recurring internal corrosion in plant floor drain piping that is within the scope of the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program. This program will include periodic inspections on this population of carbon steel piping in the floor drain systems to ensure that recurring aging effects are adequately managed.

The new Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program will be implemented prior to the period of extended operation.

As a result of the response to RAI 3.5.2.2.1.5-1 provided in Enclosure A of this letter, Table 2.4-7 beginning on on page 2.4-33 of the LRA is revised as shown below:

Table 2.4-7 **Primary Containment**
Components Subject to Aging Management Review

Component Type	Intended Function
Bolting (Containment Closure)	Structural Pressure Barrier
	Structural Support
Bolting (Structural)	Structural Support
Bolting (Vacuum Relief Line Pipe Flanges)	Structural Pressure Barrier
	Structural Support
Concrete Anchors	Structural Support
Concrete Embedments	Structural Pressure Barrier
	Structural Support
	Water retaining boundary
Concrete: Containment Wall (accessible areas - includes Buttresses)	Missile Barrier
	Shelter, Protection
	Shielding
	Structural Pressure Barrier
	Structural Support
Concrete: Containment Wall (inaccessible areas - includes Buttresses)	Missile Barrier
	Shelter, Protection
	Shielding
	Structural Pressure Barrier
	Structural Support
Concrete: Foundation, Subfoundation, Basemat (accessible areas - Tendon Access Tunnel Ceiling)	Shelter, Protection
	Structural Pressure Barrier
	Structural Support
Concrete: Foundation, Subfoundation, Basemat (inaccessible areas)	Flood Barrier
	Shelter, Protection
	Structural Pressure Barrier
	Structural Support
Concrete: Interior (Drywell Floor and Cavity Floor)	Direct Flow
	Structural Pressure Barrier
	Structural Support
Concrete: Interior (Pedestal)	Direct Flow
	Structural Pressure Barrier
	Structural Support
Concrete: Interior (Suppression Pool Columns)	Structural Support
Concrete: Reactor Cavity Contiguous Fuel Pool Walls with Tendons	Shielding

Table 2.4-7 **<u>Primary Containment</u>**
Components Subject to Aging Management Review (Continued)

Component Type	Intended Function
Concrete: Reactor Cavity Contiguous Fuel Pool Walls with Tendons	Structural Support
Doors (Reactor Shield Wall Doors)	Shielding
	Structural Support
Downcomer Jet Deflectors	Direct Flow
	Shelter, Protection
Electrical Penetration Assemblies (includes Penetration Sleeves and Closure Plates)	Shelter, Protection
	Structural Pressure Barrier
Hatches/Plugs ***(Suppression Chamber Access Hatches)***	Missile Barrier
	Shelter, Protection
	Structural Pressure Barrier
Mechanical Penetrations (includes Penetration Sleeves, Flued Heads, and Closure Plates for Pipe and Instrument Penetrations)	Shelter, Protection
	Structural Pressure Barrier
	Structural Support
Metal Components (Permanent Drywell Shielding)	Shielding
	Structural Support
Penetration Sleeves: Drywell Floor (including Closure Rings, Plates, and Caps)	Structural Pressure Barrier
	Structural Support
Personnel Airlock, Equipment Hatch: CRD Hatch	Missile Barrier
	Shelter, Protection
	Structural Pressure Barrier
Personnel Airlock, Equipment Hatch: Locks, Hinges, and Closure Mechanisms	Shelter, Protection
	Structural Pressure Barrier
Pipe Whip Restraints and Jet Impingement Shields	Pipe Whip Restraint
Prestressing System: Anchorage Components	Structural Support
Prestressing System: Grease Cap at Tendon Anchorage	Shelter, Protection
Prestressing System: Tendons	Structural Support
Seals and Gaskets	Structural Pressure Barrier
Service Level I Coatings (Containment Boundary)	Maintain Adhesion
Service Level I Coatings (Internal Structures)	Maintain Adhesion
Sliding Surfaces (Support)	Structural Support

As a result of the response to RAI 3.5.2.2.1.5-1 provided in Enclosure A of this letter, the affected portions of LRA Table 3.5.2-7 beginning on page 3.5-147 of the LRA are revised as shown below:

Table 3.5.2-7 Primary Containment

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Item	Table 1 Item	Notes
Concrete Embedments	Structural Pressure Barrier	Carbon Steel	Air – Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-46	3.5.1-35	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-46	3.5.1-35	A
			Concrete	None	None	II.B2.2.CP-114	3.5.1-41	C
			Treated Water	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-46	3.5.1-35	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-46	3.5.1-35	A
	Structural Support	Carbon Steel	Air – Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-46	3.5.1-35	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-46	3.5.1-35	A
			Concrete	None	None	II.B2.2.CP-114	3.5.1-41	C
			Treated Water	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-46	3.5.1-35	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-46	3.5.1-35	A
	Water retaining boundary	Carbon Steel	Air – Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-46	3.5.1-35	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-46	3.5.1-35	A
			Concrete	None	None	II.B2.2.CP-114	3.5.1-41	C
			Treated Water	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***

Table 3.5.2-7 Primary Containment (Continued)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Item	Table 1 Item	Notes
Hatches/Plugs ***(Suppression Chamber Access Hatches)***	Missile Barrier	Carbon Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B4.C-13***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B4.C-16	3.5.1-28	C
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B4.C-16	3.5.1-28	C
		Stainless Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B4.C-13***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.C-49	3.5.1-37	C
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.C-49	3.5.1-37	C
	Shelter, Protection	Carbon Steel	Air – Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B4.C-13***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B4.C-16	3.5.1-28	C
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B4.C-16	3.5.1-28	C
		Stainless Steel	Air – Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B4.C-13***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.C-49	3.5.1-37	C
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.C-49	3.5.1-37	C
	Structural Pressure Barrier	Carbon Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B4.C-13***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B4.C-16	3.5.1-28	C
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B4.C-16	3.5.1-28	C
		Stainless Steel	Air – Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B4.C-13***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.C-49	3.5.1-37	C
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.C-49	3.5.1-37	C

Table 3.5.2-7 Primary Containment (Continued)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Item	Table 1 Item	Notes
Personnel Airlock, Equipment Hatch: CRD Hatch	Missile Barrier	Carbon Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B4.C-13***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B4.C-16	3.5.1-28	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B4.C-16	3.5.1-28	A
	Shelter, Protection	Carbon Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B4.C-13***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B4.C-16	3.5.1-28	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B4.C-16	3.5.1-28	A
		Glass	Air - Indoor Uncontrolled	None	None	VIII.I.SP-9	3.4.1-55	A
	Structural Pressure Barrier	Carbon Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B4.C-13***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B4.C-16	3.5.1-28	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B4.C-16	3.5.1-28	A
		Glass	Air - Indoor Uncontrolled	None	None	VIII.I.SP-9	3.4.1-55	A

Table 3.5.2-7 Primary Containment (Continued)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Item	Table 1 Item	Notes
Steel Elements: Drywell Head	Missile Barrier	Carbon Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B4.C-13***	***3.5.1-9***	***C, 2***
				Fretting or Lockup	ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.C-23	3.5.1-36	A
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-46	3.5.1-35	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-46	3.5.1-35	A
	Shielding	Carbon Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B4.C-13***	***3.5.1-9***	***C, 2***
				Fretting or Lockup	ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.C-23	3.5.1-36	A
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-46	3.5.1-35	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-46	3.5.1-35	A
	Structural Pressure Barrier	Carbon Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B4.C-13***	***3.5.1-9***	***C, 2***
				Fretting or Lockup	ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.C-23	3.5.1-36	A
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-46	3.5.1-35	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-46	3.5.1-35	A
Steel Elements: Drywell Liner, Liner Anchors, Integral Attachments (accessible areas)	Structural Pressure Barrier	Carbon Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-46	3.5.1-35	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-46	3.5.1-35	A
			Concrete	None	None	II.B2.2.CP-114	3.5.1-41	C

Table 3.5.2-7 Primary Containment (Continued)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Item	Table 1 Item	Notes
Steel Elements: Drywell Liner, Liner Anchors, Integral Attachments (accessible areas)	Structural Support	Carbon Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-46	3.5.1-35	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-46	3.5.1-35	A
			Concrete	None	None	II.B2.2.CP-114	3.5.1-41	C
Steel Elements: Drywell Liner, Liner Anchors, Integral Attachments (inaccessible areas)	Structural Pressure Barrier	Carbon Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-63	3.5.1-5	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-63	3.5.1-5	A
			Concrete	None	None	II.B2.2.CP-114	3.5.1-41	C
	Structural Support	Carbon Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-63	3.5.1-5	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-63	3.5.1-5	A
			Concrete	None	None	II.B2.2.CP-114	3.5.1-41	C

Table 3.5.2-7 Primary Containment (Continued)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Item	Table 1 Item	Notes
Steel Elements: Ring Girder Assembly (includes Cone Skirt)	Flood Barrier	Carbon Steel	Air – Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-46	3.5.1-35	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-46	3.5.1-35	A
			Concrete	None	None	II.B2.2.CP-114	3.5.1-41	C
			Treated Water	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-46	3.5.1-35	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-46	3.5.1-35	A
	Structural Pressure Barrier	Carbon Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-46	3.5.1-35	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-46	3.5.1-35	A
			Concrete	None	None	II.B2.2.CP-114	3.5.1-41	C
			Treated Water	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-46	3.5.1-35	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-46	3.5.1-35	A
	Structural Support	Carbon Steel	Air – Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-46	3.5.1-35	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-46	3.5.1-35	A
			Concrete	None	None	II.B2.2.CP-114	3.5.1-41	C
			Treated Water	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-46	3.5.1-35	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-46	3.5.1-35	A

Table 3.5.2-7 Primary Containment (Continued)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Item	Table 1 Item	Notes
Steel Elements: Ring Girder Assembly (includes Cone Skirt)	Water retaining boundary	Carbon Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-46	3.5.1-35	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-46	3.5.1-35	A
			Concrete	None	None	II.B2.2.CP-114	3.5.1-41	C
			Treated Water	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-46	3.5.1-35	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-46	3.5.1-35	A
Steel Elements: Suppression Chamber Liner, Liner Anchors, Integral Attachments (accessible areas)	Structural Pressure Barrier	Stainless Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.C-49	3.5.1-37	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.C-49	3.5.1-37	A
			Concrete	None	None	VII.J.AP-19	3.3.1-120	C
			Treated Water	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.C-49	3.5.1-37	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.C-49	3.5.1-37	A
	Structural Support	Carbon Steel	Concrete	None	None	II.B2.2.CP-114	3.5.1-41	C

Table 3.5.2-7 Primary Containment (Continued)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Item	Table 1 Item	Notes
Steel Elements: Suppression Chamber Liner, Liner Anchors, Integral Attachments (accessible areas)	Structural Support	Stainless Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.C-49	3.5.1-37	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.C-49	3.5.1-37	A
			Concrete	None	None	VII.J.AP-19	3.3.1-120	C
			Treated Water	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.C-49	3.5.1-37	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.C-49	3.5.1-37	A
	Water retaining boundary	Stainless Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.C-49	3.5.1-37	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.C-49	3.5.1-37	A
			Concrete	None	None	VII.J.AP-19	3.3.1-120	C
			Treated Water	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.C-49	3.5.1-37	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.C-49	3.5.1-37	A
Steel Elements: Suppression Chamber Liner, Liner Anchors, Integral Attachments (inaccessible areas)	Structural Pressure Barrier	Stainless Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.C-49	3.5.1-37	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.C-49	3.5.1-37	A
			Concrete	None	None	VII.J.AP-19	3.3.1-120	C

Table 3.5.2-7 Primary Containment (Continued)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Item	Table 1 Item	Notes
Steel Elements: Suppression Chamber Liner, Liner Anchors, Integral Attachments (inaccessible areas)	Structural Pressure Barrier	Stainless Steel	Treated Water	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.C-49	3.5.1-37	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.C-49	3.5.1-37	A
	Structural Support	Carbon Steel	Concrete	None	None	II.B2.2.CP-114	3.5.1-41	C
		Stainless Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.C-49	3.5.1-37	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.C-49	3.5.1-37	A
			Concrete	None	None	VII.J.AP-19	3.3.1-120	C
			Treated Water	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.C-49	3.5.1-37	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.C-49	3.5.1-37	A
	Water retaining boundary	Stainless Steel	Air – Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.C-49	3.5.1-37	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.C-49	3.5.1-37	A
			Concrete	None	None	VII.J.AP-19	3.3.1-120	C
			Treated Water	***Cumulative Fatigue Damage***	***TLAA***	***II.B2.2.C-48***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.C-49	3.5.1-37	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.C-49	3.5.1-37	A
Steel Elements: Vacuum Breaker Valves, Isolation Valves, and Piping	Pressure Relief	Carbon Steel	Air – Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B4.C-13***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-46	3.5.1-35	A

Table 3.5.2-7 Primary Containment (Continued)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Item	Table 1 Item	Notes
Steel Elements: Vacuum Breaker Valves, Isolation Valves, and Piping	Pressure Relief	Carbon Steel	Air - Indoor Uncontrolled	Loss of Material	ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-46	3.5.1-35	A
		Stainless Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B4.C-13***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.C-49	3.5.1-37	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.C-49	3.5.1-37	A
	Structural Pressure Barrier	Carbon Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B4.C-13***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.CP-46	3.5.1-35	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.CP-46	3.5.1-35	A
		Stainless Steel	Air - Indoor Uncontrolled	***Cumulative Fatigue Damage***	***TLAA***	***II.B4.C-13***	***3.5.1-9***	***C, 2***
				Loss of Material	10 CFR Part 50, Appendix J (B.2.1.32)	II.B2.2.C-49	3.5.1-37	A
					ASME Section XI, Subsection IWE (B.2.1.29)	II.B2.2.C-49	3.5.1-37	A

As a result of the response to RAI 3.5.2.2.1.5-1 provided in Enclosure A of this letter, subsection 3.5.2.2.1.5 on page 3.5-22 of the LRA is revised as shown below:

3.5.2.2.1.5 Cumulative Fatigue Damage

If included in the current licensing basis, fatigue analyses of suppression pool steel shells (including welded joints) and penetrations (including penetration sleeves, dissimilar metal welds, and penetration bellows) for all types of PWR and BWR containments and BWR vent header, vent line bellows, and downcomers are TLAAs as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c). The evaluation of this TLAA is addressed separately in Section 4.6, "Containment Liner Plates, Metal Containments, and Penetrations Fatigue Analysis," of this SRP-LR.

Item Number 3.5.1-9 is applicable for LSCS. Fatigue is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c). LSCS has no penetration bellows, torus vent line, vent line header, vent line bellows, unbraced downcomers, or vent header. The evaluation of fatigue as a TLAA for the LSCS Primary Containment liner, ***Class MC Components,*** and penetration sleeves, refueling bellows, and downcomers is addressed in Section 4.6.

As a result of the response to RAI 4.2.5-1 provided in Enclosure A of this letter, the LRA Table of Contents on LRA page xiii is revised to change the titles for Tables 4.2.5-1 and 4.2.5-2, as follows:

As a result of the response to RAI 4.2.5-1 provided in Enclosure A of this letter, LRA Table 4.1-2 on LRA page 4-6 is revised as follows:

TABLE 4.1-2 SUMMARY OF RESULTS - LSCS TIME-LIMITED AGING ANALYSES		
TLAA DESCRIPTION	**Disposition**	**LRA SECTION**
IDENTIFICATION AND EVALUATION OF TIME-LIMITED AGING ANALYSES (TLAAS)		4.1
Identification of LSCS Time-Limited Aging Analyses		4.1.1
Evaluation of LSCS Time-Limited Aging Analyses		4.1.2
Acceptance Criteria		4.1.3
Summary of Results		4.1.4
Identification of Exemptions to 10CFR50.12 Associated With TLAAs		4.1.5
REACTOR VESSEL AND INTERNALS NEUTRON EMBRITTLEMENT ANALYSES		4.2
Neutron Fluence Analyses	§54.21(c)(1)(ii)	4.2.1
Upper-Shelf Energy Analyses	§54.21(c)(1)(ii)	4.2.2
Adjusted Reference Temperature Analyses	§54.21(c)(1)(ii)	4.2.3
Pressure – Temperature Limits	§54.21(c)(1)(iii)	4.2.4
Axial Weld Failure Probability Assessment Analyses	§54.21(c)(1)(ii)***&(iii)***	4.2.5
Circumferential Weld Failure Probability Assessment Analyses	§54.21(c)(1)(iii)	4.2.6
Reactor Pressure Vessel Reflood Thermal Shock Analyses	§54.21(c)(1)(ii)	4.2.7
RPV Core Plate Rim Hold-Down Bolt Loss of Preload Analysis	§54.21(c)(1)(i)	4.2.8
Jet Pump Riser Brace Clamp Loss of Preload Analysis	§54.21(c)(1)(i)	4.2.9
Jet Pump Slip Joint Repair Clamp Loss of Preload Analysis	§54.21(c)(1)(iii)	4.2.10
METAL FATIGUE ANALYSES		4.3
ASME Section III, Class 1 Fatigue Analyses	§54.21(c)(1)(iii)	4.3.1
ASME Section III, Class 2 and 3 and ANSI B31.1 Allowable Stress Analyses	§54.21(c)(1)(i)	4.3.2
Environmental Fatigue Analyses for RPV and Class 1 Piping	§54.21(c)(1)(iii)	4.3.3
Reactor Vessel Internals Fatigue Analyses	§54.21(c)(1)(iii)	4.3.4
High-Energy Line Break (HELB) Analyses Based On Fatigue	§54.21(c)(1)(iii)	4.3.5
Main Steam Relief Valve Discharge Piping Fatigue Analysis	§54.21(c)(1)(i)	4.3.6
ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIC COMPONENTS		4.4
Environmental Qualification (EQ) of Electric Components	§54.21(c)(1)(iii)	4.4.1
CONCRETE CONTAINMENT TENDON PRESTRESS ANALYSES		4.5
Concrete Containment Tendon Prestress Analyses	§54.21(c)(1)(iii)	4.5.1
PRIMARY CONTAINMENT FATIGUE ANALYSES		4.6
Primary Containment Liner and Penetrations Fatigue Analyses	§54.21(c)(1)(iii)	4.6.1
Primary Containment Refueling Bellows Fatigue Analysis	§54.21(c)(1)(i)	4.6.2
Primary Containment Downcomer Vents Fatigue Analysis	§54.21(c)(1)(i)	4.6.3
OTHER PLANT-SPECIFIC ANALYSES		4.7
Reactor Building Crane Cyclic Loading Analysis	§54.21(c)(1)(i)	4.7.1
Main Steam Line Flow Restrictors Erosion Analysis	§54.21(c)(1)(ii)	4.7.2

As a result of the response to RAI 4.2.5-1 provided in Enclosure A of this letter, LRA Section 4.2.5 starting on LRA page 4-44 is revised as follows:

4.2.5 AXIAL WELD FAILURE PROBABILITY ASSESSMENT ANALYSES

TLAA Description:

The BWRVIP recommendations for inspection of reactor pressure vessel shell welds in BWRVIP-05 (Reference 4.8.8) include examination of 100 percent of the axial welds and inspection of the circumferential welds only at the intersections of these welds with the axial welds. BWRVIP-05 contains generic analyses supporting a conclusion in the NRC SER (Reference 4.8.9) that the generic-plant axial weld failure probability is orders of magnitude greater than the 40-year end-of-life circumferential weld failure probability, and this analysis was used to justify relief from inspection of the circumferential welds as permitted in Generic Letter 98-05 (Reference 4.8.7) and as described in LRA Section 4.2.6. The ***weld*** failure ***probability***~~frequency~~ is dependent upon given assumptions of flaw density, distribution, and location. ~~Since the current axial weld failure probability assessments for the Combustion Engineering (CE) Owners Group (CEOG) reactor vessels and for the Chicago Bridge and Iron (CB& I) reactor vessels are based upon 32 EFPY fluence values associated with 40 years of operation, they have been identified as TLAAs requiring evaluation for 54 EFPY through the period of extended operation.~~

Subsequent to the original SER for BWRVIP-05, additional failure probability assessments were performed by the BWRVIP to more accurately determine the conditional failure probability of BWR axial welds. The staff reviewed these and provided separate conditional failure probability assessments in the Supplement to the Final Safety Evaluation of the BWRVIP-05 Report, dated March 7, 2000 (Reference 4.8.46). These assessments resulted in higher failure probabilities than the BWRVIP assessments and are therefore limiting. They demonstrate that the RPV failure frequency due to failure of the axial welds in the BWR fleet are no greater than 5.02E-06 per reactor year. Since the NRC staff failure probability assessments provided in the supplement to the final safety evaluation are applicable to LSCS Units 1 and 2, they have been identified as TLAAs requiring evaluation through the period of extended operation.

TLAA Evaluation:

Unit 1:

The NRC assessment provided in ***Table 3 of*** the ***Supplement to the*** Final Safety Evaluation ~~Report (FSER) to~~ ***of the*** BWRVIP-05 ***Report, dated March 7, 2000*** (Reference 4.8.***46***~~9~~) computed an axial weld failure probability value of ~~8.28E-01~~ ***5.02E-06*** for the ~~CE~~ ***limiting Combustion Engineering*** (CE~~OG~~) vessel, ***Mod 2 variant*** ~~at 64 EFPY and an axial weld failure probability of 3.82E-01 for the CB& I vessel at 64 EFPY~~. In order to evaluate axial weld failure probability assessments for 60 years for the LSCS Unit 1 CE vessel ~~and the LSCS Unit 2 CB&I vessel~~, 54 EFPY fluence values were ***obtained*** ~~derived~~ for ***the inside surface (0T) of*** the limiting axial welds ~~for each vessel~~ from the RAMA fluence projections described in Section 4.2.1. Using the~~se~~ ***bounding*** inside surface ~~(0T)~~ fluence value~~s~~ for the limiting welds (Reference~~s~~ 4.8.1 ~~and 4.8.3~~), the ~~M~~***m***ean RT_{NDT} values were determined for these welds ~~on each unit.~~***, where the mean*** RT_{NDT} ***value does not include the margin term (M) described in RG 1.99, Revision 2, consistent with the evaluation methodology***

described in Section 2.1 of the March 7, 2000 supplement to the final safety evaluation. The results are shown in LRA Table 4.2.5-1.

~~Table 4.2.5-1 provides a comparison between the NRC 64 EFPY axial weld failure probability assessment for the CE (CEOG) vessel and the 54 EFPY axial weld failure probability assessment for the LSCS Unit 1 CE vessel. Table 4.2.5-2 provides a comparison between the NRC 64 EFPY axial weld failure probability assessment for the CB&I vessel and the 54 EFPY axial weld failure probability assessment for the LSCS Unit 2 CB&I vessel. Although a conditional failure probability has not been calculated for the LSCS units at 54 EFPY, since the Unit 1 Mean RTNDT value of 139.9 degrees F is less than the NRC value of 172.4 degrees F for the CE vessel and since the Unit 2 Mean RTNDT value of 12 degrees F is less than the NRC value of 117.1 degrees F for the CB&I vessel, it is concluded that the Unit 1 and 2 conditional failure probabilities are bounded by the NRC analyses , consistent with the requirements defined in GL 98-05. Therefore, the projected mean~~ RT_{NDT} ~~values remain bounded by the NRC analyses.~~

LRA Table 4.2.5-1 provides a comparison of the limiting Unit 1 axial weld mean RT_{NDT} value at 54 EFPY of 139.9ºF (without margin) to the 114ºF mean RT_{NDT} value (without margin) determined for the limiting CE reactor, Mod 2 variant, in the March 7, 2000 supplement to the final safety evaluation. Since the 114ºF mean RT_{NDT} value from the staff assessment does not bound the limiting Unit 1 mean RT_{NDT} value of 139.9ºF, the staff failure probability of 5.02E-06 does not bound the Unit 1 axial welds at 54 EFPY. The limiting Unit 1 axial welds are projected to reach the 114ºF mean RT_{NDT} value (without margin) at approximately 39.15 EFPY. For reference, the Unit 1 cumulative neutron exposure through July 2015 is equivalent to 24.05 EFPY.

TLAA Disposition *for Unit 1*: 10 CFR 54.21(c)(1)~~(ii)~~*(iii)* ~~–~~ ~~The RPV axial weld failure probability assessments have been projected through the period of extended operation.~~ ***The effects of aging on the Unit 1 reactor pressure vessel axial welds will be managed by:***

(1) The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.1.1) aging management program, which requires periodic examination of the axial welds in accordance with the requirements of ASME Section XI, Table IWB-2500-1, Examination Category B-A, Pressure Retaining Welds in Reactor Vessel, and

(2) The Reactor Vessel Surveillance (B.2.1.20) aging management program, which manages neutron embrittlement by monitoring neutron fluence and ensuring that all neutron embrittlement analyses are updated as necessary to evaluate bounding neutron fluence values for each unit. The Reactor Vessel Surveillance program will be enhanced as follows: 1) prior to the period of extended operation, establish a monitoring limit (currently 39.15 EFPY) for neutron fluence at the limiting Unit 1 axial weld that corresponds to an axial weld failure probability of 5.02E-06 per reactor year; and 2) complete a probabilistic axial weld failure analysis for Unit 1 prior to 39.15 EFPY that demonstrates the 60-year axial weld failure probability is no greater than 5.02E-06 per reactor year.

The combination of periodic axial weld examinations by the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program and management of neutron embrittlement by the enhanced Reactor Vessel Surveillance program ensures that the loss of fracture toughness due to neutron irradiation embrittlement of axial welds will not affect the structural integrity of the Unit 1 reactor vessel through the period of extended operation.

Unit 2:

LRA Table 4.2.5-2 provides a comparison of the limiting Unit 2 axial weld mean RT_{NDT} value at 54 EFPY of 12.2ºF (without margin) to the 91ºF mean RT_{NDT} value (without margin) determined for the limiting Chicago Bridge and Iron (CB&I) reactor in the Supplement to the Final Safety Evaluation of the BWRVIP-05 Report, dated March 7, 2000. Since the 91ºF mean RT_{NDT} value from the staff assessment bounds the limiting axial weld mean RT_{NDT} value of 12.2ºF for Unit 2, the staff failure probability of 2.73E-06 is also bounding for Unit 2 for 54 EFPY. The Unit 2 axial welds will also be periodically examined in accordance with the requirements of ASME Section XI, Table IWB-2500-1, Examination Category B-A, Pressure Retaining Welds in Reactor Vessel.

TLAA Disposition for Unit 2: 10 CFR 54.21(c)(1)(ii) – The Unit 2 axial weld failure probability TLAA has been satisfactorily projected through the period of extended operation.

As a result of the response to RAI 4.2.5-1 provided in Enclosure A of this letter, LRA Tables 4.2.5-1 and 4.2.5-2, starting on LRA page 4-45 are revised as shown below.

Table 4.2.5-1
Comparison of NRC ~~64 EFPY~~ Axial Weld Failure Probability Assessment for CE ~~(CEOG)~~ RPV *(Mod 2 Variant)* to LSCS Unit 1 54 EFPY Axial Weld Failure Probability Assessment

Parameter	**NRC Staff ~~64 EFPY~~ Axial Weld Assessment [1]** **(CE ~~CEOG~~ RPV *Mod 2 Variant)***	**LSCS Unit 1 54 EFPY Axial Weld Assessment** **(CE RPV Middle Shell Axial Welds 3-308A, B, & C)**
Copper Content (%)	0.219	0.210
Nickel Content (%)	0.996	0.750
Chemistry Factor (CF)	***Not Reported*** ~~231.1~~	437
Fluence at 0T (n/cm^2)	~~4.0E+18~~ ***0.148E+19***	8.66E+17
Unirradiated Reference Temperature $RT_{NDT(U)}$ (ºF)	***-2*** ~~0~~	-30
Shift in Reference Temperature ΔRT_{NDT} (ºF)(without margin) **[2]**	***Not Reported*** ~~172.4~~	169.9
Mean RT_{NDT} (ºF) ***[4]***	***114*** ~~172.4~~	139.9
Probability of Failure Event	***5.02E-06*** ~~8.28E-01~~	**[3]**

NOTES:

[1] The NRC data is obtained from ~~BWRVIP-05 Report, "Final Safety Evaluation of the BWR vessel and Internals Project BWRVIP-05 Report, July 28, 1998 (Reference 4.8.9).~~***the Supplement to the Final Safety Evaluation of the BWRVIP-05 Report, March 7, 2000 (Reference 4.8.46).***

[2] $\Delta RT_{NDT} = CF * f^{(0.28 - 0.10 \log f)}$, where f is fluence in units of E+19 n/cm^2.

[3] Although a conditional failure probability has not been calculated***,*** ~~for LSCS~~ since the LSCS ***Unit 1*** Mean RT_{NDT} value~~s are~~ ***is not*** less than the NRC value~~s~~, it is concluded that the LSCS ***Unit 1*** conditional failure probability is ***not*** bounded by the NRC analysis~~, consistent with the requirements defined in GL 98-05~~. ***Consistent with SRP-LR Section***

4.2.3.1.5, the Reactor Vessel Surveillance aging management program is enhanced to monitor axial weld embrittlement relative to the value specified in the Supplement to the Final Safety Evaluation of the BWRVIP-05 Report, March 7, 2000.

[4] These Mean RT_{NDT} values do not include the margin term (M) described in RG 1.99, Revision 2 and the PTS Rule (10 CFR 50.61), consistent with the methodology described in Section 2.1 of the Supplement to the Final Safety Evaluation of the BWRVIP-05 Report, March 7, 2000.

Table 4.2.5-2 Comparison of NRC ~~64 EFPY~~ Axial Weld Failure Probability Assessment for CB&I RPV to LSCS Unit 2 54 EFPY Axial Weld Failure Probability Assessment		
Parameter	**NRC Staff ~~64 EFPY~~ Axial Weld Assessment [1]** **(CB&I RPV)**	**Unit 2 54 EFPY Axial Weld Assessment** **(CB&I RPV Lower-Intermediate Shell Axial Welds BD, BE, & BF)**
Copper Content (%)	0.10	0.026
Nickel Content (%)	1.08	0.920
Chemistry Factor (CF)	***Not Reported*** ~~135~~	41
Fluence at 0T (n/cm^2)	***0.676E+19*** ~~1.38E+19~~	1.14E+18
Unirradiated Reference Temperature $RT_{NDT(U)}$ (ºF)	-30	-6
Shift in Reference Temperature ΔRT_{NDT} (ºF)(without margin) **[2]**	***Not Reported*** ~~147.1~~	18.***2***
Mean RT_{NDT} (ºF) ***[4]***	***91*** ~~117.1~~	12.***2***
Probability of Failure Event	***2.73E-06*** ~~3.82E-01~~	**[3]**

NOTES:

[1] The NRC data is obtained from ~~BWRVIP-05 Report, "Final Safety Evaluation of the BWR vessel and Internals Project BWRVIP-05 Report, July 28, 1998 (Reference 4.8.9).~~***the Supplement to the Final Safety Evaluation of the BWRVIP-05 Report, March 7, 2000 (Reference 4.8.46).***

[2] $\Delta RT_{NDT} = CF * f^{(0.28 - 0.10 \log f)}$, where f is fluence in units of E+19 n/cm^2.

[3] Although a conditional failure probability has not been calculated, ~~for LSCS~~ since the LSCS ***Unit 2*** Mean RT_{NDT} value~~s are~~ ***is*** significantly less than the NRC value~~s~~, it is concluded that the LSCS ***Unit 2*** conditional failure probability is bounded by the NRC analysis, consistent with ~~the requirements defined in GL 98-05.~~ ***SRP-LR Section 4.2.3.1.5.***

[4] These Mean RT_{NDT} values do not include the margin term (M) described in RG 1.99, Revision 2 and the PTS Rule (10 CFR 50.61), consistent with the methodology described in Section 2.1 of the Supplement to the Final Safety Evaluation of the BWRVIP-05 Report, March 7, 2000.

As a result of the response to RAI 4.2.5-1 provided in Enclosure A of this letter, LRA Section 4.8, Docketed References, on LRA page 4-114, is revised as follows:

4.8.31 UFSAR Section 3.8.2.5.2.2.2, *Normal and Upset Conditions*

4.8.32 UFSAR Section 3.8.2.5.2.2.3, *Emergency Conditions*

4.8.33 UFSAR Section 3.7.3.17, *Determination of Number of Safety/Relief Valve (SRV) Discharge Cycles*

4.8.34 USNRC *Safety Evaluation Report for Relief Request CR-38*, January 28, 2004.

4.8.35 10 CFR 50 Appendix A , *General Design Criterion 31, Fracture prevention of reactor coolant pressure boundary*

4.8.36 ASME Boiler and Pressure Vessel Code, Section XI, 2001 Edition through 2004 Addenda.

4.8.37 ASME Boiler and Pressure Vessel Code, Section XI, *Non-mandatory Appendix A*

4.8.38 BWR Vessel and Internals Project, *BWR Core Plate Inspection and Evaluation Guideline (BWRVIP-25),* EPRI Report TR-107284, December 1996.

4.8.39 *Safety Evaluation for Referencing of BWR Vessel and Internals Project, BWR Core Plate Inspection and Evaluation Guideline (BWRVIP-25), Report for Compliance with the Technical Information Requirements of the License Renewal Rule,* December 7, 2000.

4.8.40 NEDC-30271, Revision 9, *LaSalle County Station Unit 1 NSSS New Loads Design Adequacy Evaluation Final Summary Report,* Table 2-3, June 1981.

4.8.41 NEDC-30272, Revision 9, *LaSalle County Station Unit 2 NSSS New Loads Design Adequacy Evaluation Final Summary Report,* Table 2-3, September 1983.

4.8.42 UFSAR Section 9.1.4.2.3, *Reactor Building Crane*

4.8.43 UFSAR Section 5.4.4, *Main Steam Line Flow Restrictors*

4.8.44 UFSAR Section 15.6.4, *Steam System Pipe Break Outside Containment*

4.8.45 UFSAR Table 15.6-8, *Steam Line Break Radiological Effects*

4.8.46 Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report, March 7, 2000

As a result of the response to RAI 4.2.5-1 provided in Enclosure A of this letter, LRA Section A.1.1 Item 20, on LRA page A-6 is revised as follows:

16. Fire Protection (Section A.2.1.16) [Existing - Requires Enhancement]

17. Fire Water System (Section A.2.1.17) [Existing - Requires Enhancement]

18. Aboveground Metallic Tanks (Section A.2.1.18) [Existing - Requires Enhancement]

19. Fuel Oil Chemistry (Section A.2.1.19) [Existing - Requires Enhancement]

20. Reactor Vessel Surveillance (Section A.2.1.20) [Existing ***– Requires Enhancement***]

As a result of the response to RAI 4.2.5-1 provided in Enclosure A of this letter, LRA Section A.2.1.20, on LRA page A-25 is revised as follows:

A.2.1.20 Reactor Vessel Surveillance

The Reactor Vessel Surveillance aging management program is an existing condition monitoring program that manages the loss of fracture toughness due to neutron irradiation embrittlement of the reactor vessel beltline materials. The program meets the requirements of 10 CFR 50, Appendix H. The program manages the surveillance capsules in each unit and ensures that the specimen exposure, capsule withdrawal, sample testing, and capsule storage meet the requirements of 10 CFR 50, Appendix H and ASTM E-185. The program evaluates neutron embrittlement by projecting Upper-Shelf Energy (USE) for reactor materials and impact on Adjusted Reference Temperature for the development of pressure-temperature limit curves. Embrittlement evaluations are performed in accordance with Regulatory Guide 1.99, Revision 2. The Reactor Vessel Surveillance program is part of the BWRVIP Integrated Surveillance Program (ISP) described in BWRVIP-86 Revision 1-A and approved by the NRC staff. The schedule for removing surveillance capsules is in accordance the timetable specified in BWRVIP-86 Revision 1-A for the current license term and for the period of extended operation.

The program monitors plant operating conditions to ensure appropriate steps are taken if reactor vessel exposure conditions are altered; such as the review and updating of 60-year fluence projections to support upper-shelf energy calculations and pressure-temperature limit curves. The program also includes condition monitoring by removal and analysis of surveillance capsules as part of the BWRVIP ISP. These measures are effective in detecting the extent of embrittlement to prevent significant degradation of the reactor pressure vessel during the period of extended operation.

The Reactor Vessel Surveillance aging management program will be enhanced to:

1. Establish a monitoring limit (currently 39.15 EFPY) for neutron fluence at the limiting Unit 1 axial weld that corresponds to an axial weld failure probability of 5.02E-06 per reactor year.

This enhancement will be implemented prior to the period of extended operation.

2. Complete a probabilistic axial weld failure analysis for Unit 1 that demonstrates the 60-year axial weld failure probability is no greater than 5.02E-06 per reactor year.

This enhancement will be completed prior to Unit 1 exceeding 39.15 EFPY.

As a result of the response to RAI 4.2.5-1 provided in Enclosure A of this letter, LRA Section A.4.2.5, starting on LRA page A-49 is revised as follows:

A.4.2.5 Axial Weld Failure Probability Assessment Analyses

The BWRVIP recommendations for inspection of reactor pressure vessel shell welds in BWRVIP-05 include examination of 100 percent of the axial welds and inspection of the circumferential welds only at the intersections of these welds with the axial welds. BWRVIP-05 contains generic analyses supporting a conclusion in the NRC Final Safety Evaluation Report (FSER) ***dated July 28, 1998,*** that the generic-plant axial weld failure ***probability***~~rates~~ is orders of magnitude greater than the 40-year end-of-life circumferential weld failure probability and used this analysis to justify relief from inspection of the circumferential welds. ~~The failure frequency is dependent upon given assumptions of flaw density, distribution, and location. Since the axial weld failure probability assessment is based on 32 EFPY fluence values associated with 40 years of operation, it has been identified as a TLAA requiring evaluation for the period of extended operation.~~ ***The staff provided separate conditional failure probability assessments in the Supplement to the Final Safety Evaluation of the BWRVIP-05 Report, dated March 7, 2000. Since these NRC staff failure probability assessments are applicable to LSCS Units 1 and 2, they are identified as TLAAs requiring evaluation through the period of extended operation.***

~~The LSCS axial weld failure probability has been projected for the period of extended operation. In order to evaluate the axial weld failure probability assessment for 60 years, 54 EFPY fluence values were derived for the limiting axial weld. Using the 54 EFPY fluence values, the LSCS Mean RT_{NDT} values were determined for each unit and compared to the NRC analytical results for 64 EFPY provided in the FSER to BWRVIP-05. Although a conditional failure probability has not been calculated for the LSCS units, the LSCS Mean RT_{NDT} values for the period of extended operation are significantly less than the NRC RT_{NDT} value used in determining the conditional failure probability. Therefore, the NRC conditional failure probability is bounding for LSCS Unit 1 and Unit 2, consistent with the requirements defined in GL 98-05. These analyses have been projected through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).~~

For LSCS Unit 1, the limiting axial weld mean RT_{NDT} value at 54 EFPY exceeds the mean RT_{NDT} value determined for the limiting Combustion Engineering (CE), Mod 2 variant, reactor vessel. Therefore, the associated axial weld failure probability for Unit 1 at 54 EFPY is not bounded by the conditional axial weld failure probability of 5.02E-06 determined for the CE reactor vessel. The effects of aging for the Unit 1 reactor vessel axial welds will be managed in accordance with 10 CFR 54.21(c)(1)(iii) by:

(1) The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (A.2.1.1) program, which requires periodic examination of the axial welds in accordance with the requirements of ASME Section XI, Table IWB-2500-1, and

(2) The Reactor Vessel Surveillance (A.2.1.20) program, which manages neutron embrittlement by monitoring neutron fluence and ensures that neutron embrittlement analyses are updated as necessary to evaluate bounding neutron fluence values for each unit. The Reactor Vessel Surveillance program is enhanced as follows: 1) prior to the period of extended operation, establish a monitoring limit (currently 39.15 EFPY) for neutron fluence at the limiting Unit 1 axial weld that corresponds to an axial weld failure probability of 5.02E-06 per reactor year; and 2) complete a probabilistic axial weld failure analysis for Unit 1 prior to 39.15 EFPY that demonstrates the 60-year axial weld failure probability is no greater than 5.02E-06 per reactor year.

The combination of periodic axial weld examinations by the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program and management of neutron

embrittlement by the enhanced Reactor Vessel Surveillance program ensures that the axial weld failure probability will not exceed 5.02E-06 and ensures that loss of fracture toughness due to neutron irradiation embrittlement will not affect the structural integrity of the Unit 1 reactor vessel through the period of extended operation.

For LSCS Unit 2, the limiting axial weld mean RT_{NDT} value at 54 EFPY is bounded by the mean RT_{NDT} value determined for the limiting axial weld in the Chicago Bridge and Iron (CB&I) reactor vessel evaluated in the supplement to the final safety evaluation for BWRVIP-05. Therefore, the NRC conditional axial weld failure probability of 2.73E-06 is bounding for LSCS Unit 2. Therefore, this analysis has been projected through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

As a result of the response to RAI 4.2.5-1 provided in Enclosure A of this letter, LRA Section B.1.5 Item 20, on LRA page B-8 is revised as follows:

14. Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (Section B.2.1.14) [Existing - Requires Enhancement]
15. Compressed Air Monitoring (Section B.2.1.15) [Existing - Requires Enhancement]
16. Fire Protection (Section B.2.1.16) [Existing - Requires Enhancement]
17. Fire Water System (Section B.2.1.17) [Existing - Requires Enhancement]
18. Aboveground Metallic Tanks (Section B.2.1.18) [Existing - Requires Enhancement]
19. Fuel Oil Chemistry (Section B.2.1.19) [Existing - Requires Enhancement]
20. Reactor Vessel Surveillance (Section B.2.1.20) [Existing ***– Requires Enhancement***]

As a result of the response to RAI 4.2.5-1 provided in Enclosure A of this letter, LRA Section B.2.1.20, on LRA page B-94 is revised as follows:

Enhancements

~~None.~~

Prior to the period of extended operation, the following enhancement will be implemented in the following program elements:

1. ***Establish a monitoring limit (currently 39.15 EFPY) for neutron fluence at the limiting Unit 1 axial weld that corresponds to an axial weld failure probability of 5.02E-06 per reactor year. Program Elements Effected: Monitoring and Trending (Element 5), Acceptance Criteria (Element 6), and Corrective Actions (Element 7)***

Prior to exceeding 39.15 EFPY on Unit 1, the following enhancement will be implemented in the following program element:

2. ***Complete a probabilistic axial weld failure analysis for Unit 1 that demonstrates the 60-year axial weld failure probability is no greater than 5.02E-06 per reactor year. Program Element Effected: Corrective Actions (Element 7)***

As a result of the response to RAI 4.2.5-1 provided in Enclosure A of this letter, LRA Appendix C, Applicant Action Item BWRVIP-74-A (12) on LRA page C-10 is revised as follows:

Action Item Description	LaSalle Response
BWRVIP-74-A (12) As indicated in the staff's March 7, 2000, letter to Carl Terry, a LR applicant shall monitor axial beltline weld embrittlement. One acceptable method is to determine that the mean RT_{NDT} of the limiting axial beltline weld at the end of the period of extended operation is less than the values specified in Table 1 of this FSER.	The Axial Weld Failure Probability Assessment Analyses ha***ve***~~s~~ been identified as ~~a~~ TLAA***s*** ~~as~~ ***that are evaluated*** ~~discussed~~ in LRA Section 4.2.5 ***and*** ~~. The TLAA evaluation shows that the mean RT_{NDT} of the limiting axial beltline weld for each unit at the end of the period of extended operation is less than the value specified in Table 1 of BWRVIP-74-A FSER as shown in LRA~~ Table***s*** 4.2.5-1~~,~~ and ~~Table~~ 4.2.5-2.
BWRVIP-74-A (13) The Charpy USE, P-T limit, circumferential weld and axial weld RPV integrity evaluations are all dependent upon the neutron fluence. The applicant may perform neutron fluence calculations using staff approved methodology or may submit the methodology for staff review. If the applicant performs the neutron fluence calculation using a methodology previously approved by the staff, the applicant should identify the NRC letter that approved the methodology.	An NRC approved methodology was used to determine fluence during the period of extended operation, as discussed in LRA Section 4.2.1. The RAMA Methodology used was approved within the SER for BWRVIP-114, 115, 117 and 121.
BWRVIP-74-A (14) Components that have indications that have been previously analytically evaluated in accordance with sub-section IWB-3600 of Section XI to the ASME Code until the end of the 40-year service period shall be re-evaluated for the 60-year service period corresponding to the LR term.	There are no components within the ASME Code Class 1 reactor coolant pressure boundary with indications that have been previously analytically evaluated until the end of the 40-year service period.

Enclosure C

LSCS License Renewal Commitment List Updates

This Enclosure identifies commitments made in this document and is an update to the LSCS LRA Appendix A, Section A.5 License Renewal Commitment List. Any other actions discussed in the submittal represent intended or planned actions. They are described to the NRC for the NRC's information and are not regulatory commitments.

Changes to the LSCS LRA Appendix A, Section A.5 License Renewal Commitment List are as a result of the Exelon response to the following RAI:

RAI 4.2.5-1

Note: To facilitate understanding, relevant portions of the previously submitted License Renewal Commitment List have been repeated in this Enclosure, with revisions indicated. Previously submitted information is shown in normal font. Changes due to this submittal are highlighted with ***bolded italics*** for inserted text and ~~strikethroughs~~ for deleted text.

Enclosure C

As a result of the response to RAI 4.2.5-1 provided in Enclosure A of this letter, LRA Appendix A, Section A.5 on page A-69 of the LRA is revised as follows:

NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE
20	Reactor Vessel Surveillance	~~Existing program is credited.~~ ***Reactor Vessel Surveillance is an existing program that will be enhanced to:*** ***1. Establish a limit (currently 39.15 EFPY) and monitor for neutron fluence at the limiting Unit 1 axial weld that corresponds to an axial weld failure probability of 5.02E-06 per reactor year.*** ***2. Complete a probabilistic axial weld failure analysis for Unit 1 that demonstrates the 60-year axial weld failure probability is no greater than 5.02E-06 per reactor year.***	~~Ongoing~~ ***Program to be enhanced prior to the period of extended operation.*** ***Analysis to be completed prior to 39.15 EFPY on Unit 1.***	Section A.2.1.20 ***Section 4.2.5*** ***Exelon Letter RS-15-223 08/26/2015***

Enclosure D

Correction to Enclosure B of Exelon letter RS-15-165

Associated with Response to Set 2 RAI B.2.1.20-2

Note: The LRA mark-up of the "Exceptions to NUREG-1801" sub-section of LRA Appendix B, Section B.2.1.20, provided on Page 14 of 33 of Enclosure B of Exelon letter RS-15-165, submitted on June 25, 2015, contained a numerical error. This error is being corrected by resubmittal of this subsection from the original letter, with the incorrect number, 8.67E+18, shown in ~~strikethrough~~ format, and the correct number, 8.67E+17, shown in ***bolded/italics*** format. No other information associated with Exelon letter RS-15-165 has been changed.

Note: See explanation on page 1 of this Enclosure.

Exceptions to NUREG-1801

The NUREG-1801, Revision 2, Chapter XI.M31, Reactor Vessel Surveillance aging management program, Element 4, Detection of Aging Effects, states that “the program withdraws one capsule at an outage in which the capsule receives a neutron fluence of between one and two times the peak reactor vessel wall neutron fluence at the end of the period of extended operation and tests the capsule in accordance with the requirements of ASTM E 185-82.” The neutron fluence values for LSCS Unit 2 ISP(E) surveillance plate and weld materials, as planned in BWRVIP-86, Revision 1-A, are not consistent with this fluence range criterion. The Unit 2 ISP(E) surveillance weld is located in a Susquehanna Unit 1 capsule that has a planned fluence exposure of ~~8.67E+18~~***8.67E+17*** n/cm^2, which is less than one times the 60-year peak (0T) fluence value of 1.14E+18 n/cm^2 projected for the Unit 2 weld. The Unit 2 ISP(E) surveillance plate is located in a River Bend reactor capsule that has a planned fluence exposure of 4.49E+18 n/cm^2, which is more than two times the 60-year peak (0T) fluence value of 1.22E+18 n/cm^2 projected for the limiting Unit 2 plate.
Program Elements Affected: Detection of Aging Effects (Element 4)