April 13, 2022

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U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
11545 Rockville Pike
One White Flint North
Rockville, MD 20852-2746

St. Lucie Nuclear Plant Units 1 and 2
Dockets 50-335 and 50-389
Facility Operating Licenses DPR-67 and NPF-16

SUBSEQUENT LICENSE RENEWAL APPLICATION REVISION 1 – SUPPLEMENT 2

References:

1. FPL Letter L-2021-192 dated October 12, 2021 – Subsequent License Renewal Application **–** Revision 1 (ADAMS Accession No. ML21285A107)
2. U.S. Nuclear Regulatory Commission (NRC) Letter dated September 24, 2021, St. Lucie Plant, Units 1 and 2 – Aging Management Audit Plan Regarding the Subsequent License Renewal Application Review (ADAMS Accession No. ML21245A305)

FPL, owner and licensee for St. Lucie Nuclear Plant (PSL) Units 1 and 2, has submitted a revised subsequent license renewal application (SLRA) for the Facility Operating Licenses for PSL Units 1 and 2 (Reference 1). During NRC's aging management audit of the SLRA with FPL (Reference 2), FPL agreed to supplement the SLRA (Enclosure 3, Attachment 1 of Reference 1) with new or clarifying information. The attachments to this letter provide that information.

For ease of reference, the index of attachment topics is provided on page 3 of this letter. In each attachment, changes are described along with the affected section(s) and page number(s) of the docketed SLRA (Enclosure 3, Attachment 1 of Reference 1) where the changes are to apply. For clarity, revisions to the SLRA are provided with deleted text by ~~strikethroughs~~ and inserted text by **bold red underline.** Revisions to SLRA tables are shown by providing excerpts from each affected table.

Should you have any questions regarding this submittal, please contact me at (561) 304-6256 or William.Maher@fpl.com.

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

Executed on the 13th day of April 2022.

Sincerely,

William Maher

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Attachments Index	
Attachment No.	PSL SLRA Revision 1 Enclosure 3 Attachment 1 Topic
1	SLRA Further Evaluation Section 3.5.2.2.2.6, Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation
2	SLRA Further Evaluation Section 3.5.2.2.2.7, Loss of Fracture Toughness Due to Irradiation Embrittlement of Reactor Pressure Vessel (RPV) Supports
End	

SLRA Further Evaluation Section 3.5.2.2.2.6, Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation

Affected SLRA Sections: 3.5.2.2.2.6, Table 3.5.2.2-1, B.2.3.33

SLRA Page Numbers: 3.5-35, 3.5-38, 3.5-39, 3.5-40, B-251

Description of Change:

Clarified the thickness of the primary shield wall (PSW) and the upper cavity walls in SLRA Section 3.5.2.2.2.6, page 3.5-35.

Provided a summary of the PSW evaluations for both units and clarified the analysis of record applicable to both units in SLRA Section 3.5.2.2.2.6, page 3.5-35.

A statement for the analysis of record superseding all other UFSAR information for PSW loading and capacities is added in SLRA Section 3.5.2.2.2.6, page 3.5-35.

Included concrete strength for various parts of the cavity walls in SLRA Section 3.5.2.2.2.6, Table 3.5.2.2-1, page 3.5-38, for a better description consistent with drawings.

The following discussion is provided to address the uncertainties associated with the neutron fluences, gamma doses and displacements per atom calculations performed by Westinghouse (Reference 12) for the PSL primary shield wall (PSW) and reactor pressure vessel (RPV) supports for the locations below:

- RPV beltline
- PSW/lower cavity concrete (LCC)
- RPV support steel

The methodology used to determine the RPV, PSW concrete and vessel support structure exposures for the PSL SLRA followed the guidance of Regulatory Guide 1.190 (Reference 2) and was consistent with the USNRC-approved methodology described in WCAP-18124-NP-A (Reference 3).

Reactor Pressure Vessel Beltline

To demonstrate the applicability of the Reference 3 methodology to PSL Units 1 and 2, the results of the plant-specific neutron transport calculations were compared with the available in-vessel surveillance capsule threshold sensor measurements. These measurement-to-calculation (M/C) and best-estimate-to-calculation (BE/C) comparisons are provided in summary letter reports LTR-REA-21-1-NP (PSL Unit 1) and LTR-REA-21-2-NP (PSL Unit 2) included in Enclosure 4 of the PSL SLRA (Reference 1) and are repeated in Table 2-1 through Table 2-4 below.

Table 2-1
Measurement-to-Calculation (M/C) Ratios for the Surveillance Capsules – PSL Unit 1

Reaction	Capsule 97	Capsule 104	Capsule 284	Average	Std. Dev.
^{63}Cu (n,α) ^{60}Co	1.40	1.11	1.17	1.23	12.5%
^{46}Ti (n,p) ^{46}Sc	1.22	0.96	--[1]	1.09	16.9%
^{54}Fe (n,p) ^{54}Mn	1.10	0.89	1.05	1.01	10.8%
^{58}Ni (n,p) ^{58}Co	1.14	0.85	1.15	1.05	16.3%
^{238}U(Cd) (n,f) ^{137}Cs	1.17	0.75	--[2]	0.96	30.9%
Average of M/C Ratios				1.07	16.1%

Note(s):
1. The normalized reaction rate for this sensor was not within three standard deviations of the Combustion Engineering (CE) in-vessel surveillance capsule database value. This sensor was therefore rejected.
2. The uranium powder in this fission monitor was contaminated with cadmium powder and could not be counted. This is not unusual for the type of surveillance capsules used at St. Lucie.

Table 2-2
Best-Estimate-to-Calculation (BE/C) Ratios for the Surveillance Capsules – PSL Unit 1

Capsule	Fast (E > 1.0 MeV) Fluence Rate		Iron Atom Displacement Rate	
	BE/C	**Std. Dev.**	**BE/C**	**Std. Dev.**
97	1.09	6.0%	1.10	6.0%
104	0.83	6.0%	0.85	6.0%
284	1.08	7.0%	1.08	6.0%
Average	1.00	14.8%	1.01	13.7%

Table 2-3
Measurement-to-Calculation (M/C) Ratios for the Surveillance Capsules – PSL Unit 2

Reaction	Capsule 83°	Capsule 263°	Capsule 97°	Average	Std. Dev.
^{63}Cu (n,α) ^{60}Co	1.27	1.18	--[1]	1.23	5.2%
^{46}Ti (n,p) ^{46}Sc	1.13	1.13	1.10	1.12	1.5%
^{54}Fe (n,p) ^{54}Mn	1.11	1.11	1.02	1.08	4.8%
^{58}Ni (n,p) ^{58}Co	1.12	1.07	1.03	1.07	4.2%
^{238}U(Cd) (n,f) ^{137}Cs	0.75	--[1]	0.77	0.76	1.9%
Average of M/C Ratios				1.06	14%

Note(s):
1. The normalized reaction rate for this sensor was not within three standard deviations of the Combustion Engineering (CE) in-vessel surveillance capsule database value. This sensor was therefore rejected.

Table 2-4
Best-Estimate-to-Calculation (BE/C) Ratios for the Surveillance Capsules – PSL Unit 2

Capsule	Fast (E > 1.0 MeV) Fluence Rate		Iron Atom Displacement Rate	
	BE/C	Std. Dev.	BE/C	Std. Dev.
83°	1.00	6.0%	1.01	6.0%
263°	1.08	7.0%	1.08	6.0%
97°	0.96	6.0%	0.97	6.0%
Average	1.01	6.0%	1.02	5.5%

The M/C and BE/C data comparisons in Table 2-1 through Table 2-4 provide a validation of the results of the plant-specific neutron transport calculations. Each of these data comparisons shows that the in-vessel measurements and calculations agree within the 20% criterion specified in Reference 2. In addition, the average M/C and BE/C results agree within the 13% (1σ) analytical uncertainty assigned in Reference 3 to fast neutron (E > 1.0 MeV) fluence values at RPV beltline locations. Additional details regarding the analytical uncertainty analysis performed for the RPV beltline region are provided in Section 4.5 of Reference 3.

PSW

An analytical uncertainty analysis associated with the neutron fluence and gamma dose at the inner surface of the PSW was not performed for the PSL SLRA. Therefore, a conservative estimate of the uncertainty associated with these results was established using the RPV extended beltline uncertainty analysis described in WCAP-18124-NP-A Revision 0 Supplement 1-P (Reference 4). Note that the level of detail in the model used for the extended beltline uncertainty analysis is commensurate with the plant-specific model for PSL. For example, the mesh sizes, treatment of anisotropic scattering, angular quadrature, modeling of internals structures, etc., are similar.

The existing RPV extended beltline analysis quantified the analytical uncertainty associated with calculated fast neutron (E > 1.0 MeV) fluence rates at the RPV inner and outer surfaces at various elevations above and below the active fuel. As part of this analysis, numerous parameters that were identified as having a potentially significant contribution to the core neutron source, reactor geometry, coolant temperature, discretization, and modeling approximation uncertainties at the RPV inner and outer surfaces were evaluated. More specifically, each parameter identified was evaluated on an individual basis by determining the maximum relative change in the base-case fluence rate that occurred as the magnitude of that parameter was varied over a bounding range of values. The net analytical uncertainty associated with a given RPV location was then determined by taking the root sum of squares of the individual parameter uncertainty values determined at that location. Given the parameters considered, the magnitudes of the parameter variations evaluated, and the relative proximity of the RPV outer surface to the PSW, the extended beltline uncertainty analysis results for the RPV outer surface were judged to provide a reasonable basis for estimating the analytical uncertainty associated with the PSW neutron and gamma exposures.

The maximum neutron fluence and gamma dose projections at the inner surface of the PSW occur at elevations that are near the core midplane. However, since the extended beltline uncertainty analysis was, by design, focused on the RPV extended beltline region only, it did not consider axial elevations near the core midplane; the elevations nearest the midplane considered were 30 cm above the top and 30 cm below the bottom of the active fuel. Therefore,

the extended beltline uncertainty analysis results determined at the RPV outer surface, 30 cm above the top of the active fuel were used as the starting point for estimating the uncertainty associated with the PSW neutron and gamma exposures. This is conservative because analytical uncertainties increase with axial distance above the top of the active fuel.

In addition to using the uncertainty of this bounding RPV location as a starting point, the concrete composition parameter uncertainty value determined at this location was increased by a factor of 2. This value was increased because it was associated with the one parameter evaluated in the RPV extended beltline analysis whose uncertainty was judged to be potentially impacted in a non-negligible manner if a detailed uncertainty analysis for the PSW were performed. Note that the standard concrete composition from the BUGLE-96 documentation (Reference 6) was used for the PSW in both the extended beltline analytical uncertainty analysis base-case calculations and the St. Lucie neutron fluence and gamma dose calculations.

Following this process, the analytical uncertainty associated with the fast neutron (E > 1.0 MeV) fluence and gamma dose results at the inner surface of the PSW was conservatively estimated to be 20%. Note that:

- the estimated 20% value is based on the extended beltline uncertainty analysis results determined at the RPV outer surface, 30 cm above the top of the active fuel, and
- analytical uncertainties at the RPV outer surface increase with distance from the core midplane elevation.

Therefore, the estimated 20% uncertainty is:

- representative for fast neutron (E > 1.0 MeV) fluence and gamma dose results determined at the PSW inner surface and axial elevations within a foot of the top and bottom of the active fuel, and
- bounding for fast neutron (E > 1.0 MeV) fluence and gamma dose results determined at the PSW inner surface and axial elevations near the core midplane.

The estimated 20% uncertainty does not explicitly account for neutrons with energies between 1.0 MeV and 0.1 MeV. However, the PSW exposures determined for St. Lucie are maximum values that occur at elevations near the core midplane, where the analytical uncertainty for fast neutron (E > 1.0 MeV) fluence at the PSW inner surface is significantly less than 20%. For example, Section 4.5 of Reference 3 documents that the analytical uncertainty for fast neutron (E > 1.0 MeV) fluence in the reactor cavity (i.e., at the RPV outer surface) at the core midplane elevation is approximately 12%. While the uncertainty associated with fast neutron (E > 0.1 MeV) fluence at the PSW inner surface and elevations near the core midplane is greater than 12%, it would not be expected to be significantly different, or greater, than the estimated uncertainty of 20% assigned to the PSW maximum exposures in Section 3.5.2.2.2.6 of Enclosure 3 of the St. Lucie SLRA (Reference 1).

Finally, the maximum PSW gamma dose in Section 3.5.2.2.2.6 of Enclosure 3 of the St. Lucie SLRA is 6.62×10^9 rad. Applying the estimated 20% uncertainty to this value would not cause it to exceed the gamma dose threshold of 1.0×10^{10} rad reported in Section 3.5.2.2.2.6 of NUREG-2192 (Reference 5). Also, as noted on the markup of SLRA page 3.5-39, applying the 20% uncertainty to the neutron fluence will not affect the conclusions presented in the SLRA when using a realistic concrete strain at ultimate strength from ACI 318-69 of 0.003. The radiation induced volumetric expansion depth when applying the 20% uncertainty would still be 0.

RPV Support Steel

An analytical uncertainty analysis associated with the neutron fluence and iron atom displacement (dpa) results for the RPV support steel was not performed for the PSL SLRA. Therefore, a conservative estimate of the uncertainty associated with these results was established using the same RPV extended beltline uncertainty analysis that was used for the PSW.

The maximum neutron fluence and iron atom displacement projections at the RPV support columns and horizontal support bottoms occur at elevations that are within 3 feet of the core midplane. However, since the extended beltline uncertainty analysis was, by design, focused on the RPV extended beltline region only, it did not consider axial elevations within 3 feet of the core midplane; the elevations nearest the midplane considered were 30 cm above the top and 30 cm below the bottom of the active fuel. Therefore, the extended beltline uncertainty analysis results determined at the RPV outer surface 30 cm above the top of the active fuel were used as the starting point for estimating the uncertainty associated with the RPV support columns and horizontal support bottoms. This is conservative because analytical uncertainties increase with axial distance above the top of the active fuel.

The maximum neutron fluence and iron atom displacement projections at the top of the 6-inch plate under the RPV nozzle foot occur at an elevation that is less than 2 ft above the top of the active fuel. Therefore, the extended beltline uncertainty analysis results determined at the RPV outer surface 90 cm above the top of the active fuel were used for the top of the 6-inch plate under the RPV nozzle foot.

In addition to using these bounding RPV locations as starting points, the concrete composition parameter uncertainty values determined at these locations were increased by a factor of 2. These values were increased because they were associated with the one parameter evaluated in the RPV extended beltline analysis whose uncertainty was judged to be potentially impacted in a non-negligible manner if a detailed uncertainty analysis for the RPV support structure were performed. Note that the standard concrete composition from the BUGLE-96 documentation (Reference 6) was used for the PSW in both the extended beltline analytical uncertainty analysis base-case calculations and the St. Lucie neutron fluence and iron atom displacement calculations.

Following this process, the analytical uncertainty associated with the neutron fluence and iron atom displacement results for RPV support columns and horizontal support bottoms was conservatively estimated to be 20%; the analytical uncertainty for the top of the 6-inch plate under the RPV nozzle foot was estimated to be 25%.

The guidance provided in the NUREG-0933 (Reference 7) GSI-15 resolution is to utilize Figure 3-1 of NUREG-1509 (Reference 8) to calculate the change in RPV support structure nil-ductility transition temperature (ΔNDTT) based on iron atom displacements from neutrons with energies greater than 0.1 MeV. However, the RPV support iron atom displacement exposures listed in Table 3.5.2.2-3 and Table 3.5.2.2-4 of Section 3.5.2.2.2.7 of the PSL SLRA (Reference 1) include the contribution from neutrons with energies less than 0.1 MeV. Excluding the contribution from these lower energy neutrons would reduce the iron atom displacement values for the support columns and horizontal support bottoms by approximately 8% and reduce the iron atom displacement values for the top of the 6-inch plate under the RPV nozzle foot by approximately 16%.

Based on the discussion of uncertainties above, exposure calculations in SLRA Section 3.5.2.2.2.6, page 3.5-39 were revised to include a 20% uncertainty at the core mid-plane.

Additional information regarding aggregates was added in SLRA Section 3.5.2.2.2.6, page 3.5-39.

A statement for radiation effects on potential cracking was added in SLRA Section 3.5.2.2.2.6, page 3.5-39. Explanation was provided for much lower IR and cracking under branch line pipe break loads in SLRA Section 3.5.2.2.2.6, page 3.5-39.

The qualitative assessment described in LTR-SDA-21-021-P/NP (Reference 9) did not consider the analytical uncertainty associated with the neutron fluence and iron atom displacement results for the RPV support structure. This is consistent with the approach taken for the Point Beach RPV support structure fracture mechanics analysis summarized in WCAP-18554-P/NP (Reference 10). The St. Lucie fracture mechanics analysis summarized in WCAP-18623-P/NP (Reference 11) included an iron atom displacement exposure uncertainty of 25% in the embrittlement calculation. As discussed in the response to Attachment 2 to this supplement, the updated comparison ratios using calculated critical flaw sizes for Point Beach and St. Lucie confirm the conclusions of the qualitative assessment in LTR-SDA-21-021-P/NP. The St. Lucie RPV support structure critical flaw sizes are larger than the ones for Point Beach, even when accounting for an estimated uncertainty of 25% in the calculated RPV support structure neutron exposures.

Added additional information regarding how the PSL ASME Section XI Inservice Inspection, Subsection IWF AMP treats all three RPV supports on each unit as one and provided the interrelationship of the three AMPs that periodically inspect the reactor cavity areas for PSL Units 1 and 2 in SLRA Section 3.5.2.2.2.6, page 3.5-40.

Added additional information to the plant specific operating experience in SLRA Section B.2.3.33, Structures Monitoring, to address the use of EMBECO 636 grout.

References:

1. Florida Power & Light Company Letter L-2021-142, “Application for Subsequent Renewed Facility Operating License,” Rev. 1, October 2021. (ADAMS Accession Package No. ML21285A107).
2. USNRC Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” Office of Nuclear Regulatory Research, March 2001. (ADAMS Accession No. ML010890301)
3. Westinghouse Report WCAP-18124-NP-A, Revision 0, “Fluence Determination with RAPTOR-M3G and FERRET,” July 2018. (ADAMS Accession No. ML18204A010)
4. Westinghouse Letter LTR-NRC-20-69 dated December 7, 2020, Submittal of WCAP-18124-NP-A Revision 0 Supplement 1-P and WCAP-18124-NP-A Revision 0 Supplement 1-NP, “Fluence Determination with RAPTOR-M3G and FERRET – Supplement for Extended Beltline Materials,” Revision 0 (Proprietary/Non-Proprietary) (ADAMS Accession No. ML20344A386)
5. USNRC Report NUREG-2192, “Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants,” July 2017. (ADAMS Accession Number ML17188A158)

6. RSICC Data Library Collection DLC-185, “BUGLE-96, Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications,” Radiation Shielding Information Computational Center, Oak Ridge National Laboratory (ORNL), July 1999.

7. USNRC Report NUREG-0933, “Resolution of Generic Safety Issues (Formerly entitled “A Prioritization of Generic Safety Issues),” Main Report with Supplements 1–34. Generic Issue No: 15, “Radiation Effects on Reactor Vessel Supports (Rev. 3).”

8. USNRC Report NUREG-1509, “Radiation Effects on Reactor Pressure Vessel Supports,” May 1996. (ADAMS Accession No. ML20112B249)

9. Westinghouse Letter LTR-SDA-21-021-P/NP, Revision 2, “St. Lucie Units 1&2 Subsequent License Renewal: Reactor Pressure Vessel Supports Assessment,” September 2021.

10. Westinghouse Report WCAP-18554-P/NP, Revision 1, “Fracture Mechanics Assessment of Reactor Pressure Vessel Structural Steel Supports for Point Beach Units 1 and 2,” September 2020.

11. Westinghouse Report WCAP-18623-P/NP, Revision 1, “St. Lucie Units 1 & 2 Subsequent License Renewal: Fracture Mechanics Assessment of Reactor Pressure Vessel Structural Steel Supports,” December 2021.

12. Westinghouse Letter CSTLM-RV000-TR-CF-000001, Revision 0, “St. Lucie Units 1&2 Subsequent License Renewal: Reactor Pressure Vessel Supports NRC Breakout Session Question Response,” March 2022.

SLRA Section 3.5.2.2.2.6, page 3.5-35 is revised as follows:

The PSL Units 1 and 2 primary shield walls (PSWs) have the same concrete strength and configuration with the exception of steel reinforcement. With regard to steel reinforcement, the PSL Unit 2 PSW uses higher grade steel resulting in additional margin above that of the PSL Unit 1 PSW. The following description applies to both PSL Units 1 and 2. The reinforced concrete **PSW** ~~primary shield wall (PSW)~~ surrounds the reactor vessel (RV) and is anchored to the lower mass concrete at Elevation 18.0 ft. The arrangement of the reactor building internal concrete is shown in Figure 3.5.2.2-1. The internal structure consists of a 33 ft. thick section of mass concrete which rests atop the 10 ft. thick reactor building base mat. The reactor cavity extends about 21 ft. into the mass concrete section. **Per the concrete design drawings and Appendix 3H of the PSL Unit 1 UFSAR,** ~~A~~**a**bove the mass concrete section, a 7 ft., 3 in. thick concrete primary shield wall surrounds the reactor **up to elevation 36 ft**. **Above the primary shield wall, 6 ft. thick upper cavity walls extend** ~~This wall continues~~ up to the operating deck and form~~s~~ a part of the refueling canal wall. The reactor support beams are embedded into the primary shield wall approximately 5 ft. above the top of the mass concrete. The 7 ft. 3 in. thick PSW and the mass concrete on which it rests (elevation 7.5' to 18'), which is identified as the lower cavity concrete (LCC), surrounds the RV where potential radiation exposure in the concrete is maximum. ~~Both the Unit 1 and Unit 2 PSW/LCC have the same configuration.~~ There is no liner on the PSW/LCC.

The PSL Unit 1 PSW was originally evaluated in four different calculations over approximately 15 years. The first calculation (dated 1970) evaluated the PSW for the loading given in Table 3.8-11 of the PSL Unit 1 UFSAR using a simplified conservative approach. A second calculation (dated 1975), which is described in Section D of Appendix 3H of the PSL Unit 1 UFSAR, used finite element analysis of the PSW under RPV support loads (Table 3H-1 of the PSL Unit 1 UFSAR) and temperature distribution (Figure 3H-15 of the PSL Unit 1 UFSAR). A third calculation (originally initiated in ADAMS Accession Nos. ML18114A219 and ML18108A562 and later finalized in 1979) evaluated the PSW using nonlinear finite element analysis under increased RPV support loads to address NRC concerns regarding asymmetric blowdown loads (i.e., North Anna loads).

The PSL Unit 1 reactor thermal shield was removed in 1984, resulting in increased operating temperatures in the PSW. A fourth calculation (dated 1984), which is documented in Section F of Appendix 3H of the PSL Unit 1 UFSAR, evaluated the PSW under the increased temperature loading (Figure 3H-32 of the PSL Unit 1 UFSAR) and the asymmetric blowdown loads (Figures 3H-34, 3H-35, and 3H-36 of Unit 1 UFSAR). This fourth calculation (dated 1984) superseded all previous PSL Unit 1 calculations and is the analysis of record.

Similarly, the PSL Unit 2 PSW was first evaluated for the loading given in Table 3.8-19 of the PSL Unit 2 UFSAR using a simplified conservative approach. A second calculation evaluated the Unit 2 PSW using finite element analysis. An evaluation specific to the PSL Unit 2 PSW was not performed for the asymmetric blowdown loads due to the fact that (a) asymmetric blowdown loads (i.e., North Anna loads) used for the PSL Unit 1 evaluations are higher than the original PSL Unit 2 RPV support loads and are applicable to the PSL Unit 2 PSW as well, (b) the PSL Unit 2 PSW has higher margin over PSL Unit 1 PSW as discussed previously in this section, and (c) the analysis of record for the PSL Unit 1 PSW is conservatively applicable to the PSL Unit 2 PSW. Note PSL Unit 2 does not have a reactor thermal shield.

In conclusion, the PSL Unit 1 PSW evaluation dated 1984 is the analysis of record for both units. The analysis of record supersedes all information for PSW loads, load combinations, loading conditions, and capacities given in UFSARs for both units. Note that all calculations mentioned above, including the analysis of record, evaluated the portion of the reactor cavity below Elevation 25.5 ft (i.e., PSW and LCC), where the load effects and radiation effects are maximum. The 6 ft thick upper cavity walls are not included in these evaluations.

SLRA Section 3.5.2.2.2.6, Table 3.5.2.2-1, page 3.5-38 is revised as follows:

Table 3.5.2.2-1
PSL Primary Shield Wall/ Lower Cavity Concrete Specifications

Ingredients	Applicable Specification	
	PSL Unit 1	**PSL Unit 2**
Strength	~~5000 psi~~ **- primary shield wall: 5000 psi** **- upper cavity walls 4000 psi** **- lower cavity concrete: mix of 4000 psi and 5000 psi**	~~5000 psi~~ **- primary shield wall: 5000 psi** **- upper cavity walls 4000 psi** **- lower cavity concrete: mix of 4000 psi and 5000 psi**
Cement	ASTM C-150 Type II	ASTM C-150 Type II
Air Entraining Agent	ASTM C-260	ASTM C-260
Water Reducing Agent	ASTM C-494 Type A and D	ASTM C-494 Type A and D
Aggregate	ASTM C-33 Fine aggregate consists of natural and/or manufactured sand Coarse aggregate consists of hard, durable crushed rock or natural gravel	ASTM C-33 Fine aggregate consists of natural and/or manufactured sand Coarse aggregate consists of hard, durable crushed rock or natural gravel
Water to Cement Ratio	0.44 to 0.60	0.42

SLRA Section 3.5.2.2.2.6, page 3.5-39, below Table 3.5.2.2-2 is revised as follows:

Based on these results, the projected end of SPEO gamma doses for PSL Unit 1 and PSL Unit 2 fall below the NUREG-2191 and NUREG-2192 concrete irradiation threshold for gamma radiation (1.0 x 10^{10} rads)**, even when a 20% uncertainty is applied (i.e., 1.2 x 6.62 x 10^9 = 7.94 x 10^9)**. Accordingly, no further evaluation of the PSL Units 1 and 2 PSW/LCCs for gamma irradiation effects is required.

Neutron fluence attenuation and radiological effects on the PSW/LCCs were determined utilizing industry guidance provided in EPRI report 3002002676 (Reference 3.5.4.4), PNNL 15870 entitled "Compendium of Material Composition Data for Radiation transport Modeling" (Reference 3.5.4.5), and EPRI Report 3002011710 (Reference 3.5.4.6).

Reference 3.5.4.6 uses attenuation ratio to determine the point into the concrete where the fluence will reach the NUREG-2192 neutron fluence damage threshold of 1 x 10^{19} n/cm^2. The attenuation ratio is defined as (threshold fluence)/(incident fluence at the surface of the concrete). The information in Reference 3.5.4.6 is representative of the PSL concrete in relation to 2 loop PWR fluence model, use of Portland cement, and use of sand and crushed rock aggregate. Using the higher Unit 2 value for neutron fluence calculated by Westinghouse **and accounting for 20% uncertainty**, the attenuation ratio to reach the neutron fluence damage threshold in NUREG-2192 would be **1/(1.2 x 1.42) = 0.59**= ~~1/1.42 = 0.70~~. Using the formula for the attenuation curve (Reference 3.5.4.6), neutron fluence would reach the damage threshold at 0.8 in. into the PSW/LCC**, and at 1.14 in. considering 20% uncertainty at the core mid-plane**. Based on this fluence and using a realistic concrete strain at ultimate strength from ACI 318-69 of 0.003, the radiation induced volumetric expansion depth is 0.

The specifications for aggregates for the PSW concrete as presented in SLRA Table 3.5.2.2-1 do not identify the origin of the aggregates. Even if the aggregates were to contain materials such as quartz or silicates, the conclusions regarding minimal radiation induced volumetric expansion (RIVE) effects when using a realistic concrete strain at ultimate strength from ACI 318-69 as presented above would be the same. Additionally, the location of the RPV support horizontal beam anchorages are sufficiently far away from the location of maximum neutron fluence (core mid-plane) that exposures would be below the NUREG-2191 and NUREG-2192 concrete irradiation damage threshold even when applying uncertainty to the 72 EFPY neutron fluence values.

The analysis of record, which is described in Section F of Appendix 3H of the PSL Unit 1 UFSAR and applicable to PSL Units 1 and 2 PSWs, concludes that the governing failure mode is the tensile failure of the vertical rebars at the inner face of the PSW/LCC with **an IR of 0.77**. Based on the irradiation effects summarized above, and the original analysis of the PSW/LCC under CLB loading conditions for both PSL Units 1 and 2, there will be minimal effect on the IR associated with the governing failure mode of 0.77. **The irradiated PSW IR of 0.77 is based on the analysis of record for the PSL Unit 1 PSW, which has smaller margin than the PSL Unit 2 PSW, and 72 EFPY exposures for the PSL Unit 2 PSW, which are larger than PSL Unit 1**

PSW exposures. Hence the IR of 0.77 is conservatively applicable for both units. The analysis of record indicates cracking of PSW concrete (Figures 3H-34 to 3H-42) under CLB loads. The long-term exposure levels are low based on their location away from the core mid-plane and will have no impact on these cracking patterns. Note that ~~this~~ **the** IR **of 0.77** is based on **conservative LOCA loads due to** a guillotine break of the main primary loop piping **and these large LOCA loads are by far the main contributor to the IR among other CLB loads.** ~~, thus the actual IR will be much lower considering b~~**B**oth PSL Units 1 and 2 have implemented leak before break **(LBB)** of the primary loop piping as part of their CLBs. **The LOCA loads based on a branch line pipe break (BLPB) (CSTLM-RCS-TM-CS-000001, Rev. 0) after implementation of LBB are smaller than 10% of the conservative LOCA loads used in the analysis of record. Thus, the actual IR would be much lower and the PSW concrete cracking shown in the analysis of record would not occur if BLPB loads were used instead of large break LOCA loads.**

SLRA Section 3.5.2.2.2.6, page 3.5-40, is revised as follows:

Conservatisms in the above evaluation were as follows:

- Exposures were based on 72 EFPY which is more than best estimate EFPY based on a 95% capacity factor of ~69 EFPY.
- Future projections included a 10% positive bias on the peripheral and re-entrant corner assemblies on the projection fuel cycle **and an additional 20% uncertainty was considered as discussed above**.
- Irradiation effects were assumed to apply to the entire vertical surface of the PSW/LCC corresponding to the active fuel region, whereas actual fluence and gamma dose would be much less at the top and bottom regions of the fuel.

There are three AMPs that periodically inspect components in the PSL Units 1 and 2 reactor cavity areas, including the RPV supports, their anchorages and the interior surfaces of the PSWs. These three AMPs are the Boric Acid Corrosion (Section B.2.3.4), Structures Monitoring (Section B.2.3.33) and ASME Section XI Inservice Inspection, Subsection IWF (Section B.2.3.30) AMPs. Inspections of the reactor cavity areas for PSL Units 1 and 2 are performed every refueling outage as part of the Boric Acid Corrosion AMP. These inspections include the PSW surfaces and the reactor cavity floor areas, including the interfaces between the RPV supports and the concrete, and provide direct input into the Structures Monitoring AMP. The condition of accessible surfaces is recorded using high-definition digital photography equipment (to meet ALARA stay time limits). Although the required inspection frequency for the Structures Monitoring AMP for each unit is every five years, the inspection report for that period will include the results of at least three inspections of the reactor cavity area from the Boric Acid Corrosion AMP.

For the RPV supports, ASME Section XI IWF, Category F-A, Item 1.40 describes the requirements for supports other than piping supports. Also, note (3) states, "For multiple components other than piping, within a system of similar design, function, and service, the supports of only one of the multiple components are required to be examined". The PSL Units 1 and 2 reactor vessels are each treated as one vessel with three separate RPV supports. Although only one support is required to be examined, the St. Lucie Inservice Inspection (ISI) Program Plan for PSL Units 1 and 2 schedules all three of the RPV supports on each unit for examination each ten-year ISI interval due to the environment and criticality of the supports. Based on operating experience to date, there has been no indication of cracking, chipping or spalling of the concrete in the reactor cavities of either unit. Additional OE related to the RPV supports is presented in SLRA Section 3.5.2.2.2.7.

SLRA Section B.2.3.33, page B-251, is revised as follows:

- The evaluation of results was performed in accordance with the program's implementing procedures, based on a review of the results of the latest program walkdowns.

PSL Structures Monitoring health reports are developed and trended. Program health reports from 2015 to present indicate that inspections, procedures, and plans meet the program requirements. The overall performance of the PSL Structures Monitoring AMP is WHITE. The path to GREEN involves reducing a backlog of mostly white and some yellow work orders.

The following review of site-specific OE demonstrates how PSL is managing aging effects associated with the PSL Structures Monitoring AMP.

- Corroded pipe support spring cans were observed during a Structures Monitoring Program walkdown for the Unit 2 Refueling Water Tank. Corrosion had been identified on the two spring supports, as documented in previous ARs. The evaluation noted that no additional degradation had occurred, and the spring cans were replaced.
- Observations during Structures Monitoring walkdowns of the Unit 1 and Unit 2 Reactor Containment Buildings included localized corroded embedded steel, abandoned corroded anchors, pop-outs, degraded weatherproofing (concrete joint sealant), and spalling. As part of each AR, an evaluation was performed on the extent of conditions, and the evaluations concluded that in each case the structural or functional integrity of the structure was not impacted. The ARs were closed to the work management process for implementing the necessary repairs.
- Observations during walkdowns of the Unit 1 and Unit 2 Component Cooling Water Buildings included localized concrete cracks, spalls, honeycomb and rust bleeding, degraded elastomer seals for concrete joints, corroded embedded steel in concrete, abandoned anchors in concrete, steel locations with corrosion and degraded coatings, and corroded supporting steel, handrail, and bolted connections on stairs. The conditions were evaluated and determined to be minor, and the structural and functional integrity of the structures were not impacted. The ARs were closed to the work management process for implementing the necessary repairs.
- Corrosion greater than 1/32 inches was observed on lighting conduit on the exterior east wall of the Unit 1 reactor auxiliary building. An evaluation determined that no immediate repairs were required, and that the condition did not impact the structural component's function, integrity, or ability to withstand a design basis event. The AR was closed to the work management process for implementing the necessary repairs.

With regard to the topic of EMBECO 636 grout, there is documentation that EMBECO 636 grout has been used at PSL. Based on review of plant drawings, the specific type of grout used below the reactor vessel support baseplates cannot be confirmed. The letter to NRC dated May 7, 2001 from the supplier of EMBECO grout (ADAMS Accession No. ML011310474) identified an issue with

the separation of aggregate and cement during grout placement. This condition would have been identified during placement and testing of the EMBECO 636. Additionally, OE reviews performed for PSL Units 1 and 2 have concluded that there has been no site-specific OE related to the use of EMBECO 636 including no aggregate/cement separation at the RPV support baseplates. Furthermore, review for the use of EMBECO 636 Plus has been performed and concluded that the material has not been used at PSL. The reactor vessel supports are designed with grouted base plates. Even if EMBECO 636 grout were used on the reactor vessel supports, the baseplate is at a low fluence region that is well below the fluence threshold in NUREG-2191 and NUREG-2192. Inspections for the grout beneath the reactor vessel supports are part of the Structures Monitoring Program.

SLRA Further Evaluation Section 3.5.2.2.2.7, Loss of Fracture Toughness Due to Irradiation Embrittlement of Reactor Pressure Vessel (RPV) Supports

Affected SLRA Section: 3.5.2.2.2.7

SLRA Page Numbers: 3.5-44, 3.5-45, 3.5-46,

Description of Change:

The following discussion provides clarification regarding loading conditions considered for the PSL Units 1 and 2 RPV support evaluation. Sections 3.8.3.3.2.3 of the PSL Unit 1 UFSAR and Table 3.8-12 of the PSL Unit 2 UFSAR present general loading conditions which apply to containment steel structures. Each specific steel structure has design requirements and/or specifications associated with it that give the specific loading each structure must be qualified to resist. Since all loading conditions and types are not applicable to all steel structures, the specifications for the individual structures and components ultimately list the relevant loading conditions that the design must withstand. For the RPV supports, the loading conditions applicable to the structure are contained in the design specifications (Reference 12). These applicable loading conditions are reflected in the legacy qualification analyses of the supports, and the same loading conditions are considered in the SLR analysis for PSL Units 1 and 2. The methodology employed with the given loads is to apply static loading for all conditions, including dynamic conditions such as earthquakes. Applying bounding static loads in all applicable directions for each load is a conservative method for stress analysis.

For the RPV supports, ASME Section XI IWF, Category F-A, Item 1.40 describes the requirements for supports other than piping supports. Also, note (3) states, "For multiple components other than piping, within a system of similar design, function, and service, the supports of only one of the multiple components are required to be examined". The PSL Units 1 and 2 reactor vessels are each treated as one vessel with three separate RPV supports. Although only one support is required to be examined, the St. Lucie Inservice Inspection (ISI) Program Plan for PSL Units 1 and 2 schedules all three of the RPV supports on each unit for examination each ten-year ISI interval due to the environment and criticality of the supports. As noted in SLRA Section 3.5.2.2.2.7, page 3.5-46, inspections performed to date on both units have not identified any areas requiring further evaluation.

The following discussion provides additional information regarding the PSL RPV sliding plates, their lubricant and the potential for stress corrosion cracking (SCC). Per PSL drawing 8770-369 (Reference 13), the socket (concave plate) is made of ASTM A-283-67 structural steel plates. According to the PSL Units 1 and 2 Containment Systems Design Basis Document (DBD) (References 5, 6), the slide (convex plate) and expansion plates were originally designed with ASTM B-22 Alloy E, a manganese bronze alloy, lubricated with Lubrite type AE-2 lubricant. Based on test data, the maximum temperature at the base of the reactor vessel lubrication plate is limited to 300°F for the design friction coefficient of 0.15. During the plant construction stage, a new lubrication plate material, Meehanite (a cast iron alloy with controlled graphite dispersion) was tested and approved for use at other Combustion Engineering designed plants. The Meehanite material would allow operation temperature up to 600°F. However, PSL Units 1 and 2 decided to keep the original manganese bronze alloy material with Lubrite lubricant since the Meehanite material requires additional analyses and replacing components that had already been provided for the plants. The 300°F limit noted above is consistent with the SLRA Section 2.3.3.12, Page 2.3-57 discussion that the reactor support cooling system limits the temperature at the

bottom of the lubrication plate between the reactor and support leg to 300°F for the Lubrite. The 300°F limit is also consistent with CN-SEE-II-09-39 (Reference 15) which determined that the base of lubrication plate remains below 300°F at EPU conditions. Therefore, the slide and expansion plates are confirmed to be ASTM B-22 manganese bronze alloy, and the contact surfaces are lubricated with Lubrite type AE-2 (References 5, 6) as shown on PSL drawing 8770-369 (Reference 13).

As discussed in Section 3.3.1.1 of the NRC Final Safety Evaluation Report for WCAP-14422, Rev. 2 (Reference 7), the three key factors for SCC to occur are the use of high strength material, moist environment, and a high level of sustained tensile stress. In the absence of any one of these factors, SCC is unlikely to occur. The support shoe (socket/slide assembly) is located in a dry environment. The lubricated surfaces of the socket and slide, as well as the expansion plates on the side of the support shoe are in compression. There is no condition that would result in high level of sustained tensile stress in these components. Therefore, SCC is unlikely to occur. Page 3.5-45 of the SLRA is revised accordingly.

The following discussion provides clarification regarding whether the SLRA evaluation methodology for the embrittlement of the RPV supports accounted for all of the re-evaluation criteria of NUREG-1509. The qualitative assessment in LTR-SDA-21-021 (Reference 2) used estimated PSL faulted loads to calculate finite element stresses and the fracture toughness values were calculated without analytical uncertainties associated with the methodology used to calculate embrittlement. The normalized comparison ratios for PSL vs. Point Beach (PBN) RPV supports components in Table 7-1 and Table 7-2 of LTR-SDA-21-021 (Table 3.5.2.2-5 of the SLRA) are all greater than 1 and concluded that the PSL RPV support components have critical flaw sizes greater than the limiting PBN components reported in WCAP-18554 (Reference 3). A detailed, PSL plant-specific fracture mechanics evaluation for the RPV supports was performed and documented in WCAP-18623 (Reference 4) which included calculation of plant-specific critical flaw sizes with PSL loading conditions and analytical uncertainties for embrittlement. The evaluations in WCAP-18623 follow the general guidance of ASME Section XI to investigate brittle fracture of the structural steel supports per NUREG-1509 (Reference 11). In addition, the assumptions and conservatism in the PSL RPV steel support evaluation are provided in Section 7 of WCAP-18623 (Reference 4).

Table 1 and Table 2 below present the updated ratio of critical flaw sizes, in terms of flaw depth (*a)* ratios (in/in) between PSL and PBN based on the results from WCAP-18623. Note that critical flaw size ratios in Table 1 and Table 2 are lower than the normalized ratios predicted in the qualitative assessment (Reference 2). One conservatism in the detailed critical flaw size evaluation, WCAP-18623 (Reference 4), is that an additional +25% iron dpa was used in calculating embrittled fracture toughness to address analytical uncertainties associated with the methodology used to calculate embrittlement. The qualitative assessment does not include the additional +25% iron dpa. Nevertheless, the updated critical flaw size ratios considering the PSL plant-specific detailed fracture mechanics evaluations remained greater than 1 (see Table 1 and Table 2). Thus, the qualitative assessment conclusion in Reference 1 remains valid. PSL RPV support critical flaw sizes for the plates are larger than PBN. As predicted in LTR-SDA-21-021 (Reference 2), the critical flaw sizes calculated in WCAP-18623 (Reference 4) for all PSL bolts have large margins compared to the ASME Section XI (Reference 16) allowable flaw sizes and are bounded by PBN bolt results.

Table 1: PSL vs. PBN Ratio Comparison for Top Horizontal Support Plates

	PSL Unit 1		PSL Unit 2	
Plate Material	A-441 Plate	A-533 Cl.2 Gr. B Plate	A-441 Plate	A-533 Cl.2 Gr. B Plate
Normalized Ratio in Table 7-1 of Ref. 2	4.34	2.56	7.09	7.3
Critical flaw size ratio in terms of flaw depth [in/in] PSL(1) / PBN	2.53	1.93	4.07	5.18

Note:

(1) PSL critical flaw size is calculated using fracture toughness which considered +25% iron dpa to account for analytical uncertainties associated with the methodology used to calculate embrittlement.

Table 2: PSL vs. PBN Ratio Comparison for Bottom Horizontal Support Plates

	PSL Unit 1		PSL Unit 2	
Plate Material	A-441 Plate	A-533 Cl.2 Gr. B Plate	A-441 Plate	A-533 Cl.2 Gr. B Plate
Normalized Ratio in Table 7-2 of Ref. 2	4.07	2.52	4.31	2.58
Critical flaw size ratio in terms of flaw depth [in/in] PSL(1) / PBN	2.11 (2)		2.11 (2)	

Notes:

(1) PSL critical flaw size is calculated using fracture toughness which considered +25% iron dpa to account for analytical uncertainties associated with the methodology used to calculate embrittlement.

(2) LTR-SDA-21-021 (Reference 2) assumed the postulated flaws were located in the 3” thick A-441 plate and the 5” thick A-533 Cl.2 Gr. B plate. However, the detailed fracture mechanics evaluation in WCAP-18623 determined that the 4” thick A-441 plate is limiting for the bottom horizontal support plates and is reported in this table.

Page 3.5-46 of the SLRA is revised accordingly.

The following points address the conservatisms present in the structural model that produced stresses used for PSL Units 1 and 2.

1. All supports for both units were represented by one model. This was accomplished by taking the “worst case” geometry from each support and incorporating it into one model.

Where the supports may have slightly different member lengths, or slightly different geometric details to accommodate local interfaces, these details were incorporated to create a bounding conservative geometry. The result was a final geometric model that had the least desirable features with regard to stress output. No single support contains all of the features represented in the model, thus the stress analysis over-estimates the stress on every support.

2. All loads are applied statically, and in a direction intended to increase the resultant stress. Furthermore, the largest magnitude load for each direction for each support was applied simultaneously in one load case. Thus, the final loads applied in each load case consider a combination of loads that is not postulated to exist at any single support, since the highest loads from each support are used, regardless of whether they occur at the same support.

3. Use of static loading, in general, is a conservative method for stress analysis, particularly for dynamic loads such as earthquake transients and pipe breaks. These two transients last for different lengths of time and are driven by fundamentally different resonant frequencies in the base structure. The peak loading from these transients does not occur at the same time, nor does the structure respond to the dynamic input the same way for each transient. Using the peak loading over each transient as a static load, all combined together to result in a final static load, is a conservative approach.

4. Often, earthquake loads and pipe break loads are combined via the square-root-sum-of-the-squares method. This is, in part, a recognition of the time-phase difference between the two transients. However, this method was not employed in this analysis; the earthquake and pipe break loads are added together as an absolute sum.

Regarding the fixity of the metal to the concrete in the model, where the metal enters the concrete as embedded steel, the boundary condition is not totally fixed at the interface. The interface between the embedded metal and the concrete is considered a frictionless support in the finite element model. Practically, this means that the embedded metal plates cannot displace perpendicular to their faces (i.e., the plates cannot displace into the adjacent volume where the concrete would be), but the plates are able to displace axially (i.e., the metal is free to stretch longitudinally, or away from the concrete). This allows for minor rotations of the material at the concrete boundary that would not be possible in a fully fixed condition.

For concrete degradation to be a concern, large areas of concrete would have to be compromised to alter the boundary condition to a point where the structural analysis performed for this project was inaccurate or unconservative. Review of Appendix H of the PSL Unit 1 UFSAR indicates the following with regards to concrete degradation:

1. Concrete cracking was postulated so that the impact of such degradation on the stiffness of the support structure and the distribution of loading among the concrete volume could be adequately assessed. The use of postulated concrete cracking was not implemented as part of an investigation of reactor vessel support steel stresses.

2. PSL Unit 1 UFSAR Figures 3H-34, 3H-35, and 3H-36 show the postulated cracks that were developed in the concrete in the area of the steel support. These cracks are not present at the surface of the concrete, at the interface between the support and the

steel, and are not present to such an extent that would fundamentally alter the stress distribution in the steel portions of the support.

3. Additional information regarding the lower concrete IR and limited cracking when considering reduced branch line pipe break loads is provided in Attachment 1 to this supplement.

The following provides clarification regarding the critical flaw comparative approach. As discussed above, Westinghouse performed a detailed fracture mechanics evaluation for PSL Units 1 and 2 in WCAP-18623 (Reference 4), which consisted of calculating critical flaw sizes. The calculated critical flaw size ratios for PSL and PBN in Table 1 and Table 2 above concluded that the qualitative assessment conclusion in Reference 2 remains valid. PSL RPV supports critical flaw sizes are greater than PBN.

The following discusses the potential for crack growth with regard to cyclic loading. Cyclic loading, as commonly understood for stress analysis, is not applicable to the support steel per the support specifications (Reference 12) and is not mentioned in the PSL UFSAR sections referenced above. Steady state perturbations in reactor vessel temperature and pressure are generally not considered cyclic loadings with regards to the steel supports. Furthermore, any applied loads to the supports that could potentially be considered cyclic (e.g., the thermal loading due to heat up and cool down of the vessel) do not occur with a frequency over the postulated life of the plant, even considering life extension, that would be of concern for causing fatigue induced cracking.

The following discusses the magnetic particle examination (MT) of the RPV support shoe and the evaluation of the sliding plate lubricant. The MT performed was on the hot leg reactor support foot which is above the socket/slide shoe assembly.

As discussed above, the PSL Units 1 and 2 RPV support socket and slide assembly are lubricated with Lubrite. The qualitative assessment in LTR-SDA-21-021 (Reference 2) performed a sensitivity study assuming a complete loss of function for the lubricant. The structural effect of the lubricant in the detailed fracture mechanics evaluation in WCAP-18623 (Reference 4) was conservatively addressed by increasing the lateral friction loads assuming complete loss of function of the lubricant. Additionally, there is no concern of SCC occurring due to the dry environment and lack of high levels of sustained tensile stress in the lubricated components.

The MT of the hot leg reactor support foot does not affect the RPV slide plate lubricant degradation. The MT area is entirely above the slide plate.

NASA studied the dynamic friction and wear of a solid film lubricant during exposure in a nuclear reactor (Reference 8). The film lubricant consisted of molybdenum disulfide and graphite in a sodium silicate binder. Radiation levels of fast neutrons ($E \geq 1MeV$) were fluxes up to 3.5×10^{12} n/cm^2-sec (intensity) and fluences up to 2×10^{18} n/cm^2 (total exposure). The amount of total exposure (fluence) was found to not affect lubrication behavior as severely as the radiation intensity (flux) during sliding. Wear life was severely reduced at high flux (above 3×10^{12} n/cm^2-sec, $E \geq 1MeV$) and slightly increased at low flux (below 3×10^{11} n/cm^2-sec, $E \geq 1MeV$). Friction coefficients for low level flux (2×10^{11} to 4×10^{11} n/cm^2-sec) were within the range of unirradiated coatings. The PSL average neutron flux is approximately 9.3×10^7 n/cm^2-sec, far lower than the NASA study findings. Furthermore, the detailed fracture

mechanics evaluation conservatively considered loss of lubricant function with increased friction load. Page 3.5-44 of the SLRA is revised accordingly.

The following clarifies the RPV support weld inspections. Per design specifications (Reference 9), all full penetration tee weld root pass and final weld layers were magnetic particle tested (MT) or liquid penetrant tested (PT). Ultrasonic tests (UT) were performed for completed welds. This is consistent with PSL drawing 2998-5378 (Reference 14) which indicates that all full penetration welds are MT and UT. Additionally, all welds were "*carefully examined to ensure that there are no slag inclusions, craters, cracks or undercuts. Defects shall be removed by chipping or grinding and then rewelded.*"

The cutlines illustrated in Figure 6-4 of LTR-SDA-21-021 (Reference 2) correspond to PSL drawing 8770-G-794, Sheet 2 (Reference 10.a), Rev. 5, Zone F-12 and PSL drawing 2998-G-794, Sheet 2 (Reference 10.b), Rev.3, Zone B-12. Although the welding symbols on the Unit 2 design drawing corresponding to the cutline locations in Figure 6-4 of LTR-SDA-21-021 are unclear, the corresponding weld symbols on the Unit 1 design drawing and the manufacturer's construction drawing for PSL Unit 2, 2998-5378, suggest the welds in question are full penetration welds. Furthermore, from a structural design and welding practicality point of view, typical good practice is to use full penetration welds for connecting thick (3 to 6-inch) plates. Therefore, the welds in question would receive the NDE examinations as described above.

The MT, PT and UT specified in Reference 9 are pre-service examinations.

The design specifications for PSL RPV steel supports that specify the examinations are listed in Reference 9.

References:

1. St. Lucie, Units 1 and 2, Application for Subsequent Renewed Facility Operating Licenses, Rev. 1, NRC ADAMS Accession Number ML21285A307, October 12, 2021.
2. Westinghouse Letter, "St. Lucie Units 1&2 Subsequent License Renewal: Reactor Pressure Vessel Supports Assessment," September 2021.
    a. LTR-SDA-21-021-P, Rev. 2
    b. LTR-SDA-21-021-NP, Rev. 2
3. Westinghouse Report, "Fracture Mechanics Assessment of Reactor Pressure Vessel Structural Steel Supports for Point Beach Units 1 and2," September 2020.
    a. WCAP-18554-P, Rev. 1
    b. WCAP-18554-NP, Rev. 1
4. Westinghouse Report, "St. Lucie Units 1 & 2 Subsequent License Renewal: Fracture Mechanics Assessment of Reactor Pressure Vessel Structural Steel Supports," December 2021.
    a. WCAP-18623-P, Rev. 1
    b. WCAP-18623-NP, Rev. 1
5. St. Lucie Unit 1 Containment Systems Design Basis Document, DBD-CNTMT-1, Rev. 7, pages 31-32.
6. St. Lucie Unit 2 Containment Systems Design Basis Document, DBD-CNTMT-2, Rev. 7, pages 31-32.

7. WCAP-14422, Rev. 2-A, "Licensing Renewal Evaluation: Aging Management for Reactor Coolant System Supports," December 2000.
8. NASA Technical Note, NASA TN D-6940, "Dynamic Friction and Wear of a Solid Film Lubricant During Radiation Exposure in a Nuclear Reactor," September 1972.
9. EBASCO Structural Steel Specifications Electronically Attached to PSLWEC-21-0066, Rev. 0, "Design Input Transmittal for WS06b-c RPV Supports to Support the St. Lucie Unit 1 and Unit 2 Subsequent License Renewal Rev. 1," June 15, 2021.
    a. FLO-8770-761, Rev. 2, "Structural Steel - Saint Lucie Unit No. 1," December 18, 1970.
    b. FLO-8770-761, Rev. 2, Addendum, "Addendum to EBASCO FLO-8770-761 For Class I Structures and Components," January 7, 1971.
    c. FLO-2998-761, Rev. 6, "Structural Steel - Saint Lucie Unit No. 2," February 18, 1987.
10. EBASCO Drawings Electronically Attached to PSLWEC-21-0066, Rev. 0, "Design Input Transmittal for WS06b-c RPV Supports to Support the St. Lucie Unit 1 and Unit 2 Subsequent License Renewal Rev. 1," June 15, 2021.
    a. 8770-G-794, Sheet 2, Rev. 5, "St Lucie Plant Unit 1 Reactor Building Equipment Supports."
    b. 2998-G-794, Sheet 2, Rev. 3, "St Lucie Plant Unit 2 Reactor Building Equipment Supports."
11. U.S. Nuclear Regulatory Commission, NUREG-1509, "Radiation Effects on Reactor Pressure Vessel Supports," May 1996.
12. Combustion Engineering Specifications:
    a. E-13172-220-001, Rev. 3, "Reactor Coolant System Support Design Loads."
    b. E-19367-220-001, Rev. 2, "Reactor Coolant System Support Design Loads."
    c. 19367-31-1, Rev. 5, "Engineering Specification for a Reactor Vessel Assembly for Florida Power and Light Co. Hutchinson Island Plant Unit No. 1 1973 – 890 MW Installation Contract No. 19367."
    d. 13172-31-1, Rev. 6, "Project Specification for a Reactor Vessel Assembly for St. Lucie Unit 2."
13. 8770-369, Rev. 0, "Reactor Vessel Support Details" (a.k.a. Combustion Engineering drawing, E-STD-220-002, Rev. 01)
14. 2998-5378, Rev. 2, "Reactor Supports."
15. CN-SEE-II-09-39, Rev. 0, "Thermal Analysis of St. Lucie Unit 1 Reactor Cavity under EPU Conditions," February 15, 2010.
16. American Society of Mechanical Engineers (ASME) Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2007 Edition with 2008 Addenda.

SLRA Section 3.5.2.2.2.7, page 3.5-44, first paragraph is revised as follows:

> Westinghouse performed a qualitative assessment of the PSL Units 1 and 2 structural steel RPV supports in References 3.5.4.7 and 3.5.4.8. A comparison of the key inputs to ASME Section XI critical flaw size calculations was made between PSL and Point Beach Nuclear (PBN) in order to ascertain the acceptability of the PSL RPV Supports for the subsequent period of extended operation (SPEO) with consideration of irradiation aging effects. These key inputs consist of the fracture toughness and stresses of the RPV support components and were combined into a comparative ratio term based on the general form of stress intensity factor. This comparative ratio effectively normalizes the fracture toughness and stress relative to PBN as it pertains to the calculation of critical flaw sizes. A normalized ratio greater than 1 indicates the PSL critical flaw size would be larger than that of PBN and therefore, the conclusions contained within the detailed PBN fracture mechanics evaluation (References 3.5.4.9 and ML21111A155) can be applied to PSL. **Furthermore, the ratios of RPV supports critical flaw sizes calculated in WCAP-18623-P/NP for PSL and WCAP-18554-P/NP for PBN are also greater than one, confirming the qualitative assessment conclusions.**

SLRA Section 3.5.2.2.2.7, page 3.5-45 is revised as follows:

The slide and expansion plates of the RPV support shoe assembly are made of ASTM B-22 Alloy E, a manganese bronze alloy, and the contact surfaces were lubricated with Lubrite type AE-2. Westinghouse assessed the RPV slide plate lubricant for degraded conditions such as a decrease in viscosity due to radiation effects. Neutron flux was identified as the key parameter in irradiation aging effects of dry film lubricants such as that employed at PSL. The average flux at the RPV slide plate for 72 EFPY was calculated and shown to be below the **flux level in a NASA study** ~~threshold~~ where degradation is observed. Therefore, the functionality of the lubricant is not adversely affected by the radiation exposure in the SPEO. **In addition, the reactor support cooling system limits the temperatures in the vicinity of the support shoe assemblies to less than 300°F. Furthermore, the Westinghouse evaluation of the RPV supports conservatively increased the friction load, assuming the lubricant is not providing a lubrication function and the surfaces have metal to metal contact. Stress corrosion cracking (SCC) is unlikely to occur in the lubricated plates due to the dry environment and lack of high level of sustained tensile stress in the plates.**

SLRA Section 3.5.2.2.2.7, page 3.5-46 is revised as follows:

Table 3.5.2.2-5
Normalized Ratio Comparison for Most Limiting ~~Faulted Loads~~ RPV Supports Components

		PSL Unit 1		PSL Unit 2	
Plate Material	**Comparison Method (1)**	**A-441**	**A-533 Cl. 2 Gr. B**	**A-441**	**A-533 Cl. 2 Gr. B**
Top Horizontal Support Plate	**Qualitative Assessment**	4.34	2.56	7.09	7.30
	Fracture Mechanics Evaluation	**2.53**	**1.93**	**4.07**	**5.18**
Bottom Horizontal Support Plate	**Qualitative Assessment**	4.07	2.52	4.31	2.58
	Fracture Mechanics Evaluation	**2.11 (2)**		**2.11 (2)**	

Notes:

(1) Qualitative Assessment, LTR-SDA-21-021-P normalized ratios are based on stresses and fracture toughnesses between PSL and PBN. The fracture mechanics evaluation ratios are based on the critical flaw sizes calculated in WCAP-18623-P for PSL and WCAP-18554-P for PBN, which considered all conditions including faulted. The PSL fracture mechanics evaluation considered +25% iron dpa to address analytical uncertainties associated with the methodology used to calculate embrittlement. The PBN critical flaw sizes does not include the +25% iron dpa in the embrittlement calculation.

(2) LTR-SDA-21-021-P assumed the postulated flaws were located in the 3” thick A-441 plate and the 5” thick A-533 Cl.2 Gr. B plate. However, the detailed fracture mechanics evaluation in WCAP-18623-P determined that the 4” thick A-441 plate is limiting for the bottom horizontal support plates and is reported in this table.

Summary of Results

The **qualitative** comparative ratios calculated for the PSL RPV supports are greater than one indicating that the projected critical flaw sizes for the PSL supports would be greater than those for PBN. See Table 3.5.2.2-5 for a comparative ratio of the most limiting regions for the most limiting branch line pipe break (BLPB) as concluded in **the qualitative assessment,** LTR-SDA-21-021-P (Reference 3.5.4.8). **Furthermore, the ratios of RPV supports critical flaw sizes calculated in WCAP-18623-P/NP for PSL and WCAP-18554-P/NP for PBN are also greater than one, confirming the qualitative assessment conclusions.** Additionally, OE shows that VT-3 inspections were performed in accordance with ASME Section XI, and results have met the acceptance criteria of Subsection IWF-3410. Therefore, the conclusions in WCAP-18554**-P/NP** can be conservatively applied to PSL. Based on the discussions above the RPV supports at PSL Units 1 and 2 are structurally stable (i.e., flaw tolerant) considering 80 calendar years (72 EFPY) of radiation embrittlement effects. Additionally, there is sufficient level of flaw tolerance demonstrated to justify continuing the current visual examinations (VT-3) of the RPV structural steel supports as part of the PSL ASME Section XI, Subsection IWF Inservice Inspection Program. Based on these results, a plant specific AMP or enhancements to an existing AMP are not required to manage loss of fracture toughness due to irradiation embrittlement of the RPV supports at PSL.