**St. Lucie SLRA: Breakout Questions**
SLRA FE Sections 3.5.2.2.2.6 and 3.5.2.2.2.7:
TRP: 76

Note: Breakout Questions are provided to the applicant and will be incorporated into the publicly-available audit report.

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| **Technical Reviewer** | A Prinaris<br>D Dijamco<br>J Dean | 01/05/22<br>01/06/22<br>01/21/22 |
| **Technical Branch Chief** | J Colaccino<br>A Buford<br>R Lukes | Concurrence Date: 01/13/22<br>Concurrence Date: 01/10/22<br>01/21/22 (through Jeremy Dean) |
| **Breakout Sessions** | *Date/Time* | *To be filled in by PM* |

| **Applicant Staff** | **NRC staff** |
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| *To be filled out by PM during breakout* | |

**76a: SLRA Section 3.5.2.2.2.6**

| Question Number | SLRA Section | SLRA Page | Background/Issue (As applicable/needed) | Discussion Question/Request | Outcome of Discussion |
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| 1<br>AP | 3.5.2.2.2.6 | 3.5-35<br>3.5-39 | The SLRA states that "[t]he 7 ft. 3 in. thick [Primary Shield Wall] 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 RPV where potential radiation damage in the concrete is maximum. Both the Unit 1 and Unit 2 PSW/LCC have the same configuration." Table 3.5.2.2-1, "PSL Primary Shield Wall/Lower Cavity | By sharing a screen, review the aforementioned As Built Drawings (as well as other relevant drawings), discuss, and clarify what are the compressive strengths of reactor cavity concretes. Concrete compressive | |

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| | | | Concrete Specifications," of the SLRA states that the compressive strengths of Units 1 and 2 PSW and LCC concretes are 5,000 psi. Section 6.1 (pg. 39) of ePortal document 8770-16346, Revision 3, "EBASCO Backfit Engineering, St. Lucie Units 1 & 2 Civil Engineering Design Criteria," states that the reactor internal structures' concrete strengths are 5,000 and 4,000 psi. ePortal As Built Drawing 8770-G-518, Revision 3, notes: "Concrete shall be Class 'AA' (4,000 psi)" and indicates the reactor cavity concrete compressive strength to be 5,000 psi from approximate elevations of 18 feet to 36 feet. ePortal As Built Drawings 8770-G-529, Revision 2, and 2998-G-529, Revision 4, notes that concrete shall be class AAA (5,000 psi) and Class AA (4,000 psi) and indicate that only portion of the reactor cavity concrete strength to be 5000 psi.<br><br>It is not clear what are the actual/As Built concrete compressive strengths of the reactor cavity concretes (PSW/LCC). | strengths and associated mechanical properties are reduced when exposed to radiation. Concrete material properties are of importance particularly in areas where RPV support beams frame into reactor cavity concrete wall(s). | |
| 2<br>AP | 3.5.2.2.2.6 | 3.5-35 | The SLRA states that a "7 ft., 3 in. thick concrete primary shield wall surrounds the reactor. […] The 7 ft. 3 in. thick PSW and the mass concrete on which it rests […] surrounds the RPV where potential radiation damage in the concrete is maximum." It also states that "[b]**oth the Unit 1 and Unit 2 PSW/LCC have the same configuration"** (emphasis added).<br><br>Section 3.8.3.4, "Design and Analysis Procedures," of the UFSAR states that the "primary shield wall is a 6 feet thick cylinder and is analyzed as a thick cylinder." Section B.2, "Reactor Support Structure," of Unit 1 UFSAR Appendix 3H and Unit | a) By sharing a screen, show construction drawings of the Unit 1 and Unit 2 PSWs to clarify what is the actual (As Built) thickness of the concrete PSW cylinder at each Unit.<br><br>b) If the As Built PSW cylinders are thicker than that analyzed, particularly in the area where the RPV support system (assembly) beams frame into the | |

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| | | | 2 UFSAR Section 3.8.3.1.1, "Primary Shield Wall," however, confirm that the PSW concrete thickness for both Units is 7 ft., 3 in as stated in the SLRA.<br><br>It is not clear whether the PSW concrete has variable thickness for Unit 1, whether the analysis for both Units were done for a 6 feet thick cylinder but the Units' PSWs were built as 7 ft., 3 in. thickness cylinders because of differing loading conditions. It is also not clear whether a 7 ft., 3 in. thick cylinder has been used for conservatism in the PSL PSW Unit 1 and Unit 2 irradiated concrete safety evaluations (e.g., those in ePortal document NEESL00008-REPT-098 that reference the CLB analysis) although their CLB analysis may have varied. | cavity concrete, discuss the applicability of the 6-feet thick concrete cylinder analysis to that of a 7 feet 3 in. concrete cylinder construct. | |
| 3<br>AP | 3.5.2.2.2.6 | 3.5-39 | Portions of the PSL RPV beam supports are embedded in concrete. PSL in ePortal document NEESL00008-REPT-098, "St. Lucie Units 1 and 2 Subsequent License Renewal Primary Shield Wall Irradiation Evaluation," uses interaction ratios (IRs) to assess the capacity of the reactor cavity concrete to carry imposed loads including those from the embedded RPV support beams. To this end the SLRA states, that the governing failure mode at the reactor cavity wall is tensile failure of the vertical rebars at the inner face of the PSW/LCC. It also states that based on the irradiation effects summarized in the section 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." It then states: | a) Clarify whether the loading factor of 1.15 was considered in the faulted loading condition for the PSW IR evaluation in ePortal document NEESL00008-REPT-098.<br><br>b) Discuss whether loads and loading conditions in UFSAR Unit 1 and 2 Tables 3.8-11 and 3.8-19, respectively have been considered when calculating PSW IR(s) in ePortal document NEESL00008-REPT-098. If not considered, provide | |

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| | | | this IR is based on a guillotine break of the main primary loop piping; thus the actual IR will be much lower considering both PSL Units 1 and 2 have implemented leak-before-break of the primary loop piping as part of their CLBs.<br><br>ePortal document NEESL00008-REPT-098, (hereinafter referred as the basis document) references Specifications 8770-16346, Revision 3, for St Lucie CLB reinforced concrete loading combinations and capacity reduction factors (φ). Section 2.2, “Current Licensing Basis Loading Condition,” of the basis document references Section 4 of Specification 8770-16346, Revision 3 and recounting CLB calculations determines that the “Pipe Break or Accident with Design Basis Earthquake” is the governing loading combination for the reactor cavity concrete subject to exposure. It then references UFSAR Unit 1 Appendix 3H page 20 for the definition of the SLRA listed IR of 0.77. For Unit 1, it is apparent that this IR is calculated based on the tensile stress of a vertical rebar of 27.6 ksi divided by the allowable yield strength of the rebar (strength reduction factor of rebar 0.9 X 40 ksi). However, the relevant loading condition in Appendix 3H of the UFSAR Unit 1 assigned for the definition of 0.77 IR includes a load factor of 15 percent increase for all loads (except for loads associated with temperature effects). Furthermore, when considering Unit 1 UFSAR Table 3.8-11, the governing PSW loading condition and its loading factors differ from that listed in NEESL00008-REPT-098. A similar | an explanation justifying their exclusion.<br><br>c) Discuss the impact of an increase in exposure uncertainty by 20 percent on an IR of 0.77, the governing mode of tensile failure, and the RPV support beam embedment in concrete. | |

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| | | | observation is made for Unit 2 PSW governing loading condition and its loading factors (see Unit 2 UFSAR Table 3.8-19). Furthermore, applied loads and resulting loading capacities for Units 1 and Unit 2 PSWs also differ (see corresponding UFSAR Unit 1 and Unit 2 Tables 3.8-11 and 3.8-19, respectively).<br><br>It is not clear whether the basis document considered all of the above governing loading conditions when discussing the IR of 0.77. If so, it is not clear whether the 0.77 IR considered also the 15 percent loading factor increase discussed above, and whether the 15 percent loading factor was used in the evaluation with the faulted loading condition. Furthermore, it is not clear whether the tensile mode of failure for the IR considered, accounted for effects of radiation (if any). If so, it is not clear what would be the IR value, particularly when radiation exposure uncertainty consistent with conclusions reached at Point Beach is maximized to 20 percent. | | |
| 4<br>AP | 3.5.2.2.2.6 | | ePortal Drawings 8770-G-794 and 2998-G-794 indicate that RPV structural steel assembly column base plates are supported at elevation 2.92 feet to concrete through a 3 inches thick concrete grout. Section 4.0 “Material Specification Data” of ePortal Document PSL-110389-001-M04, “Summary & Results Report for Reactor Head Drop Analysis,” Attachment B (page B5) states that the grout used in column anchorage is EMBECO 636. In a letter dated May 7, 2001(ADAMS Accession No. ML011310474) the supplier of EMBECO 636 notified NRC of a “Possible Nonconformance | a) Identify which type of EMBECO 636 grout has been used on RPV structural system (assembly) supports.<br><br>b) Discuss if there is any OE related to the performance of the EMBECO 636 grout.<br><br>c) Discuss measures to be taken to ensure adequacy | |

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| | | | Related to the Performance of EMBECO 636 (CMTR) Grout" also noted in Event Notification Number 37685. It is not clear whether there is any Operating Experience (OE) at PSL regarding the performance of the EMBECO 636 grout to date.<br><br>In addition, it is noted that the PLUS version (type) of the grout contains metallic and quartz aggregates (see https://marbri.net/embeco-636-plus-grout) which could be affected by boric acid corrosion and radiation effects.<br><br>Since the EMBECO 636 grout is identified as a nuclear safety related product, it is not clear, consistent with the reevaluation criteria of NUREG-1509, whether the environment in the reactor cavity has been examined as to the future performance of the EMBECO 636 grout during the subsequent period of extended operation. | of grout performance during the subsequent period of extended operation, particularly if the grout contains metallic and quartz aggregates and exposed to radiation and corrosive (boric acid) environment. | |
| 5<br>AP | 3.5.2.2.2.6<br>3.5.2.2.2.1 | 3.5-35<br>3.5-28<br>3.5-38 | SLRA Table 3.5.2.2-1, "PSL Primary Shield Wall/Lower Cavity Concrete Specifications," states that concrete mix "[f]ine aggregate consists of natural and/or manufactured sand [and c]oarse aggregate consists of hard, durable crushed rock or natural gravel." Although the SLRA discusses qualification of aggregates (e.g., SLRA Section 3.5.2.2.2.1) consistent with ASTM and ACI standards, it does not identify whether the concrete mix had quartz or limestone aggregates.<br><br>The SLRA also states that the inner surface of the PSW concrete does not have a steel liner. However, construction drawings (e.g., 8770-G-794 SH2, Revision 3 and 2998-G-794 SH2, Revision 5) | a) Clarify the origin of the aggregates and discuss if there is a material certification indicating pit/quarry origin or provide evidence that the aggregates are not quartz, and if they are, discuss that their potential RIVE swelling is of no concern.<br><br>b) Discuss condition of PSW/LCC inner surface. State if there are areas of concrete spalling. Provide RPV cavity concrete | |

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| | | | indicate the inclusion of a liner in areas where RPV steel beam supports frame into the PSW concrete.<br><br>It is not clear what is the origin of the manufactured sand. Typically, manufactured sand is silica based and mostly derived from quartz or granite. Given the uncertainty in radiation exposure, RIVE maybe a factor contributing to spalling of PSW concrete surface particularly if the concrete aggregates were of quartz origin. | (PSW/LCC) pictures indicating that no spalling or other aging effects are occurring.<br><br>c) Discuss and clarify the purpose of the liner installed at location of the RPV beams framing into the PSW concrete. Was the liner installed to mitigate potential concrete spalling at location of RPV beam support framing into the PSW concrete? | |
| 6<br>AP | 3.5.2.2.2.6 | 3.5-35 | FPL letter (page 3) to NRC (ADAMS accession No ML18108A562) states that "[w]ith the exception of the reactor cavity wall, the Unit 2 reactor coolant system support design duplicates the Unit 1 design. With regard to cavity wall design, the Unit 2 design has been modified […] to provide additional margin above that shown to be acceptable for Unit 1." The SLRA states that **"[b]oth the Unit 1 and Unit 2 PSW/LCC have the same configuration**," which is indicative that there are no differences in PSW/LCC and/or reactor cavities for the two PSL Units.<br><br>It is not clear what are the reactor cavity modifications at Unit 2 PSW/LCC. It is also not clear what precipitated these modifications and whether their examination is addressed in inspection procedures (current and those proposed for the subsequent period of extended operation) | a) By sharing screen show relevant Unit 1 and Unit 2 construction drawings illustrating differences in areas of said modifications in the PSW/LCCs.<br><br>b) Clarify the cause, need, and importance for such modifications. Discuss to what extent (if any) such modifications may have helped improve radiation shielding, maintenance, or aging effects management of PSW/LCC concrete and RPV structural steel system support assembly (e.g., of the RPV steel support components | |

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| | | | particularly if they are in areas of increased radiation. | encased in concrete) and further addressed in inspection procedures.<br><br>c) Discuss whether such modifications would not alter the previously discussed IR(s) for Unit 2. | |
| 7<br>AP | 3.5.2.2.2.6 | | ePortal document NEESL00008-REPT-098, "St. Lucie Units 1 and 2 Subsequent License Renewal Primary Shield Wall Irradiation Evaluation," (aka basis document) references (page 23) the original CLB analysis for safety evaluation of the PSW/LCC in the subsequent period of extended operation. It is not clear whether the referenced CLB analysis regarding the PSW has been summarized in FPL letters to NRC (ADAMS Accession Nos.: ML18114A219 and ML18108A562) and in UFSAR Appendix 3H. | Clarify whether the CLB analysis referenced in the basis document is summarized in the aforementioned FPL letters to NRC. If not summarized and not discussed in Appendix 3H of the UFSAR, summarize the relevant sections of the CLB analysis, particularly in areas of high exposure and where the RPV support beams (assembly) frame into the PSW concrete. | |
| 8<br>JD | 3.5.2.2.2.6 and 3.5.2.2.2.7 | 3.5-39 | SLRA Table 3.5.2.2-2, "End of Subsequent Period of Extended Operation (72 EFPY) Exposures for PSL Concrete," provides fluence and gamma dose values for PSL Units 1 and 2. The SLRA then states:<br><br>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 damage threshold for gamma radiation ($1.0 \times 10^{10}$ rads). Accordingly, no further evaluation of | Clarify the below:<br><br>What is the estimate of the uncertainty associated with the neutron fluence and gamma dose or displacements per atom (dpa) results for:<br><br>• Reactor Pressure Vessel (RV) beltline<br>• Biological Shield Wall (PSW/LCC) | |

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| | | | the PSL Units 1 and 2 PSW/LCCs for gamma irradiation effects is required<br><br>Uncertainties in radiation exposure are the issue at hand for the RV beltline, RV steel support assembly, and BSW | • RV support steel (ring girder and support columns) | |
| 9<br>JD | 3.5.2.2.2.6 and 3.5.2.2.2.7 | 3.5-38 | The SLRA states:<br><br>Unlike what was done for initial license renewal, future projections for PSL SLR included a 10% positive bias on the peripheral and re-entrant corner assemblies on the projection fuel cycle. Peripheral assemblies have one or more faces exposed to the core baffle plates and re-entrant corner assemblies have one corner exposed the core baffle plates.<br><br>The 10% positive bias applied to the projection cycle peripheral and re-entrant corner assembly relative powers is intended to account for normal cycle-to-cycle variations that have been observed in past PSL core designs and are expected to occur in future ones as well.<br><br>It is not clear whether fuel loading restrictions are accounted in the above. | Clarify the below:<br><br>a) Based on the projected margin in the SLRA, are there any fuel loading restrictions needed? This should include any shield assemblies and their designs if needed.<br>b) If so, please state the loading restrictions and how they are being implemented/managed? | |
| 10<br>JD | 3.5.2.2.2.6 and 3.5.2.2.2.7 | 3.5-34 through 3.5-47 | There are no discussions of insulation in SLRA Sections 3.5.2.2.2.6 and 3.5.2.2.2.7 relevant to fluence calculations. | Clarify the types, locations, and results of and inspections of any insulation materials relevant to fluence calculations. Areas of focus are reactor coolant piping, Primary Shield Wall (PSW), and BSW. This should include: | |

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| | | | | • Reflective Metal Insulation (RMI) – reference (IN) 2007-21, Supplement 1<br>• Any other flexible, rigid, or metallic insulations | |
| 11<br>JD | 3.5.2.2.2.6 | 3.5-34 through 3.5-40 | Table 3.5.2.2-2 of SLRA Section 3.5.2.2.2.6 indicates exceedance of fluence limits (see SRP-SLR 3.5.2.2.2.6) on concrete for PSL Units 1 and 2 for 72 EFPY. The section however does not include a discussion of radiation-induced volumetric expansion (RIVE) calculation and its effects on concrete other than the statement:<br><br>Localized cracking and spalling of the concrete at the peak areas of neutron fluence due to radiation-induced volumetric expansion are not expected. | Discuss how any radiation-induced volumetric expansion (RIVE) effects are evaluated in combination with other structural stresses that may be present for all the aging management systems under evaluation. | |
| 12<br>JD | 3.5.2.2.2.6 | 3.5-34 through 3.5-40 | The SLRA section states:<br><br>Although the PSL PSW/LCCs do not have a liner plate on their outside surfaces, localized cracking and spalling of the concrete at the peak areas of neutron fluence due to radiation-induced volumetric expansion are not expected. The reactor cavity areas for both units will continue to be inspected as part of the Structures Monitoring AMP (Section B.2.3.33).<br><br>Since there is no liner to inhibit any concrete spalling that could affect safety related SSCs, the issue at hand is adequacy of inspections for the PSL PSW/LCCs. | Clarify whether there are any components of the PSW/LCC that require any additional or augmented inspections/actions over the subsequent period of extended operation based on the results of OE and information provided in this SLRA. | |

**76b: SLRA Section 3.5.2.2.2.7**

| Question Number | SLRA Section | SLRA Page | Background / Issue (As applicable/needed) | Discussion Question / Request | Outcome of Discussion |
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| 1<br>AP | 3.5.2.2.2.7 | 3.5-44<br>3.5-46 | The SLRA states that "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." SLRA Reference 3.5.4.7 (WCAP LTR-SDA-21-021-NP, Revision 1) addresses the specifics of the assessment and states that "in order to perform the comparison assessment with PBN RPV support evaluation in WCAP-18554 the branch line pipe break (BLPB) for PSL are required and calculated." It also states that the PSL BLPB load development is not complete as the reactor coolant system models from the analysis of record need to be updated. Hence, "conservatively estimated loads are used instead" based on faulted loading condition (including e.g., thrust/reactor vessel internals (RVI) blowdown pressure waves loads) to predict the most stressed location of the RPV steel support system (assembly).<br><br>FPL letters to NRC (ADAMS Accession Nos.: ML18114A219 and ML18108A562) include analyses to determine PSL RPV support system (assembly) loads and evaluation of supports' restraint capability based on a faulted loading condition. They conclude that the support system (assembly) safety margins are adequate. Subsequent analyses summarized in PSL UFSAR Unit 1 Appendix 3H (e.g., pages 3H-3, 3H-19, etc.) with augmented thermal loads contributing to vertical thermal growth of the RPV steel support | Discuss/explain/justify that there was no need to consider the loading conditions listed in Section 3.8.3.3.2.3 and Table 3.8-12 of Unit 1 and Unit 2 UFSARs, respectively in loads development and that the PSL Unit 1 UFSAR Appendix 3H faulted loading condition(s) is/are adequate for a conservative estimate of loads to predict the most stressed location of the RPV steel support system (assembly). | |

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| | | | assembly with displacement to limit its loading capacity (displacement control).<br><br>Although Section 3.8.3.3.2.3 of the Unit 1 UFSAR and Table 3.8-12 of Unit 2 UFSAR list both service load and factored loading conditions for the RPV steel support system (assembly), it is not clear whether PSL in developing the unitless ratios in SLRA Table 3.5.2.2-5 considered these as well. If so, which one? Without updating the PSL RCS models, it is not clear how the plant specific loading conditions used for calculating flaw tolerance for each of the PSL Units would align with those of PBN to afford such a comparison (e.g., see PSL Unit 1 Appendix 3H thermal loads, LOCA break times discussed in ePortal document Book C142, North Anna Syndrome Analysis, etc.) in an irradiated environment.<br><br>1. References:<br>WCAP LTR-SDA-21-021-NP/P, Revision 1, St. Lucie Units 1&2 Subsequent License Renewal: Reactor Pressure Vessel Supports Assessment, June 24, 2021.<br>2. WCAP-18554-NP/P, Rev. 1, "Fracture Mechanics Assessment of Reactor Pressure Vessel Structural Steel Supports for Point Beach Units 1 and 2," September 2020 | | |
| 2<br>AP | 3.5.2.2.2.7 | 3.5-42<br>3.5-44<br>3.5-46 | SLRA Figure 3.5.2.2-3 shows that the RPVs for Units 1 and 2 are supported by 3 beam-column assemblies. The SLRA states that the guidance of NUREG-1509 is used for the embrittlement evaluation of the PSL RPV supports. In its OE segment of the Section, the SLRA states that | a) Provide history of ASME Section XI, subsection IWF inspections for the Unit 1 and Unit 2 RPV structural support system (assembly) to date. | |

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| | | | consistent with Section 4.3.1.1 of NUREG-1509, physical examination of the RPV supports is essential to the evaluation. The NUREG also states that the assessment includes a “mandatory, systematic” reevaluation of the current condition of the supports with an emphasis placed on components supports in tension, as well as “observations and predictions of their possible future weaknesses.”<br><br>Such examinations in subsequent license renewals are performed through the ASME Section XI, Subsection IWF Inservice Inspections (ISIs) programs with additional guidance of GALL-SLR AMP XI.S3. Consistent with ASME Section XI, Subsection IWF Table IWF-2500-1, they include 100 percent (visual) VT-3 examination of supports. However, for multiple components, within a system of similar design, function, and service, the supports of only one of the multiple components are required to be examined.<br><br>Table IWF-2500-1 “Category F-A, Supports” of ePortal PSL 4th ISI Interval Program Plan for Unit 2 states that only one of multiple components, within a system of similar design, function, and service are to be examined. The ePortal 5th ISI Program Plan for Unit 1, however, does not provide such clarification. The SLRA indicates that existing OE for Uni1 included VT-3 of hot and cold leg accessible portions of support, while for Unit 2, it states that VT-3 inspections were performed on all accessible areas of hot and cold leg supports to the extent possible. | b) Clarify whether any of the past inspections were consistent with the requirements of ASME Section XI, Subsection IWF or deviated/augmented as noted in the FPL letter to NRC. If so, state what prompted such deviations and whether PSL plans to continue implementing these along with the guidance provided in GALL-SLR AMP XI.S3 so as to avoid possible future weaknesses during the subsequent period of extended operation.<br><br>c) Discuss whether differences in PSL ASME Section XI Programs for Subsection IWF of Units 1 and 2 RPV structural support systems (assemblies) inspections are just limited to sampling. | |

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| | | | Furthermore, FPL letter to NRC (ADAMS Accession No. ML18114A219) (page 3) indicates that FPL considered augmented NDEs within the cavity beyond those of Section XI inspection interval requirements. Past OE discussed in the SLRA states that the "Unit 2 RPV supports were not being examined." ePortal AR 01716657 (page 65) dated 12/15/2011 discusses Unit 1 RPV supports nonconformance to ASME Code Section XI, Subsection IWF and states that the ISI Program of Unit 1 did not include examination of the RPV Class 1 components supports as required. NUREG-1779, dated July 2003, however, indicates that the applicant's in its initial LRA (page 3.5-36) did include AMRs for ASME Section XI Subsection IWF and Boric Acid inspections of the RPV supports.<br><br>It is not clear when the first ASME Section XI inspection of Unit 1 and Unit 2 RPV structural support assemblies took place and whether any of the past inspections were consistent with the requirements of ASME Section XI, Subsection IWF or augmented as noted in the FPL letter. It is also not clear whether current Unit 1 ISI Program follows the sampling requirements of ASME Code Section XI, Subsection IWF Table-2500-1 "Category F-A, Supports" or has augmented NDE as noted above. | | |
| 3<br>AP | 3.5.2.2.2.7 | 3.5-46<br>B-247<br>2.3-57 | ASME Section XI, Subsection IWF-1300, "Support Examination Boundaries," in its Figure IWF-1300-1 identifies the extent of support examination boundaries and states that the "boundary of an integral support (C) connected to a building structure (E) is the surface of the building | a) Discuss whether IWF and/or the SMP aging management programs (or both) inspect the RPV support beams and columns and whether such | |

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| | | | structure." In addition to the ASME Section XI, Subsection IWF inspection requirements, the GALL-SLR AMP S6, "Structures Monitoring," provides guidance for inspection of structures including those within the reactor containment.<br><br>SLRA AMP B.2.3.33 states that the Structures Monitoring Program (SMP) includes inspection of containment internal structures. SLRA Section 2.3.3.12, "Ventilation," describes the necessity to ventilate the reactor annulus so that concrete is not dehydrated and cracks and thermal growth of the reactor vessel supporting steelwork is limited to 3/16-inch. ePortal IWF Supports includes folders for Units 1 and 2 supports that include inspection reports and numerous photos.<br><br>It is not clear whether the aforementioned concrete-steel interfaces at the reactor cavity are routinely examined and inspected for cracking due to potential thermal effects (including those that could develop from potentially added gamma dose exposure). It is also not clear whether there is OE (e.g., ARs, CRs, and WO) associated with such inspections and examinations. | inspections include examination of the concrete surface where the RPV steel support components frame into the cavity concrete/basemat. For the RPV support beams that would include inspections of the PSW surface at about 18 feet of elevation and for the column at the base plate resting on the 3 inches thick EMBECO 636 grout.<br><br>b) Provide relevant OE and discuss cracking of concrete surface/grout (if any) witnessed during IWF ISI/SMP inspections and examinations. Present pictures detailing condition of concrete surface where the RPV support beam(s) frame into the PSW and of the grout at column base plate(s). | |
| 4<br>AP | 3.5.2.2.2.6 | 3.5-45<br>2.3-57 | The SLRA states that Westinghouse assessed the RPV slide plate lubricant for degraded conditions such as a decrease in viscosity due to radiation effects. In addition to irradiation, the lubricant is exposed to an elevated temperature discussed in Section 2.3.3.12 of the SLRA which states that the "temperature at the bottom of the lubrication plate | a) Discuss what is the lubricant used in the sliding plate and how it is applied to the plate.<br><br>b) Clarify whether the sliding shoe assembly has been evaluated for potential | |

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| | | | between the reactor and support leg to be as high as 300°F (degrees Fahrenheit).<br><br>Although, neutron flux was identified as the key parameter in irradiation aging effects of the dry film lubricant it is not clear what the lubricant is and whether the combined effect of radiation and elevated temperatures could lead to increased frictional forces, excessive wear, and potentially loss of sliding shoe intended mechanical function. In addition, it is not clear whether the support shoe assembly components are exposed to an overall environment conducive to stress corrosion cracking (SCC). | increased frictional forces (and to this end the RPV structural steel beams), excessive wear, and loss of its intended mechanical function due to ineffectiveness of the lubricant caused by its exposure to radiation and elevated temperatures.<br><br>c) Clarify whether the support shoe assembly components are exposed to an overall environment conducive to SCC. | |
| 5<br>AP | 3.5.2.2.2.7 | 3.5-44 | The SLRA states:<br><br>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 steel 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 […] therefore, the conclusions | Discuss whether the SLRA proposed evaluation methodology for the embrittlement of the RPV beam-column support system (assembly) has accounted for all of the prescribed reevaluation criteria of NUREG-1509 (see examples provided) for the subsequent period of extended operation.<br><br>In particular, in order for the staff to determine that the comparative ratios in SLRA Table 3.5.2.2-5 would hold true for the PSL RPV steel supports and provide the same level of assurance of protection against | |

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| | | | contained within the detailed PBN fracture mechanics evaluation (References 3.5.4.9 and ML21111A155) can be applied to PSL.<br><br>LTR-SDA-21-021-NP/P, Revision 1 details the formula used to evaluate PSL RPV structural steel system supports to radiation effects. Both the numerator as well as the denominator of the formula are ratios. The numerator is a ratio of mode I critical fracture toughness (Kic, a measure of flawed steel material's ability to resist fracture to applied loads/stresses) of irradiated material used for the PSL and PBN steel support systems, while the denominator is a ratio of max stresses/tractions presumably applied normal to potential PSL and PBN cracks.<br><br>It is not clear whether the proposed methodology of mixing irradiated material states with stresses fulfills the guidance of NUREG-1509 for plant specific evaluation of the RPV steel supports for each plant, which should also consider; **for example**:<br><br>1. Assessment of the existing condition of the supports at the time of reevaluation;<br>2. Comparison with the initial construction conditions, fabrication procedures (i.e., pre-service examinations of the plates and welds that comprise the RPV steel support assembly);<br>3. Degree of degradation predicted by the end of plant life;<br>4. Original design and safety margins; | cracking due to loss of fracture toughness as the PBN RPV supports, assumptions and conservatisms for the PSL RPV steel support evaluation similar to the assumptions and conservatisms in the PBN RPV steel support evaluation, as discussed in the PBN SLRA Supplement 1 (ADAMS Accession No. ML21111A155, Attachment 21, pages 12 through 16), need to be discussed. Therefore, discuss such assumptions and conservatisms for the PSL RPV steel support evaluation. As was done for the PBN SLRA, the discussion of these assumptions and conservatisms would need to be brought upfront into the SLRA. | |

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| | | | 5. Original design methodology, load combinations for which the supports were designed, allowable stresses and their margins with respect to the actual stresses in the members, and codes governing the original design.<br><br>It is not clear whether the above noted criteria for reevaluation have been considered (implicitly or otherwise) in the implementation of the proposed methodology for evaluation of embrittlement of the PSL RPV beam-column supports. | | |
| 6<br>AP | 3.5.2.2.2.7 | 3.5-43<br>3.5-39 | The Westinghouse analysis in LTR-SDA-21-021-NP/P, Revision 1 assumes full fixity at the RPV beam supports framing into the PSW concrete for calculating its critical stresses and its Mode I fracture toughness. Essentially. the full fixity at the supports results in a typical short stubby support beam (see SLRA Figure 3.5.2.2.-4) with max tensile stresses developing on the exposed portion of the support beam.<br><br>The concrete analysis of the PSW Unit 1 discussed in its UFSAR Unit 1 Appendix 3H, however, indicates that tensile cracking is the potential governing mode of failure (see also SLRA pg. 3.5-39) of the PSW. Summarized Finite Element Analyses in Appendix 3H indicate also potential cracking within the concrete where the RPV support beams are anchored into the concrete is also possible. It is not clear how fixity for the RPV support beam can be assumed under these conditions | In view of the information contained in the UFSAR Unit 1 Appendix 3H, clarify the RPV beam support fixity into the PSW concrete and discuss the conservatism of performed structural analysis of the RPV beam-column support system (assembly). Also discuss the potential for cracking of the RPV steel support beams within the concrete. | |

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| | | | to evaluate embrittlement of the exposed support beam presented in LTRSDA-21-021-P/NP, Revision 1. | | |
| 7<br>DD | 3.5.2.2.2.7 | 3.5-44 page 10 of Ref 3.5.4.8 | The first paragraph in Section 5.0 of LTR-SDA-21-021-NP, Revision 1 (ADAMS Accession No. ML21215A320, Enclosure 4, Attachment 3 to the SLRA) states that the "PSL unit-specific fluence values are taken into consideration for the embrittlement using Figure 3-1 of NUREG-1509 [2] upper bound curve." | Clarify the source(s) for the fluence values. Are they from the appropriate tables in LTR-REA-21-1-NP and -2-NP, Revision 1 (ADAMS Accession No. ML21215A320, Enclosure 4, Attachments 1 and 2 to the SLRA)? | |
| 8<br>DD | 3.5.2.2.2.7 | 3.5-44 page 10 of Ref 3.5.4.8 | Section 5.0 of LTR-SDA-21-021-NP/P, Revision 1 discusses the fracture toughness determination for the plant Saint Lucie (PSL) Units 1 and 2 reactor pressure vessel (RPV) steel supports. The staff needs clarification on the step-by-step determination of fracture toughness for each unit as described in Section 5.0 of LTR-SDA-21-021-NP/P, Revision 1. Note that Sections 5.1.1.1 and 5.1.1.2 of WCAP-18554-P (Ref. 4 of LTR-SDA-21-021-NP/P, Revision 1) discusses the step-by-step determination of fracture toughness of the RPV steel supports for the Point Beach nuclear plant (PBN).<br><br>***** Because a majority of the discussion in the referred section is proprietary, the discussion during the breakout session would very likely cover proprietary information. ***** | a) Clarify the step-by-step determination of fracture toughness for PSL Unit 1 as described in the enclosed brackets in Section 5.0 of LTR-SDA-21-021-NP, Revision 1.<br><br>b) Clarify the step-by-step determination of fracture toughness for PSL Unit 2 as described in the enclosed brackets in Section 5.0 of LTR-SDA-21-021-NP, Revision 1. Discuss why the referenced publicly available documents in this discussion of PSL Unit 2 are redacted<br><br>c) Show sample fracture toughness calculations shown in Table 5-1 of LTR-SDA-21-021-NP, Revision 1 for PSL Unit 1 and PSL Unit 1. | |

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| | | | | d) Discuss how the fracture toughness values for the welds in Table 5-1 of LTR-SDA-21-021-NP, Revision 1 were obtained. | |
| 9<br>DD | 3.5.2.2.2.7 | 3.5-44 | In the referenced section and page, the SLRA states the following about the key inputs for the comparative ratio approach:<br><br>*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.*<br><br>Details of the comparative ratio approach are discussed in Section 7.1 of LTR-SDA-21-021-NP, Revision 1 (details redacted in "NP" version). | To confirm the results of the comparative ratio approach, did you attempt to solve for the critical flaw depth iteratively using the equation on page 18 of LTR-SDA-21-021-NP, Revision 1 for PSL and for PBN (for at least one of the cases reported in SLRA Table 3.5.2.2-5)? Note that in the equation, fracture toughness and stress are known (Tables 5-1 and 6-1 of LTR-SDA-21-021-NP, Revision 1, respectively), but the shape factor "F" must be iteratively determined in order to solve for the critical flaw depth. | |
| 10<br>DD | 3.5.2.2.2.7 | 3.5-44 page 16 of Ref 3.5.4.8 | Figure 6-3 of LTR-SDA-21-021-NP, Revision 1 shows stress contour plot (redacted as shown) of only one T-shaped RPV steel support assembly. | a) Does the FEA stress contour plot in Figure 6-3 of LTR-SDA-21-021-NP, Revision 1 represent the highest stressed T-shaped support for each unit?<br><br>b) Do the other T-shaped assemblies have similar stress plots as in Figure 6-3 or are the stress contour completely different? | |

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| 11<br>DD | 3.5.2.2.2.7 | 3.5-44 page 18 of Ref 3.5.4.8 | The staff couldn't verify from the references provided on post-weld heat treatment (PWHT) discussion on page 18 of LTR-SDA-21-021-NP, Revision 1 that PWHT was performed for the RPV steel supports of both PSL units. | a) For PWHT of PSL Unit 1 RPV steel supports, page 18 of LTR-SDA-21-021-NP, Revision 1 states that reference "[9.a] has no indication or notes of any field welding." Clarify in the drawing in Ref 9.a that there is no field welding at the high stress locations shown in Figure 6-4 of LTR-SDA-21-021-NP, Revision 1.<br><br>b) For the PSL Unit 2 RPV steel supports, page 18 of LTR-SDA-21-021-NP, Revision 1 states that Ref. 14 has information on PWHT, but the staff couldn't verify from Ref. 14 (PSLWEC-21-0066 dated June 15, 2021) that PWHT was performed for the RPV steel supports. Clarify from Ref. 14 (or other references) that PWHT was performed for the PSL Unit 2 RPV steel supports. | |
| 12<br>DD | 3.5.2.2.2.7 | 3.5-44 pgs 13-14 of Ref 3.5.4.8 | The staff couldn't determine in the discussion of loads and stress in Section 6.0 of LTR-SDA-21-021-NP, Revision 1 if there are any cyclic loadings (from thermal or other loads) that affect the RPV steel supports of PSL Units 1 and 2. | Discuss cyclic loadings (from thermal or other loads), if any, that could cause postulated cracks to grow in the RPV steel supports of PSL Units 1 and 2. | |
| 13<br>DD | 3.5.2.2.2.7 | 3.5-45 | The staff needs clarification regarding SLRA Tables 3.5.2.2-3 and 3.5.2.2-4. | Are these two components listed in the referenced tables, ***Support Column*** and ***Bottom of horizontal support (top of column) (56" below top of 6" plate)***, coincident locations? | |

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| 14<br>DD | 3.5.2.2.2.7 | 3.5-37<br>3.5-41<br>3.5-44<br>3.5-45<br>pages<br>10, 20<br>to 21 of<br>Ref<br>3.5.4.8 | The staff needs clarifications on the support shoes and socket/slide assembly (how they work and whether its components are susceptible to stress corrosion cracking due to presence of lubricant) described in the referenced pages. | a) Go over the support shoe socket and slide assembly configuration and how it works. See Figure 4-4 of LTR-SDA-21-021-NP, Revision 1, for the configuration (the SLRA says the support shoe is welded to the underside of the RPV nozzle); if any photographs of the configuration is available, please show during the discussion.<br><br>b) Page 3.5-46 of the SLRA states that magnetic particle examination was performed for the nozzle support. Discuss the latest examination results for both units.<br><br>c) Go over the evaluation of effects of lubricant, as discussed in the Section 7.3 of LTR-SDA-21-021-NP/P, Rev. 1. *** **Discussion would likely include proprietary information**. ***<br><br>d) How is the above (part b) tied to the discussion on page 3.5-45 of the SLRA on the RPV slide plate lubricant degradation (decrease in viscosity). Also, page 3.5-45 of the SLRA mentions a flux threshold: what | |

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| | | | | is the threshold value and document/specs for this value? | |
| 15 DD | 3.5.2.2.2.7 | Drawings in ePortal | Drawings of the PSL RPV steel supports assemblies in the ePortal (EBASCO Drawing 8770-G-794, Rev. 5 for Unit 1 and EBASCO Drawing 2998-G-794, Rev. 3 for Unit 2) indicate that the plates of the assemblies are joined by fillet or groove welds. If the size of the fillet or groove weld is small compared to the thickness of the joined plates, only a partial penetration welded joint can result instead of a full penetration welded joint. | a) Discuss the impact of partial penetration welds on the stresses used in the evaluation of the PSL RPV steel supports. Were there sensitivity studies performed on the finite element analyses (FEA) with the welds joining the plates modeled as partial penetration welds versus the welds modeled as full penetration welds (i.e., as fully bonded joint)?<br><br>b) Even though a weld joint is shown as full penetration in the drawings (for example, Sector B-12 of EBASCO Drawing 2998-G-794, Rev. 3 for Unit 2), the welding symbol in the drawing shows what appears to be a ¼-inch double-bevel groove weld for the 4-inch thick plate, which would result in a partial penetration weld. (Hard to understand why a 4-inch plate would be joined with a ¼-inch weld, so could the drawing be incorrect? What are we missing?) If the FEA is based on fully-bonded joints, is there a physical way to confirm that the groove (or fillet) welded joints are full penetration, especially the locations of high stress | |

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| | | | | (Figure 6-4 of LTR-SDA-21-021-NP, Revision 1)? | |
| 16<br>DD | 3.5.2.2.2.7 | Drawings in ePortal | Some drawings in the ePortal (such as those with filenames *8770-4306_001_2* and *2998-5376_001_2*) seem to indicate some welds of the plates comprising the PSL RPV steel support assemblies have received magnetic particle testing (MT) and ultrasonic testing (UT) examinations. | Clarify these MT and UT examinations of the welds of the plates comprising the PSL RPV steel support assemblies:<br><br>a) Which welds received MT and UT examinations? Did the welds at the locations of high stress (Figure 6-4 of LTR-SDA-21-021-NP, Revision 1) receive them?<br><br>b) Are these MT and UT examinations pre-service examinations?<br><br>c) Confirm that the design specifications for the PSL RPV steel supports that specify these MT and UT examinations are:<br>• FLO-8770-761, Rev 2 for PSL Unit 1<br>• FLO-2998-761, Rev 6 for PSL Unit 2 | |
| 17<br>DD | 3.5.2.2.2.7 | 3.5-44 page 21 of Ref 3.5.4.8 | Section 7.6 of LTR-SDA-21-021-NP, Revision 1 discusses the anchor bolts at the base plates and that the fluence at the base plate is relatively low. | What does “relatively low” means in terms of a fluence value (compare with the exposure levels in SLRA Tables 3.5.2.2-3 and 3.5.2.2-4). | |
| 18<br>JD | 3.5.2.2.2.7 | 3.5-40 through 3.5-47 | The SLRA provides Interaction Ratios (IRs) for concrete, however no IRs and details on their | Provide a summary of the IRs and material strength values for each component/system being | |

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| | | | calculation are stated for the RV steel support assembly specific components/systems. | evaluated. Also include the method/reference for these values any whether they include radiation effects. | |