Point Beach Subsequent License Renewal Aging Management Audit Staff Breakout Questions

SLRA Section B.2.3.1 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	B.2.3.1	B-420	As described in AR 01714837 dated December 23, 2011, there was an active leak in fittings at the seal table of the flux thimble tubes. Based on the eight operating experience files on the portal, there were more similar cases discovered during outage inspections in recent years and the issue does not seem to have been mitigated. Provide an AR search of related cases identified since 2011 and post on the portal prior to the breakout session. Discuss corrective actions taken to ensure leakage integrity. Discuss whether an enhancement to the ISI program is needed to ensure leakage integrity of the subject components.	
2	B.2.3.1	B-420	The staff reviewed the operating experience files on the portal and did not find any program assessment of the Point Beach ISI program. Provide on the portal results of any peer-assessment and self-assessment performed since 2011. As result of any assessment findings for improvement, discuss corrective actions that were required and timetable for implementation.	
3	B.2.3.1	B-420	Provide an AR search of any missed ASME Code-required inspections/examinations and post on the portal prior to the breakout session. If there have been any missed examinations, discuss in each case corrective actions taken to address the issue.	

SLRA Section B.2.3.3 Reactor Head Closure Stud Bolting

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	B.2.3.3	B-55	Show (and discuss as needed) documented ASME Code Section XI IWB volumetric examination results of the reactor closure head studs (B6.20) and threads-in-flange (B6.40) from their last examination (5th 10-year ISI examination expected).	The ASME Code Section XI IWB volumetric examination results should support the plant-specific determination via the CAP database search of no degradation (pg. B-55 of the SLRA). Also, the examination results would provide further assurance of effective protection against SCC and/or IGSCC since volumetric examinations are effective in detecting cracks due to SCC and/or IGSCC.

Reactor Coolant Pump Flywheel Fatigue Crack Growth TLAA

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	4.7.4	4.7-6	Discuss the examination history of the reactor coolant pump flywheel, including examination results.	We need to know whether the RCP flywheel contains flaws to verify that the applicant's fatigue crack growth calculation represents the condition in the field.

P-T Limits and Low Temperature Overpressure Protection (LTOP) Setpoints TLAA

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	4.2.5	4.2-23	Technical Requirements Manual (TRM) 2.2, Reference 5.5 cites the Westinghouse report WCAP-16669 Revision 1, "Point Beach Units 1 and 2 Heatup and Cooldown Limits Curves for Normal Operation." Provide the report.	The staff was not able to find WCAP-16669 in the applicant's portal. The staff need this information to verify the P-T Limits.
2			TRM 2.2, Reference 5.4 cites two NRC letters regarding the pressure temperature limit report dated June 30, 2014. Provide the ADAMS accession numbers for the two NRC letters.	The staff needs to review the history of the pressure temperature limit report. The staff was not able to find these two NRC letters in

				the ADAMS nor in the portal.

Section B.2.3.27, Table 3.3.2-8 Buried Pipe

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	B.2.3.27	B-195	Scope of program – Cementitious materials SLRA Section B.2.3.27, "Buried and Underground Piping and Tanks," states "[t]he cementitious materials associated with the intake structure forebay and discharge piping are managed by the PBN Inspection of Water Control Structures Associated with Nuclear Power Plants AMP."	Clarification needed regarding cementitious materials associated with discharge piping (i.e., cementitious piping, steel piping with mortar lining, steel piping embedded in concrete, steel piping in concrete vault, etc.). Discussion to clarify if there is in-scope cementitious piping at Point Beach.
2	N/A	N/A	Scope of program – Ductile Iron materials The 2015 cathodic protection survey on ePortal states "fire water pipe at this plant is constructed from ductile iron pipe that was reportedly continuity bond[ed] when it was installed."	The SLRA Table 2's do not include ductile iron piping. Discussion needed to clarify if there is in-scope ductile iron piping at Point Beach. [Note: This is the same issue as Selective Leaching Topic #1]
3	B.2.3.27 Table 3.3.2-8	B-196 3.3-221	Scope of program – Underground Environment SLRA Section B.2.3.27 states that the EDG fuel oil storage tanks are located in an underground concrete vault associated with the diesel building. SLRA Table 3.3.2-8, "Emergency Power System – Summary of Aging Management Evaluation," states that the EDG fuel oil storage tanks are exposed to a concrete external environment.	Clarification needed regrading if the subject tanks are exposed to an underground environment (i.e., below grade but are contained within a tunnel or vault such that they are in contact with air and are located where access for inspection

				is limited) or embedded in concrete.
4	N/A	N/A	Preventive Actions – Cathodic Protection GALL-SLR Table XI.M41-1, "Preventive Actions for Buried and Underground Piping and Tanks," recommends cathodic protection for buried steel piping and tanks exposed to soil. The SLRA includes steel piping exposed to soil (in the Service Water, Emergency Power, and Fire Protection systems) and a steel tank exposed to soil (half buried Emergency Fuel Oil Storage Tank). FPLCORP00036-REPT-058 states PBN does not have any buried tanks that are cathodically protected. AR2158241 notes that cathodic protection systems at Point Beach were originally installed to provide corrosion control for the containment structures and buried circulating water piping. SLRA B.2.3.27 enhancements related to cathodic protection do not state which components will be cathodically protected.	Clarification needed regarding in all in-scope steel piping and tanks exposed to soil will be cathodically protected consistent will GALL-SLR Table XI.M41-1.
5	B.2.3.27	B-203	Preventive Actions – Uncoated Fire Protection Piping GALL-SLR Table XI.M41-1 recommends external coatings for buried metallic components exposed to soil. SLRA Section B.2.3.27 states "[t]he [NRC] inspectors noted that the only uncoated buried piping was fire water piping that had been installed within the previous 10 years."	Portions of buried piping in the fire protection system are uncoated. GALL-SLR Report AMP XI.M41 states applicants provide justification when coatings are not provided.
6	B.2.3.27	B-202	Detection of Aging Effects – Direct Inspections of Buried Fire Protection Piping. SLRA Section B.2.3.27 states "[t]he commitment for SLR is to perform a fire protection buried piping inspection (excavation) at least every 10 years."	The staff notes that one inspection of buried fire protection piping will be performed in each 10-year period during the PEO (40-60 years); however, it is not clear that this

			Based on a review of SEM 8.0 on the ePortal, the staff noted a susceptible location in the fire protection system will be inspected once prior to the PEO and at least every 10 years during the PEO.	activity will continue during the SPEO (60-80 years). The staff could not identify a specific commitment to inspect this piping during the SPEO.
7	B.2.3.27	B-196 B-198 B-200	Detection of Aging Effects – Air-to-soil interface SLRA Section B.2.3.27 states the following: • "[i]ndustry OE shows that buried and underground piping and tanks are subject to corrosion. The critical areas appear to be at the interface where the component transitions from above ground to below ground." • Emergency Fuel Oil Storage Tank (T-072) is a bitumastic coated carbon steel tank that is half buried. • "[e]xaminations of the buried tank T-072 are conducted from the external surface of the tank using visual techniques or from the internal surface of the tank using volumetric techniques. A minimum of 25% of the buried surface is examined. This area includes at least some of both the top and bottom of the tank. If the tank is inspected internally by volumetric methods, the method must be capable of determining tank wall thickness and general and pitting corrosion and qualified at PBN to identify loss of material that does not meet acceptance criteria."	Examinations of this tank do not appear to account for increase susceptibility for corrosion at the air-to-soil interface. This is a unique configuration not specifically addressed in the GALL-SLR Report.
8	UFSAR Section 16.2.2.27	A-32	Detection of Aging Effects – Preventive Action Category C UFSAR Section 16.2.2.27, "Buried and Underground Piping and Tanks," states "[b]ased on the PBN OE and the preventive design features in place, the buried steel piping at PBN meets the criteria for Preventive Action Category C. Thus, the number of inspections for each	Future rectifier current/voltage output and cathodic protection potential measurements will determine if Preventive Action Category C is appropriate for buried steel piping.

			10-year inspection period, commencing 10 years prior to the SPEO, based on the inspection quantities identified in GALL-SLR Table XI.M41-2 (adjusted for a 2-unit plant site) is two.”	Preventive Action Categories E or F may be appropriate based on future OE and cathodic protection efficacy.
9	B.2.3.27 UFSAR Section 16.2.2.27	B-196 B-199	Acceptance Criteria – Cathodic Protection SLRA Section B.2.3.27 states the following: • “cathodic protection system will meet the requirements of GALL SLR Section XI.M41, including the polarized potential criteria of NUREG-2191” (i.e., -850 mV instant-off and other alternative criteria) • “[c]riteria for pipe-to-soil potential when using a saturated copper/copper sulfate (CSE) reference electrode is as stated in NUREG-2191 Table XI.M41-3.” (i.e., -850 mV instant-off) UFSAR Section 16.2.2.27 states “[f]or steel components, where the acceptance criteria for the effectiveness of the cathodic protection is other than -850 mV instant off potential (i.e., the electrode’s polarized half-cell potential taken immediately after stopping the cathodic protection current), loss of material rates are measured.	Clarification needed regarding if alternatives to the -850 mV instant-off criterion for steel piping cited in GALL-SLR Table XI.M41-3 will be utilized. If electrical resistance corrosion rate probes will be used, the application should identify (1) the qualifications of the individuals that will determine the installation locations of the probes; and (2) how the impact of significant site features and local soil conditions will be factored into placement of the probes.
10	N/A	N/A	Corrective Actions – Unacceptable cathodic protection survey results. The “corrective actions” program element of GALL-SLR Report AMP XI.M41 states unacceptable cathodic protection survey results are entered into the plant corrective action program. The Buried and Underground Piping and Tanks program enhancements do not include this this recommendation.	Could not identify where this is addressed in current procedures. Based on a review of SEM 8.0, it could be in 56945-02, 56945-03, and/or 56945-04.
11	B.2.3.16	B-132	Submerged Fire Protection System Piping The OE discussion associated with the Fire Water System AMP states:	Clarification needed regarding how aging management of submerged fire water system piping will be addressed. Fire

			"[i]n November 2012, the north and south fire protection supply headers were constantly submerged in water and exhibited surface corrosion at the above ground to below ground transition pits inside the pumphouse. A determination was made that the exterior should be examined, then cleaned and coated and a work order was initiated to perform the corrective action. In January 2019, water was once again identified in these fire protection piping chases, and another work order was initiated to pump out the chase and perform a repair."	water system piping aging management review items only cite soil, concrete, and air external environments.
12	B.2.3.27	B-196 B-197	Preventive Actions – Limiting critical potential of -1,200 mV.to prevent damage to external coatings. SLRA Section B.2.3.27 states that PBN is committed to meeting the cathodic protection system requirements of NACE SP016-2013 (with the exception of Section 6, "Criteria and Other Considerations for Cathodic Protection") and that the information from NACE SP0169-2007 will be used instead of NACE SP0169-2016 for Section 6. NACE SP0169-2007, "Control of External Corrosion on Underground or Submerged Metallic Piping Systems," (specifically Section 6.2.3.2.1) includes a limiting critical potential of -1,200 mV, but it only applies to aluminum piping.	Clarification needed regrading if the -1,200 mV provision will also apply to non-aluminum buried piping. There is not a 2016 version of NACE SP0169.

SLRA Section 4.7.1 Leak-Before-Break of Reactor Coolant System Loop Piping
SLRA Section 4.7.2 Leak-Before-Break of Reactor Coolant System Auxiliary Piping

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	4.7.1	4.7-1	In the TLAA Evaluation on page 4.7-1, it states that the SG inlet and outlet nozzles that contain Alloy 82/182 welds have been "repaired" using Alloy 52/152 inlay to mitigate PWSCC. If these welds experienced a problem, then the wording is correct. However, if not, consider changing wording "to mitigate PWSCC due to the existence of Alloy 82/182 welds", Alloy 52/152 inlays have been applied to mitigate PWSCC."	The words "repair" and "mitigate" provide two different meanings.

2	4.7.1	4.7-2	In the TLAA Evaluation on page 4.7-2, it states that water hammer should not occur in the reactor coolant system piping because of system design, testing and operational considerations. Is this because the LBB evaluation of the RCS piping compared the water hammer loads to the SSE loads at each location identified and the results showed that the water hammer loads were below the SSE loads? If not, please provide additional specific information as to why water hammer should not be considered as an active degradation mechanism and is not a potential source for pipe rupture for the extended period of operation.	The discussion needs to be expanded to discuss why water hammer should not be considered. The expanded discussion will enhance the quality of the final SE.
3	4.7.1	4.7-2	In the TLAA Evaluation on page 4.7-2, it states that the effects of low and high cycle fatigue on the integrity of the primary piping are negligible. Is this statement based on the using stress reduction factors for ASME BPV Code, Section III? The ASME Code Section III defines the stress intensification factors for various piping components under fatigue loading. If not, please provide additional specific information why low and high cycle fatigue of the primary piping is negligible and not a concern for the extended period of operation.	In the SLR, it; is stated that WCAP-14439-P, Revision 4 provides an analysis which justifies the elimination of reactor coolant system primary loop pipe breaks from the structural design basis for the 80 year plant life and provides a statement that "the effects of low and high cycle fatigue on the integrity of the primary piping are negligible." The applicant should expand this statement that is in the WCAP to justify why the effects of low and high cycle fatigue is negligible. This will strengthen the staff final SE.
4	4.7.1		Please provide a description of how the leak detection system complies with RG 1.45, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage."	If the leak detection system complies with RG 1.45, it is helpful to reference this in the final the staff's SE.

5	4.7.1		In the TLAA, there is no mention of the likelihood of cleavage type failure as is described in SRP 3.6.3. Please verify that the likelihood of cleavage type failure remains low due to the piping is made of austenitic stainless steel and 82/182 welds have been mitigated with Alloy 52/152 weld inlays to the safe end welds of the SG primary nozzle that are exposed to the primary coolant. Therefore, this material is highly ductile, and the likelihood of failure from cleavage type rupture is low. If not, please provide justification for why cleavage type failure is not a concern for the extended period of operation.	This addition will strengthen the development of the final SE and also that the TLAA is in compliance with SRP 3.6.3.
6	4.7.1		In the TLAA, there is no discussion which addresses the specifics of thermal aging of stainless steel materials used at PBN and specifically for the 80-year plant life. Please identify if the ferrite numbers (FNs) is consistent with the guidance in RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," and identify what the maximum FN of the weld filler metals are limited to in the PBN piping systems. If representative tensile and fracture toughness data are available for the stainless steel welds, discuss the data or relevant references in comparison with the base materials to confirm that the tensile properties of the base materials are more limiting and brittle fracture would not be an issue for the weld materials.	Addition of the specifics to thermal aging of stainless steel base materials and welds will help strengthen the discussion in the final SE.
7	4.7.1, 4.7.2		In the PBN SLR portal, PBN Report 039 appears to be a summary of the operating experience related to specific Action Requests (AR's) for subjects like SSC's and Mechanical Systems and components, etc. Upon reviewing the report, there does not appear to be OE's related to leakage or cracks pertaining to the RCS loop piping or the reactor coolant system auxiliary piping. Please provide verification that this statement is correct.	PBN is requested to provide clarification for the staff review for the development of the final SE.

SLRA Section B.2.3.28 Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	B.2.3.28	B-207	The exception in SLRA Section B.2.3.28 states "[t]he internal coating applied to the T-175A and T-175B EDG Fuel Oil Storage Tanks will be inspected opportunistically as opposed to periodically. This exception is consistent with the PBN Fuel Oil Chemistry AMP, which takes an exception to the requirement to periodically drain, clean, and inspect T-175A and T-175B. The T-175A and T-175B internal coatings will be inspected opportunistically when an internal inspection of the tanks is deemed necessary based on the results of fuel oil sample analysis or as recommended by the system engineer." SLRA Table 3.3.2-8, "Emergency Power System – Summary of Aging Management Evaluation," states that the internal surfaces of the subject tanks will be managed by the Fuel Oil Chemistry and One-Time Inspection programs. There are no AMR items associated with loss of coating integrity for these tanks.	Clarification needed regarding how fuel oil sample analysis can detect loss of coating integrity. Transport of coating material to downstream components can results in downstream effects such as reduction in flow, reduction in pressure, or reduction of heat transfer.
2	N/A	N/A	GALL-SLR Report Table XI.M42-1, "Inspection Intervals for Internal Coatings/Linings for Tanks, Piping, Piping Components, and Heat Exchangers," recommends an inspection interval between 4 and 12 years based on certain conditions. Based on its review of FPLCORP00036-REPT-082, the staff noted (a) MIC is suspected to be an issue on the raw water side of CCW heat exchangers as exhibited by routine discovery of 'soft spots' in the coating, which when pierced exude a black material along with pitting of the carbon steel; (b) the CCW heat exchangers are currently inspected on a 3 year frequency. Several ARs posted on ePortal associated with coating inspections associated with the CCW heat exchangers reference a 2-year inspection frequency.	Based on plant-specific operating experience, the CCW heat exchangers are inspected every 2-3 years. The staff seeks clarification regarding why the less frequent inspections prescribed in GALL-SLR Report Table XI.M42-1 are appropriate for these components.

SLRA Section B.2.3.33 Masonry Walls

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	B.2.2.33	B-236	In the Point Beach Units 1 and 2 Subsequent License Renewal Aging Program Basis Document – Masonry Walls (FRLCORP00036-REPT-073, Rev. 0), Section 4.6 states "The PBN Masonry Walls AMP acceptance criteria and guidelines for structural inspections are documented in procedure N7.7.9," and "For masonry walls, documented areas are compared to previously reported conditions to determine the extent of any new degradation or changing condition. If a structure or component does not meet the acceptance criteria stated in the inspection procedure, the structure/component will be entered into the corrective action system and evaluated to determine if it can meet its intended function." The procedure N7.7.9, attachment D "Degradation Effects and Acceptance Criteria," Section 2.0 "Degradation Acceptance Criteria by Examination Categories," lists degradation effect or mechanism, non-recordable degradation first-tier (absence of any signs), and recordable degradation second-tier(presence of) for masonry block. It is not clear how this acceptance criteria is used for the inspection of masonry walls. Please clarify the acceptance criteria used for masonry walls and explain the procedure to assess observed degradation against evaluation basis.	To clarify the acceptance criteria used for the Masonry Walls AMP and the procedure used for masonry wall inspection.
2	B.2.2.33	B-236	Elements 3 and 6 of Masonry Walls AMP describe the enhancements to "monitor and inspect for spalling, scaling, shrinkage and/or separation as well as loss of material at the mortar joints, and gaps between the supports and masonry walls that could potentially impact the intended function or potentially invalidate its evaluation basis" and "include specific assessment of the acceptability of crack widths and lengths and gaps between supports and masonry walls", respectively. GALL-SLR XI.S5 – Acceptance Criteria recommends that <u>observed degradation</u> (e.g., shrinkage and/or separation, cracking of masonry walls, cracking or loss of material at	To verify the adequacy of enhancements for the Masonry Walls AMP.

			the mortar joints and gaps between the supports and masonry walls) are assessed against the evaluation basis to confirm that the degradation has not invalidated the original evaluation assumptions or impacted the capability to perform the intended functions. Please explain how these enhancements will be consistent with NUREG-2921 XI.S5, "Masonry Walls".	
3	B.2.2.33	B-236	Correction needed: SLRA states "The following review of plant-specific OE demonstrates how PBN is managing aging effects associated with the PBN Masonry Walls AMP (Section B.2.3.34)". "Section B.2.3.34" shall be "Section B.2.3.33" for Masonry Walls AMP.	

SLRA Section B.2.3.14 Compressed Air Monitoring

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	B.2.3.14	B-118	SLRA Section B.2.3.14, states, "The PBN Compressed Air Monitoring AMP, with enhancements, will be consistent with the ten elements of NUREG-2191, Section XI.M24, "Compressed Air Monitoring." Program Basis Document (PBD), FPLCORP00036-REPT-065, specifically, element 3 states, "Periodic operability checks for the EDG system in part confirm DA system components are operating appropriately. GALL-SLR, Element 3, "Parameters Monitored or Inspected" states, "[p]eriodic air samples are taken and analyzed for moisture content and corrosive contaminants. Opportunistic visual inspections of accessible internal surfaces are performed for signs of corrosion and abnormal corrosion products that might indicate a loss of material within the system. It is not clear to the staff if periodic operability checks are done in-lieu of monitoring air quality for moisture and corrosive contaminants, which is recommended by GALL-SLR. What is	May not be consistent with GALL-SLR, which states, "Periodic air samples are taken and analyzed for moisture content and corrosive contaminants. Opportunistic visual inspections of accessible internal surfaces are performed for signs of corrosion and abnormal corrosion products that might indicate a loss of material within the system."

			<u>the justification for deviating from the GALL-SLR recommendations?</u>	

SLRA Section B.2.3.27 Outdoor and Large Atmospheric Metallic Storage Tanks

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	B.2.3.17	B-136	SLRA Section B.2.3.17, states “The PBN Outdoor and Large Atmospheric Metallic Storage Tanks AMP, with enhancements, will be consistent with the 10 elements of NUREG-2191, Section XI.M29, “Outdoor and Large Atmospheric Metallic Tanks.” Additionally, SLRA Section B.2.3.17, states, “Direct periodic (10-year) visual inspections of an RWST’s nonwetted surface for evidence or loss of material and cracking.” The GALL-SLR Table XI.M29-1, “Tank Inspection Recommendations,” which states that external side of tank exposed to air, surface examination is recommended to detect cracking. <u>The staff’s concern is that visual inspection alone may not be enough to detect cracking, this is also not consistent with the GALL-SLR’s recommendations. What is the justification for deviating from the GALL-SLR recommendations?</u>	May not be consistent with GALL-SLR Table XI.M29-1, “Tank Inspection Recommendations,” which states that external side of tank exposed to air, surface examination is recommended to detect cracking.

SLRA Section B.2.3.21 Selective Leaching

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	N/A	N/A	Scope of Program – Ductile Iron	The SLRA Table 2’s do not include ductile iron piping.

			The staff reviewed FPLCORP00036-REPT-080 and noted the PBN Selective Leaching AMP includes components made of ductile iron. The staff reviewed LR-AMP-024-OTINSP which states "[d]uring the aging management review process it was not always possible to determine whether the cast iron components in question were constructed of gray or ductile cast iron" and "[s]hould further material evaluation determine that the material is ductile cast iron and therefore not susceptible to selective leaching, these components will be removed from the selected set of components to be inspected."	Discussion needed to clarify if there is in-scope ductile iron piping at Point Beach. [Note: This is the same issue as Buried Piping Topic #2]
2	N/A	N/A	GALL-SLR Report AMP XI.M33, "Selective Leaching," allows for the external surfaces of buried components to be excluded from scope based on the condition of external coatings and cathodic protection efficacy. Based on its review of the SLRA and initial license renewal safety evaluation report, portions of buried fire protection system piping are uncoated. The staff reviewed FPLCORP00036-REPT-080 and noted (a) a majority of potentials measured during the 2015 cathodic protection survey did not meet the -850 mV polarized potential criterion; and (b) cathodic protection system performance will provide operating experience to determine whether the Selective Leaching program may exclude buried and cathodically protected components from scope.	Based on the condition of external coatings in the fire protection system and cathodic protection efficacy noted in the 2015 survey, the staff seeks clarification regarding why excluding buried components from the scope of the Selective Leaching program is appropriate.
3	B.2.3.21	B-160	SLRA Section B.2.3.21 states "[e]ach of the one-time and periodic inspections for the various material and environment populations at each unit comprises a 3 percent sample or a maximum of 10 components." NUREG-2222, "Disposition of Public Comments on the Draft Subsequent License Renewal Guidance Documents	Based on recent industry operating experience, the staff seek clarification with respect to using the reduced sample size (i.e., 3 percent with a maximum of 10 components) for gray

			NUREG–2191 and NUREG–2192," provides the basis for reducing the extent of inspections for selective leaching during the subsequent period of extended operation (i.e., 3 percent with a maximum of 10 components per GALL-SLR guidance) when compared to the extent of inspections for selective leaching during the initial period of extended operation (i.e., 20 percent with a maximum of 25 components per GALL Report, Revision 2 guidance). Part of the basis for reducing the extent of inspections is that industry OE has not identified instances of loss of material due to selective leaching which had resulted in a loss of intended function for the component. The NRC issued Information Notice (IN) 2020-04, "Operating Experience Regarding Failure of Buried Fire Protection Main Yard Piping," to inform the industry of OE involving the loss of function of buried gray cast iron fire water main yard piping due to multiple factors, including graphitic corrosion (i.e., selective leaching), overpressuration, low-cycle fatigue, and surface loads. As noted in the IN, a contributing cause to the failures of buried gray cast iron piping at Surry Power Station (SPS) was the external reduction in wall thickness at several locations due to graphitic corrosion.	cast iron components exposed to soil.

SLRA Section B.2.3.25 Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	B.2.3.25	B-184	SLRA Section B.2.3.25, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components," states "[t]his AMP is also used to manage cracking due to stress corrosion cracking (SCC) in aluminum and stainless steel (SS) components…"	The program is also credited for managing cracking in copper alloy components in the waste disposal system. SLRA Section B.2.3.25 does not address cracking of copper alloy components.

2	N/A	N/A	Detection of Aging Effects – Eddy Current Testing The SLRA states "[e]ddy current testing is performed on the RHR heat exchanger tubes through the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.3.25) AMP based on plant specific OE." This is also discussed in the plant specific operating experience section of SLRA Section B.2.3.25. The last plant specific operating experience example in SLRA Section B.2.3.25 states "[e]ddy current testing (ECT) of the Unit 2 containment fan cooler tubes discovered that 7 U-tubes needed to be plugged. The tube degradation was from internal corrosion."	Eddy current testing is not addressed in the program description section of SLRA Section B.2.3.25 (or in the corresponding UFSAR section). The internal surfaces of the containment fan cooler tubes appear to be managed by the Open-Cycle Cooling Water program (see SLRA page 3.3-233). Clarification needed regarding which AMP(s) are credited for managing the internal surfaces of these tubes.
3	Table 3.3-1	3.3-41	For item 3.3.1-55 (steel piping, piping components, tanks exposed to condensation) the discussion states "[n]ot applicable. There are no steel piping, piping components, tanks exposed to condensation in the Auxiliary Systems."	There is steel piping exposed to condensation in SLRA Table 3.3.2-10; however, the item cites 3.3.1-249. Clarification needed regarding not used vs. not applicable designation for item 3.3.1-55.
4	Table 3.3-1	3.3-80	For item 3.3.1-258 (metallic, elastomer, fiberglass, HDPE piping, piping components exposed to waste water; flow blockage due to fouling) the discussion states "[n]ot applicable. There are no metallic, elastomer, fiberglass, HDPE piping, piping components exposed to waste water in the Auxiliary Systems."	There are components exposed to waste water in auxiliary systems that cite flow blockage due to fouling; however ,the items cite 3.3.1-91 or 3.3.1-95. Clarification needed regarding not used vs. not applicable designation for item 3.3.1-258.
5	Table 3.3.2-5	3.3-158	For the boric acid waste evaporator vacuum system tubes, loss of material on the internal surfaces will be managed by	Since the tubes perform a pressure boundary intended

			the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program. The item cites Table 1 item 3.3.1-134, which also cites flow blockage due to fouling (in addition to loss of material).	function, the staff seeks clarification regarding why flow blockage due to fouling is not cited.

SLRA Section B.2.3.9 Bolting Integrity

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	B.2.3.9	B-84	MoS2 Lubricants. Has MoS2 lubricant ever been used or is still being used at Point Beach NPP? The SLRA provides an enhancement to the AMP to ensure that this lubricant is not used for future pressure retaining bolting applications. However, it is not clear if this lubricant has been used or is being used considering that this enhancement will be a new restriction moving forward.	Per GALL-SLR Report, XI.M18. The use of molybdenum disulfide (MoS2) as a lubricant has been shown to be a potential contributor to SCC and should not be used. If they are being used, additional actions may be necessary to adequately manage them during the PEO.
2	B.2.3.9	B-84	Corrective Actions when Leakage is Identified. Clarify how the existing procedure will be consistent with the corrective actions recommended by the GALL-SLR Report when leakage is detected/identified, and the recommended increased inspections. Is enhancement No. 7 intended to capture this (i.e. to follow the specific criteria stated in the GALL-SLR Report)? Per the AMP Basis Document Section 4.7, paragraph no. 4, an enhancement was proposed to update the procedure to ensure that recommendations from the GALL-SLR Report for corrective actions be taken into consideration when leakage is detected. However, based on the enhancement in the SLRA (as written), it is not clear what specific criteria will be followed by the AMP enhancement to demonstrate that the aging effects willadequately managed during the PEO.	Per GALL-SLR Report, XI.M18 – Corrective Actions. If a bolted connection for pressure retaining components is reported to be leaking, follow-up periodic visual inspections are conducted in accordance with plant-specific procedures until the leak is corrected. If the leak rate is increasing, more frequent inspections are warranted. The effects of leakage from bolted connections that have an intended function identified in 10 CFR 54.4(a)(2) are

				evaluated for its impact on components with an intended function identified in 10 CFR 54.4(a)(1) and located within the vicinity of the leaking bolted connection. …The number of increased inspections is determined in accordance with the site's corrective action process; however, there are no fewer than five additional inspections for each inspection that did not meet acceptance criteria, or 20 percent of each applicable material, environment, and aging effect combination is inspected, whichever is less. The number of increased inspections is determined in accordance with the site's corrective action process; however, there are no fewer than five additional inspections for each inspection that did not meet acceptance criteria, or 20 percent of each applicable material, environment, and aging effect combination is inspected, whichever is less..etc.
3	B.2.3.9	B-84	<u>Similarity of Operating Condition at Each Unit</u>. Describe the basis for why the operating conditions at each Unit are	<u>Per GALL-SLR Report, XI.M18 – Detection of Aging</u>

			similar enough (e.g., chemistry) to provide representative inspection results. • Are there any systems which have had an out-of-spec water chemistry condition for a longer time period, or out-of-spec conditions which have occurred more frequently? • For lubricating or fuel oil systems, are there any components that were exposed to the more severe contamination levels? • For raw water systems, is the water source from different sources where one or the other is more susceptible to microbiologically influenced corrosion or other aging mechanisms? SLRA Enhancement no. 4 proposes a maximum of 19 inspections per unit instead of the 25 generally recommended. Which might be okay for a two-unit site provided that the applicant describes in the SLRA the basis for why the conditions at each unit are similar enough, as stated in the GALL-SLR Report. After reviewing the SLRA AMP it is not clear what were the basis and/or potential differences that were considered to justify the acceptability of this reduction in number of inspections.	Effects. In order to conduct 17 or 19 inspections at a unit in lieu of 25, the applicant states in the SLRA the basis for why the operating conditions at each unit are similar enough (e.g., chemistry) to provide representative inspection results. The basis should include consideration of potential differences…
4	16.2.2.9 B.2.3.9	A-18 B-84	Bolting Integrity Program FSAR Supplement. In SLRA Section 16.2.2.9 (3rd paragraph) and the AMP description section states, in part, that "…periodic system walkdowns and inspections are performed at least once per refueling cycle to provide reasonable assurance that indications of loss of preload (leakage), cracking, and loss of material are identified before leakage becomes excessive." • How is excessive defined? Was it meant to state "… before the leakage	It is not clear how this statement in the FSAR Supplement (as written) is consistent with the requirement in 10 CFR § 54.21(a)(3). Excessive leakage is a qualitative measurement and does not seems to establish a clear requirement to ensure that the intended function is maintained

			<u>could result in loss of intended function</u>"? As stated in the GALL-SLR Report, this program monitors the effects of aging on the <u>intended function</u> of closure bolting.	consistent with the current licensing basis (CLB) regardless of how excessive the leakage (big or small) may or may not appear to be. 10 CFR § 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed <u>so that the intended function(s) will be maintained</u> consistent with the CLB for the period of extended operation.
5	B.2.3.9	B-84	<u>Inconsistency between SLRA Commitment No.13(d) and associated AMP enhancement</u>. Per the 4th enhancement in B.2.3.9, the enhancement seeks to add the acceptance criteria for alternate means of testing, and to specify the required representative sample of inspection. However, no commitment (under Commitment No. 13) was identified to include/capture these criteria as stated in the enhancement. Can it be clarified why it is missing as a commitment?	It is not clear how the SLRA commitments aligns with the proposed enhancements to the AMP and to be consistent with the GALL-SLR Report.
6	B.2.3.9 Table 1s	B-84 Several	<u>High Strength Bolts</u>. • Describe how it was verified/confirmed that there are no high-strength bolts present or being used in PBN. Note that items in SLRA Table 1s (items 3.2-1, 012; 3.3-1, 010; and 3.4-1, 007) states that there are there is no high-strength steel closure bolting in the engineered safety features systems, auxiliary systems, and in in the steam and power conversion systems.	To verify consistency with the GALL-SLR Report for those not applicable items and with the FSAR Supplement as it relates to High Strength Bolts

			• Enhancement No. 3 includes actions for when high-strength bolting is used. However, the FSAR Supplement does not appear to consider/include this action in its description as recommended by the GALL-SLR Report in Table XI-01. The GALL-SLR Report recommends that the program description include the volumetric examinations of high-strength closure bolting to detect indications of cracking.	
7	General		<u>Submerged Bolting.</u> Currently, are there any submerged bolting within the scope of the Bolting Integrity Program? Or exposed to water?	To understand the scope of the AMP as it relates to the enhancements associated with submerged bolting.

SLRA Section B.2.3.5 Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components Aging Management Program (AMP)

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
	B.2.3.5	B-63	"An engineering change removed the Alloy 600 (Alloy 82/182) weld filler metal and heat-affected zone for the Point Beach Unit 1 Steam Generator Channel Head Bowl drains. The welds were replaced with partial-thickness Alloy 690 (Alloy 52) welds." Recent plant specific operating experience for Point Beach Unit 1(LER266/2020-001-00) is related to a through-weld leak identified on the 1B SG channel head drain isolation weld. The LER stated that the cause of the event was high cycle fatigue that instated from a small weld defect at the root of the socket fillet weld. Preliminary staff review of recently posted ARs related to this event seem to suggest that there was also a through-weld leak at the channel head to bowl drain partial penetration weld.	Recent operating experience from October 2020, may be relevant to the applicant's AMP
1			Clarify the locations, sizes and identify the weld metals related to this LER.	The question is related to identifying which AMPs, if any

				the specific operating experience is applicable to at Point Beach
2			a) Discuss the causal analysis/root cause analysis related to the events from LER 2020-001. b) Discuss other similar configurations/locations at Point Beach.	Establish the degradation mechanism and other susceptible locations
3			Discuss how this plant specific information will be used to inform/enhance the Point Beach Nickel Alloy AMP.	Identify potential enhancements to the AMP

SLRA Section B.2.3.22 ASME Code Class 1 Small-Bore Piping

#	SLRA Section	SLRA Pages	Question / Issue	Why are we asking?
	B.2.3.22	B-165 & B-166	SLRA states that “The PBN ASME Code Class 1 Small-Bore Piping AMP is an existing condition monitoring program for detecting cracking in small-bore ASME Code Class 1 piping (page B-165). Under Enhancements on page B-166 the SLRA states “Create new procedure to do the following: Perform the new one-time inspections of small-bore piping using the program methods, frequencies, and acceptance criteria included in a new program procedure.”	
1			Please clarify if the current program is an existing program.	Clarification
	B.2.3.22	B-168	SLRA states, “The site-specific OE for ASME Code Class 1 Small-bore Piping indicates that no age-related cracking has been identified, thus PBN remains a Category A plant per NUREG-2191, Table XI.M35-1. However: LER 90-008-00 related to Point Beach Unit 1, in part is related to RCS leakage on the upstream weld on B steam generator channel head drain line isolation valve 1RC-526B.	

			LER 1999-012-00, is also related to Point Beach Unit 1, and RCS leakage on the upstream weld of the A steam generator channel head drain line isolation valve 1RC-526A. Point Beach Unit 1 LER 2020-001-00, states in part "a through-weld leak was identified on the steam generator (SG) 1 B drain line [AB] at the drain isolation valve, 1RC-00526B weld."	
2			In light of the above, explain the conclusion that "no age-related cracking has been identified," and how a one-time inspection for Unit 1 is adequate.	Clarification why a one-time inspection of small-bore piping for Unit 1 is adequate.
3			LER 2017/-003-00 is related to RCS leakage for a Point Beach Unit 1 instrument tubing welded joint. Please confirm that the RCS leakage identified in the above LER is not applicable to the Point Beach Unit 1, ASME Code Class 1 small-bore piping population.	Clarification/Confirmation
4			NRC issued IN-2007-21, "Pipe Wear Due to Interaction of Flow-induced Vibration and Reflective Metal Insulation (ML071159951)," which was related ASME Class 2 piping. More recently the NRC revised the IN due to recent operating experience from US nuclear plants where additional instances of wear were identified (ML20225A204) and some of the wear marks were deeper and extended nearly 360 degrees around the circumference of ASME Code Class 1 small-bore piping. Is the recent Industry operating experience with reflective metal insulation applicable to the small-bore piping at Point Beach Unit 1 and 2?	Related to recent industry operating experience for small-bore piping.

SLRA Sections 2.4.15, 4.7.6, and B.2.3.13 Inspection of Overhead Heavy Load and Light Load (Related to Refueling, Crane Load Cycle Limit

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	2.4.15 4.7.6	2.4-29 4.7-10	Based on the review of the following documents, it is not clear to the staff whether the applicant accounts for all	1- To verify whether all the all the PBN cranes, hoist and

			of the PBN cranes, hoist and lifting devices in the SLR application. For example: 1. some of the *"Special lifting devices,"* in the UFASR were not addressed in the SLR Application, such as NUHOMS OS197-PB Lift Beam, NUHOMSS197-1 Transfer Cask, etc. 2. *"Emergency Diesel Generator G-03 and G-04 Cranes"* were not addressed in *"Control of Safe Load Path and Rigging Manual,"* 3. *"the Emergency Diesel Generator G03 and G04 Cranes,"* were not addressed in the UFSAR. Therefore, the staff is requesting a list of the PBN cranes, hoist and lifting devises considered in the SLR, as well as a list of the cranes that are excluded from SLR. This list should also justify whether a TLAA assessment is necessary for the cranes.	lifting devises are accounted for in the PBN SLR application. 2- To clarify whether "the Emergency Diesel Generator G03 and G04 Cranes," should be included in the TLAA assessment.
2	Table 2.4-15 Table 3.3-1 Table 3.5.2-15	2.4-31 3.3-41 3.3-68 3.5-141	In Table 2.4-15, *"Cranes, Hoists, and Lifting Devices Subject to Aging Management Review,"* the applicant tabulated the component types as *"Bridge and Trolley Framing, Crane Rail, Lifting Devices and Rail Hardware."* In Table 3.3-1, *"Summary of Aging Management Evaluations for the Auxiliary Systems,"* the applicant identifies *"Item 3.3-1, 199,"* as *"Cranes: steel structural <u>bolting</u> exposed to air,"* which are "Consistent with NUREG-2191," under column "Discussions." In Table 3.5.2-15, "Cranes, Hoists, and Lifting Devices – Summary of Aging Management Evaluation," the applicant identifies component types	To clarify whether the "Rail Hardware," includes steel structural bolting for cranes, hoists and lifting devices.

			associated with "Item 3.3.-1, 199" as "Rail Hardware." Clarify whether the *"Rail Hardware,"* in Table 3.5.2-15 includes *"Cranes: steel structural bolting."*	

SLRA Table 3.3-1, 3.3.2-8 Non-metallic

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	Table 3.3-1 Table 3.3.2-8	Page 3.3-53 Page 3.3-216	Table 3.3.2-8 cites no aging effects for plastic piping exposed to air-indoor uncontrolled. The corresponding Table 1 item (i.e., item 3.3.1-119) cites the following materials: nickel alloy, PVC, and glass.	Need confirmation that plastic piping cited in Table 3.3.2-8 is PVC. If the material is not PVC, it is unclear why citing no aging effects is appropriate. The GALL-Report cites aging effects (e.g., hardening or loss of strength) for polymeric piping exposed to air.

TLAA Identification (SLRA Section 4.1, Pages 4.1-1 – 4.1-8)

#	SLRA Section	SLRA Page(s)	Question / Issue	Why are we asking?
1	4.1, including Tables 4.1.5-1, 4.1.5-2 and 4.1.5-3-2	4.1-1 – 4.1-8	TLAA Identification Methodology, Screening Bases and Results	
		4.1-4 – 4.1-5 for T4.1.5-1 4.1-6 for T4.1.5-2 4.1-7 – 4.1-8 for T4.1.5-3	*High energy line break (HELB) analyses.* SLRA Table 4.1.5-1 (and Note 1 for the table) states that the HELB analyses for PBNP Units 1 and 2 are not TLAAs because the "HELB methodology does not involve time-limited assumptions defined by the current operating term." • The staff requests that the applicant discuss the basis for its conclusion that the S_A maximum allowable thermal expansion stress parameter defined and used in FSAR Section A.2.5 for intermediate piping segments is not considered to involve a time-dependent parameter defined by the current	According to SRP-SLR Section 4.1, The HELB analysis for a licensed PWR may qualify as time-limited aging analysis (TLAA) per criteria of TLAAs in 10 CFR 54.3(a). Thus, the staff needs validation that the Point Beach HELB analyses do not include any evaluation criteria for ANSI B31.1 class intermediate piping segments that are based

#	SLRA Section	SLRA Page(s)	Question / Issue	Why are we asking?
			operating term, as defined in Criterion 3 of 10 CFR 54.3a. *References*: SLRA Table 4.1-2; FSAR Appendix A, Section A.2 (including Subsection A.2.5)	on time-dependent parameters defined by the current operating term for the facility.
			Assessment of Flaw growth due to fatigue or stress corrosion cracking – relative to potential MNSA or ½-nozzle repair designs. • Staff seeks confirmation whether the design of the ASME Class 1 or Class A systems at PBNP includes any mechanical nozzle seal assemblies (MNSAs) or ½-Nozzle repair design modifications of the Primary Coolant System. *References*: St. Lucie LRA Section 4.6.4 (as a NextEra Energy Fleet Comparison); NUREG-1779. Section 4.6.4	The staff seeks confirmation/validation that the Point Beach unit designs do not currently include any MNSA or ½-nozzle assemblies where the design bases for installing the repair assemblies may have included performance of a time-dependent flaw growth or fracture mechanics assessments defined by the current operating term.
2	4.1.4	4.1-3	Identification of Exemptions. In SLRA Section 4.1.4, the applicant states that it did not identify any regulatory exemptions granted under 10 CFR 50.12 that remain in effect for the CLB and are based on a TLAA (or that need evaluation in accordance with 10 CFR 54.21(c)(2)). Discuss the LTOP methodology basis that is currently adopted in the most recently submitted PTLR (per Tech Spec 5.6.5.c) for the establishment of the LTOP system pressure lift and arming temperature setpoints. Explain why either: (1) the existing regulatory exemption for use of ASME Code Case N-514 is not considered to be an exemption previously granted in accordance with 10 CFR 50.12 that is based on a TLAA, or (2) the exemption granted for Code Case N-514	The staff needs confirmation that the exemption granted under 10 CFR 50.12 and issued by the staff on January 27, 1997 for use of Code Case N-514 either is not based on a TLAA or does not remain in effect for the CLB. This is needed for validation that the exemption for use of Code Case N-514 does not need to be listed and evaluated in the SLRA pursuant to the requirements in 10 CFR 54.21(c)(2).

#	SLRA Section	SLRA Page(s)	Question / Issue	Why are we asking?
			is no longer considered to remain in effect for the CLB. *Reference(s)*: FSAR Page 4.3-4; TS 5.6.5; Wisconsin Energy Exemption Request of April 20, 1933 (ADAMS ML20035G121); NRC Evaluation of January 27, 1997 (ADAMS ML021970302); Nuclear Management Company Exemption Request of July 14, 2000 (ADAMS ML003734904); NRC Evaluation of October 6, 2000 (ADAMS ML003758475)	

Point Beach SLRA Breakout Question Section B.2.3.24 Flux Thimble Tube Inspection

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	B.2.3.24	B-180 through B-182	**Discussion Topic and Questions:** Discussion Topics: Please be prepared to discuss how NextEra Energy uses wear rate data to project the amount of wear that may occur in the thimbles at the next inspection, and also the type of wear rate projection and associated wear-rate value that is currently used for the eddy current testing reinspection interval for the tubes. Question: Do the inspections for the thimble tubes in each unit (especially for the limiting worn tube in each unit) demonstrate that wear may be occurring in tubes under an increasing, non-linear wear rate?	Wear rate is a key factor in the calculation of inspection frequency. If wear rates are increasing and non-linear, that would result in non-conservative inspection frequencies. NRC staff observed during review of documents on the portal that the limiting wear rate for Unit 1 was calculated to be 14% based on observations during the first 3 years of operation. It is unclear how these wear rates have changed over time.

			Issue: HX-02 Rev 11 “Thimble Tube Condition Assessment Program”, dated January 20,2012, states that the previous thimble tube inspection for Unit 1 was performed in March 2010. HX-02 also states that the next scheduled thimble tube inspection for Unit 1 would be performed in Spring 2013. HX-02 states that the worst thimble tube from Unit 1 was 1-G6 with a minimum estimated life of 5.5 years. HX-02 recommended that program requirements would indicate a 3 year interval (1 cycle before 4.5 years) based on tube 1-G6. FPLCORP00036-REPT-057 “Aging Management Program Basis Document-Thimble Tube Inspection”, dated November 6, 2020, mentions Unit 1 thimble tube inspections performed in the Spring of 2010 and the Spring of 2017. There is no mention of any inspection in the Spring of 2013. Questions: What is the status of tube 1-G6 (i.e. in-service, plugged)? Were the Unit 1 thimble tubes inspected in the Spring of 2013? If not, explain how this is consistent with the thimble tube condition assessment program requirements as stated in HX-02?	If Unit 1 scheduled inspections were not performed in 2013, the resulting 7 year inspection interval does not agree with the recommended 3 year inspection frequency specified in the licensee’s basis documents. Also, if the wear rate for tube 1-G6 met the worst case projections, it is possible that it might have been taken out of service. If tube 1-G6 was taken out of service, it will affect inspection frequency calculations going forward.
			Issue: HX-02 Rev 11 (issued in January 2012) states that the previous thimble tube inspection for Unit 2 was performed in March/April 2011. HX-02 also states that the next scheduled thimble tube inspection for Unit 2 would be performed in Fall 2012. HX-02 states that the worst thimble tube from Unit 2 was 31-J12 with a minimum estimated life of 10.4 years. HX-02 recommended that program requirements would indicate a 6 year interval for Unit 2. REPT-057 mentions thimble tube inspections performed in the Spring of 2011 and the Fall of 2017. There is no mention of any inspection in the Fall of 2012.	If Unit 2 scheduled inspections were not performed in 2012, the resulting 6-1/2 year inspection interval does not agree with the recommended minimum 6 year inspection frequency specified in the licensee’s basis documents.

			Question: Were the Unit 2 thimble tubes inspected in the Fall of 2012? If not, were they inspected at any other time between Spring 2011 and Fall 2017? If not, explain how this is consistent with the thimble tube condition assessment program requirements that inspection intervals should not exceed 6 years?	
			Issue: In the Plant Specific Operating Experience section of REPT-057, the licensee mentions concerns related to inspection deferrals, calculating methodology and record retention, and references the following ARs (AR00004630, AR00004646, AR00004658). However, when the NRC staff looked up records AR00004630, AR00004646, and AR00004658 on the portal (under folder ARs WOs Etc), the records did not appear to address inspection deferrals, calculating methodology and record retention. These ARs instead discussed boric acid buildup on a packing gland, discrepancies in documents related to electrical safety, and license renewal review of the leak detection system. Question Please be prepared to discuss the concerns related to inspection deferrals, calculating methodology and record retention. If documents relevant to these topics can be found on the portal, please demonstrate where these may be found, and briefly explain the contents of these documents.	Issues of inspection deferrals, calculation methodology, and record retention are relevant to the NRCs safety evaluation. NRC staff need to be able to audit and understand these ARs that are referenced in the basis document and are relevant to the safety evaluation.

Point Beach SLRA Breakout Session Sections 3.5.2.2.2.6 and 3.5.2.2.2.7 Fluence Concrete and Steel

#	SLRA Section(s)	SLRA Pages	Question / Issue	Why are we asking?

1 KH AP	TLAA 4.2.2	4.2-7	Topic: Radiation fields in concrete/steel SLRA Table 4.2.2-1, "RTPTS Calculations for PBN Units 1 and 2 Reactor Vessel Beltline Materials," states that the surface fluence at the intermediate shell forging for E > 1.0 MeV at EFPY 72 is 7.80 E19 n/cm^2. Nuclear Management Company, LLC, 2004, Application for Renewed Operating Licenses, Point Beach Nuclear Plant Units 1 & 2 (Docket Nos. 50-266 and 50-301, Hudson, WI) states that the fluence at EFPY 53 for E > 1.0 MeV in that location would be 5.39 E19 n/cm^2. Projecting the 53 EFPY radiation to 72 EFPY, for E > 1.0 MeV would yield a fluence of about 7.32 E19 n/cm^2. The value reported in the 2004 LRA does not appear to have a 10% bias applied. Application of a 10% bias yields a projected 72 EFPY fluence value of 8.05 E19 n/cm^2, which differs from the SLRA reported value of 7.80 E19 n/cm^2 at that location. Provide discussion or clarification of the primary factors attributable to this difference.	To further validate Westinghouse-derived radiation estimates on BSW concrete and RV steel support structure for the subsequent period of extended operation
2 AP	FE 3.5.2.2.2.6	3.5-37	Topic Irradiated BSW concrete damage The SLRA states that the "latest research data presented in Reference 3.5.4.5 indicates that the threshold for damage to concrete from gamma dose may be higher than 1 E10 rads." It then compares applicant's UFSAR Section 5.3.2.2 containment	To assess applicability of applicant's claim of conservatism for gamma heating on of BSW (considering they exceed SRP-SLR limits for concrete), PSW concrete and ensure their structural integrity and capacity to maintain their intended function(s) for the subsequent period of extended operation. The

			ventilation system temperature of 105°F during normal operations to that of referenced EPRI Report 3002011710 of 150°F for gamma heating. There is no discussion of the added synergistic effects of thermal gradients considered in the original design (see Bechtel Book 29), to the conclusion reached in Reference 3.5.4.5 for concrete exposed to gamma radiation. Discuss applicability of gamma heating results obtained in EPRI Report No. 3002011710, "Irradiation Damage of the Concrete Biological Shield Wall for Aging Management" (Reference 3.5.4.5) to PBN BSW and PSW concrete. To what extent various concretes referenced in the EPRI Report are directly comparable to the plant specific cast in place PSW, BSW concrete at PBN taking into consideration synergistic effects of plant specific CLB (design) cavity temperature of 125°F (noted in Bechtel Book 29).	10 CFR 54.21(a)(3) requires for the identified SSCs that the effects of aging will be adequately managed such that their intended functions are maintained consistent with the current licensing basis (CLB) for the subsequent period of extended operation.
3 AP	FE 3.5.2.2.2.6	3.5-34 and 3.5-35	Topic Irradiated BSW concrete damage The SLRA states that ASTM C-194 Type D water reducing agent was used in the construction of BSW and PSW. However, it also states that the w/c ratio was 0.6. Discuss whether the BSW concrete could be considered as wet concrete that would imply an increased moderation and thermalization of neutrons	To assess conservatism of estimated damage to BSW (considering they exceed SRP-SLR limits for concrete), PSW concrete due to radiation and ensure their structural integrity and capacity to maintain their intended function(s) during the subsequent period of extended concrete.

			due to its higher hydrogen content. Clarify whether shielding effects of high w/c ratio considered in evaluating the effects of fluence and gamma dose damage inside the BSW concrete?	The 10 CFR 54.21(a)(3) requires for the identified SSCs that the effects of aging will be adequately managed such that their intended functions are maintained consistent with the current licensing basis (CLB) for the subsequent period of extended operation.
4 AP	FE 3.5.2.2.2.6	3.5-35 thru 3.5-39	Topic Irradiated BSW concrete damage The SLRA states that the “BSW performs no structural function and that it was evaluated for irradiation effects to ensure it will maintain its structural integrity.” It also states that that LOCA loads consistent with CLB are included in Faulted-2 loading condition. It further states that the “neutron fluence would reach the damage threshold in NUREG-2192 at 3.35 inches” and subsequent to volumetric expansion (RIVE) damage would escalate to 3.92 inches into the BSW. Bechtel Book 29 discusses CLB LOCA (guillotine failure) loads and force impingement on the BSW. It further states that seismic loads on the wall are not controlling its design. It then states that concrete from elevation 16’-2 and1/8” to 26’-4.5” to be considered as “useless due to radiation.” Although large LOCAs and force impingements are eliminated from the design basis (see Westinghouse WCAP-14439-P, Revision 4, “Technical Justification for	To clarify capacity of BSW irradiated concrete to maintain its structural integrity and intended function(s) during the period of extended operation. The 10 CFR 54.21(a)(3) requires for the identified SSCs that the effects of aging will be adequately managed such that their intended functions are maintained consistent with the current licensing basis (CLB) for the subsequent period of extended operation

			Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Point Beach Nuclear Plant Units 1 and 2 for the Subsequent License Renewal Program (80 Years)"), it is not clear what loads and loading conditions consistent with CLB and the ultimate strength design criteria of ACI 318-63 were used for the re-evaluation of the BSW design (including loading conditions such as Faulted-2 and irradiation effects) so that it can continue to maintain its structural integrity during the period of extended operation. Clarify what loads and loading conditions and their adequacy have been used in the re-evaluation of the BSW (including loading conditions such as Faulted-2 and irradiation effects) so that it can continue to maintain its structural integrity during the period of extended operation.	
5 AP	FE 3.5.2.2.2.6	3.5-36	Topic Irradiated BSW concrete damage The SLRA states that the neutron fluence would reach the damage threshold in NUREG-2192 including radiation induced volumetric expansion (RIVE) at 3.92 inches into the BSW. It also states that the liner plates covering the surface of the BSW are welded to each other and are anchored to the concrete with steel angle sections, thus enabling composite action with the concrete wall." Bechtel	To clarify capacity of BSW irradiated concrete to maintain its structural integrity and intended function(s) during the period of extended operation. The 10 CFR 54.21(a)(3) requires for the identified SSCs that the effects of aging will be adequately managed such that their intended functions are maintained consistent with the current licensing basis (CLB) for the

			Drawing C-326, Revision 3, “Containment Structure Biological Shield Liner Plate,” indicates that the liner is secured to concrete via embedded structural steel angles. In addition, Bechtel Book 29 states that BSW concrete from elevation 16-2 and1/8” to 26’-4.5” to be considered as “useless due to radiation.” It then states that vertical reinforcing steel bars exist at about 4” from the BSW surface. DRAFT NUREG/CR, “Radiation Evaluation Methodology for Concrete Structures,” indicates that higher-energy neutrons moderated and thermalized by successive scatters may induce a thermal peak inside Type 04 concrete (see ANSI 6.4,-2006, “nuclear analysis and design of concrete radiation shielding for nuclear power plants”) to a depth of ~12 cm or about 5 inches. a. Given the state of BSW estimated damage based on estimates of fluence and gamma dose, demonstrate that the liner plate would remain sufficiently anchored to the BSW concrete for the subsequent period of extended operation. b. Clarify that the vertical reinforcing bars close to the surface of BSW would not be affected (including their bond	subsequent period of extended operation

			strength) from radiation induced internal stress and strain fields further augmented by thermal expansion associated with cavity temperatures of 125 ^{o}F (or more) potentially displacing these outwards	
6 AP	FE 3.5.2.2.2.6	3.5-35	Topic Irradiated BSW concrete damage The SLRA States that "Westinghouse calculated the maximum neutron fluence (E > 0.1 MeV), gamma dose and displacements per iron atom (dpa) for the end of the SPEO on the BSW, and RV support components based on the reactor models and radiation transport calculations performed for the PBN SLR RV neutron exposure." Clarify the above statement on displacements per iron atoms (dpas) that are only applicable to steel RV support components as the BSW, PSW concrete structures are made from amorphous materials that lack a crystalline structure.	To clarify applicant's statement in the application regarding damage assessment of BSW, PSW concretes.
7 AP	FE 3.5.2.2.2.6	3.5-36	Topic: Irradiated BSW concrete The SLRA states that the "governing failure mode of the wall is pure tension stress in the wall tangential (horizontal) direction under the combination of accident pressure and thermal load." By sharing a screen, clarify load transfers (BSW to	To get a visual of connectivity of BSW, PSW concrete walls essential in understanding applicant's statement on PSW governing mode of failure and ensure that involved structures will maintain their intended functions during the

			PSW) and identify in a Bechtel/Westinghouse appropriate drawing(s), where a. are the shear keys connecting the BSW to PSW? b. is the location of the secondary shield	subsequent period of extended operation. The 10 CFR 54.21(a)(3) requires for the identified SSCs that the effects of aging will be adequately managed such that their intended functions are maintained consistent with the current licensing basis (CLB) for the subsequent period of extended operation
8 AP	FE 3.5.2.2.2.6	3.5-37	Topic: Irradiated BSW concrete The SLRA states that “comparing with the un-irradiated concrete (where the maximum interaction ratio (IR) was calculated as 0.84, the maximum IR for the irradiated concrete was determined to be 0.89 which has been increased but remains less than 1.0.” Discuss and clarify calculation of derived tensile and shear failure interaction ratios (IRs) in FPLCORP00036-CALC-002 and FPLCORP00036-CALC-003 for PSW concrete with particular emphasis on loads and load combinations, stress calculations and potential applicability to BSW.	To obtain a better understanding of pre and post period of extended operation applicant derived tensile and shear failure IRs of concrete and assess their validity so that a reasonable assurance of structural adequacy of irradiated PSW concrete to maintain its intended functions during the period of extended operation can be made. The 10 CFR 54.21(a)(3) requires for the identified SSCs that the effects of aging will be adequately managed such that their intended functions are maintained consistent with the current licensing basis (CLB) for the subsequent period of extended operation
9 KH	FE 3.5.2.2.2.6	3.5-35	Topic: Calculation of Fluence and Gamma dose to BSW concrete What are the uncertainties associated with the calculated maximum neutron fluence and gamma dose values presented in Table	Biases and uncertainties can play a critical role in assessing whether calculated results are adequately bounded by acceptance criteria.

			3.5.2.2-2? How were these uncertainties quantified?	

Point Beach SLRA Breakout Session Sections 3.5.2.2.2.6 and 3.5.2.2.2.7 Fluence Concrete and Steel (continued)

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1 DD	FE 3.5.2.2.2.7	3.5-41	Topic: SLRA RV steel support structure evaluation SLRA Table 3.5.2.2-5 shows that the dpa exposures at the top and bottom of the RV support columns are in the dpa range in Figure 3-1 of NUREG-1509 where the embrittlement shift is insignificant. The staff needs clarity and confirmation of the locations of the top and bottom of the columns: drawing 34535 appears to indicate that the top and bottom of the columns are welded connections. Confirm that these welded connections (i) are the approximate locations of the top and bottom of the columns referred to in SLRA Table 3.5.2.2-5, and (ii) their irradiated fracture toughness is therefore bounded by the fracture mechanics evaluation of the column itself.	To confirm that the welded connections at the ends of the columns are bounded by the fracture mechanics evaluation of the column itself for which a large embrittlement shift is determined due to the maximum column dpa exposure reported in SLRA Table 3.5.2.2-5.
2 DD	FE 3.5.2.2.2.7	3.5-42	Topic: SLRA RV steel support structure evaluation Of the selected components evaluated for critical flaw sizes in SLRA Table 3.5.2.2-6 that have welds (i.e., the T-1 materials of the box ring girder and I-beam), explain how the base metal fracture toughness bounds the fracture toughness of the	To ensure that the fracture mechanics evaluation considers the material performance of weldments, not just the base metal.

			associated welds. ASME Code Section III NF-2331 specifies that impact testing should include weld material, NF-2331(b)(3), so shouldn't there be test reports for the welds also from which impact testing properties can be obtained (and thus fracture toughness can be derived for the welds)?	
3 DD	FE 3.5.2.2.2.7	3.5-42	Topic: SLRA RV steel support structure evaluation Regarding the critical flaw size results in SLRA Table 3.5.2.2-6 for the box ring girder bolts: a. Confirm that the calculated critical 360-deg surface circumferential flaw sizes could be closer to the ASME Code Section XI allowable flaw size of 9.4% if some of the conservatisms were removed in the calculation of the critical flaw sizes. b. Table 6-1 of WCAP-18554-NP shows the maximum allowable length (value is proprietary) of a surface, straight and semi-circular front flaw. The specification for the box ring girder bolts (ASTM A490 per Section 3 of WCAP-18554-NP, which is included in Enclosure 4, Attachment 2 of the SLRA) states that magnetic particle testing was used to inspect for circumferential-oriented flaws. Discuss if the maximum length of the straight and semi-circular front flaw in Table 6-1 of WCAP-18554-NP would be	a. To confirm understanding of the general approach and intent of the comparison of the calculated critical flaw sizes with those of the ASME Code Section XI allowable law sizes. b. To ensure that smaller surface flaws, not just the longer 360-deg circumferential flaws, would have also been detected during procurement of bolts (and thus bolts with unacceptable indications would have been rejected for use).

			detected by the magnetic particle testing technique.	
4 DD	FE 3.5.2.2.2.7	3.5-42	Topic: SLRA RV steel support structure evaluation One of the components analyzed for critical flaw sizes in SLRA Table 3.5.2.2-6 is the support shoe box. The sliding motion on the bearing plates (of the RPV nozzles that goes on top of the support shoes) is summarized in Section 3 of WCAP-18554-NP (Enclosure 4, Attachment 2 of the SLRA). Confirm the sliding motions in the RPV support shoe assemblies, i.e., that there is relative motion between the following components and no lubricants are used in both: • Nozzle and shim plate • Shim plate and support shoe	Lubricants can promote stress corrosion cracking (SCC), which is always a potential degradation mechanism that needs to be managed for long term operation. This is to confirm that lubricants are not used to facilitate the sliding motion in the support shoe assemblies, which would confirm no SCC in RPV supports.
5 DD	FE 3.5.2.2.2.7	3.5-42	Topic: SLRA RV steel support structure evaluation SLRA Table 3.5.2.2-6 shows the box ring girder and I-beam as two of the three components that are limiting in terms of critical flaw sizes. Section 3 of WCAP-18554-NP (the source of SLRA Table 3.5.2.2-6) states that the box ring girder and I-beam are made of T-1 steels, which are ASTM A-514-65 or ASTM A-517-65 Type F steels. The CMTRs of the T-1 steels provided in the ePortal show copper and nickel contents of about 0.24	To determine the applicability of Figure 3-1 of NUREG-1509 for predicting embrittlement of the PBN RPV supports, especially the box ring girder and I-beam which are exposed to high dpa values and have relatively higher copper and nickel content than the materials included in Figure 3-1 of NUREG-1509.

			and 0.75, respectively (which are within the ASTM specs). The staff noted that Figure 3-1 of NUREG-1509 was used to determine the ΔNDTT for the box ring girder and I-beam, as discussed in Section 5.1.3 of WCAP-18554-NP. The staff noted that the steels used in Figure 3-1 of NUREG-1509 are mostly ASTM A212-B material and noted that the material specification for A212-B does not specify copper and nickel content, and one sample material made of A212-B had very low copper and nickel contents compared to those of the T-1 material. Since copper and nickel content are known embrittlement agents in low alloy or carbon steels, and the data in Figure 3-1 of NUREG-1509 are mostly for A212-B that could have low copper and nickel content, discuss the adequacy of Figure 3-1 of NUREG-1509 to predict the ΔNDTT of the T-1 steels of the box ring girder and I-beam.	
6 KH	FE 3.5.2.2.2.7	3.5-41	Topic: Calculation of Fluence and Displacements per iron atom to reactor vessel support components What are the uncertainties associated with the calculated neutron fluence and displacements per iron atom values presented in Table 3.5.2.2-5? How were	Biases and uncertainties can play a critical role in assessing whether calculated critical flaw sizes are appropriately compared to ASME Section XI allowable flaw sizes.

			these uncertainties quantified?	
7 AP	FE 3.5.2.2.2.7	3.5-43	Topic SLRA RV steel support structure OE The SLRA under Operating experience for Unit 1 states that VT-3 inspections of accessible portions of the Unit 1 PBN RV supports were performed in 2010 and those for Unit 2 in 2009. Nextera letter PBNWEC-20-0023, dated April 27, 2020 to Westinghouse states that the Unit 1 PBN RV supports were examined in 2005, 2007, and 2016, while the Unit 2 supports were examined 2006, 2008, and 2005. Discuss and resolve the discrepancy between the SLRA and the PBNWEC letter.	To reconcile inspection dates provided in the application with internal Nextera correspondence with Westinghouse assuring the intended function of the RV steel support structure is maintained during the period of subsequent operation consistent with 10 CFR 54.21(a)(3)
8 AP	WCAP-14422, Rev. 2a	ML010660324	Topic: SLRA RV steel support structure evaluation Westinghouse Report, “Licensing Renewal Evaluation: Aging Management for Reactor Coolant System Supports,” WCAP-14422, Rev. 2a, states that the “applicant must also evaluate that the plant-specific component includes any protective measures assumed in the GTR (coatings, cathodic protection, etc.).” It then states that for loss of material “[t]he effects of corrosion could result in reduced load capacity, strength, or loss of movement between sliding surfaces.”	To assure that effects of corrosion if any, have been properly accounted for in fracture mechanics calculations so that the intended function of the RV steel support structure is maintained during the period of subsequent operation consistent with 10 CFR 54.21(a)(3)

			Clarify whether the effects of aging for loss of material (if any) of the RV support system have been accounted for in the fracture mechanics evaluation subsequent period of extended operation. Reference: Generic Technical Report (GTR) WCAP-14422, "License Renewal Evaluation: Aging Management for Reactor Coolant System (RCS) Supports," Revisions 0 and 1.	
9 AP	EPortal Irradiation of Concrete PBN and SLRA Enclosure 5 Attachment 2		Topic: SLRA RV steel support structure evaluation Nextera letter dated PBNWEC-20-003 states that “the bearing pads are in direct contact with the insulated RPV support bracket … The water passing through the nozzles at the two remaining support bracket locations maybe at ~ 613 ^{o}F at full power operation. The ring beam is not constrained and it's free to expand and contract due to temperature changes; therefore, it is not subject to cyclic thermal expansion stresses.” WCAP-18554-P, Revision 1, “Fracture Mechanics Assessment of Reactor Pressure Vessel Structural Steel Supports for Point Beach Units 1 and 2,” includes several details of the ring-beam/girder,	To assure that the stress calculations and fracture mechanics evaluation are adequate and account for the effects of high temperatures and potential constraining forces so that intended function of the RV steel support structure is maintained during the period of subsequent operation consistent with 10 CFR 54.21(a)(3)

			including its attachments to supporting embedded and partially embedded columns. Given the details presented in WCAP-18554-P, it is not clear whether the ring-girder can freely move in all directions. It is also not clear whether localized forces developed at the support bracket locations due to elevated temperatures are accounted for in box ring-girder (or other relevant structural members) critical flaw sizes reported in SLRA Table 3.5.2.2-6, "Summary of Critical Flaw Sizes for 72 EFPY." By sharing a screen clarify that load transfers from the nozzle brackets would not impose localized forces to the ring girder (or other relevant structural members) that could affect the reported the critical flaw sizes.	
10 AP	FE 3.5.2.2.2.7	3.5-40	Topic: SLRA RV steel support structure evaluation SLRA Table 3.5.2.2-4, "Summary of RPV Support Component Stress Interaction Ratios [IRs]," summarizes the RV support load and load combinations. It shows that the IRs are < 1.0 for the subsequent period of extended operation. A review of Attachment 5, to LAR 261 EPU Licensing Report dated April 2009, list the same IRs for RV loads, load combinations, and presumably same	To clarify and assure that the effects of flaw sizes have been considered on the IRs so that the intended function of the RV steel support structure is maintained during the period of subsequent operation consistent with 10 CFR 54.21(a)(3)

			allowables. Identifying "stability" with flaw tolerance of irradiated RV structural steel components, the SLRA concludes that "the PBN reactor vessel support components continue to be structurally stable. It is not clear whether the controlling modes of failure identified in Table 3.5.2.2-4 of the SLRA (if applicable, i.e. those in tension and shear) have been used for flaw tolerance analysis in WCAP-18554-P, Revision 1 and the critical flaw sizes identified in the in SLRA Table 3.5.2.2-6, "Summary of Critical Flaw Sizes for 72 EFPY". It is also not clear whether the considered flaw sizes in Table 3.5.2.2-6, for all of the RV structural components would modify the IR entries in Table 3.5.2.2-4. Discuss effects of flaw sizes on the IRs reported in SLRA Table 3.5.2.2-4.	

Metal Fatigue of Non-Class 1 Components

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?

1	4.3.3	4.3-8	SLRA Section 4.3.3 addresses the fatigue analysis waiver for non-Class 1 components in accordance with the ANSI B31.3 code. In the SLRA section, Table 4.3.3-2 discusses the projected numbers of equivalent full temperature cycles to demonstrate that the total transient cycle for each evaluated piping system does not exceed the 7000 cycle threshold that allows no reduction to the allowable stress range. Specifically, Table 4.3.3-2 indicates that, in accordance with the design transient cycles specified in UFSAR supplement (SLRA Appendix A) Table 4.1-8, the 7000 cycle threshold is not exceeded by the following systems: (1) feedwater and condensate system; (2) main and auxiliary steam system; (3) reactor coolant system (Non-Class 1); and (4) safety injection system. In comparison, paragraph 102.3.2 of ANSI B31.1 indicates that, if the range of temperature change varies for transient cycles, the equivalent full temperature cycles may be computed by considering the lesser temperature change ratios based on the partial temperature changes of the transient cycles. However, SLRA Section 4.3.3 does not clearly discuss which design transients can contribute to the equivalent full temperature cycles for the evaluated systems in addition to the plant heatup and cooldown transients. Therefore, the contributions of the other transients to the full temperature cycles are not clearly addressed. 1. Clarify which design transients, other than plant heatup and cooldown transients, in USFAR supplement Table 4.1-8 and SLRA Table 4.3.1-1 can contribute to the equivalent full temperature cycles for each of the following systems: (1) feedwater and condensate system; (2) main and auxiliary steam system; (3)	There is a need to clarify which design transients other than plant heatup and cooldown transients can contribute to the equivalent full temperature cycles for the following systems: (1) feedwater and condensate system; (2) main and auxiliary steam system; (3) reactor coolant system, non-Class 1; and (4) safety injection system. In addition, information is needed to clarify how the other transients and their effects on the equivalent full temperature cycles are evaluated in the fatigue wavier analysis.

			reactor coolant system, non-Class 1; and (4) safety injection system. 2. Considering the design transients identified in item 1 and their cycles, confirm that the equivalent full temperature cycles do not exceed the 7000 cycle threshold in each system.	
2	4.3.3	4.3-11	SLRA Section 4.3.3 and Table 4.3.3-2 indicate that the equivalent full temperature cycle number for the feedwater and condensate system does not exceed the 7000 cycle threshold that allows no reduction to the allowable stress range in the stress analysis. SLRA Table 4.3.3-2 states that a specific hot leg sample evaluation was performed for the PBN license renewal for 60 years of operation (first license renewal) and that it resulted in a total of 6702 cycles for 60 years of operation. The SLRA section also states that the critical assumption in the evaluation that would be used for the additional 20 year operating period is thermal cycling of the hot leg sample line occurs only 10 times per year. Accordingly, the applicant estimated additional 200 cycles for the hot line sample line for the subsequent period of operation, resulting in a total number of 6902 cycles for the 80-year operation. During the audit, the staff noted that the technical basis of the significantly reduced thermal cycle frequency (i.e., 10 cycles per year) was descried in the following license renewal document (Reference: LR-TR-516, Estimate of Thermal Cycles for the RCS Hot Leg Sample Line). The document indicates that PBN Unit 1 conservatively represents the thermal cycles of the hot leg sampling lines for both units at PBN. The reference also states that the hot leg sampling line is	Additional information is needed to confirm the cycle projections are supported by the recent operating experience.

			the most limiting sampling line in terms of thermal cycles. In addition, the reference document explains that the installation of a post-TMI reactor monitoring system at PBN changed the hot leg sampling line operation from off-on-off flow approach to continuous flow approach and that the installation is conservatively assumed to have occurred in 1983. Based on the operational change of the hot leg sampling line, only 10 thermal cycles per year is assumed after the installation of the new reactor monitoring system. However, the SLRA and reference document do not clearly discuss whether the assumed annual cycle number (10 cycles per year) is supported by the current operating experience regarding the thermal cycles of the hot leg sampling line. Clarify whether the current operating experience regarding the thermal cycles of the hot leg sampling line supports the assumed annual cycle number (10 cycles per year). If not, provide the updated thermal cycle estimation for the hot leg sampling line based on the observed thermal cycles of the piping system.	

Section 4.3.2 ASME Code, Section III, Class 1 Component Fatigue Waivers

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	4.3.2	4.3-8	SLRA Section 4.3.2 addresses the fatigue analysis waivers for the following ASME Code, Section III, Class 1, steam generator components: (1) shop installed weld tube plugs; (2) ribbed mechanical tube plugs; and (3) tube wall undercut. The SLRA section explains that these components do not require a detailed fatigue analysis in accordance with the fatigue exemption provisions in N-415-1 of ASME Code, Section III, 1965 Edition through the summer 1966 Addenda. SLRA Section 4.3.2 also indicates that the design transients for these components are consistent with the design transients identified in SRLA Table 4.3.1-1. The SLRA section further discussed that, since the original 40-year design cycle numbers are bounding for the 80-year allowable transient cycle numbers, the fatigue exemption for the SG components remain valid for 80 years of operation. However, SLRA Section 4.3.2 does not clearly describe the specific fatigue exemption provisions that involve the time-limited assumptions of the fatigue TLAA for the SG components. The staff also found a need to confirm that the transients evaluated to determine the fatigue analysis wavier are included in SLRA Table 4.3.1-1. 1. Describe the fatigue exemption provisions that specifically involve the time-limited assumptions of the fatigue TLAA for the SG components in order to demonstrate that the associated fatigue exemption criteria continue to be met for 80 years of operation. 2. Clarify whether the transients of the SG components that are evaluated in the fatigue exemption analysis are represented and bounded by those in SLRA Table 4.3.1-1.	SLRA Section 4.3.2 does not clearly describe the fatigue exemption provisions and their acceptance criteria that are applied to the time-limited assumptions of the fatigue TLAA for SG components. There is a need to confirm that all the transients evaluated to determine the fatigue analysis wavier are included in SLRA Table 4.3.1-1.

Section 4.3.4 Environmentally Assisted Fatigue (EAF)

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	4.3.4	4.3-12	SLRA Section 4.3.4 addresses the EAF TLAA. GALL-SLR AMP X.M1, “Fatigue Monitoring,” and SRP-SLR Section 4.3.2.1.2 recommend that an EAF TLAA evaluate the component and piping locations identified in NUREG/CR-6260. SLRA Section 4.3.4 includes EAF evaluations for most of the sample locations addressed in NUREG/CR-6260 for the Westinghouse-designed reactor. However, the SLRA does not clearly address the following locations identified in NUREG/CR-6260: (1) pressurizer surge line piping locations other than the reactor coolant loop surge line nozzle and pressurizer vessel surge line nozzle; and (2) RHR system Class 1 piping locations other than the RHR tee and accumulator safe injection nozzle locations. Specifically, the existing fatigue analysis for the pressurizer surge line described in WCAP-13509 indicates that the maximum CUF value is estimated at a surge line piping location but not at the reactor coolant loop surge nozzle or the pressurizer vessel surge nozzle. However, the SLRA does not clearly address this location identified in WCAP-13509 in the evaluation of the NUREG/CR-6260 locations. In addition, SLRA Section 4.3.4 does not clearly discuss whether the RHR tee and accumulator safe injection nozzle locations are the lead EAF locations for the RHR piping system. 1. Clarify whether the nozzle locations of the pressurizer surge line are more limiting than the surge line piping locations in the EAF analysis. As part of the discussion, discuss the EAF evaluation results for the pressurizer surge line piping locations compared to	Additional information is needed to clarify whether the nozzle locations of the pressurizer surge line are more limiting than the surge line piping locations in the EAF analysis. In addition, addition discussion is needed on how the RSR tee and accumulator safety injection nozzle location (nozzle to the reactor coolant loop) are the lead locations in the EAF analysis for the RHR system class 1 piping.

			those for the nozzle locations (i.e., reactor coolant loop surge line nozzle and pressurizer vessel surge line nozzle). 2. Clarify whether the RHR tee and accumulator safe injection nozzle locations are the lead EAF locations for the RHR system Class 1 piping. If not, explain why the other piping locations of the RHR piping are not evaluated in the EAF TLAA.	
2	4.3.4	4.3-14	SLRA Section 4.3.4 indicates that the original code of record for the PBN piping systems is ANSI B31.1. Since the start of the operation, some piping systems were additionally analyzed in accordance with the fatigue analysis provisions in ASME Code, Seciton III. SLRA Section 4.3.4 explains that the environmental CUF (CUF_{en}) values for the screening purpose were determined by using conservative input values (e.g., conservative stresses, strain rates, and water chemistry conditions) when the CUF_{en} calculation method in NUREG/CR-6909, Rev. 1 is applied. If the screening CUF_{en} values are determined to be less than 1.0, the corresponding locations are screened out with no further evaluations to be performed. The screening CUF_{en} value is equal to the acceptable criteria for CUF_{en} (1.0) and is, therefore, very high. Accordingly, the staff found a need for additional information to confirm that the level of conservatism associated with the screening CUF_{en}	Additional information is needed to confirm that the screening CUF_{en} calculations involve sufficient conservatism given that the threshold for the screening CUF_{en} is equal to the CUF_{en} design limit (1.0).

			calculations is sufficiently high to use the very high threshold (1.0) for screening CUF_{en}. Discuss the level of conservatism associated with the screening CUF_{en} values. As part of the discussion, if available, discuss the approximate value of the refined final CUF_{en} that may correspond to the threshold value (1.0) of screening CUF_{en} to demonstrate that the screening CUF_{en} calculations involve sufficient conservatism as a screening parameter for the EAF analysis.	
3	4.3.4	4.3-16	SLRA Section 4.3.4 provides a subsection that addresses the further evaluation of ASME Code, Section III components to be done after EAF screening. The subsection refers to the application of more appropriate fatigue strength reduction factors (FSRFs) as part of the discussion on more realistic refinement of CUF_{en} values. It is not so clear how the FSRFs are used to refine the CUF_{en} values by eliminating excessive conservatism on CUF_{en}. Clarify how the FSRFs are used to refine the CUF_{en} values.	It is unclear how the FSRFs are used to refine the CUF_{en} values by eliminating excessive conservatism.

4	4.3.4	4.3-16	GALL-SLR AMP X.M1, "Fatigue Monitoring," and SRP-SLR Section 4.3.2.1.2 recommend that an EAF analysis include the evaluation of the sample locations identified in NUREG/CR-6260 and that, if plant-specific locations are more limiting than those sample locations, the plant-specific location should be also evaluated in the EAF TLAA. SLRA Section 4.3.4 explains that the piping systems of PBN were originally designed and constructed in accordance with the ANSI B31.1 code. After the commence of the operation, some piping systems were evaluated in accordance with the fatigue analysis provisions of ASME Code, Section III. The SLRA also indicates that the sample locations specified in NUREG/CR-6260 and the additionally identified pressurizer spray piping are evaluated in the EAF TLAA to address the piping systems that were designed per ANSI B31.1 but not analyzed per ASME Code, Section III. The applicant did not perform a detailed screening analysis for the ANSI B31.1 piping systems. However, the applicant used the previous EAF screening results of similar Westinghouse-designed reactors as the basis of selecting the NUREG/CR-6260 locations and pressurizer spray piping for further EAF evaluation. Therefore, for the piping systems designed per ANSI B31.1 but not analyzed per ASME Code, Section III, the staff found a need for additional information to confirm the adequacy of the applicant's approach that the EAF TLAA does not evaluate plant-specific locations other than the sample locations of NUREG/CR-6260 and the pressurizer spray piping. 1. Among the piping systems or lines equivalent to ASME Code, Class 1, describe the systems or lines that meet the following criteria: (1) were designed	For the piping designed for ANSI B31.1 and not analyzed per ASME Code, Section III, no plant-specific locations are analyzed in the applicant's EAF TLAA in addition to the sample locations of NUREG/CR-6260 and pressurizer spray piping. The main basis of selecting these locations is the previous EAF screening results of similar Westinghouse-designed reactors. Therefore, additional information is needed to confirm the adequacy of the applicant's approach for selecting the lead locations.

			per ANSI B31.1; (2) have not been analyzed for EAF per ASME Code, Section III and NUREG/CR-6909, Revision 1; and (3) are not represented by the NUREG/CR-6260 sample locations or the pressurizer spray piping in the EAF analysis. 2. If any piping systems or lines are identified in the evaluation above, discuss how EAF is addressed for the piping systems of lines for the subsequent period of extended operation. As part of the discussion, clarify why reactor coolant pumps are not addressed in the EAF TLAA in SRLA Section 4.3.4.	

5	4.3.4	4.3-18 4.3-24	SLRA Section 4.3.1 addresses the ASME Section XI, Appendix L analysis for the pressurizer spray piping. The detailed analysis is provided in the following document (Reference: Westinghouse LTR-SDA-20-064-NP, Revision 1, "ASME Section XI Appendix L Evaluation Results for the Point Beach Units 1 and 2 Pressurizer Spray Nozzles," October 7, 2020, as submitted with the SLRA). The SLRA and the reference document indicate that the two analysis locations (ASN 3 and ASN 2) were considered in the Appendix L fracture mechanics analysis. The reference document also provides the following information: ASN 3 (safe end) was identified as the fatigue sensitive location on the pressurizer spray nozzle with respect to environmental fatigue. However, the ASME Code, Section XI inspection program considers only the safe end to pipe weld region (ASN 2); therefore, in order to justify flaw tolerance of ASN 3 (fatigue sensitive location), which is not within the inspection zone, the Appendix L fracture mechanics needs to evaluate and justify ASN 2 (inspectable weld location) to bound ASN 3 in the Appendix L analysis. SLRA Table 4.3.4-2, which summarizes the Appendix L analysis results, shows that the ASN 2 location involves in the more limiting inspection frequency based on the higher fatigue crack growth. For the liming crack orientation (axial orientation), the ANS 2 location is estimated to have an allowable operating period of 12 years, compared to 53 years for the ANS 3 location. The staff found a need to clarify which location is the more limiting in terms of CUF_{en} (fatigue crack initiation) between the ASN 3 (safe end) and ANS 2 (weld) locations. 1. Clarify which location is the more limiting between the ANS 3 and ANS 2 locations	Additional information is needed to clarify whether the ANS 3 location (safe end) is the more liming than the ANS 2 location (safe end weld) in terms of the EAF (i.e., CUF_{en} representing crack initiation). In addition, clarification is needed on whether an initial baseline inspection will be performed on the ANS 3 (safe end) location to confirm the absence of an unacceptable flaw. Clarification is also need on when an initial inspection will be performed on the ANS 2 location that will be periodically inspected in the ISI program. There is a need to confirm that the ANS 3 location will be also inspected if a flaw is detected at the ANS 2 location.

			in terms of CUF_{en} (fatigue crack initiation). As part of the discussion, provide the CUF_{en} values for these locations at 60 years of operation entering the subsequent period of extended period. 2. Appendix L, L-3410(a) of ASME Code, Section XI requires that the absence of any flaw larger than the applicable acceptance standard referenced in Table IWB-3410-1, at the location of concern, shall be verified by surface or volumetric examination. Since the ANS 3 location is not a location that are periodically inspected, clarify whether the verification discussed above will be conducted for the ANS 3 (safe end) location prior to entering the subsequent period of extended operation (SPEO; e.g., during the 6 years before the SPEO). If not, discuss why such verification is not necessary.	

6	4.3.4	4.3-18 4.3-24	With respect to the ASME Code, Section XI, Appendix L analysis for the pressurizer spray piping, SLRA Table 4.3.4-2 lists the values of the Appendix L calculated aspect ratio. Additional information is needed to clarify how these values are used in the analysis for crack growth and flaw tolerance. 1. Describe how these crack aspect ratios are calculated. In addition, the following reference uses the term, "maximum calculated aspect ratios" (Ref.: Westinghouse LTR-SDA-20-064-NP, Revision 1). Explain the meaning of the term, maximum, that is used to describe the crack aspect ratios. 2. Clarify whether these aspect ratio values are used to define both the initial postulated crack sizes (ASME Code, Section XI, Appendix L, L-3211) and the end-of-evaluation allowable crack sizes (Appendix C, C-5000 and C-6000). If not, discuss how the aspect ratios are determined for the analyses specified in Appendix L, L-3211 (initial postulated cracks) and Appendix C, C-5000 and C-6000 (final allowable cracks).	Additional information is needed to clarify how the crack aspect ratios are calculated and how the calculated aspect values are used in the postulation of the initial cracks and the determination of the allowable crack sizes.
7	4.3.4	4.3-18 4.3-24	Section 3.3 of Westinghouse LTR-SDA-20-064-NP, Revision 1 addresses the welding residual stress for fatigue crack growth analyses. Clarify whether the welding residual stress is applied to both the ANS 2 (safe end weld) and ANS 3 (safe end) locations. If not, explain the basis for why the weld residual stress is not applied to both locations.	Clarification is needed to whether the weld residual stress is applied to both the ANS 2 and ANS 3 locations.
8	4.3.4	4.3-18 4.3-24	Section 2.5 of the following document indicates that the representative reference plant F_{en} values of the pressurizer components for carbon steel and stainless steel locations were calculated in accordance with Revision 0 and draft Revision 1 of NUREG/CR-6909, respectively (Reference: LTR-SDA-II-20-13-P, Environmentally Assisted Fatigue Evaluation Results for the Point Beach Unit 1 and Unit 2 Pressurizer Lower Head for Subsequent License	Additional information is needed to clarify which version of NUREG/CR-6909 is used to determine the F_{en} and fatigue design curves for the PBN pressurizer components. In addition, there is a need to confirm that the F_{en} calculation follows the

			Renewal). The document also indicates that the F_{en} equations for stainless steel from Revision 1 of NUREG/CR-6909 are bounded by the equations in draft Revision 1. In contrast, Section 2.5.1 of the reference document states that, for Point Beach Nuclear Plant (PBN), the F_{en} for each of the carbon steel and stainless steel location was calculated by using the equations in Revision 1 of NUREG/CR-6909. This discussion suggests that the reference plant evaluation mentioned above were modified for PBN by using the F_{en} per NUREG/CR-6909, Rev. 1. Therefore, the staff found a need to clarify which version of NUREG/CR-6909 is used to determine the F_{en} and fatigue design curves. In addition, GALL-SLR AMP X.M1 recommends that, if NUREG/CR-6909, Rev. 0 is used, the average temperature should be used in accordance with the clarification in Rev. 2 (e.g., page A-2 of the NUREG). This aspect needs to be clarified, too. 1. Clarify which revision of NUREG/CR-6909 is used to determine the F_{en} and fatigue design curve for the carbon and stainless steel locations of the pressurizer components in the PBN EAF analysis. 2. If Revision 0 of NUREG/CR-6909 is used, clarify whether the F_{en} calculation follows the clarification and recommendation regarding the use of the average temperature (e.g., guidance on page A-2), consistent with GALL-SLR AMP X.M1.	clarification on the average temperature, as provided in Rev. 1 of NUREG/CR-6909 (e.g., on page A-2), when the guidance in Rev. 0 is used.

Section 4.3.1 Metal Fatigue of Class 1 Components

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	4.3.1	4.3-5 4.3-7 B-26	Note: This item is also related to the Fatigue Monitoring Program. SLRA Section 4.3.1, in part, addresses the fatigue analysis for reactor vessel internal (RVI) components. The section indicates the following: (1) Table 2.2.3-3 of the PBN EPU license amendment request provides a summary of the CUFs for the RVI components; (2) All CUFs are below the acceptance criteria of 1.0; and (3) As shown in Table 4.3.1-1, the 40-year design cycles (CLB cycles) bound the projected cycles for the 80-year SPEO. However, the operating experience evaluation in SLRA Section B.2.2.1 explains that, after the EPU project, a Westinghouse corrective action was taken to use correct fatigue curves in the CUF calculations for the baffle former bolts. The section also indicates that, as a result of the updated fatigue analysis, some of allowable design cycle values were reduced. The reduced allowable design cycle values for the RVI baffle former bolts are described in Table 4.1-8 of SLRA Appendix A, UPSAR supplement. However, the reduced design cycle values are not clearly reflected in Table 4.3-1 or in SLRA Section 4.3.1. Clarify why SLRA Table 4.3-1 and Section 4.3.1 do not address the reduced design cycle numbers for the RVI baffle former bolts in comparison to the less limiting 40-year design cycle numbers.	The staff needs to clarify why SLRA Section 4.3.1 and Table 4.3.1-1 do not address the reduced allowable transient cycles for the RVI baffle former bolts in comparison with the less limiting 40-year design cycles.
2	4.3.1	4.3-7	SLRA Table 4.3.1-1 describes the allowable 80-year transient cycle numbers in comparison with the original 40-year design cycle numbers for the applicant's fatigue analyses. For the "control rod drop," "excessive FW [feedwater] flow," and OBE [operating basis earthquake] transients, SLRA Table 4.3.1-1	There is a need to clarify the following: (1) the basis of the 80-year allowable cycle numbers for "control rod drop," "excessive FW flow," and OBE transients;

			does not identify existing design allowable cycle numbers. However, the table lists specific 80-year allowable cycle numbers for these transients (table items 32, 33 and 34). The staff needs to clarify the basis for the 80-year allowable cycles. For the “feedwater cycling at hot standby” and “boron concentration equilibrium” transients, the existing design allowable cycle numbers are described as 2000 and 23360 respectively in SLRA Table 4.3-1. However, the table does not provide the 80-year allowable design cycles for these transients (table items 26 and 27). The staff needs to clarify why 80-year allowable cycles are not specified. 1. Describe the basis for the 80-year allowable cycles for the “control rod drop,” “excessive FW flow,” and OBE transients. 2. Explain why SLRA Table 4.3.1-1 does not identify the 80-year allowable cycle numbers for the “feedwater cycling at hot standby” and “boron concentration equilibrium” transients as part of fatigue monitoring activities.	and (2) why SLRA Table 4.3-1 does not identify 80-year allowable cycle numbers for the “feedwater cycling at hot standby” and “boron concentration equilibrium” transients.

3	4.3.1	4.3-6	SLRA Section 4.3.1 states that the cumulative usage factor (CUF) for the pressurizer spray piping has been projected to the end of the subsequent period of extended operation. The SLRA section also indicates that stratification cycles are conservatively projected based on thermocouple data with a leaking spray control valve that is assumed to leak throughout 80 years of plant operation. In addition, SLRA Section 4.3.1 explains that the fatigue analysis for the piping results in a CUF value of 0.369 for 80 years of operation. The section states that, due to the conservatism applied to this analysis, cycle monitoring is not required for the piping. Discuss how the previous inspections and future planned inspections support that cycle monitoring is not needed when considering that this location is sensitive to fatigue and thermal stratification.	Addition information is needed regarding the inspection results and activities that are applied to the pressurizer spray piping to support the conservatism of the fatigue analysis.

Section B.2.2.1: Fatigue Monitoring Program
(note: some items are related to the Fatigue TLAAs)

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	B.2.2.1	B-22	Note: This question is applied to the Fatigue Monitoring Program and the fatigue TLAA in SLRA Section 4.3.1. SLRA Section B.2.2.1 states that SLRA Table 4.3.1-1 identifies the design cycles utilized in the component fatigue analyses and concludes that the projected cycles through the SPEO will not exceed the design cycles assumed in the analyses. SLRA Table 4.3.1-1 is provided in SLRA Section 4.3.1 that, in part, addresses the applicant’s fatigue analyses for reactor vessel internal (RVI) components, including those that were evaluated in the previous EPU project such as core plates and upper support plates. In a similar manner, the fatigue transients and their allowable cycle numbers are provided in SLRA Appendix A, UFSAR supplement, Table 4.1-8. Generally speaking, the UFSAR	Clarification is necessary on the allowable transient cycles for the RVI baffle former bolts. Additional information is also needed to reconcile the difference between SLRA Table 4.3.1-1 and SLRA Appendix A, Table 4.1-8.

			supplement table in SLRA Appendix A is less comprehensive than SLRA Table 4.3.1-1 in terms of the design transients identified in the tables. For example, SLRA Appendix A, UFSAR supplement, Table 4.1-8 does not identify the accumulator safety injection, loss of charging flow, loss of letdown flow, or pressurizer heatup transient as a design transient, while these transients are included in SLRA Table 4.3.1-1. However, SLRA Appendix A, UFSAR supplement, Table 4.1-8 specifically identifies the more limiting allowable cycle numbers for RVI baffle bolts (also called baffle former bolts) in addition to the general limits to the design transient cycle numbers that are applied to the other RVI and piping components. The specific allowable cycle numbers for the baffle former bolts are not described in SLRA Table 4.3.1-1. For example, SLRA Appendix A, UFSAR supplement, Table 4.1-8 indicates that the allowable cycle number of the "plant loading at 5% of full power per minute" transient for the baffle former bolts is 2485 cycles compared to 11600 cycles for the other components. In contrast, SLRA Table 4.3.1-1 lists only 11600 cycles as the allowable cycle number for the transient without a specific note for the baffle former bolts. 1. Reconcile the difference between SLRA Table 4.3-1 and SLRA Appendix A, Table 4.1-8 regarding the allowable transient cycles for the RVI baffle former bolts. 2. Explain why the design transients listed in SLRA Table 4.1.3-1 and SLRA Appendix A, Table 4.1-8 are different. If the difference cannot be justified, identify a consistent design transient table for both SLRA Section 4.3.1 and SLRA Appendix A.	

2	B.2.2.1	B-26	The "monitoring and trending" program element of GALL-SLR AMP X.M1, in part, indicates that the monitoring of plant operating conditions or water chemistry parameter conditions is performed to provide inputs for components with stress-based fatigue calculations or environmental fatigue calculations. In comparison, NextEra used the FatiguePro computer program to perform stress-based fatigue monitoring and analysis for certain ASME Code Class 1 and 2 components, as described in Structural Integrity Associates, Inc. (SIA) Calculation Package No. 2000088.303, Revision 0. The SIA report and implementing procedure (NP 7.7.19, Attachment B) for fatigue monitoring indicate the stress-based method is used to monitor and evaluate the fatigue for certain Class 1 and 2 components such as steam generator feedwater nozzles, pressurizer shells and hot leg nozzles for pressurizer surge lines. The method monitors operating conditions such as flow rates and temperatures. SLRA Section B.2.2.1 also indicates that the applicant contracted with SIA to review the FatiguePro program and received an updated report that incorporated FatiguePro data through 2019 for 80-year life projections. The SLRA further states that, going forward, the applicant will no longer use FatiguePro and instead will use the manual count method. In addition, SLRA Section B.2.2.1 and implementing procedure for fatigue monitoring do not clearly discuss how frequently the monitored cycles and data are evaluated to confirm the fatigue CUF values meet the acceptance criteria. 1. Discuss how the stress-based fatigue monitoring will be performed by using the manual count method that is planned to be implemented for the subsequent period of extended operation. 2. Clarify how frequently the monitored fatigue cycles and related operating conditions are evaluated to confirm that the fatigue CUFs meet the acceptance criteria. In addition,	Additional information is needed to clarify how the stress-based fatigue monitoring if the applicant does not continue to use the existing fatigue monitoring computer program but does use only the manual method to monitor and evaluate the fatigue cycles and conditions. In addition, information is needed on the frequency of the fatigue cycle and operating condition evaluation and its basis.

			discuss the basis for the frequency of the periodic fatigue evaluations.	
3	B.2.2.1	B-24	SLRA Section B.2.2.1 specifies a program enhancement to the “Parameters Monitored or Inspected” program element, which updates the AMP governing procedure to monitor the chemistry parameters that provide inputs to F_{en} factors used in CUF_{en} calculations. In the enhancement, these chemistry parameters include dissolved oxygen and sulfate and are controlled and tracked in accordance with the PBN Water Chemistry AMP. The Fatigue Monitoring program is an existing program and the environmentally assisted fatigue (EAF) analysis is also included in the existing fatigue analyses. Given that environmental effects are already evaluated in the existing Fatigue Monitoring program and associated fatigue analyses, clarify why this program enhancement is needed to monitor the water chemistry parameters.	Clarification is needed on why the monitoring of the chemistry parameters is identified as a program enhancement even though the Fatigue Monitoring program is an existing program and the environmental effects are evaluated in the exiting fatigue analyses.

Section B.2.3.7 PWR Vessel Internals

#	SLRA Section(s)	SLRA Pages	Question / Issue	Why are we asking?
1	Table 3.1.2-2	3.1-75 – 3.1-86	The staff would like to discuss the following aging management review (AMR) line items in SLRA Tables 3.1.1 and 3.1.2-2 involving PWR reactor vessel (RVI) internal components that are within the scope of license renewal and subject to an AMR: • Note J Items for the clevis wear surfaces and upper core plate alignments pins (SLRA pages 3.1-75 and 3.1-76)	ISG now includes GALL-SLR line items for the clevis wear surface and upper core plate alignment pins.
			• Line items for core barrel outlet nozzles on SLRA page 3.1-80 and core barrel upper girth welds on SLRA page 3.1-82 (i.e., in general, the staff would like to discuss the AMR line items for Expansion components linked to the core barrel upper flange welds as the designated Primary components)	Based on the AMR items for the core barrel outlet nozzles in SLRA Table 3.1.2-2, staff needs clarification which inspection category is being applied to the outlet nozzles and which inspection category is being applied to the core barrel upper girth welds.
			• Line items for lower support column bodies at the bottom of SLRA page 3.1-77 – Basis why loss of fracture toughness (as linked to irradiation embrittlement) is not included among the aging effects listed in the line item for "loss of material" of the column bodies	EPRI MRP cites irradiation embrittlement (IE) as an applicable aging mechanism for lower support column bodies in Item W4.4 of Table 4-6 in MRP-227, Rev. 1-A.
			• Applicant's response bases to Table 1 AMR Item 3.1.1-028 and 3.1.1-087 in SLRA Table 3.1.1	Staff has questions on the non-applicability bases for 028 and 087.
			• For the Table 3.1.2-2 AMR line items on the CRGT lower flange welds on SLRA pages 3.1-78 and 3.1-79 , why don't the line items distinguish between the flange welds in the peripheral CRGT assemblies as "Primary" category components versus those in the non-peripheral CRGT assemblies as "Expansion" category components? The SLRA AMR item setup for the CRGT lower flange welds (LFWs) seems to	The way the SLRA sets the AMRs up for the CRGT LFWs is not quite consistent with the way they are treated in MRP-227, Rev. 1-A.

			treat all of the CRGT (LFWs as EPRI MRP "Primary" category components. However, MRP-227, Rev. 1-A only identifies that the CRGT LFWs in the peripheral assemblies are "Primary" category components. The CRGT LFWs in the non-peripheral assemblies are "Expansion" category components.	
2	SLRA Appendix C, Table C.A1	C-22	*MRP-based Inspection Category for the Point Beach internals hold-down springs.* SLRA Table C.A1 states in that the hold-down springs at PBNP Units 1 and 2 are made from Type 403 stainless steel, and that the springs are placed in the EPRI MRP "No Additional Measures" category. Document LR-AMP-015-RVINT indicates that the Unit 2 hold-down spring are made from quenched and tempered Type 403 martensitic stainless steel and that the measurements of the spring were sufficient for prior 60-year basis. Staff seeks clarification whether the Unit 1 hold-down spring is made from the same material and whether the past measurement inspection of the Unit 1 hold-down spring yielded the same result (i.e., no change in the configuration of the spring from loss of material, loss of preload, or changes in dimension). *Reference(s)*: SLRA Appendix C, Table C.A1, Item W8; Basis Document No. LR-AMP-015-RVINT	Necessary for staff confirmation that the Unit 1 and Unit 2 hold-down springs are both made from Type 403 stainless steel and can be placed in the "No Additional Measures" category.
3	B.2.3.7	B-69 – B-77	*WCAP-17096 as a supplemental methodology for the Reactor Vessel Internals Program.* In general, the applicant should be prepared to confirm that it uses WCAP-17096-NP as a supplemental methodology for SLRA AMP B.2.3.7, "Reactor Vessel Internals," and that Rev. 2-A is the current version of the WCAP being applied to the program. If	GALL-SLR AMP XI.M16A, "PWR Vessel Internals," as updated in SLR-ISG-2021-01-PWRVI (ML20217L203) allows supplemental methodologies to be used in addition to MRP-227, Rev. 1-A. If WCAP-17096 is used

			Rev. 2-A is the current version of the WCAP being applied to the program, applicant should be prepared to discuss how it will address or resolve those applicant/licensee actions items identified by the staff its safety evaluation for Rev. 2 of the WCAP report (i.e., SE dated May 3, 2016; ADAMS Accession No. ML16061A243). *Reference(s)*: AR No. 02242495-1; AR No. 02242495-2	as a supplemental methodology, the report may include staff-issued actions items on use of the WCAP methodology.
4	B.2.3.7 And Appendix C	B-69 – B-77 C-21 – C24	Projected neutron fluence exposure estimates for Point Beach RVI component-specific locations that are evaluated in MRP-227, Rev. 1-A and comparisons to the EPRI MRP fluence band estimates for those components in MRP-2018-022. Applicant should be prepared to discuss the methodology (according to the basis document, it is WCAP-18124-NP-A, Rev. 0 [ADAMS ML18204A010]) that it applied for development of the 80-Year Point Beach RVI component-specific fluence levels in support of the gap analysis provided in SLRA Appendix C, and how those 80-year component-specific fluence projections compared to the 80-year RVI component-specific fluence bands and categories developed by the EPRI MRP for the 80-year assessment. *Reference(s)*: AR No. 02245034-1;	Staff seeks verification that 80-Year component-specific fluence projections for the RVI components are within fluence bands established by the EPRI MRP for the components.
5	B.2.3.7 And Appendix C	B-69 – B-77 C-21 – C24	Subsequent renewed I&E bases (going forward) for RVI components with known operating experience or that may be subject to site-specific programmatic criteria. The staff's primary focus is on NextEra Energy's inspection and evaluation (I&E) bases for the RVI components with known specific operating experience (OpE) in the rows for Item 5 that follow.	

			Baffle-to-Former Bolts (BFBs). : • Briefly summarize the past inspection results for "Primary inspections performed on the BFBs (e.g. bolt with worst case through-wall cracking result, total number of failed bolts, number of degraded bolts in clustered areas) and whether the past inspections results of the BFBs prompted the need for performing "Expansion" inspections of the lower support column bolts (as the Primary Expansion Component for the BFBs) or barrel-to-former bolts (as the Secondary Expansion Component for the BFBs). If "Expansion" category inspections were necessary, clarification on whether the "Expansion" inspections have been performed to date (and the associated results of the "Expansion inspections) or the schedule for the future inspections of the "Expansion" components (if the Expansion inspections have yet to be performed). As part of the summary, identify any interim guidelines currently being applied to the BFB re-inspection basis. • Clarify whether the flaw evaluation referenced in AR No. 02442495-03 has been performed to support reinspection of the BFB's on a 10-Year augmented ISI interval basis. If so, the staff requests that the evaluation be posted on the audit portal. *References*: SLRA Appendix C, Table C.A1, Item W6; WO No. 40201462-01; WO No. 40201463-01; AR No. 02193700; AR No. 02442495-03; AR No. 02245034-02	High profile operating experience component for PWR RVI management program. Staff seeks validation that the BFBs can be re-inspected (using UT) on a 10-Year augmented ISI basis during the SPEO.
			Control Rod Guide Tube (CRGT) Assembly Guide Plates (Guide Cards): • Summary of inspection results of past inspections on CRGT guide cards at Unit s1 and 2 (e.g., number of guide cards with wear indications in each	Another component with generic OpE. Staff seeks validation of the inspection and evaluation basis for performing visual re-

			unit, %wear in worst case guide card with wears) • Frequency for re-inspection of the CRGT guide cards for future operations, including those during the SPEO. • Clarification on whether any of the following interim guidelines are being applied to CRGT guide card inspections: (1) MRP 2018-007, (2) PWROG OG-18-46, (3) NSAL-17-1 • Clarification whether Westinghouse Class 2 Proprietary Report WCAP-17451 is being applied as the basis for CRGT guide card examinations, and if so, the version of the WCAP that is being used for the inspections. Clarification whether the inspections of the CRGT guide cards will expand to the **[ ]** if the worst-case as-found degree of wear exceeds the acceptance criterion for this in the WCAP. *NRC, SLRA and Audit Portal References*: SLRA Appendix C, Table C.A1, Item W1; AR No. 01953761; AR No. 02165156; AR No. 02253333	inspections of the CRGT guide cards during the SPEO.
			CRGT Support Pins (Split Pins): • Identification of the material of fabrication, replacement status, and EPRI MRP (MRP-227) inspection category for the CRGT split pins during the SPEO. *NRC, SLRA and Audit Portal References*: NUREG-2192, Volume 1 (GALL-SLR Volume 1); NUREG-2192 (SRP-SLR); SLR-ISG-2021-01-PWRVI	The staff needs the information for confirmation on whether the CRGT split pins can be placed into the “No Additional Measures” category of components.
			Clevis Insert Assemblies (Including Dowels; Bolts, and Surfaces): • Summarize the inspection results for the most recent past VT-3 visual inspections on clevis insert assembly bolts, dowels, and wear surfaces in Units1 and 2 (e.g., number	To address generic OpE with clevis insert assemblies and its potential applicability to the Point Beach Reactor Vessel Internals Program, including potential

			of components with crack or wear indications in each unit, %wear or cracking in worst case component with known flaw or wear indications). Based on past inspection results, identify the frequency that will be used to perform re-inspections of the clevis insert assembly bolts, dowels, and wear surfaces during the SPEO. • Clarify whether the past inspections of the assemblies have revealed any evidence of distortion occurring in a clevis insert assembly at Point Beach Unit 1 or 2. *NRC, SLRA and Audit Portal References*: SLRA Appendix C, Table C.A1, Items W10 and W14	applicability of a recent report of distortion that was detected in a U.S. clevis insert assembly. In spite of the generic operating experience, staff seeks confirmation that the clevis insert assemblies and their components can still be re-inspected on a 10-year basis.
			Core Barrel Middle Axial Welds and Lower Axial Welds: • Clarification on whether the interim guidance in MRP-2019-009 has been included as a supplemental report for the Reactor Vessel Internals Program, and if so, whether the one-time VT-3 examinations of the core barrel middle axial welds (MAWs) and lower axial welds (LAWs) at Units 1 and 2 have been performed. ➢ If the one time VT-3 inspections of the MAWs and LAWs have been performed, the date of the examinations and a summary on whether the inspections revealed any evidence of cracking occurring in the core barrel MAWs and LAWs. ➢ If one-time VT-3 inspections of the welds will be performed but have yet to be performed, the approximate schedule for performing the one-time VT-3s of the MAWs and LAWs (i.e. if the	Generic OpE with cracking of MAWs and LAWs in the core barrel of one of the St. Lucie units. Application of MRP-2019-009

			schedule has been established by NextEra Energy). • Applicant should be prepared to discuss its resolution of RAI B2.1.7.3, Parts 1 and 2, in Section 3.4 of Report FPLCORP00036-REPT-046 (as posted in the audit portal), and discuss whether achievement of a reduced weld coverage (< 75% weld coverage) for potential EVT-1 expansions to the core barrel the MAWs and LAWs needs to be identified as an "Exception" to the "Detection of Aging Effects," "Monitoring and Trending", and "Acceptance Criteria" elements in GALL AMP XI.M16A, "PWR Vessel Internals," particularly if the maximum achieved weld coverage for any one axial weld will be less than 50% weld coverage (refer to obstruction and weld coverage statements in Notes 5 and 6 in Table 4-6 of MRP-227, Rev. 1-A). *NRC, SLRA and Audit Portal References*: SLRA Appendix C, Table C.A2, Items W4.2 and W4.3; AR No. 02235758 (summarizing the St. Lucie OpE); AR No.02321810	
			Fuel Alignment Pins (in Upper and Lower Assemblies) and Radial Support Keys: • Summary on whether past EPRI or ASME Section XI defined VT-3 visual inspections of the fuel pins and radial support keys have revealed any evidence of wear or cracking occurring in the pins or keys *References*: SLRA Appendix C, Table C.A3, Items W11, W15, W16, and W17	Confirmation that here has been no significant cracking or occurrences of wear in the fuel alignment pins and that re-inspections are justified on a 10-Year basis.
			Thermal Shield Flexures: • Summary of inspection results of past inspections on thermal shield flexures at Unit s1 and 2 (e.g., number of flexures with cracking or wear	Confirmation that significant cracking of the thermal shield flexures has yet to be detected in the thermal shield flexures

			indications in each unit, % cracking/wear in worst-case flexure with indications). Discuss basis why future VT-3 inspections of the thermal shield flexures can continue to be performed on a 10-Year re-inspection frequency per Item W9 in SLRA Table C.A1. • Is Westinghouse Technical Bulletin (TB) 19-5 being applied as part of the basis for the thermal shield inspections? *NRC, SLRA and Audit Portal References*: SLRA Appendix C, Table C.A1, Item W1; AR No. 02253333	of Point Beach Units 1 and 2 and that re-inspections of the thermal shield flexures are still justified on a 10-Year basis.
6	SLRA Appendix C, including Tables C.A1, C.A2, and C.A3	C-1 – C-28	Reports for Past Inspections or Evaluations of RVI Upper Internals Assembly Components, Core Barrel Assembly Components, and Interfacing or Alignment Components in the EPRI MRP “Primary” or “Existing Program” Categories The staff has observed that AR No. 2241723-23 in the audit portal references a number of inspection result reports or evaluation reports for BFB, lower internals assembly components, and CRGT guide cards that were previously inspected by the applicant (See additional reports below in this table that the staff is requesting for placement on the audit portal), but not any corresponding inspection reports or evaluation for the RVI upper internals assembly, core barrel assembly, and alignment or interfacing components being inspected as “Primary” or “Existing Program” components for the program. Does NextEra Energy have the corresponding reports for the past inspection of these components at PBNP Units 1 and 2 as well? If so, the staff requests these reports also be posted to the audit portal as well.	Requested reports for placement on the audit portal may have consolidated and more concise inspection result summaries in lieu of the numerous lists of inspection results included in AR No. 02241723.

			References: SLRA Appendix C, Tables C.A1, C.A2, and C.A3; AR No. 02241723	

SLRA Sections 3.1 B.2.3.10 Steam Generators

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	3.1	3.1-101	Section 3.1, "Program Overview and Background," of FPLCORP00036-REPT-049, "Point Beach Units 1 and 2 Subsequent License Renewal Aging Management Program Basis Document – Steam Generators," states, "The aging of steam generator pressure vessel welds, supporting secondary system components and piping including steam generator feedrings, J-nozzles, and feedring supports are managed by other programs…," and goes on to cite Aging Management Program (AMP) XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," and AMP XI.M2, "Water Chemistry." However, SLRA Table 3.1.2-5, "Steam Generators – Summary of Aging Management Evaluation," states that the aging effects of the feedwater feedring and support structure and the feedwater j-nozzle are managed by AMP XI.M19 and AMP XI.M2. SLRA Table 3.1.2-5 states that the intended function for the feedwater feedring and support structure and feedwater j-nozzle is "structural integrity (attached)." SLRA Table 2.1.5-1, "Passive/Structure/Component Intended Function," defines "structural integrity (attached)" as non-safety related components that maintains mechanical and structural integrity to provide structural support to attached safety-related systems, structures, and components. While the feedwater feedring support structure appears to meet this definition, the feedwater feedring and feedwater j-nozzle typically have an intended function such as flow distribution. However, the definition of "flow distribution" in SLRA Table 2.1.5-1 is "Provide a passageway for the distribution of the reactor coolant flow to the reactor core."	The U.S. Nuclear Regulatory Commission (NRC) staff is seeking clarification that the aging effects of the feedwater feedring and support structure and the feedwater j-nozzle are managed by AMP XI.M19 and AMP XI.M2. In addition, the staff is seeking clarification of the intended function for the feedwater feedring and feedwater j-nozzle.

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
			Please confirm, consistent with SLRA Table 3.1.2-5, that the aging effects of the feedwater feedring and support structure and the feedwater j-nozzle are managed by AMP XI.M19 and AMP XI.M2. In addition, please discuss the intended function of the feedwater feedring and feedwater j-nozzle.	
2	3.1, 16.2.2.20, 16.4, B.2.3.10, and B.2.3.20	3.1-1, A-26, A-64, B-88, and B-152	Section 3.1.2.2.11, “Cracking Due to Primary Water Stress Corrosion Cracking,” of NUREG-2192, “Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants” (SRP-SLR) (ADAMS Accession No. ML17188A158), recommends actions to manage aging of divider plate assemblies depending on the material of the divider plate assemblies and whether industry analyses (Electric Power Research Institute (EPRI) 3002002850, “Steam Generator Management Program: Investigation of Crack Initiation and Propagation in the Steam Generator Channel Head Assembly,” dated October 2014) are bounding for the applicant's unit(s). Since the divider plate assemblies in the PBN Unit 1 steam generators (SGs) are fabricated of Alloy 600 material, the following recommendations from SRP-SLR are potentially applicable for PBN Unit 1: • For units with divider plate assemblies fabricated of Alloy 600 or Alloy 600 type weld materials, if the analyses performed by the industry (EPRI 3002002850) are applicable and bounding for the unit, a plant-specific AMP is not necessary. • For units with divider plate assemblies fabricated of Alloy 600 or Alloy 600 type weld materials, if the industry analyses (EPRI 3002002850) are not bounding for the applicant's unit, a plant-specific AMP is necessary, or a rationale is necessary for why such a program is not needed. A plant-specific AMP (one beyond the primary water chemistry and the steam generator programs) may include a one-time inspection that is capable of detecting cracking to verify the effectiveness of the	The NRC staff is seeking clarification on whether the industry analyses (EPRI 3002002850) are bounding for the PBN Unit 1 SGs. In addition, the staff is requesting that the applicant’s evaluation be made available for their review.

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
			water chemistry and steam generator programs and the absence of primary water stress corrosion cracking in the divider plate assemblies. The following SLRA sections and tables appear to state that the evaluation of the industry analysis (EPRI 3002002850) to determine whether it is or is not bounding for PBN is not complete. • SLRA Section 3.1.2.2.11, "Cracking due to Primary Water Stress Corrosion Cracking" • No. 14 in SLRA Table 16-3, "List of SLR Commitments and Implementation Schedule" • SLRA Section B.2.3.10, "Steam Generators" However, the following SLRA sections and tables appear to state that the evaluation of the EPRI analysis is complete and is not bounding for PBN. • SLRA Table 3.1.2-5 and Plant Specific Note 1 • SLRA Section 16.2.2.20, "One-Time Inspection" • No. 24 in SLRA Table 16-3 • SLRA Section B.2.3.20, "One-Time Inspection" The NRC staff also notes that SLRA Section 3.1.2.2.11.1 does not, for completeness, state why a plant-specific AMP is not needed for the PBN Unit 2 SGs. It is important to note that the NRC staff cannot complete its review until the applicant completes its evaluation to determine whether the industry analyses are bounding. Please provide the status of the evaluation regarding EPRI 3002002850. If the evaluation is complete, discuss whether the	

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
			relevant industry analyses are bounding for the PBN Unit 1 SGs. Please provide the evaluation for NRC staff review. In addition, please revise all SLRA sections and tables to be consistent.	
3	16.2.2.20 and B.2.3.20	A-26 and B-152	SRP-SLR Section 3.1.2.2.11 recommends actions to manage aging of divider plate assemblies depending on the material of the divider plate assemblies and whether industry analyses EPRI 3002002850 are bounding for the applicant's unit(s). Since the divider plate assemblies in the PBN Unit 1 SGs are fabricated of Alloy 600 material, SRP-SLR Section 3.1.2.2.11 states that a plant-specific AMP is necessary, or a rationale is necessary for why such a program is not needed. SRP-SLR Section 3.1.2.2.11 goes on to state that a plant-specific AMP may include a one-time inspection is capable of detecting cracking to verify the effectiveness of the water chemistry and steam generators programs and the absence of primary water stress corrosion cracking in the divider plate assemblies. There are inconsistencies in how the intent of the one-time inspection of the PBN Unit 1 SG divider plate assemblies are described (SLRA Sections 16.2.2.20, B.2.3.10, B.2.3.20, and Table 16-3). For instance, verifying the effectiveness of the Steam Generators Program is not always stated, and in most causes discussion of verifying the absence of primary water stress corrosion cracking is not stated. Please confirm that the intent of the one-time inspection of the PBN Unit 1 SG divider plate assemblies is to verify the effectiveness of both the Water Chemistry and Steam Generators Programs, and to verify the absence of primary water stress corrosion cracking in the divider plate assemblies. In addition, please revise all SLRA sections to be consistent.	The NRC staff is seeking clarification that the one-time inspection of the Point Beach Unit 1 SG divider plate assemblies will verify the effectiveness of both the Water Chemistry and Steam Generators Programs, and to verify the absence of primary water stress corrosion cracking in the divider plate assemblies.
4	B.2.3.10	B-88	Figure 4 of SEM 7.11.20, “Eddy Current Testing of the Unit 1 Steam Generators,” Revision 15, includes three welded alloy 600	The NRC staff is seeking clarification regarding the tube

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
			plugs in SG A of PBN Unit 1 that were installed during manufacture. Section 4.8.2 of NP 7.7.17, “Requirements for Steam Generator Primary Side Activities,” Revision 21, states that all alloy 600 plugs were replaced with alloy 690 plugs during refueling outage 26 (U1R26) for PBN Unit 1. SLRA Section B.2.3.10 states, “Primary side preventive maintenance activities include replacing corrosion susceptible plugs with corrosion resistant materials and preventively plugging tubes susceptible to degradation.” Please confirm there are no alloy 600 plugs remaining in the PBN Unit 1 SGs.	plug material in the PBN Unit 1 SGs.
5	3.1	3.1-32	The Discussion column for Item Number 3.1-1, 025, in SLRA Table 3.1-1, “Summary of Aging Management Evaluations for the Reactor Vessel Internals, and Reactor Coolant System,” states that it is not applicable and points to a further evaluation in SLRA Section 3.1.2.2.11. However, SLRA Table 3.1.2-5 includes Table 1 Item 3.1-1, 025, for the nickel alloy divider plates in the PBN Units 1 and 2 SGs, and for the tube-to-tubesheet weld for the PBN Unit 2 SGs. Given that Table 1 Item 3.1-1, 025, appears to be applicable, please explain the statement in SLRA Table 3.1-1 that it is not applicable.	The NRC staff is seeking clarification on the statement in SLRA Table 3.1-1 that Table 1 Item 3.1-1, 025, is not applicable.
6	3.1	3.1-99	SLRA Table 3.1.2-5 does not cite a program(s) to manage wall thinning due to flow-accelerated corrosion in the carbon steel blowdown piping nozzles and secondary side shell penetrations. Blowdown piping nozzles and secondary side shell penetrations are captured together in the “Component Type” column of SLRA Table 3.1.2-5. NUREG-2191, “Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report – Final Report,” dated July 2017 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML1787A031 (Vol. 1) and ML17187A204 (Vol. 2), identifies wall thinning due to flow-accelerated corrosion as an	The NRC staff is seeking clarification on which program(s) will be used to manage wall thinning due to flow-accelerated corrosion in the carbon steel blowdown piping nozzles and secondary side shell penetrations.

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
			applicable aging effect for steel pressure boundary and structural components/structures exposed to secondary feedwater or steam. For example, GALL-SLR Item IV.D1.R-37, 3.1-1, 061, identifies wall thinning due to flow-accelerated corrosion of steel exposed to secondary feedwater or steam as an applicable aging effect to be managed using AMP XI.M17. Please explain which program(s) will be used to manage wall thinning due to flow-accelerated corrosion in the carbon steel blowdown piping nozzles and secondary side shell penetrations or state the basis for why a program is not necessary.	
7	3.1	3.1-101	SLRA Table 3.1-1 states that Item Number 3.1-1, 074, is consistent with the GALL-SLR. The “Component” column for Item Number 3.1-1, 074, in SLRA Table 3.1-1 and the “Structure and/or Component” column in Table D1, “Steam Generator (Recirculating),” in the GALL-SLR include the feedwater inlet ring and support. SLRA Table 3.1.2-5 includes Note D for managing wall thinning due to flow-accelerated corrosion by AMPs XI.M19 and XI.M2 of the carbon steel feedwater feedring and support structure exposed to treated water or steam. Note D is used when the component is different; material, environment, aging effect, and AMP are consistent with the GALL-SLR; and the AMP has exceptions to the GALL-SLR description. Given that it appears that the component is the same, please explain why Note D was cited in SLRA Table 3.1.2-5, rather than Note B.	The NRC staff is seeking clarification on Note D used in SLRA Table 3.1.2-5 for managing wall thinning due to flow-accelerated corrosion by AMPs XI.M19 and XI.M2 of the carbon steel feedwater feedring and support structure exposed to treated water or steam.
8	3.1	3.1-106	SLRA Table 3.1.2-5 does not cite a program(s) to manage loss of material due to boric acid corrosion in the carbon steel seismic lugs, transition cone, steam outlet nozzle, and feedwater nozzle. The GALL-SLR identifies loss of material due to boric acid corrosion as an applicable aging effect for external surfaces of steel SG	The NRC staff is seeking clarification on which program(s) will be used to manage loss of material due to boric acid corrosion in the carbon steel

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
			components exposed to air with borated water leakage. For example, GALL-SLR Item IV.D1.R-17, 3.1-1, 049, identifies loss of material due to boric acid corrosion as an applicable aging effect for external surfaces of steel SG components exposed to air with borated water leakage as an applicable aging effect to be managed using AMP XI.M10, "Boric Acid Corrosion." Please explain which program(s) will be used to manage loss of material due to boric acid corrosion in the carbon steel seismic lugs, transition cone, steam outlet nozzle, and feedwater nozzle or state the basis for why a program is not necessary.	seismic lugs, transition cone, steam outlet nozzle, and feedwater nozzle.
9	3.1	3.1-110	Plant Specific Note 2 appears to be missing from SLRA Table 3.1.2-5. Plant Specific Note 2 is cited for loss of material of the carbon steel transition cone welds (new welds) exposed to treated water.	The NRC staff is seeking clarification regarding Plant Specific Note 2 to SLRA Table 3.1.2-5.
10	3.1.2.2.2.2	3.1-9	SLRA Section 3.1.2.2.2.2 states, "The new circumferential closure welds will be managed by the Water Chemistry (B.2.3.2) AMP." This section also discusses the one-time inspection of these welds. SLRA Table 3.1-1 states, "The new transition cone circumferential welds on Units 1 and 2 will be managed using the Water Chemistry (B.2.3.2) and One-Time Inspection (B.2.3.20) AMPs." However, SLRA Table 3.1.2-5 cites GALL-SLR Item IV.D1.RP-368, 3.1-1, 012, for the "Transition cone welds (new welds)" with Note A for AMP XI.M1 (consistent). Therefore, SLRA Table 3.1.2-5 appears to state that loss of material for the new transition cone welds will be managed by AMP XI.M1, M2, and M32, "One-Time Inspection." Please clarify whether loss of material of the new transition cone welds will also be managed by AMP XI.M1. If not, please discuss the basis. In addition, please revise all SLRA sections to consistently reflect management of the new transition cone welds.	The NRC staff is seeking clarification on the programs managing loss of material for the new transition cone welds.

Section B.2.3.35 Inspection of Water-Control Structures Associated with Nuclear Power Plants (TRP 47)

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	B.2.3.35	B-246	SLRA operating experience indicates that the roof of the Circulating Water Pumphouse (CWPH) was noted to be in degraded condition in 2013. The evaluation concluded that there was no operability of functional concern and the CWPH remained fully operable. However, there is evidence of water leakage from the roof of the CWPH where the diesel fire pump exhaust stack penetrates the roof in AR 2250436. 1. Provide WO to demonstrate the completeness of corrective actions for the CWPH roof leakage. 2. Supplement the SLRA to capture this operating experience for the CWPH roof leakage.	To verify if corrective actions are taken for the CWPH roof leakage and ensure the accuracy of operating experience in the SLRA.
	B.2.3.35	B-244	Based on NUREG-2191, Section XI.S7, the preventive actions for storage, lubricants, and stress corrosion cracking potential discussed in Section 2 of RCSC (Research Council for Structural Connections) publication “Specification for Structural Joints Using ASTM A325 or A490 Bolts,” will be used if the structural bolting consists of ASTM A325, ASTM F1852, and/or ASTM A490 bolts. Both the PBN ASTM Section XI, Subsection IWE and ASTM Section XI, Subsection IWF AMPs provide the enhancement for this, however, the PBN Water-Control Structures AMP does not include this enhancement. Please provide the same enhancement for the Element 2 of Water-Control Structures AMP to ensure the consistency with NUREG-2191, Section XI.S7, or provide the justification why this enhance is not needed.	To ensure the consistency between the PBN Water-Control Structures AMP and NUREG-2191.

SLRA B.2.3.29 ASME Section XI, Subsection IWE (GALL-SLR AMP XI.S1)

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	B.2.3.29 Table 16-3 (item 33)	B-215 A-101	<u>Program element 4 enhancement (LR Commitment 33(d)) - Supplemental Surface Examination to Detect Fatigue Cracking</u>. SLRA enhancement states: “Augment existing procedures to implement periodic supplemental surface examinations (or other appropriate examination / evaluation methods) at intervals no greater than other	To verify consistency with GALL-SLR, to get clarity/specificity, and evaluate adequacy of enhancement.

			IWE inspections to detect cracking due to cyclic loading of non-piping penetrations (hatches, electrical penetrations, etc.).” The enhancement/commitment appears to lack clarity and specificity with regard to other appropriate examination methods that will be used (in lieu of surface examinations), the specific examination interval that will be used, and whether the enhancement covers “all” non-piping penetrations without exception. Considering these, provide a revised enhancement that addresses the following: (a) For the option “or other appropriate / evaluation methods” to be included in the enhancement/commitment, explicitly specify and justify the adequacy of each “other appropriate examination / evaluation methods” that will specifically be used by the applicant in lieu of surface examination to detect cracking due to cyclic loading. Or, alternatively remove the phrase “(or other appropriate / evaluation methods)” in the enhancement. (b) State the specific inspection interval (or frequency) for the supplemental examination and justify its adequacy. (c) Confirm if the enhancement applies to “all non-piping penetrations” without exception or additions (e.g. dissimilar metal welds (DMWs), steel, stainless steel etc.). If there are exceptions or additions, explicitly specify the components to which the enhancement applies. DMWs do not appear to be addressed for cracking in Table 3.5.2-1. Why are dissimilar metal welds of electrical or other penetrations not included?	
2	B.2.3.29 Table 16-3 (item 33), 3.5.2.2.1.6	B-215, A-101,	<u>Program elements 4 & 7 enhancement (LR Commitment 33(e)) - Supplemental One-Time Inspection to Confirm absence of SCC.</u>	To verify consistency with GALL-SLR and help evaluate adequacy of the enhancement.

		3.5-22, -23	SLRA enhancement states: “Augment existing procedures to implement supplemental one-time inspections, performed by qualified personnel using methods capable of detecting cracking due to SCC, comprising (a) a representative sample (two) of the stainless steel penetrations or dissimilar metal welds associated with high-temperature (temperatures above 140°F) stainless steel piping systems in frequent use on each unit; and (b) the stainless steel fuel transfer tube on each unit. If SCC is detected as a result of the supplemental one-time inspections, additional inspections will be conducted in accordance with the site’s corrective action process.” The FE 3.5.2.2.1.6 (by just stating that it will be addressed in corrective actions process) and related above enhancement does not appear to state: (i) the specific examination “method(s) capable of detecting cracking due to SCC” that will be used by the applicant, and (ii) the objectives (e.g., per XI.M32 “One-Time Inspection,” extent of condition, extent of cause, and need for periodic inspections if absence of SCC cannot be confirmed) and outcome (e.g., periodic inspections at specific intervals using appropriate methods if SCC is determined to be an applicable aging effect) of additional inspections that will be conducted if SCC is detected as a result of the supplemental one-time inspections. Also, see Turkey Point SER Section pages 3-141/142, 3-276/277 and A-28/29 for similar issue. Considering the above issues, clarify the components that will be subject to the one-time inspection, and provide a revised enhancement/LR commitment 33(e) that addresses the following: (a) State and justify the specific methods capable of detecting SCC that will be used for the one-time inspection.	SLRA enhancement / Commitment 33(e) states:

			(b) Considering the objectives and outcome of additional inspections, state the specific actions that will be taken with regard to additional inspections (and up to and including implementing periodic inspection (method, interval, criteria)) if SCC is detected as a result of the one-time inspection and absence of the aging effect was not confirmed.	
3	B.2.3.29 Table 16-3 (item 33)	B-215 A-101	<u>Program enhancement and LR Commitment 33(f) – One-Time Supplemental Volumetric Inspection of Liner Plate based on OE Trigger.</u> SLRA enhancement states: "Augment existing procedures to implement a one-time supplemental volumetric inspection of metal liner surfaces that samples randomly selected as well as focused locations susceptible to loss of thickness due to corrosion from the concrete side if triggered by plant-specific OE identified through code inspections after the date of issuance of the first renewed license for each unit." The SLRA does not appear to provide a positive confirmatory statement as to whether the OE-trigger has occurred to date since issuance of the first renewed license. The enhancement/commitment does not include the statistical-based acceptance criteria for sampling specifications in the GALL-SLR program (i.e., that the sample size, locations and any needed scope expansion for this one-time volumetric examination shall demonstrate statistically with 95 percent confidence that 95 percent of the accessible portion of the containment liner is not experiencing corrosion degradation with greater than 10 percent loss of nominal thickness). The trigger specified in the GALL-SLR is the site-specific occurrence or recurrence of the stated plant-specific OE without regard	To verify consistency with GALL-SLR and help evaluate adequacy of Enhancement/LR Commitment.

			to the method by which (how) it is identified. Contrary to this, the SLRA enhancement/commitment states that the triggering OE would be specific to that identified through “code” inspections. Point Beach is a two-unit plant. The enhancement/commitment does not address treatment of the two units with regard to conducting the supplemental volumetric examination if the triggering OE occurs in one of the units. Considering the above issues, provide a revised enhancement/Commitment 33(f) addressing the questions below: a) Confirm affirmatively whether or not a plant-specific OE of through-wall containment liner corrosion has occurred at Point Beach Units 1 or 2 since the issuance of the first renewed license. b) Provide the statistical-based acceptance criteria for the sampling size and locations and method independence of identifying the triggering OE consistent with the GALL-SLR, and how it is captured in the enhancement/commitment. c) Describe the applicant’s specific considerations with supporting justification of the treatment of the two PBN Units when the triggering OE occurs in one with regard to the conduct of the one-time OE-triggered volumetric examination of the containment liner and how is it appropriately captured in LR Commitment 33(f) in SLRA Table 16-3. (see additional discussion in Turkey Point SER, ML19191A057, pages 3-141/142, A-28/29). d) Also, provide an implementation schedule (relative to the date of occurrence of the triggering OE) for implementation of the one-time supplemental volumetric examination	

			that covers the case where the triggering OE occurs during the SPEO.	
4	App. A, 16.2.2.29	A33 – A34	The FSAR Supplement program description appears to be not consistent with the SLRA B.2.3.29 program description and recommended description in GALL-SLR Table XI-01 for AMP XI.S1, specifically with regard to the program including supplemental surface examination to detect cracking due to cyclic loading (or SCC) for specific components that are subject to cyclic loading and have no CLB fatigue analysis. a) Provide a revised FSAR supplement description to cover supplemental surface examination or appropriate App J test to detect cracking due to cyclic loading (or SCC) for specific containment pressure-retaining components for which a CLB fatigue analysis does not exist. Identify the appropriate method (e.g., surface examination, appropriate App J Type B test) that will be used to detect cracking and identify specific components to which each method applies.	To verify adequacy/consistency of the FSAR summary description with the SLRA AMP B.2.3.29 description and recommended FSAR supplement description for program XI.S1 in GALL-SLR Table XI-01.
5	B.2.3.29 Table 3.5.2-1	B-215, B-216, 3.5-90	Regarding PBN's evaluation of NRC IN 2014-07 & RIS 2016-07 regarding leak-chase interface components that serve a moisture barrier function, Section B.2.3.29 states "…, the plant does have accessible capped lines for the leak chase. The locations of the leak chase channel vents were documented on applicable PBN drawings and included in IWE visual examinations. RIS 2016-07, ……. PBN has been proactively performing exams on moisture barriers once each Period during the 2nd Interval, and this RIS has been incorporated into the 3rd IWE Interval program plan." However, Table 3.5.2-1 AMR results does not include line item(s) that correspond to the above components that serve a moisture barrier function for certain	To verify SLRA AMR results reflects aging management for containment components that serve a moisture barrier function per industry OE in IN 2014-07 and RIS 2016-07.

			inaccessible areas of containment liner and that correspond to the above statements made in the SLRA related to OE described in IN 2014-07 & RIS 2016-07. a) Clarify, by illustrating from applicable IWE program documents and drawings, if accessible leak-chase channel interface components at or near the containment floor that serve a moisture barrier function (i.e., prevent intrusion of moisture into inaccessible portions of liner) are monitored in the IWE program, consistent with the recommendations in NRC IN 2014-07 and RIS 2016-07. b) Provide Table 2 AMR line items in the SLRA for above containment components that serve a moisture barrier function per NRC IN 2014-07 & 2016-07.	
6	Table 3.5.2-1	3.5-79, 3.5-96, 3.5-54, 3.5-53	Table 3.5.2-1 on p3.5-79 includes two AMR line items with "copper alloy" material corresponding to Table 1 AMR items 3.5.1-027 & -028 with generic/plant-specific notes F,1 which states "Copper alloy is not addressed as a structural component in NUREG-2191. However, the environment, aging effects (cracking and loss of material) and AMPs for steel air lock, hatch components are conservatively also applicable to the copper alloy airlock bushings." a) Explain how PBN determined that it has identified all applicable aging effects for this component, material, and environment combination related to subject AMR items with generic Note F; and, where is it documented?	This is a finding the staff needs to make in the SER for an AMR line item with generic Note F.
7	Tables 3.5.1 & 3.5.2-1; FE 3.5.2.2.1.5	3.5-48, -49, -53, -87	AMR Item 3.5-1, 009 related to cumulative fatigue damage states in discussion column: "Consistent with NUREG-2191 for liner and <u>most</u> mechanical penetration assemblies."	For clarity and accounting of which components are covered by which AMR line items for the aging effect of

	& 3.5.2.2.1.6		a) Clarify where cumulative fatigue damage or cracking due to cyclic loading is addressed for the other mechanical penetrations that are not covered by item 3.5-1, 009. State which specific mechanical penetrations are covered by 3.5.1, 027. b) With reference to FEs 3.5.2.2.1.5 & 3.5.2.2.1.6, clarify by walking through the “specific” components subject to cracking due to cyclic loading or due to SCC that are covered by items 3.5-1, 009, and -027, and for SCC 3.5-1, 010; show the corresponding AMR line items in Table 3.5.2-1. For components that do not have a CLB fatigue analysis, are there any for which an appropriate Type B leak test is credited to monitor for cracking due to cyclic loading? c) Clarify the inconsistency in the discussion column of Table 3.5.1 for AMR items 3.5-1, 009, -010, and -027, and specifically in that they do not include the fuel transfer tube. Additionally, in Table 3.5.2-1 on p3.5-87, fuel transfer tube is credited under item 3.5-1, 010 only. d) Clarify the intent of the statement “…These same stainless-steel penetrations, and any DMWs, are also susceptible to cyclic loading. ..” in the first full paragraph on p3.5-23 of FE 3.5.2.2.1.6 related to SCC.	cumulative fatigue damage (cracking due to cyclic loading) and SCC.
8	Table 3.5.2-1	3.5-96	Table 3.5.2-1 has a plant-specific note 6, which states: “Penetration assemblies for high temperature stainless steel piping systems only, whereas other mechanical penetration sleeves/assemblies are addressed for cumulative fatigue damage.” However, note 6 is not assigned to any of the AMR line items in Table 3.5.2-1. a) Identify the AMR line item(s) in Table 3.5.2-1 to which plant-specific note 6 applies.	Need clarification on an apparent omission of information in the SLRA.

9	General		Confirm if LR Commitments 71 and 72 from the first license renewal for the IWE program will continue to be implemented for the subsequent period of extended operation.	Confirmation.

SLRA Section 2.4 Structural Scoping and Screening

#	Document	Section / Page #	Question / Issue	Why are we asking?
1	SLRA	Section 2.4.5	The “boundary” writeup in SLRA Section 2.4.5 notes that joint and penetration seals are conservatively included in the current structure boundary; however, this component type is not included in Table 2.4-5 or Table 3.5.2-5. This same issue occurs in Sections 2.4.6, 2.4.7, 2.4.10, and 2.4.12.	To understand if “joint and penetration seals” are in scope.
2	SLRA	Section 2.4.11	The “boundary” writeup in SLRA Section 2.4.11, “Yard Structures,” notes that masonry walls are conservatively included in the current structure boundary; however, this component type is not included in associated Table 2.4-11 or Table 3.5.2-11.	To understand if masonry walls are in scope of the “Yard Structures” group.

Containment Liner Plate and Penetrations Fatigue TLAA

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	4.6, 16.3.6, Table 3.5-1	4.6-3, A-58, A-59	The subject SLRA Sections state: “PBN has been <ins>*unable to locate the original fatigue analysis or confirm if a fatigue waiver exists for the PBN containment penetrations other than piping penetrations, piping penetrations with dissimilar metal welds, and for the expansion’s joints of the containment structure reactor fuel transfer tube*</ins>. Therefore, consistent with NUREG-2192, Table 3.5-1, Item 3.5-1, 027, cracking due to cyclic loading of non-piping containment penetrations (i.e. personnel airlocks, equipment hatch, personnel hatch, electrical penetrations, piping penetrations with dissimilar metal welds, and the expansions joints of the containment structure fuel	To clarify SLRA 4.6 partial TLAA disposition of components inconsistent with the definition of TLAAs in Part 54 regulations; and to get a proper accounting of components that require addressing of the aging effect of cracking due to cyclic loading (or cumulative fatigue damage).

			transfer tube will be managed by the ASME Section XI, Subsection IWE AMP (Section B.2.3.29) and periodic supplemental surface examinations incorporated into and consistent with the frequency of the 10 CFR Part 50, Appendix J AMP (Section B.2.3.32).” “*The fatigue analyses associated with the containment personnel airlocks, equipment hatch, personnel hatch, electrical penetrations, piping penetrations with dissimilar metal welds, and expansions joints of the containment structure fuel transfer tube* will be managed by the ASME Section XI, Subsection IWE AMP (Section 16.2.3.29) and the 10 CFR Part 50, Appendix J AMP (Section 16.2.3.32) *in accordance with 10 CFR 54.21(c)(1)(iii)*.” Issue: • SLRA Chapter 4 should be addressing only TLAAs that meet all the six criteria in the definition of TLAA in 10 CFR 54.3 (identified per methodology in SLRA 4.1) and thereby qualify to be dispositioned in accordance with 54.21(c)(1); *Fatigue analyses that cannot be located or confirmed (as claimed) or do not exist in the CLB* do not qualify to be dispositioned as TLAAs in SLRA Chapter 4, and cannot be dispositioned in accordance with 54.21(c)(1)(iii) as stated in the SLRA excerpts above as it is inconsistent with the above cited Part 54 regulations and related NRC guidance. • The first License Renewal Application for Point Beach, Section 4.3.11 “Containment Liner Plate Fatigue Analysis” and the corresponding NRC SER did not document the containment non-piping penetration components listed above (in underlined italics) to have CLB fatigue TLAAs dispositioned in accordance with 54.21(c)(1). The TLAA disposition therein was limited to containment liner plate and piping penetrations.	

			• Subsection IWE and Appendix J AMPs (GALL-SLR XI.S1 & XI.S4) are not TLAA AMPs. TLAA AMPs are those in GALL-SLR Chapter X. Further, SRP-SLR Table 3.5-1, AMR Item 027 applies specifically to containment pressure-retaining components for which CLB fatigue analysis does not exist in order to address cracking due to cyclic loading using aging management actions. While it is acceptable to manage cracking due to cyclic loading through appropriate aging management actions (examination/test methods capable of detecting cracking, interval) in accordance with 54.21(a)(3) using IWE and/or App J programs, this should be addressed in Section 3.5.2.2.1.5 and the credited SLRA Appendix B AMPs; but should not be dispositioned as TLAA. A TLAA disposition would only correspond to components for which Table 3.5-1, AMR item 009 applies. <u>Questions</u>: a) For the specific PBN containment penetrations other than piping penetrations, piping penetrations with dissimilar metal welds, and expansion's joints of fuel transfer tube for which SLRA 4.6 and 16.3.6 claimed to be unable to locate the original fatigue analysis or confirm if a fatigue waiver exists but dispositioned as TLAA per 54.21(c)(1)(iii), clarify and provide objective evidence that fatigue analyses per definition in 54.3 exists, and upload such documents on the ePortal. b) If CLB fatigue analyses do not exist for these components and they are managed for the aging effect cracking due to cyclic loading using AMR item 3.5.1, 027 (as it appears to be), then revise SLRA Sections 4.6 and 16.3.6 accordingly to remove the TLAA disposition of certain components in accordance with 54.21(c)(1)(iii) (Note that should for such components should be addressed only in appropriate SLRA Section 3.5 and/or the	

			AMPs credited for managing the aging effect. c) In continuation of related breakout question for the B.2.3.29 IWE AMP, for the above subject components, please walk-through the examination methods, frequency and their adequacy to manage cracking due to cyclic loading, and where addressed in the SLRA. d) SLRA Table 3.5-1, item 3.5-1, 027 in the Discussion column states the fuel transfer tube is addressed by item 3.5-1, 009, which applies to components for which a CLB fatigue analysis exists. However, as noted above SLRA 4.6, fuel transfer tube is among the components for which fatigue analyses could not be located. Clarify the inconsistency and correct errors, if any.	
2	4.6	4.6-1 thru 4.6-3	The following documents are referenced in SLRA Section 4.6 in support of the TLAA. (i) Structural Integrity Associates (SIA) Calculation PBCH-06Q-301, Revision 1, “Containment Penetration Fatigue Evaluation,” 11/11/2003 (including Encl 4, Attachment 15) (ii) SIA Report No. 2000088.401, Revision 2, “Cycle Counts and Fatigue Projections for 80 Years of Plant Operation for Point Beach Nuclear Units 1 and 2,” 10/19/2020 (iii) Calculation or documents in the CLB documenting fatigue analyses for Containment Liner Plate Request: a) Please provide the above documents on the ePortal to enable staff audit verification of TLAA analyses described in SLRA 4.6	Request for supporting documents for audit

ASME Section XI, Subsection IWL

#	Document	Section / Page #	Question / Issue	Why are we asking?
1	AR01216214 / AR02021710		AR01216214 discusses tendons (D2-229 and D2-230) that had significant voids of grease in	To understand how past operating experience has

			1999. The issue was corrected and found acceptable. AR02021710 discusses areas of containment concrete that show grease stains / minor grease seepage. Has there been any indication that tendons D2-229 and D2-230 (or other tendons) have had significant grease leakage since 1999? If so, how has the issue been addressed/resolved?	informed the program.
2	AR02316032		This AR discusses groundwater leakage in the U2 tendon gallery, specifically around tendon can V-333. 1) What has been done to ensure the water is not impacting the tendon? Has the cap been removed to examine the tendon? 2) More generically, how significant is groundwater leakage in the tendon galleries? Does it impact the tendons and how has this been determined?	To understand past operating experience related to water ingress in the tendon galleries and if this water has impacted the containment tendons.
3	FPLCORP0036-REPT-039 SLRA	Attachment 4 B.2.3.30 / B-222	SLRA states that plant specific operating experience has shown that degradation has occurred, including failed tendon wires, missing or broken components found in the tendon hardware, and degraded concrete in containment structure. Review of the OE in the SLRA and the referenced document identified examples of tendon OE; however, no examples were provided related to general containment concrete degradation. Please provide a summary of recent OE related to the containment concrete visual exams or provide the examination results. Be prepared to discuss any significant OE.	To understand the extent of containment degradation and how identified degradation has been addressed.

4	FPLCORP0036-REPT-071 (IWL Basis Doc) 3rd Interval-IWL-PB-1/2-Program Plan LR-AMP-028-IWEL	Page 27 of 33 Page 32 of 34 Page 8 of 28	These documents state that the program is consistent and include the appropriate requirements; however, both documents are very high-level. The actual inspection procedures for the IWL visual inspections do not appear to be on the portal. Please provide the inspection procedures for the IWL visual inspections.	To verify consistency of the existing program with the GALL Report AMP.

AMP B.2.2.3 – Concrete Containment Unbonded Tendon Prestress / TLAA 4.5 – Concrete Containment Tendon Prestress

#	Document	Section	Question / Issue	Why are we asking?
1	FPLCORP00036-REPT-062 SLRA	4.6 B.2.2.3	The enhancement for element 6 notes that the 80-year prestress calculation will be used ***with*** or in place of the current 60-year acceptance limits. Why is it acceptable to use the acceptance limits from the 60-year calculations as opposed to the limits that have been updated for 80 years?	To understand how tendon prestress limits will be determined in the subsequent period of extended operation.
2	SLRA	4.5	SLRA Figures 4.5-1 and 4.5-4 show measured tendon forces that fall below the minimum required value (MRV). What corrective actions were taken to address these values below the MRV?	To understand how the low stress values were addressed in the tendons.

10 CFR Part 50, Appendix J

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	B.2.3.32	B-231	"Scope of Program" program element of GALL-SLR XI.S4 states, in part, "[t]he aging effects associated with containment pressure-retaining boundary components within the scope of subsequent license renewal and excluded from Type B or C Appendix J testing must still be managed. Other programs may be credited for	To verify the consistency of the "program scope" element of PBN Appendix J AMP with GALL-SLR XI.S4.

			managing the aging effects associated with these components; however, the component and the proposed AMP should be clearly identified”. SLRA Section B.2.3.32 (10 CFR Part 50, Appendix J) provides no discussion on the containment pressure-retaining components excluded from local leakage rate testing (LLRT). “Scope of Program” program element in the PBN AMP basis document of 10 CFR Part 50, Appendix J (FPLCORP00036-REPT-072) provides a list of the AMPs that support management of applicable aging effects associated with the components excluded from LLRT, however, no identification for these excluded components. Please identify the components within the scope of SLR and excluded from LLRT, and associated AMPs in keeping with “Scope of Program” element of GALL-SLR XI.S4.	
2	B.2.3.29	B-214	The PBN ASME Section XI, Subsection IWE (B.2.3.29) is used to manage cracking due to cyclic loading of non-piping penetrations. The AMP basis document, FPLCORP00036-REPT-070, states that cracking due to cyclic loading of non-piping penetrations will be managed by supplemental surface examinations or other appropriate examination/evaluation methods at intervals no greater than other IWE inspections and the PBN 10 CFR Part 50, Appendix J (B.2.3.32) AMP. It is unclear to the staff what “other appropriate examination/evaluation methods” are and how these methods are related to the PBN IWE and/or Appendix J.	To clarify what examination methods will be used to manage cracking due to cyclic loading of non-piping penetrations and how these methods are associated with the PBN IWE and/or Appendix J.
3	Table 3.5.2-1	3.5-80	Table 3.5.2-1 AMR line item, “Air locks, Equipment hatches and accessories” components “Elastomer” material associate with Table 1 item 3.5-1, 033, assigns both “Fire barrier” and “Pressure boundary” as intended functions. The AMR line item identifies 10 CFR Part 50, Appendix J (B.2.3.32) AMP to manage aging effect. However, Fire Protection AMP (B.2.3.15) is not included. It is also noted that Table 3.5.2-14 (page 3.5-139) does not include AMR line items for structural commodities identified in Table 3.5.2-1 (Air locks, Equipment hatches and accessories) to be fire barriers. Please justify if the selected	To justify if the Fire Protection AMP is needed for managing the effects of aging for fire resistant materials that serve a fire barrier function.

			Table 1 item 3.5-1, 033 may be adequate to manage the effects of aging of select components with a designated fire barrier intended function.	

Corrosion-Structural

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	SLRA Table 3.5.2-1	3.5-88	Table 3.5.2-1 AMR line item includes component type "Liners (refueling cavity) and covers (sand box, Unit 1 sump A strainer)" associated with Table 1 item 3.5-1, 100 that credits the Structures Monitoring AMP to manage potential loss of material, cracking aging effects. However, the table does not include similar AMR line items for other sumps and Unit 2. Please clarify if similar entries exist.	To clarify possible omission of similar AMR line items for other sumps and Unit 2.
2	SLRA Tables 3.5.2-1, 3.5.2-8, 3.5.2-11, 3.5.2-13	3.5-88 3.5-121 3.5-130 3.5-135 3.5-137	Several Table 2 AMR line items (Fuel transfer tube, New fuel storage Racks, Miscellaneous structural components, Anchorage/Embedment, Insulation) associated with Table 1 item 3.5-1, 100 only assigns "Loss of material" as the aging effect that needs to be managed. Table 3.5-1, 100 item relates aging effects of loss of material due to pitting and crevice corrosion, cracking due to SCC for aluminum, stainless steel support members; welds; bolted connections; support anchorage to building structure exposed to air or condensation. Please clarify whether cracking due to SCC is a potential aging effect for all the components associated with Table 1 item 3.5-1, 100.	To clarify possible omission of cracking aging effect for Table 2 AMR line items associated with Table 1 item 3.5-1, 100.
3	3.5.2.2.2.4	3.5-88, 3.5-93. 3.8-139	Section 3.5.2.2.2.4 states that stainless steel/aluminum cracking due to SCC and loss of material due to pitting and crevice corrosion is managed with the External Surfaces Monitoring of Mechanical Components (B.2.3.23) AMP that interfaces with Structures Monitoring (B.2.3.34), Fire Protection (B.2.3.15), and ASME Section XI, Subsection IWE (B.2.3.29) AMPs. Although one of the intended functions for components selected to be managed by AMR line item 3.5-1, 100 is fire barrier, the	To clarify adequacy of referenced AMPs to manage the aging effects for fire resistant materials that serve a fire barrier function consistent with GALL-SLR AMP XI.M26.

			components assigned to it in Table 3.5.2-1 of the SLRA (pages 3.5-88 and 3.5-93) do not include the Fire Protection AMP. Table 3.5.2-14 (page 3.5-139) does not include AMR line items for structural commodities identified in Table 3.5.2-1 (fuel transfer tube, liners, radiant energy shields) to be fire barriers. The selected Table 1 item 3.5-1, 100 may not be adequate to manage the effects of aging of select components with a designated fire barrier intended function.	

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
4	SLRA Table 3.5-1	3.5-77	The applicant states that Table 1 item 3.5-1, 099 is not used. Please confirm that there are no aluminum and SS support members, welds, bolted connections or anchorage to structure for ASME Class 1, 2, 3 or MC components, and specify the material used for these supports and anchorage for ASME piping and components and associated Table 1 item(s).	To clarify possible omission of AMR items associated with Table 1 item 3.5-1, 099.

GALL-SLR AMP XI.S3 ASME Section XI, Subsection IWF

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	B.2.3.31	B-226	The IWF AMP includes an exception to the "scope of program" element, which states in part: "Inspection of supports for Class MC components under ASME XI – IWF is not required by 10 CFR 50.55a and PBN does not include such inspections in its IWF ISI program." a) Clarify whether PBN has supports for components classified Class MC that are within the scope of license renewal and subject to aging management review. b) If so, identify the AMP(s) managing the effects of aging for such Class MC component supports pursuant to 10 CFR 54.21(a)(3).	To evaluate adequacy of program taking exception to its scope with regard to supports for Class MC components.

			c) Provide an adequate justification for the exception to the IWF AMP given that, regardless of 10 CFR 50.55a requirements, for license renewal 10 CFR 54.21(a)(3) requires the applicant to demonstrate that the effects of aging of structures and components subject to AMR are adequately managed such that intended functions are maintained consistent with the CLB.	
2	B.2.3.31 16.2.3.31	B-227 A-103	Does Point Beach have or plan to use ASTM F2280 bolts (twist-off equivalent of A490 bolts)? If so, for such bolts why they are not included in the "preventive actions" program element enhancement (LR Commitment 35(d)) consistent with RCSC (Research Council for Structural Connections) publication "Specification for Structural Joints Using ASTM A325 or A490 Bolts," for storage, use of lubricants, and prevention of SCC.	To verify program element consistency and adequacy of enhancement with that of GALL-SLR Report AMP XI.S3.
3	B.2.3.31 16.2.3.31	B-227 A-103	The "detection of aging effects" program element of GALL-SLR AMP XI.S3 states: "For ***all*** high-strength bolting [actual measured yield strength greater than or equal to 150 ksi (1,034 MPa)] in sizes greater than 1 inch nominal diameter (including ASTM A490 and equivalent ASTM F2280), volumetric examination comparable to that of ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1 should be performed at least once per interval to detect cracking in addition to the VT-3 examination." The enhancement (LR Commitment 35(i) in Table 16-3) to B.2.3.31 AMP states, in part: "Augment existing procedures to specify that, for ***NSSS component supports***, high-strength bolting greater than one inch nominal diameter, volumetric examination comparable to that of ASME Code, Section XI, Table IWB-2500-1,	To verify program element consistency with GALL-SLR Report AMP XI.S3.

			Examination Category B-G-1 will be performed to detect cracking in addition to the VT-3 examination. ….” a) Clarify the distinction between the terminology “RC Class 1 Supports” used elsewhere in the SLRA (e.g. SLRA 2.4.1, Table 3.5.2-1, etc.) and “NSSS Component Supports” used in enhancement (LR Commitment 35(i) to the “detection of aging effects” element (related to volumetric examination of high strength bolting) of B.2.3.31 AMP. b) Justify how the above SLRA enhancement (LR Commitment 35(i) in Table 16-3) is consistent with the corresponding GALL-SLR program element which recommends ***all*** high-strength bolting greater than 1 inch diameter in the scope of the IWF program to be sampled for volumetric examination, whereas the enhancement requires high strength bolting (greater than 1 inch diameter) only in ***NSSS component supports*** to be sampled for volumetric examination. c) Noting that supports in the scope of the IWF program include Class 1, 2, 3 piping supports and supports other than piping supports, clarify how the sampling of high strength bolting (greater than 1” diameter) from NSSS component supports can be considered representative for volumetric examination of the entire population of such high strength bolting? .	
4	B.2.3.31 16.2.3.31	B-227 A-103	The “preventive actions” program element of GALL-SLR AMP XI.S3 states: “Operating experience and laboratory examinations show that the use of molybdenum disulfide (MoS2) as a lubricant is a potential contributor to stress corrosion cracking (SCC),	To verify program element consistency with GALL-SLR Report AMP XI.S3 and adequacy of related enhancements.

			especially when applied to high-strength bolting. Thus, molybdenum disulfide and other lubricants containing sulfur should not be used." Maintenance Instruction (MI) 29.1, Rev. 14, "Use of Thread Lubricants and Sealants," Section 3.1.3 (p3) states the solid components [of Thread Lubricants] may include, among others, Molybdenum disulfide. Section 3.3.5 (p5) states the Molybdenum disulfide based thread lubricants should NOT be used unless evaluated on a case-by-case basis with consideration given for the potential of causing SCC. Attachment A (p11) includes Molykote® as a proprietary brand of thread lubricants used at Point Beach, which is of molybdenum disulfide material. The "preventive actions" program element of B.2.3.31 AMP does not discuss molybdenum disulfide lubricant used in bolting at Point Beach. A program enhancement (LR Commitment 35(i) in Table 16-3) specifies volumetric examination of a sample of high strength bolting of NSSS component supports once in 10 years during the subsequent period of extended operation (SPEO) for SCC but does not discuss lubricants. It appears that Molybdenum disulfide lubricant may have been used at Point Beach in the past. Also, the MI does not prohibit other lubricants containing sulfur. a) Clarify if lubricants using ***molybdenum disulfide and others containing sulfur*** have been or will be used at Point Beach. If the use of such lubricants is prohibited in the future, the applicant's plan not to use molybdenum disulfide or other lubricants containing sulfur appears to be not adequately described.	

			Justify adequacy of the enhancement (LR Commitment 35(i) in Table 16-3) in managing aging effect of SCC in bolting that may have used molybdenum disulfide lubricant in the past or prior to entering the SPEO. b) Since volumetric examinations per the enhancement are planned for some time into the SPEO (could be almost 10 years), provide information on whether and how the aging effect of cracking due to SCC will be detected for the population of HS bolts such that this aging effect can be managed from the start of the SPEO. c) Discuss how the program will assess the adequacy of the HS bolting sample to inspect for cracking due to SCC when additional HS bolts are installed. (see PTN RAIs B.2.3.32-1 & B.2.3.32-2, ML 19050A420)	
Ye5	B.2.3.31 (General)	B-228 thru B-230	a) Discuss the most significant plant-specific operating experience, in the assessment of the IWF program Owner at PBN, for supports within the scope of the B.2.3.31 IWF AMP and how it was addressed. If included in the ARs placed on the ePortal, identify the AR #s up to and including resolution of issue; Otherwise provide the ARs that describe and addressed the OE identified issue to resolution.	Assess significance of plant-specific OE. This is a general interview question to the Program Owner.
6	Table 3.5.2-1 Procedure ISI CL 1, 2, 3 Program, Rev. 15	3.5-93 119-120, 146-147 of 156 (F-A Supports)	Table 3.5.2-1 on p3.5-93 includes 3 AMR line items (loss of material, loss of preload, cracking) corresponding to Table 1 items 3.5.1-68, -087, -091 for RC Class 1 support bolting of steel / high-strength steel material with generic note B and plant-specific Note 8, which states: *“As described in the RAI responses/supplements for the first 2 PWRs with renewed licenses for 80*	To clarify the adequacy of the applicant’s IWF AMP to manage the specific aging effect for which it is credited. Also, to note that if additional aging management

			years, ***thermal embrittlement of the steel reactor vessel support structure columns and beams requires analysis. Existing inspections, through the ASME Section XI, Subsection IWF (B.2.3.31) AMP, manages the condition of the reactor vessel support."*** a) Clarify what inspections (method, frequency) in the credited IWF AMP will be used to manage thermal embrittlement and how is it adequate (method, frequency) to manage the aging effect of thermal embrittlement of the reactor vessel support structure? Is thermal embrittlement the same as the aging effect of loss of fracture toughness due to irradiation embrittlement evaluated in SLRA Section 3.5.2.2.2.7? If not, define it and state why this aging effect is not included in appropriate program elements or as an enhancement of the SLRA Section B.2.3.31 (IWF AMP). Subsequently, state how the effects of radiation on RV steel support are to be managed. b) Clarify the purpose of plant-specific note 8 in 3 AMR line items in Table 3.5.2-1 (p3.5-93) for loss of material, loss of preload, and cracking due to SCC when they appear consistent with SRP/GALL-SLR Table 1 items 3.5.1-68, -087, -091 for these aging effects. Do the items that are credited for thermal embrittlement (due to irradiation?) need to be separate Table 3.5.2 AMR line items? c) Clarify if the reactor vessel steel supports are classified in the IWF program as Class 1 supports or as supports other than piping supports. State if the reactor vessel steel supports are inspected in the IWF program at least once in a 10-year	actions are required for managing loss of fracture toughness, due to irradiation embrittlement, of reactor vessel steel supports based on staff evaluation of FE 3.5.2.2.2.7 (it is possible that further enhancements to the applicant's IWF program may become necessary).

			ISI interval. Provide examination results (data sheet, ARs) of the latest (with dates) reactor vessel steel support inspection.	
7	Table 3.5-1	3.5-64	Table 3.5-1, AMR item 3.5.1, 057 states in "Discussion" column that the AMR item is " Consistent with NUREG-2191 with exception. The SLRA B.2.3.31 AMP is credited with managing loss of mechanical function for constrain [constant] and variable load supports exposed to an uncontrolled indoor air environment, as described in Table 3.5.2-13. However, neither Table 3.5.2-13 nor any of the other Table 3.5.2s include structural AMR line items corresponding to item 3.5.1, 057 as dispositioned above. a) Provide Table 3.5.2s AMR line items for item 3.5.1, -57 consistent with its disposition in SLRA Table 3.5-1. b) Or, Clarify and correct any errors, omissions or inconsistency between 3.5 Table 1 and Table 3.5.2s with regard to AMR item 3.5.1, 057.	To verify if components included in SLRA item 3.5.1, 057 are subject to AMR and consistent with GALL-SLR as claimed in Table 3.5-1.
8	Tables 3.5.-1 & 3.5.2-1	3.5-70, 3.5-94	Table 3.5-1, AMR item 3.5.1, 075 states in "Discussion" column that the AMR item is "Consistent with NUREG-2191 with exception. The ASME Section XI, Subsection IWF (B.2.3.31) AMP is credited with managing aging effect of sliding surfaces exposed to indoor air environment inside the Containment." However, neither Table 3.5.2-1 nor any of the other structural Table 3.5.2s include AMR line items corresponding to item 3.5.1, 075 as dispositioned above. It is also noted that Table 3.5.2-1 includes an AMR item for sliding surfaces corresponding to item 3.5.1,	To verify if components included in SLRA item 3.5.1, 075 and 3.5.1, 074 are subject to AMR and consistent with GALL-SLR as claimed in Table 3.5-1.

			074 crediting the Structures Monitoring Program (B.2.3.34). This line item as used is inconsistent with SRP-SLR for managing the effects of aging for sliding surfaces with the IWF AMP. a) Provide Table 2 AMR line items for item 3.5.1, -075 (loss of mechanical function of sliding surfaces) consistent with its disposition in SLRA Table 3.5-1. b) Clarify and correct any errors, omissions or inconsistency between Table 3.5.1 and Table 3.5.2s with regard to items 3.5.1, 074 and 3.5.1, 075.	
9	Tables 3.5-1, 3.5.2-9	3.5-72, 3.5-124	Table 3.5-1, AMR item 3.5.1, 085 (loss of material for stainless steel bolting exposed to treated water) states in "Discussion" column that the AMR item is "Consistent with NUREG-2191 with exception for Water Chemistry. The Water Chemistry (B.2.3.2) AMP and ASME Section XI, Subsection IWF (B.2.3.31) AMP are credited with managing loss of material for stainless steel bolting exposed to treated borated water in the spent fuel pool." Table 3.5.2-9 on p3.5-124 includes one AMR line item corresponding to item 3.5.1, 085 with generic note A corresponding to NUREG-2191 item III.A6.TP-221 and AMPs Water Chemistry (B.2.3.2) & One-Time Inspection (B.2.3.20). This NUREG-2191 AMR item (III.A6.TP-221, which in fact applies to water control structures and corresponds to SRP-SLR item 3.5.1-083 of steel bolting exposed to water flowing or standing) does not address the material, environment and AMPs consistent with disposition of SLRA item 3.5-1, 085 and appears to be in error. a) Provide Table 3.5.2 AMR line items for item 3.5.1, -085 (loss of material	To verify if components included in item SLRA 3.5.1, 085 are subject to AMR and consistent with GALL-SLR as claimed in Table 3.5-1 and Table 3.5.2-9.

			of stainless steel structural bolting exposed to treated water) consistent with its disposition in SLRA Table 3.5-1. b) Clarify and correct any errors, omissions or inconsistencies between Table 3.5.1 and Table 3.5.2s and GALL-SLR with regard to AMR item 3.5.1, 085.	

SLRA Section B.2.3.2 Water Chemistry

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	B.2.3.2	B-48	<u>Scope of Program - Enhancements</u> The table of enhancements to the scope of Water Chemistry lists two enhancements: • Incorporate monitoring of critical chemistry parameters for the Heating Steam System according to ASME standard ISBN-0-7918-1204-9. • Perform a one-time inspection to verify the effectiveness of incorporating the monitoring according to the ASME standard. According to Attachment C to Report No. FPLCORP00036-REPT-043, Rev. 0 (the B.2.3.2 AMP basis document), both enhancements will be implemented six months prior to the SPEO. It is unclear how a one-time inspection can evaluate the effectiveness of the chemistry control monitoring at the same time the monitoring is implemented. Please clarify the schedule for implementation of the enhanced water chemistry controls and one-time inspection for the Heating Steam System.	There appears to be an inconsistency in the description of enhancements and the implementation schedule.
2	B.2.3.2	B-48	<u>Clarification of Enhancements</u> The enhancements to the Water Chemistry address water chemistry control for the	It is not clear how the water chemistry requirements for the Heating Steam

			Heating Steam System. For primary and secondary water chemistry, there are several PBN documents that discuss strategy, controls, and procedures. It is not clear to the staff how the enhanced chemistry requirements for the Heating Steam System are incorporated into plant documents. Therefore, please address the following questions: a) In what procedures or programs are the current and future specifications for the Heating Steam System documented other than the SLRA? The chemistry for this system does not appear to be addressed by any procedure documents in the AMP folder. b) ASME standard ISBN-0-7918-1204-9 has chemistry guidelines for six different boiler types. How are guidelines in the standard applied to the PBN boilers and how is this documented? c) What water chemistry controls are currently being applied to the Heating Steam System?	System are being implemented in plant procedures.
3	N/A	N/A	<u>Technical Requirement Manual, Section 3.4.5, “Reactor Coolant Oxygen, Chloride and Fluoride Concentrations”</u> The PBN limits for oxygen, chloride, and fluoride in the reactor coolant (100 ppb, 150 ppb, 150 ppb, respectively) correspond to Action Level 2 limits in the EPRI primary water chemistry guidelines rather than Action Level 1 limits (Table 3-3). This would appear to allow operation above the EPRI Action Level 1 indefinitely. Please explain the basis for the limits in the surveillance requirements (TSR 3.4.5.1, 3.4.5.2, and 3.4.5.3) with respect the limits in the EPRI primary water chemistry guidelines.	Explanation is requested because of a potential inconsistency between the Technical Requirements Manual and the EPRI guidelines referenced in the Water Chemistry program.
4	N/A	N/A	<u>Water Chemistry Health Reports</u>	The availability of hydrazine injection affects the ability to meet the Water

			An issue with loss of hydrazine injection was identified in at least the following health reports. • Q1 2018 – loss of hydrazine feed pump manual venting • Q3 2019 – hydrazine feed pump manual • Q1 2020 – loss of hydrazine addition Because hydrazine injection may be required for meeting the water chemistry requirements, please provide the following information: a) Please describe the current status of the loss of hydrazine injection apparently requiring manual venting of the feed pump. b) What is the current system for ensuring availability of hydrazine injection?	Chemistry program requirements.
5	N/A	N/A	<u>Water Chemistry AMP OE Interview</u> The end of the Water Chemistry AMP OE interview refers to “report of Steam Chem Cleaning on U1 which had significant issues.” The staff did not find information about these chemical cleaning issues in other documents. Because chemical cleaning issues could affect water chemistry or components such as steam generator tubes, please describe the Unit 1 chemical cleaning issues, including when the events occurred and any effects on degradation of components or ongoing effects on aging management.	Issues with chemical cleaning could affect the water chemistry or degradation.

6			Background The Water Chemistry Program folder: "XI.M2 Wtr_Chem/Ref/Procedures Specs Etc." has a document that appears to be specific for the Turkey Point Nuclear Plant: 0-ADM-651, "Nuclear Chemistry Parameters Manual," Revision 17, 6/2/20, Florida Power & Light Company, Turkey Point Nuclear Plant. The document Purpose states, "Instructional guidance for implementing water chemistry controls in accordance with EPRI guidelines, Turkey Point Plant TS, and vendor/industry recommendations." The response to Turkey Point RAI B.2.3.2-3 (Attachment 20 in ML18296A024) states that O-ADM-651 is the controlling document for primary water chemistry values. I have not seen a reference to 0-ADM-651 in the Point Beach documents that would indicate it is the basis for the Point Beach program. Questions: • What is the purpose of this document being in the folder - does it apply to Point Beach? • If it does apply to Point Beach, where is that stated in the Point Beach documents? • If it does not apply to Point Beach, is there a comparable document for Point Beach, other than those already on the portal, that controls primary water chemistry values?	

Structures Monitoring

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	B.2.3.34	B-238	Inspector Qualifications. Clarify how the existing inspectors' qualifications requirements from Section 3.5 of NP 7.7.9 is consistent with the qualification requirements specified in ACI 349.3, Revision 2. Section 3.5 of NP 7.7.9 states, in part: • Responsible Engineer: shall be a registered professional engineer OR	GALL-SLR Report Section XI.S6 – Detection of Aging Effects. The GALL-SLR recommends that inspector qualifications be consistent with industry guidelines and standards and

			• a degreed civil engineer… <u>two years minimum</u> experience in structural engineer for nuclear structures <u>Chapter 7 of ACI 349.3</u> Revision states, in part: <u>Responsible Engineer</u>: shall be a licensed professional engineer OR Civil or structural engineering graduate of an ABET accredited college or university <u>with at least 10 years' experience</u> in the design, construction, and inspection of concrete structures, and with knowledge of the performance requirements of nuclear safety-related structures and potential degradation processes.	guidelines for implementing the requirements of 10 CFR 50.65. Where the qualifications of inspection and evaluation personnel specified in ACI 349.3R (Revision 2) are found acceptable for inspection of concrete structures. SLRA claims that the AMP will be consistent with the GALL-SLR Report with no exceptions.
2	B.2.3.34	B-238	<u>Inaccessible Areas</u>. • Describe how and when inaccessible areas are <u>evaluated</u> for acceptability. • Do all inaccessible areas have an accessible area available for inspections (i.e., with similar conditions to those inaccessible areas)? Otherwise, how these inaccessible areas without a similar accessible area are being manage for aging effects? Section 4.2.4 of NP 7.7.9 discusses how inaccessible areas are monitored and states, in part, that "accessible areas subject to similar conditions may be <u>evaluated in lieu</u> of inaccessible areas." However, it is not clear how this process is consistent with GALL-SLR Report recommendation. The GALL Report seeks to <u>evaluate the (actual) inaccessible areas</u> when condition exists in similar accessible areas may indicate (as results from their inspections) the presence or results in degradation of the similar inaccessible area. The AMP (as written) seems to suggest that only the accessible areas will be one evaluated and not the (actual) inaccessible area when the accessible area indicate/suggest an ongoing degradation may be present in a similar inaccessible	<u>GALL-SLR Report Section XI.S6 – Detection of Aging Effects</u>. The GALL-SLR recommends, in part, "<u>evaluating the acceptability of inaccessible areas</u> when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas" – for sites with nonaggressive groundwater/soil environments. SLRA claims that the AMP will be consistent with the GALL-SLR Report with no exceptions.

			area – Was the program process meant to say “be inspected in lieu” as opposed to “be evaluated in lieu”?	
3	B.2.3.34	B-239	Water In-Leakage. Describe how the program will monitor the concrete when in-leakage or groundwater infiltration is identified? It is noted that an SLRA enhancement to the AMP’s “parameters monitored or inspected” program element states that the AMP will be enhanced to ensure that “concrete will be monitored to confirm the absence of water in-leakage” However, it is not clear how this enhancement aligns with the GALL-SLR Report recommendation.	GALL-SLR Report Section XI.S6 – Parameters Monitored of Inspected. The GALL-SLR recommends that “if through-wall leakage or groundwater infiltration is identified, leakage volumes and chemistry are monitored and trended for signs of concrete or steel reinforcement degradation.” SLRA claims that the AMP will be consistent with the GALL-SLR Report with no exceptions.
4	B.2.3.34	B-239	Epoxy Grouted Anchors. • Are any epoxy grouted bolts/anchors used in components that are in the scope of the SLR or safety related applications for Point Beach? Please specify. • If there are, demonstrate how the Structures Monitoring Program manages these types of components. Section 4.1 of the AMP basis document indicates that epoxy grouted anchors are within the scope of the Structures Monitoring Program. However, the NRC staff was unable to confirm that NP 7.7.9 already manages these epoxy grouted anchors/bolts and/or if any of the program elements associated for this type of component (parameters monitored, preventive actions, aging effects inspection criteria, etc.) were addressed for the program since the	GALL-SLR Report Section XI.S6 – Scope of Program. The scope of the program includes all SCs, component supports, and structural commodities in the scope of license renewal that are not covered by other structural AMPs. SRP-SLR. Aging management review results not consistent with or not addressed in the GALL-SLR Report needs to follow the acceptance criteria described in Appendix A.1 of the SRP-SLR. Specifically, the

			GALL-SLR Report XI.S6 does not generically include/address this type of component. Also, SLRA B2.3.34 does not include enhancements to ensure that the AMP will manage this component during the period of extended operation.	scope of the program should, in part, identify the specific structures and/or components, identify the aging effects that the program manages, and describe the acceptance criteria that will be used to ensure that the intended function(s) are maintained consistent with the CLB during the subsequent period of extended operation.
5	B.2.3.34	B-238	<u>Quantitative Baseline Inspection Data</u>. • Describe how the AMP ensures that the inspectors provide quantitative measurements (when available) as part of their recorded and trended information during the inspections. • Are there any plans to enhance previously performed inspections with quantitative inspection data prior to the subsequent period of extended operation? Per the staff review of prior inspections reports, it was noted that the use of quantitative measure was minimal and documented results were mostly based on qualitative information which makes it difficult to track and trend degradations.	<u>GALL-SLR Report Section XI.S6 – Monitoring and Trending</u>. The GALL-SLR Report recommends that quantitative baseline inspection data be established per the acceptance criteria described in the AMP prior to the subsequent period of extended operation. SLRA claims that the AMP will be consistent with the GALL-SLR Report with no exceptions.
6	Table 3.5-1, Table 3.5.2-1	Page 3.5-70 Page 3.5-94	<u>Sliding Surfaces</u>: Clarify what sliding surfaces are in scope of SLR and what AMP manages these components. • Are there any sliding surfaces within the scope of LR that will be manage by the Structures Monitoring Program? Per Section 4.1 of AMP basis document, there are no sliding surfaces located outside the containment in the scope of SLR. Thus, sliding surfaces inside containment are managed by the IWF program. However, an SLRA Table 2 line item exists for sliding surfaces as being managed by the SMP, not the IWF program.	<u>There are some inconsistencies identified in the SLRA</u>. • SLRA Table 3.5-1, item 3.5-1, 074 states that this item is <u>Not Applicable</u>. • However, SLRA Table 3.5.2-1 states that Sliding Surfaces <u>will be monitored</u> for loss of

			Also, no Sliding Surface (table 2 item) is being proposed by the SLRA to be managed by the IWF Program as indicated in the basis document. Therefore, Table 2 (component – sliding surfaces) line item associated with Table 1 item 3.5-1, 075 is missing from the SLRA.	mechanical function by the Structures Monitoring Program • SLRA Table 3.5-1, item 3.5-1, 075 states that it is consistent with the GALL-SLR Report. However, no SLRA Table 2 item is associated with item 3.5-1, 075.
7	Table 3.3-1 Table 3.5.2-1	Page 3.3-51 Page 3.5-88 Page 3.5-90	New Fuel Storage Racks. Clarify if Point Beach NPP has a new fuel storage racks made of steel or stainless steel material, and if so, how this component will be managed during the period of extended operation. It is not clear if Point Beach NP has this component, material and environment combination and/or how it will be managed during the period of extended operation. Some inconsistency was noted between the GALL-SLR Report intent for this line item and how the SLRA use it. The SLRA appears to repurpose the 3.3-1, 111 line item to manage the liner in the reactor cavity (SLRA Page 3.5-88) and other miscellaneous structural components in the containment building / internal structural components (SLRA page 3.5-90). However, this type of component appears not to be directly related to the new fuel storage racks component for which this line item was intended.	There are some inconsistencies identified in the SLRA. • SLRA Table 3.3-1, states that item 3.3-1, 111 is consistent with the GALL-SLR Report. • However, this line item per the GALL-SLR Report is associated with new fuel storage racks made out of steel material exposed to an air – indoor uncontrolled environment. (Page VII A1-2, Table A1)
8	Table 3.5.2-11 B.2.3.34	Page 3.5-130 Page B-238	Styrofoam Material in Manway Insulation Boards. Clarify/Demonstrate how the existing AMP will manage the aging effects of Styrofoam materials in manway insulation boards from yard structures. After reviewing NP 7.7.9, it is not clear if the Styrofoam material is within the scope of the SMP, and/or how the AMP will manage age related degradation of this material within the SMP (aging effect/acceptance criteria). Also, no enhancement was identified in SLRA	GALL-SLR Report Section XI.S6 – Scope of Program. The scope of the program includes all SCs, component supports, and structural commodities in the scope of license renewal that are not covered by other structural AMPs.

			Section B.2.3.34 to demonstrates that this component and material combination will be adequately managed by the SMP during the period of extended operation.	SRP-SLR. Aging management review results not consistent with or not addressed in the GALL-SLR Report needs to follow the acceptance criteria described in Appendix A.1 of the SRP-SLR. Specifically, the scope of the program should, in part, identify the specific structures and/or components, identify the aging effects that the program manages, and describe the acceptance criteria that will be used to ensure that the intended function(s) are maintained consistent with the CLB during the subsequent period of extended operation.
9	Table 3.5-1, 095	Page 3.5-75	Galvanized Steel Support Members. Clarify if there are galvanized steel support members; welds; bolted connections; support anchorage to building structure associated with ASME Class 1, 2, 3 and MC components. If so, demonstrate how they are being managed during the period of extended operation. SLRA Table 3.5-1, item 3.5-1, 095 states that the line item was not used since galvanized steel is included with steel components supports addressed in item 3.5-1, 092. However, no Table 2 items associated with item 3.5-1, 092 appears to be related to ASME Class 1, 2, 3 and MC components (as intended by Table 1 item 3.5-1, 095). Therefore, it is not clear how galvanize steel structural components in Class 1, 2, 3 and MC systems are being manage for the period of extended operation (if within the scope of SLR).	Per the GALL-SLR Report, item 3.5.1-095 addresses galvanize steel in ASME Class 1, 2, 3 and MC components among other general ones. SLRA states that the components associated with this line item are addressed under line item 3.5.1-092.

10	B.2.3.34	Page B-241	OE associated with Water Intrusion/Seepage. Discuss the general site-specific operating experience related to water intrusion/seepage. • How there are being identified? • How they have been tracked/monitored? • Major impacts to component, equipment, structures. • What different corrective action has been taken? Review of the OE Report has identified numerous conditions where water seeping have been identified coming thru cracks, groundwater infiltrations, and water sipping from walls and ceiling.	To understand how the existing AMPs is bounding the identified age-related degradation.
11	B.2.3.34 Table 16-3	Page B-239 Page A-105	Inconsistency between SLRA Commitment No.38 and associated AMP enhancements. Per second enhancement in B.2.3.34, the enhancement seeks to update procedures "*to include preventive actions to ensure bolting integrity for replacement and maintenance activities by specifying proper selection of bolting material and lubricant, and…*" However, no commitment (under Commitment No. 38) was identified to address/capture this enhancement. Can it be clarified why it is missing as a commitment?	The regulation in 10 CFR 54.21(d) requires the SLRA to include a FSAR Supplement summary description for each AMP or aging management activity and each TLAA that is included in the SLRA Per SRP-SLR, this includes incorporation of all FSAR Supplement summary descriptions that were included in the FSAR Supplement for AMPs, aging management activities, and TLAAs included in the SLRA, and any programmatic or TLAA-related enhancements or commitments that may have been included in the FSAR Supplement, including those for implementing the AMPs, aging management activities and TLAAs

				during the subsequent period of extended operation.

Point Beach SLRA Breakout Question Section B.2.3.24 Flux Thimble Tube Inspection

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	B.2.3.24	B-180 through B-182	**Discussion Topic and Questions:** Discussion Topics: Please be prepared to discuss how NextEra Energy uses wear rate data to project the amount of wear that may occur in the thimbles at the next inspection, and also the type of wear rate projection and associated wear-rate value that is currently used for the eddy current testing reinspection interval for the tubes. Question: Do the inspections for the thimble tubes in each unit (especially for the limiting worn tube in each unit) demonstrate that wear may be occurring in tubes under an increasing, non-linear wear rate?	Wear rate is a key factor in the calculation of inspection frequency. If wear rates are increasing and non-linear, that would result in non-conservative inspection frequencies. NRC staff observed during review of documents on the portal that the limiting wear rate for Unit 1 was calculated to be 14% based on observations during the first 3 years of operation. It is unclear how these wear rates have changed over time.
			Issue: HX-02 Rev 11 “Thimble Tube Condition Assessment Program”, dated January 20,2012, states that the previous thimble tube	If Unit 1 scheduled inspections were not performed in 2013, the resulting

			inspection for Unit 1 was performed in March 2010. HX-02 also states that the next scheduled thimble tube inspection for Unit 1 would be performed in Spring 2013. HX-02 states that the worst thimble tube from Unit 1 was 1-G6 with a minimum estimated life of 5.5 years. HX-02 recommended that program requirements would indicate a 3 year interval (1 cycle before 4.5 years) based on tube 1-G6. FPLCORP00036-REPT-057 “Aging Management Program Basis Document-Thimble Tube Inspection”, dated November 6, 2020, mentions Unit 1 thimble tube inspections performed in the Spring of 2010 and the Spring of 2017. There is no mention of any inspection in the Spring of 2013. Questions: What is the status of tube 1-G6 (i.e. in-service, plugged)? Were the Unit 1 thimble tubes inspected in the Spring of 2013? If not, explain how this is consistent with the thimble tube condition assessment program requirements as stated in HX-02?	7 year inspection interval does not agree with the recommended 3 year inspection frequency specified in the licensee’s basis documents. Also, if the wear rate for tube 1-G6 met the worst case projections, it is possible that it might have been taken out of service. If tube 1-G6 was taken out of service, it will affect inspection frequency calculations going forward.
			Issue: HX-02 Rev 11 (issued in January 2012) states that the previous thimble tube inspection for Unit 2 was performed in March/April 2011. HX-02 also states that the next scheduled thimble tube inspection for Unit 2 would be performed in Fall 2012. HX-02 states that the worst thimble tube from Unit 2 was 31-J12 with a minimum estimated life of 10.4 years. HX-02 recommended that program requirements would indicate a 6 year interval for Unit 2. REPT-057 mentions thimble tube inspections performed in the Spring of 2011 and the Fall of 2017. There is no mention of any inspection in the Fall of 2012. Question: Were the Unit 2 thimble tubes inspected in the Fall of 2012? If not, were they inspected at any other time between Spring 2011 and Fall 2017? If not, explain how this is consistent with the thimble tube condition	If Unit 2 scheduled inspections were not performed in 2012, the resulting 6-1/2 year inspection interval does not agree with the recommended minimum 6 year inspection frequency specified in the licensee’s basis documents.

			assessment program requirements that inspection intervals should not exceed 6 years?	
			Issue: In the Plant Specific Operating Experience section of REPT-057, the licensee mentions concerns related to inspection deferrals, calculating methodology and record retention, and references the following ARs (AR00004630, AR00004646, AR00004658). However, when the NRC staff looked up records AR00004630, AR00004646, and AR00004658 on the portal (under folder ARs WOs Etc.), the records did not appear to address inspection deferrals, calculating methodology and record retention. These ARs instead discussed boric acid buildup on a packing gland, discrepancies in documents related to electrical safety, and license renewal review of the leak detection system. Question Please be prepared to discuss the concerns related to inspection deferrals, calculating methodology and record retention. If documents relevant to these topics can be found on the portal, please demonstrate where these may be found, and briefly explain the contents of these documents.	Issues of inspection deferrals, calculation methodology, and record retention are relevant to the NRCs safety evaluation. NRC staff need to be able to audit and understand these ARs that are referenced in the basis document and are relevant to the safety evaluation.
2	B.2.3.24	B-180 through B-182	Issue: The Recommendations in HX-02 Rev. 11 stated that program requirements would indicate a 3 year interval (1 cycle before 4.5 years) for Unit 1 thimble tube inspections based on 1-G6, the limiting tube for Unit 1. Npm 2013-0123 stated that the Unit 1 thimble tubes were routinely eddy current tested in March 2013 and recommended that they be inspected again in Spring 2016. FPLCORP00036-REPT-057 stated that Unit 1 thimble tube inspections were performed in the Spring of 2017. There is no mention of any inspection in the Spring of 2016. Question:	If Unit 1 scheduled inspections were not performed in 2016, the resulting 4 year inspection interval does not agree with the recommended 3 year inspection frequency specified in the licensee's basis documents.

			Were the Unit 1 thimble tubes inspected in the Spring of 2016? If not, explain how this is consistent with the thimble tube condition assessment program requirements as stated in HX-02?	

Fire Protection

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	2.3, 3.3	2.3-43, 2.3-45, 3.3-153, 3.3-177	SLRA Tables 2.3.3-5, "Service Water System Components Subject to Aging Management Review," and 2.3.3-6, "Fire Protection System Components Subject to Aging Management Review," include hose reel. In addition SLRA Tables 3.3.2-5, "Service Water System – Summary of Aging Management Evaluation," and 3.3.2-6, "Fire Protection System – Summary of Aging Management Evaluation," include programs for managing loss of material, long-term loss of material, and flow blockage of the carbon steel hose reel exposed to air – indoor uncontrolled (ext) and raw water (int). Please discuss whether the carbon steel hose reel is a component subject to aging management review in both the Service Water and Fire Protection Systems.	The U.S. Nuclear Regulatory Commission (NRC) staff is seeking clarification on whether the carbon steel hose reel is a component of both the Service Water and Fire Protection Systems.
2	3.5	3.5-16,	Section 3.1, "Program Overview and Background," of FPLCORP00036-REPT-052, "Point Beach Units 1 and 2 Subsequent License Renewal Aging Management Program Basis Document – Fire Protection," cites Flamemastic as a fire-resistant material that serves a fire barrier function. Flamemastic is a water-based thermoplastic resin used as fireproofing. SLRA Section 3.5.2.1.14, "Fire Barrier Commodity," and SLRA Table 3.5.2-14, "Fire Barrier Commodity Group – Summary of Aging Management Evaluation," does not specifically cite Flamemastic. Please discuss whether Flamemastic is used at PBN and whether it should be specifically cited in SLRA Section 3.5.2.1.14 and Table 3.5.2-14.	The NRC staff is seeking clarification on whether Flamemastic is used at PBN and whether it should be specifically cited in SLRA Section 3.5.2.1.14 and Table 3.5.2-14.
3	3.3, 3.5	3.3-82, 3.5-17,	SLR-ISG-2021-02-Mechanical, "Updated Aging Management Criteria for Mechanical	The NRC staff is seeking clarification

		3.5-16, 3.5-33, 3.5-139,	Portions of Subsequent License Renewal Guidance" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20181A434) added AMR Items VII.G.A-805, VII.G.A-806, and VII.G.A-807 to Table VII.G in NUREG-2191, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," and Table 3.3-1 in NUREG-2192, "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants." The aging effects for cementitious coatings, silicates, and subliming compounds used as fireproofing/fire barriers exposed to air are loss of material, change in material properties, cracking/delamination, and separation. These aging effects are consistent with Section 6, "Fire Barriers," of EPRI 3002013084, "Long-Term Operations: Subsequent License Renewal Aging Affects for Structures and Structural Components (Structural Tools)," November 2018. Table 5-3, "Structural Tools Comparison with GALL-SLR-Structural Concrete Members," in Section 5, "Structural Concrete Members," of EPRI 3002013084 provides applicability criteria for aging effects/mechanisms for concrete structures and concrete components. Table 5-3 notes that change in material properties due to elevated temperature is applicable for concrete structures and concrete components when the general area temperature exceeds 150°F (65.6°C) or when the local area temperature exceeds 200°F (93.3°C). Therefore, applicants need to make a plant-specific determination of whether concrete structures and concrete components are exposed to temperatures exceeding these values. This is consistent with further evaluations related to concrete exposed to elevated temperatures recommended in Chapter 3.5, "Aging Management of Containments, Structures, and Component Supports," of NUREG-2192. Table 5-3 of EPRI 3002013084 also	on the aging effects for fireproofing and fire stops and wraps.

			notes that temperatures exceeding these values potentially result in loss of material and cracking of concrete structures and concrete components. Table 6-3, “Structural Tools Comparison with GALL-SLR-Fire Barriers,” in Section 6 of EPRI 3002013084 provides applicability criteria for aging effects/mechanisms for fire barrier materials. Table 6-3 notes that change in material properties due to gamma irradiation exposure is applicable for cementitious coatings, rigid fire boards (subliming compounds), and fibrous fire wraps (silicates) when exposures exceed 10^6 rads. A plant-specific determination would need to be made on whether cementitious coatings, subliming compounds, and silicates used as fireproofing/fire barriers would be exposed to greater than 10^6 rads. Section 6 of EPRI 3002013084 cites “cracking/delamination” as aging effects. SLRA Table 3.3-1, “Summary of Aging Management Evaluations for the Auxiliary Systems,” does not include change in material properties or delamination for GALL-SLR Items VII.G.A-805, SRP Item 3.3-1, 268; and VII.G.A-806, SRP Item 3.3-1, 269. SLRA Section 3.5.2.1.14 does not include change in material properties and delamination as aging effects associated with fire barrier commodities. In addition, SLRA Table 3.5.2-14 does not include loss of material properties and delamination for the applicable fireproofing and fire stop and wrap materials exposed to air. 1. While SLRA Section 3.5.2.2, “AMR Results for Which Further Evaluation is Recommended by the GALL Report,” discusses the general and local area temperature limits, please discuss whether cementitious coatings used as fireproofing will be in general areas where the temperature exceeds 150°F (65.6°C) or in local areas where the temperature exceeds 200°F (93.3°C). If cementitious coatings used as fireproofing will be in areas	

			exceeding the general and local area temperatures, then explain why change in material properties due to elevated temperature is not an applicable aging effect/mechanism that requires management. 2. SLRA Section 3.5.2.2.2.6, "Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation," states, "Furthermore, Group 4 structures that are commodities inside containment are located outside the primary shield wall. As such, Group 4 structures that are commodities will not experience cumulative fluence or gamma irradiation above the thresholds; thus, reduction of strength and mechanical properties of concrete or embrittlement of steel due to irradiation is not an applicable aging effect for the component support or fire barrier commodities or cranes located inside containment." Please discuss whether the gamma irradiation exposure for any fire stops and wraps exposed to air will exceed 10^6 rads. If the gamma irradiation exposure for any fire stops and wraps will exceed 10^6 rads, then explain why change in material properties due to gamma irradiation exposure is not an applicable aging effect/mechanism that requires management. 3. While Table 1, "Aging Effects/Mechanisms Managed by this Program," of FPLCORP00036-REPT-052 cites delamination as an aging mechanism for loss of material (aging effect), please discuss why delamination was not identified as an applicable aging effect for fireproofing and fire stops and wraps exposed to air given that EPRI 3002013084 cites "cracking/delamination" as aging effects.	
4	N/A	N/A	Section 4.4, "Detection of Aging Effects," of FPLCORP00036-REPT-052 states that visual inspection and periodic testing of the halon fire suppression system is performed annually. In addition, Section 4.8.2, "Surveillance of Fire Protection Systems," of NP 1.9.14, "Fire Protection Plan," includes	The NRC staff is seeking clarification on the frequency of the visual and functional testing of the halon fire suppression system.

			the following test frequencies for the halon gaseous suppression system: • 6 Months – halon quantity verification • Yearly – functional test • Yearly – visual header and nozzle inspection However, procedures, such as TS 78, “Semi-Annual Halon 1301 Fire Suppression System Surveillance Test,” appear to be semiannual. Please confirm the frequency of the visual and functional testing of the halon fire suppression system.	
5	3.5	3.5-139	Sections 4.1, “Scope of Program,” and 4.4, “Detection of Aging Effects,” of FPLCORP00036-REPT-052 states that the PBN Fire Protection program inspects for cracking, loss of material, and shrinkage on fire wrapping. However, SLRA Table 3.5.2-14 does not cite shrinkage as an aging affect requiring management for fire wraps. The fire wrap materials cited in SLRA Table 3.5.2-14 are calcium silicate board, and ceramic fiber, board, and mat. In EPRI 3002013084, shrinkage is cited as an aging mechanism for cracking/delamination and separation of elastomer fire stops. Please discuss whether shrinkage is an aging affect requiring management for fire wraps. If so, please discuss for what specific fire wrap material(s) it is an applicable aging affect.	The NRC staff is seeking clarification on whether shrinkage is an aging affect requiring management for fire wraps, including which specific fire wrap material(s).
6	3.5	3.5-32, 3.5-139	SLRA Section 3.5.2.2.2.4, “Cracking Due to Stress Corrosion Cracking [SCC] and Loss of Material Due to Pitting and Crevice Corrosion,” states that the environment for stainless steel fire barrier penetration seals is not expected to be aggressive enough to cause cracking or localized loss of material. This SLRA section goes on to state: In addition, stainless steel structural components are limited in number in	The NRC staff is seeking clarification on whether cracking is an applicable aging affect for the stainless steel fire barrier penetration seals.

			comparison to the amount of stainless-steel mechanical components. Furthermore, there has been no site operating experience of cracking or localized corrosion of stainless steel or aluminum SSCs. As such, cracking due to SCC and loss of material due to pitting and crevice corrosion is conservatively an applicable aging effect at PBN for stainless steel and aluminum and is managed with the External Surfaces Monitoring of Mechanical Components (B.2.3.23) AMP, which will interface with the Structures Monitoring (B.2.3.34) AMP, the Fire Protection (B.2.3.15) AMP and the ASME Section XI, Subsection IWE (B.2.3.29) AMP if degradation is detected in the mechanical components. SLRA Table 3.5.2-14 does not cite a program(s) for managing cracking in stainless steel fire barrier penetration seals. In addition, the table cites the Fire Protection AMP for managing loss of material of the stainless steel fire barrier penetration seals and includes Generic Note E and Plant Specific Note 2. Plant Specific Note 2 states, "Stainless steel that is exposed to air-indoor uncontrolled during normal plant operation are inspected under the Fire Protection AMP with the structural equivalent of the NUREG-2191 XI.M36, Externals Surfaces Monitoring of Mechanical Components AMP." Please discuss whether cracking is an applicable aging effect for the stainless steel fire barrier penetration seals that requires management. If it does, please identify the program(s) that will manage cracking.	
7	Appendix A, Appendix B	A-64, B-122	Section 4.0, "Aging Management of Program Elements," of FPLCORP00036-REPT-052 includes the following enhancement to Elements 1, 3, 4, and 6 of the Fire Protection program: Include inspections of the oil collection channels, trenches, and skids credited to mitigate the spread of combustible liquids for cracking and loss of material at least once every 18 months.	The NRC staff is seeking clarification on the inspection of the oil collection channels, trenches, and skids. [Note: The NRC staff noted that "preforming" is used instead of

			SLRA Section B.2.3.15, “Fire Protection,” and Table 16-3, “List of SLR Commitments and Implementation Schedule,” do not include this enhancement and commitment. Please confirm whether inspection of the oil collection channels, trenches, and skids credited to mitigate the spread of combustible liquids for cracking and loss of material at least once every 18 months is an enhancement to the PBN Fire Protection program and an SLR commitment.	“performing” in the enhancement related to qualified personnel in SLRA Section B.2.3.15.]
8	3.5	3.5-139	GALL-SLR Item VII.G.A-789, SRP Item 3.3-1, 255 identifies the aging effects for fire damper assemblies as loss of material due to general, pitting, crevice corrosion; cracking due to SCC; hardening, loss of strength, shrinkage due to elastomer degradation. The term “fire damper assembly” includes both the frame and the damper as evidenced by the aging effects requiring management as cited in GALL-SLR Item VII.G.A-789. For example, hardening and loss of strength would not be applicable aging effects if the intent of the GALL-SLR were to only manage aging effects associated with housings, which are typically constructed of steel materials. SLRA Table 3.5.2-14 cites GALL-SLR Items III.B1.1.TP-3, SRP Item 3.5-1, 089 and VII.G.A-789, SRP Item 3.3-1, 255 only for the “housing” of the fire dampers and louvers as a component with aging effects requiring management. The NRC staff notes that fire dampers and louvers are listed together in the Component Type column of SLRA Table 3.5.2-14. The NRC staff notes that “frames,” “housing,” “assemblies,” and “assembly” are all used in the SLRA. SLRA Section 2.3.3.6, “Fire Protection,” uses “fire damper housings.” SLRA Section 2.4.14, “Fire Barrier Commodity,” uses “fire damper and louver housings.” SLRA Table 2.4-14, “Fire Barrier Commodity Group Subject to Aging Management Review,” and Table 3.5.2-14 use “fire damper and louver frames.” SLRA Table 3.5-1, “Containment Building Structure	The NRC staff is seeking clarification on the materials of construction of the fire damper assemblies and the applicable aging affects for the fire damper assemblies.

			and Internal Structural Components – Summary of Aging Management Programs," uses "fire dampers and louvers." However, SLRA Table 3.3-1, Section 16.2.2.15, "Fire Protection," Table 16-3, and Section B.2.3.15 use "fire damper assemblies" and/or "fire damper assembly." Please state the material of construction for the fire damper assemblies other than the housing that perform their intended isolation function in the closed position and the basis for why the aging effects cited in GALL-SLR Item VII.G.A-789, SRP Item 3.3-1, 255 are not applicable to portions of the fire damper assembly other than the housing.	
9	N/A	N/A	The PBN Program Owner Interview Form, noted "Corrosion issue is recurring." Please provide additional detail regarding the recurring corrosion, including components, location, extent of corrosion, and any corrective actions taken. The PBN Program Owner Interview Form also noted that the fire probabilistic risk assessment (PRA) was unsatisfactory. Please provide the current status.	The NRC staff is seeking clarification on the recurring corrosion issue noted during the Fire Protection program owner interview. In addition, the staff is seeking the current status of the fire PRA.
10	N/A	N/A	Section 5.8.2, "Wrapped Conduits and Cable Trays and HVAC Duct," of Routine Maintenance Procedure (RMP) 9057, "Fire Barrier Penetration Fire Seal Surveillance," states, "If the aluminum or stainless steel wrapping has been punctured or damaged, then record on the survey forms and contact Supervisor." The NRC staff notes that aluminum is not included in SLRA Section 3.5.2.1.14 or Table 3.5.2-14. Please discuss whether aluminum is used as a fire wrap at PBN.	The NRC staff is seeking clarification on whether aluminum is used for fire wrap.
11	N/A	N/A	Action Request 02152508 is related to fire penetration seals not being inspected. It appears that many were not inspected due to access limitations. It also appears that the Fire Protection Engineer would be required to approve fire penetration seals not being inspected.	The NRC staff is seeking clarification on the process used when fireproofing/fire barriers cannot be inspected.

			Please walk the NRC staff through the process for when fireproofing/fire barriers cannot be inspected.	
12	3.5	3.5-127	GALL-SLR Item VII.G.A-90, SRP Item 3.3-1, 060 is for managing cracking and loss of material of reinforced concrete structural fire barriers (walls, ceilings, and floors) exposed to air by AMP XI.M26, “Fire Protection,” and XI.S6, “Structures Monitoring.” SLRA Table 3.3-1 states consistency with GALL-SLR Item VII.G.A-90, SRP Item 3.3-1, 060. Table 3.5.2-10, “Turbine Building Structure – Summary of Aging Management Evaluation,” cites GALL-SLR Item VII.G.A-90, SRP Item 3.3-1, 060 for concrete: interior walls and ceilings with intended functions of shelter, protection, structural support. Fire barrier was not cited as an intended function. Please discuss whether fire barrier should be cited as an intended function.	The NRC staff is seeking clarification on the intended functions of the reinforced concrete interior walls and ceilings in the turbine building structure.
13	3.5	3.5-93	Radiant Energy (Heat) Shield is defined in NP 1.9.14 as “A noncombustible or fire resistive barrier installed to provide separation protection of redundant cables, equipment and associated non-safety circuits within containment.” Section 2.2.3, “Electrical Raceway Fire Barrier System (ERFBS),” states, “Inside containment cable wrap may also be used to provide a non-combustible radiant energy shield…” Section 2.2.5, “Cable Tray Covers and Ceramic Blankets in Cable Trays,” appears to describe the radiant energy shield as ceramic fiber blanket on top of the cables and metal tray covers on top and bottom of each cable tray. Stainless steel radiant energy shields are included in SLRA Table 3.5.2-1, “Containment Building Structure and Internal Structural Components – Summary of Aging Management.” Please discuss whether fireproofing/fire barrier materials other than stainless steel are considered radiant energy shields in containment.	The NRC staff is seeking clarification on whether materials other than stainless steel are considered radiant energy shields in containment.

14	3.3	3.3-177	GALL-SLR Item VII.G.A-423, SRP Item 3.3-1, 142 is for managing loss of material of stainless steel and steel closure bolting exposed to raw water. SLRA Table 3.3.2-6 does not cite the raw water environment for either the carbon steel or stainless steel bolting in the Fire Protection System. Please discuss whether raw water is an applicable environment that the carbon steel and stainless steel bolting in the Fire Protection System is exposed to.	The NRC staff is seeking clarification on whether the carbon steel and stainless steel bolting in the Fire Protection System is exposed to raw water.
15	3.3	3.3-82	Attachment A – NEI 04-02 Table B-1 Transition of Fundamental Fire Protection Program & Design Elements of License Amendment Request 271, Transition to 10 CFR 50.48(c) – NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition (ADAMS Accession No. ML13182A353) states, "All ERFBS credited for NFPA 805 Chapter 4 compliance at PBNP is 3M Interam E-50 series fire wrap." However, the Discussion Column for Item Number 3.3-1, 267 in SLRA Table 3.3-1 states, "Not applicable. There are no subliming compounds (Thermo-lag®, Darmatt™, 3M™ Interam™, and other similar materials) exposed to air in the Auxiliary Systems." Please discuss whether 3M™ Interam™ E-50 series fire wrap is used for electrical raceway fire barrier systems. If it is, please discuss the basis for stating Item Number 3.3-1, 267 is not applicable.	The NRC staff is seeking clarification on whether 3M™ Interam™ E-50 series fire wrap is used for electrical raceway fire barrier systems.

Fire Water System

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	3.3	3.3-57, 3.3-178	The Discussion Column for Item Number 3.3-1, 136 in SLRA Table 3.3-1, "Summary of Aging Management Evaluations for the Auxiliary Systems," states, "The Fire Water System (B.2.3.16) AMP [Aging Management Program] is used to manage loss of material in steel fire water storage tanks and gray cast iron compressor casings exposed to	The U.S. Nuclear Regulatory Commission (NRC) staff is seeking clarification on whether the gray cast iron compressor casing in the Fire

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
			uncontrolled air and raw water." However, SLRA Table 3.3.2-6, "Fire Protection System – Summary of Aging Management Evaluation," only states that loss of material for the gray cast iron compressor casing exposed to air – indoor uncontrolled (ext) and air – indoor uncontrolled (int) will be managed by the External Surfaces Monitoring of Mechanical Components program and the Fire Water System program, respectively. Please discuss whether the gray cast iron compressor casing in the Fire Protection System will be exposed to raw water.	Protection System is exposed to raw water.
2	3.3	3.3-44, 3.3-51, 3.3-179	NUREG-2191, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," includes GALL-SLR Items VII.G.A-51 and VII.G.AP-31, Standard Review Plan (SRP) Item 3.3-1, 072 for managing loss of material due to selective leaching of gray cast iron piping, piping components exposed to raw water and soil by the Selective Leaching program. GALL-SLR Item VII.I.AP-198, SRP Item 3.3-1, 109 is for managing loss of material of steel piping, piping components exposed to soil and concrete by the Buried and Underground Piping and Tanks program. SLRA Table 3.3-1 states consistency with NUREG-2191 for Item Numbers 3.3-1, 072 and 3.3-1, 109. However, SLRA Table 3.3.2-6 does not cite either of these GALL-SLR Items for the gray cast iron fire hydrant. Please discuss whether the gray cast iron fire hydrant is exposed to concrete, raw water, or soil.	The NRC staff is seeking clarification on whether the gray cast iron fire hydrant is exposed to concrete, raw water, or soil.
3	3.3	3.3-184	GALL-SLR Item VII.C1.A-409, SRP Item 3.3-1, 126 is for managing wall thinning due to erosion of metallic piping, piping components exposed to raw water. SLRA Table 3.3.2-6 does not cite a program(s) for managing wall thinning due to erosion of gray cast iron pump casing exposed to raw water.	The NRC staff is seeking clarification on whether wall thinning due to erosion is an applicable aging effect that requires management for the gray cast iron pump casing exposed to

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
			Please discuss whether wall thinning due to erosion is an applicable aging effect that requires management for the gray cast iron pump casing exposed to raw water. In addition, the NRC staff noted that fuel oil is not an applicable environment for the gray cast iron pump casing. Please confirm that the fuel oil transfer pump that transfers fuel to the diesel fire pump fuel oil day tank is part of the Emergency Power System and not the Fire Protection System.	raw water. In addition, the staff is seeking confirmation that the fuel oil transfer pump is part of the Emergency Power System and not the Fire Protection System.
4	N/A	N/A	Action Request (AR) 01860519 identified two locations that are susceptible to the wet-dry cycle as described in NRC Information Notice 2013-06, “Corrosion in Fire Protection Piping Due to Air and Water Interaction.” The two locations are the Whse #2 Dry-Pipe System and G-05 and 1/2X-04 Deluge System. It was recommended that a new procedure be developed to routinely inspect a branch section of piping of the Whse #2 Dry-Pipe System. During routine testing of G-05 and 1/2X-04 Deluge System, it was stated that water does not flow through the system, however, a few years ago, the system was accidently actuated and introduced water into the system. Given that water may still be in the low points of the system, a one-time inspection of a low-point section of this piping was recommended. Work Request (WR) 94075251 and AR01874286 were created to perform the one-time inspection. Please discuss the status of this one-time inspection. In addition, please confirm, consistent with AMP XI.M27, “Fire Water System,” that the one-time inspection is a 100-percent visual of the internal surface of piping segments that cannot be drained or piping segments that allow water to collect.	The NRC staff is seeking an update on the one-time inspection of the Whse #2 Dry-Pipe System that was accidently actuated and may still have water in the low points.
5	Appendix A, Appendix B	A-77, B-130	There are minor inconsistencies in the table of Enhancements in SLRA Section B.2.3.16, “Fire Water System,” and the Commitments related to the Fire Water System in SLRA	The NRC staff would like to discuss additional details being added to the

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
			Table 16-3, “List of SLR Commitments and Implementation Schedule.” Specifically, using references from SLRA Table 16-3, Commitments (c), (d), (g), and (i). Table 4.4-1, “Fire Water System Inspections and Tests,” of FPLCORP00036-REPT-066, “Subsequent License Renewal Aging Management Program Basis Document – Fire Water System,” provides a description of the specific changes necessary for the Fire Water System inspections and tests to be consistent with AMP XI.M27, “Fire Water System,” Table XI.M27-1. However, the SLRA does not include a similar level summary of changes to demonstrate how existing procedures will be updated to be and how new procedures will be consistent with AMP XI.M27, Table XI.M27-1. The NRC staff would like to discuss additional details being added to the SLRA related to the description of specific changes in order to demonstrate that the procedures will be consistent with the inspections and tests described in AMP XI.M27, Table XI.M27-1. For example, changes to procedures related to fire hydrant draining.	SLRA related to the description of specific changes in order to demonstrate that the procedures will be consistent with the inspections and tests described in AMP XI.M27, Table XI.M27-1.
6	3.3	3.3-186	AMP XI.M27 states, “Fire water storage tank bottom surfaces exposed to soil or concrete are inspected in accordance with GALL-SLR Report AMP XI.M29, “Outdoor and Large Atmospheric Metallic Storage Tanks,” Table XI.M29-1. For indoor fire water storage tanks exposed to concrete, this only applies if the tank bottom to concrete interface surface is periodically exposed to moisture.” Section 3.4, “Subsequent License Renewal RAI Responses and Supplements/Amendments Addressed,” in FPLCORP00036-REPT-066 states, in part, “The PBN T-073 fire water tank does not have a history of moisture at the concrete-to-tank interface...”	The NRC staff seeks clarification on the applicable environments for the fire water tank.

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
			This information does not appear in the SLRA and SLRA Table 3.3.2-6 does not include concrete as an applicable environment for the fire water tank; only air – indoor uncontrolled (ext) and raw water (int) are cited. Please discuss whether concrete (or any other environment) is an applicable environment for the fire water tank.	
7	3.3	3.3-188	Item VII.C1.A-473, SRP Item 3.3.1-160 in NUREG-2191, Volume 1 addresses cracking due to stress corrosion cracking for copper alloy (>15% Zn or >8% Al) piping components exposed to closed-cycle cooling water, raw water, waste water to be managed by several programs including AMP XI.M38, “Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components.” SLRA Table 3.3.2-6 states that cracking for copper alloy >15% Zn valve bodies exposed to raw water will be managed by the Fire Water System program. The corresponding aging management review item (3.3.1-160) cites Standard Note E (consistent with GALL-SLR but different program credited) for the use of the Fire Water System program in lieu of the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program. AMP XI.M38 notes that periodic surface examinations are conducted for managing cracking in stainless steel and aluminum components and states, “Visual inspections for leakage or surface cracks are an acceptable alternative to conducting surface examinations to detect cracking if it has been determined that cracks will be detected prior to challenging the structural integrity or intended function of the component.” AMP XI.M27 does not provide additional guidance for managing cracking, whereas AMP XI.M38 does provide additional guidance for managing cracking. SLRA Section B.2.3.16 does not describe how the	The NRC staff is seeking clarification on how the Fire Protection System program will manage cracking for copper alloy >15% Zn valve bodies exposed internally to raw water.

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
			Fire Water System program inspections and testing performed in accordance with NFPA 25 will manage cracking of copper alloy >15% Zn valve bodies exposed internally to raw water. Describe how the Fire Water System program will manage cracking of copper alloy >15% Zn valve bodies exposed internally to raw water. Specifically discuss whether surface examinations will be performed or whether analyses will be performed to demonstrate that surface cracks can be detected by leakage prior to a crack challenging the intended function of the component, such that visual inspections would suffice. Alternatively, propose the use of a different aging management program that already includes comparable guidance.	
8	3.3	3.3-180, 3.3-188	SRP Item 3.3-1, 114 is applicable to copper alloy piping and piping components, not copper alloy >15% Zn piping and piping components. Table 3.3.2-6 cites SRP Item 3.3-1, 114 for copper alloy > 15% Zn nozzles exposed to air – indoor uncontrolled (int) and copper alloy > 15% Zn valve bodies exposed to air – indoor uncontrolled (int) and gas (int). Item VII.G.A-405a, SRP Item 3.3.1-132 in NUREG-2191, Volume 1 addresses cracking due to stress corrosion cracking (SCC) for copper alloy (>15% Zn or >8% Al) piping, piping components exposed to air, condensation. The SLRA does not state the basis for why cracking is not an applicable aging effect for copper alloy > 15% Zn nozzles and valve bodies exposed to air – indoor uncontrolled (int). While GALL-SLR does not address copper alloy >15% Zn components exposed to gas, the SLRA does not state the basis for why cracking is not an applicable aging effect for the valve bodies exposed to gas (int) (e.g.,	The NRC staff is seeking the basis for why cracking is not an applicable aging effect for copper alloy >15% ZN nozzles and valve bodies exposed to air – indoor uncontrolled (int), and copper alloy >15% ZN valve bodies exposed to gas (int).

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
			ammonia-based compounds not present in the gas, the valve bodies will not be wetted). Please state the basis for why cracking is not an applicable aging effect for copper alloy >15% ZN nozzles and valve bodies exposed to air – indoor uncontrolled (int), and copper alloy >15% ZN valve bodies exposed to gas (int).	
9	3.3	3.3-180	Section 4.4, "Detection of Aging Effects," in FPLCORP00036-REPT-066 states that the deluge systems surrounding transformers are "found outside and are therefore more vulnerable to degradation than the deluge systems located within the plant buildings. They are therefore conservatively representative of the condition of all of the deluge system nozzles." SLRA Table 3.3.2-6 does not cite air – outdoor as an applicable environment for nozzles. Please discuss whether air – outdoor should be added cited in SLRA Table 3.3.2-6 as an applicable environment for nozzles.	The NRC staff is seeking clarification on whether air – outdoor should be added cited in SLRA Table 3.3.2-6 as an applicable environment for nozzles.
10	N/A	N/A	The NRC staff reviewed several ARs, WOs, and other documents related to sprinkler head replacement/testing. Given the statement in the Program Owner Interview Form for the Fire Protection program – "Sprinkler head replacement is going poorly," the staff would like a status update.	The NRC staff is seeking a status update on sprinkler head replacement/testing. [Note: The NRC staff noted that AR1307840 and AR1307837 reference WOs 396117 and 386117, respectively, for replacing sprinkler heads and inspecting selected sprinkler piping. This appears to be a typographical error.]
11	N/A	N/A	The first sentence of the last paragraph in the Requirements Enhancements Column for Sprinkler inspections in Table 4.4-1 of FPLCORP00036-REPT-066 states, in part,	The NRC staff is seeking clarification on the sprinkler inspection and

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
			"The inspection and testing procedures listed blow will also be revised…" However, only a new procedure related to sprinkler testing is listed below. The staff's understanding is that the inspection and testing procedures that will be revised to include information regarding trending deposit amounts are those listed in the Implementing Procedure Column for Sprinkler inspections in Table 4.4-1. Please confirm the staff's understanding. If the staff's understanding is incorrect, please identify the specific procedures that will be revised.	testing procedures that will be revised to include information regarding trending deposit amounts.
12	N/A	N/A	The Required Enhancements Column for Standpipe and Hose Systems flow tests in Table 4.4-1 of FPLCORP00036-REPT-066 states, in part, "This test procedure will be enhanced…" However, the Implementing Procedure Column indicates that this is a new procedure. Please discuss whether this is a new procedure or an existing procedure that will be enhanced. If this is an existing procedure, then please identify the existing procedure.	The NRC staff is seeking clarification on the Standpipe and Hose Systems flow tests procedure.

Boric Acid Corrosion

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1			AR22262609 notes "another common issue appeared to be that many Fleet owners did not document any site-specific OE searches." For the BAC program, discuss whether this has to be addressed.	
2			FPLCORP00036-REPT-39, Rev 1, Aging Effects OpEx. AR2176346 & AR2176303 Bronze valve body was corroding in borated water and causing corrosion affecting valve lift issues. Disposition: This AR identifies loss of material due to corrosion of brass/bronze in an environment of treated water. This aging effect is identified in the Mech Tools (EPRI3002011822) and is applicable to PBN.	

			The environment appears to be treated/borated water. GALL does not have an item for this material environment combination. This seems to be similar to issue at Turkey Point where carbon steel containment spray piping had an internal environment of treated/borated water, but it was not recognized as not having a corresponding AMR item in the GALL. Although the specific valves were replaced with stainless steel, provide a discussion about whether there are other components susceptible to boric acid corrosion that are internally exposed to treated borated water.	

Recurring Internal Corrosion
[Further Evaluation Sections 3.2.2.2.7, 3.3.2.2.7, & 3.4.2.2.6]

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	3.3.2.2.7	3.3-24	SLRA states"…there have been no corrosion issues that meet the criteria for recurring internal corrosion." SRP-SLR RIC criteria: a) a 10-year search reveals three or more; or b) a 5-year search reveals two or more component not meeting plant-specific acceptance criteria or experiencing a reduction in wall thickness greater than 50% (regardless of min-wall thickness). Review of the 2018 and 2017 Service Water ISI Program Annual Reports (NPM 2019-0087 and NPM 2018-0023) include four ARs that identify service water leaks. The staff notes that other Service Water ISI Annual Reports posted on the portal include other through wall leaks. Discuss how the leaks discussed in the Service Water ISI Annual Reports are not considered as meeting the criteria in SRP-SLR 3.3.2.2.7.	
2			NP 7.7.22, Service Water and Fire Protection Inspection Program. This seems to be a plant-specific program, but it does not appear to be specifically cited within the SLRA. Provide a discussion about details for the program as they relate to recurring internal corrosion.	

3			There appeared to be a number of leaks associated with the chlorination system (CD), but it was not clear whether these leaks are due to internal corrosion or some other degradation mechanism. Provide a discussion about whether recurring internal corrosion should be considered in the CD system.	

Above Ground Tanks

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	B.2.3.17	B-136	The GALL-SLR Table XI.M29-1, “For outdoor tanks, sealant or caulking is applied at the interface between the tank external surface and concrete or earthen surface to mitigate corrosion of the tank by minimizing the amount of water and moisture penetrating the interface. Program Basis Document (PBD), FPLCORP00036-REPT-053, states, “the RMWT and RWST are not outdoors, but rather are within the Façade.” Additionally, the PBD states that caulking or sealant to the RMWT and RWSTs will not be required. AR2132309 U1 Façade roof leak. Roof has multiple leaks causing gallons of water to accumulate. In the drumming room. Condition is being tracked within the structures monitoring program. The staff is aware the façade roof has been replaced. What actions are being taken to look for possible roof leakage in the future? Because of the existing OE, what preventative actions are in place to mitigate corrosion of the tank by minimizing the amount of water and moisture penetrating the interface due to long term exposure to moisture.	Although façade roof leakage may not have immediate impact on the RMWT and the RWST, ongoing leakage should be considered as part of degradation due to long term exposure to moisture.

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
2	N/A	N/A	Portal – For the RWST and RMWT. Please provide pictures and drawings for these tanks. If they do exist on portal, please share location.	

SLRA Section 2.3.3.6 – Scoping for Fire Protection

<table>
<tr><th>#</th><th>SLRA Section</th><th>SLRA Page</th><th>Question / Issue</th></tr>
<tr><td>1</td><td>2.3.3.6</td><td></td><td>
<u>Background:</u>
For Point Beach Nuclear Plants Units 1 and 2, the staff reviewed the license renewal application (LRA), NUREG-1839, “Safety Evaluation Report Related to License Renewal of Point Beach Nuclear Plant Units 1 and 2, <u>December 2005,</u> ADAMS Accession Nos. ML053420134 and ML053420137, SLRA drawings, Updated Final Safety Analysis Report (UFSAR), Section 9.10 “Fire Protection System (FP)” and the Point Beach NFPA 805 amendment, ADAMS Accession No. ML16196A093.

<u>Issue:</u>
The following boundary drawings show the fire protection systems/components as not within the scope of license renewal (i.e., not colored in magenta):

<table>
<tr><th><u>SLRA Drawing</u></th><th><u>Systems/Components</u></th><th><u>Location</u></th></tr>
<tr><td>SLR-M-208-Sh 1</td><td>Condensate Drain</td><td>F6</td></tr>
<tr><td></td><td>1” pipe HB 22</td><td>H5</td></tr>
<tr><td></td><td>Two caps and piping</td><td>D3</td></tr>
<tr><td colspan="3">Note 7 – Print does not show factory installed operating piping on deluge valves and sprinkler systems. Refer to tech manual as required. Water for deluge valve priming connection supplied from piping upstream of deluge valve. Staff is unable to find these drawings</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td>SLR-M-208-Sh 2</td><td>Valve FP584 and screw cap</td><td>H9</td></tr>
<tr><td colspan="3">Note 7 – Print does not show factory installed operating piping on deluge valves and sprinkler systems. Refer to tech manual as required. Water for deluge valve priming connection supplied from piping upstream of deluge valve. Staff is unable to find these drawings</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td>SLR-M-208-Sh 4</td><td>Section “A-A” Service water pump area sprinkler system</td><td>G6</td></tr>
<tr><td></td><td>1.5 inch pipe from FP Valve 272 to Charcoal filter F-13</td><td>G4</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td>SLR-M-208-Sh 8</td><td>Valve 365</td><td>Staff is unable to locate</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td>SLR-M-208-Sh 12</td><td>Auxiliary drain</td><td>E8</td></tr>
<tr><td></td><td></td><td></td></tr>
</table>
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</table>

#	SLRA Section	SLRA Page	Question / Issue
			SLR-M-208-Sh 13 | Pendent sprinkler head and pipe | F5 SLR-M-208-Sh 15 | Two 1 inch drains | F9 | 1 inch cap | F9 | 1 inch drain | E6 | 1 inch drain | F6 | 1 inch drain | E5 | 1 inch drain | C7 | 1 inch drain | C6 <u>Request</u>: Verify whether the fire protection components listed above are within the scope of license renewal in accordance with 10 CFR 54.4(a) and whether they are subject to an AMR in accordance with 10 CFR 54.21(a)(1), because these components appear to be necessary to meet the requirements for 10 CFR 50.48. If they are not within the scope of license renewal and are not subject to an AMR, the staff requests that the applicant provide justification for the exclusion.
2			In License Amendment Request 271, Transition to 10 CFR 50.48(c) - NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition (ADAMS Accession No. ML13182A351) Attachment C, "NEI 04-02 Table B-3- Fire Area Transition," indicates that that a dry chemical extinguishing system protects fire area A01-E, Fire Zones 322 and 547. The staff is unable to find information on this system in the fire protection drawings, or information regarding aging management in the application. <u>Request</u>: Verify whether the dry chemical extinguishing system is within the scope of license renewal in accordance with 10 CFR 54.4(a) and whether it is subject to an AMR in accordance with 10 CFR 54.21(a)(1), because it appears to be necessary to meet the requirements for 10 CFR 50.48. If it is not within the scope of license renewal and is not subject to an AMR, the staff requests that the applicant provide justification for the exclusion.

Appendix A (A-17,67,114), Appendix B (B-79) Flow-Accelerated Corrosion (FAC)

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
1	Appendix A, Appendix B	A-17, A-67, A-114, B-79	SLRA Section 16.2.2.8, "Flow-Accelerated Corrosion," appears to state that the PBN FAC program is based on EPRI 3002000563, "Recommendations for an Effective Flow-	The U.S. Nuclear Regulatory Commission (NRC) staff is seeking

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
			Accelerated Corrosion Program (NSAC-202L-R4)," November 2013. However, Reference 61 of the Updated Final Safety Analysis Report Supplement is for NSAC-202L, Revision 3, dated August 2007. Commitments a) and d) for No. 12, Flow-Accelerated Corrosion (16.2.2.8), in SLRA Table 16-3, "List of SLR Commitments and Implementation Schedule," only references NSAC-202L; not a specific revision. The Enhancements to Element 1, "Scope of Program," and Element 4, "Detection of Aging Effects," as described in SLRA Section B.2.3.8, "Flow-Accelerated Corrosion, only references NSAC-202L; not a specific revision. Please confirm which revision of NSAC-202L the PBN FAC program incorporates and discuss whether Revision 4 of NSAC-202L should be referenced in SLRA Table 16-3 and in the Enhancements of Table of SLRA Section B.2.3.8.	confirmation on which revision of NSAC-202L the PBN FAC program incorporates. In addition, the staff is seeking whether Revision 4 of NSAC-202L should be referenced in SLRA Table 16-3 and in the Enhancements of Table of SLRA Section B.2.3.8.
2	Appendix A, Appendix B	A-67, B-79	Commitment d) for No. 12 in SLRA Table 16-3 states, "Revise or develop procedural guidance relative to erosion based on the results that includes –" This statement is unclear to the NRC staff. Commitment e) for No. 12 in SLRA Table 16-3 and the first Enhancement to Element 5, "Monitoring and Trending," as described in SLRA Section B.2.3.8 state, "<u>Wall thickness should be trended</u> to adjust the monitoring frequency and to predict the remaining service life of the component for scheduling repairs or replacements." However, Aging Management Program (AMP) XI.M17, "Flow-Accelerated Corrosion," in NUREG-2191, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report – Final Report," Volume 2 states, "For erosion mechanisms, <u>the program includes trending of wall thickness measurements</u> to adjust the monitoring frequency and to predict the remaining service life of the component for scheduling repairs or replacements." The	The NRC staff is seeking clarification regarding the statement "…based on the results…" In addition, the staff is seeking confirmation that the PBN FAC program will trend wall thickness measurements for erosion mechanisms.

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
			expectation is that the FAC program will trend wall thickness measurements for erosion mechanisms. Please discuss what is meant by "…based on the results…" In addition, please confirm that the PBN FAC program will trend wall thickness measurements for erosion mechanisms.	
3	3.4	3.4-48, 3.4-70	Section 4.1, "Scope of Program," of FPLCORP00036-REPT-047, "Subsequent License Renewal Aging Management Program Basis Document – Flow-Accelerated Corrosion," states that the PBN FAC program "predicts, detects, monitors and minimizes FAC degradation in high energy carbon and low alloy steel piping, such as associated with the Main and Auxiliary Steam, Feedwater and Condensate." Section 2.1.1, "FAC Resistant Material," of 17-0299-TR-001, "Flow Accelerated Corrosion Program – System Susceptibility Evaluation," states, in part, "…low-alloy steel with chromium content equal to or greater than 1.25% is FAC resistant." Section 2.1.1 goes on to state, "It should be noted, however, that resistance to FAC does not ensure against other erosion mechanisms such as cavitation or impingement." NUREG-2191, Volume 1 identifies wall thinning due to erosion metallic piping, piping components exposed to steam and treated water (e.g., GALL-SLR Item VII.B1.S-408, Standard Review Plan (SRP) Item 3.4-1, 060 and GALL-SLR Item VII.D1.S-408, SRP Item 3.4-1, 060). SLRA Table 3.4.2-1, "Main and Auxiliary Steam – Summary of Aging Management Evaluation," does not cite wall thinning – FAC as an applicable aging effect for any components. In addition, this table does not cite wall thinning – erosion for low-alloy steel piping, piping and piping components, or valve bodies with a treated water (int) environment. In addition, SLRA Table 3.4.2-2, "Feedwater and Condensate – Summary of Aging	The NRC staff is seeking clarification on whether all the low-alloy steel components in the Main and Auxiliary Steam systems and the Feedwater and Condensate systems have chromium contents equal to or greater than 1.25%. In addition, the staff is seeking clarification on whether wall thinning – erosion is an applicable aging affect that requires management for the low-alloy steel components in these systems.

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
			Management Evaluation," does not cite wall thinning – FAC as an applicable aging effect for any components. Please discuss whether all the low-alloy steel components in the Main and Auxiliary Steam systems and the Feedwater and Condensate systems have chromium contents equal to or greater than 1.25%. In addition, please discuss whether wall thinning – erosion is an applicable aging affect that requires management for the low-alloy steel components in these systems.	
4	Appendix A, Appendix B	A-17, B-78	SLRA Section 16.2.2.8 states that the PBN FAC program is an existing program that manages wall thinning due to erosion mechanisms. This section also states that the PBN FAC program is based on NSAC-202L-R4 and industry operating experience. SLRA Section B.2.3.8 states that the PBN FAC program is an existing program that manages wall thinning due to erosion mechanisms. The section also states that the FAC program relies on implementation of NSAC-202L-R4. Section 4.1 of FPLCORP00036-REPT-047 states, The PBN FAC AMP also inspections components for wall thinning caused by erosion mechanisms such as cavitation, flashing, and liquid drop impingement in accordance with NSAC-202L Rev. 4." Section 1.3.1.C of ER-AA-111-1000, "Flow-Accelerated Corrosion (FAC) Activities," and Section 5.0.1.C of ER-AA-111, "Flow-Accelerated Corrosion (FAC) Program," state, "In accordance with NSAC-202L Rev 4, components will be inspected for erosion mechanisms such as cavitation, flashing, and liquid drop impingement." The Section 1, "Introduction," of NSAC-202L-R4 states, "It does not cover other thinning mechanisms, such as cavitation, microbiologically-influenced corrosion (MIC), and erosive wear." In addition, Section 4.2.2, "Exclusion of Systems from Evaluation," of	The NRC staff is seeking clarification on how erosion mechanisms will be monitored in accordance with NSAC-202L-R4.

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
			NSAC-202L-R4 states, in part, "…cavitation erosion, flashing erosion, liquid drop impingement…and solid particle erosion…are not part of a FAC program and should be evaluated separately." Given that NSAC-202L-R4 does not address erosion, please discuss how erosion mechanisms will be monitored in accordance with NSAC-202L-R4.	
5	3.3	3.3-55	The Discussion Column for Item Number 3.3-1, 126 in SLRA Table 3.3-1, "Summary of Aging Management Evaluations for the Auxiliary Systems," states, "Erosion is not an applicable aging effect in treated water or treated borated water environments in the Auxiliary Systems." However, SLRA Section 3.3.2.1.7, "Heating Steam," includes wall thinning due to erosion as an aging effect requiring management. In addition, SLRA Table 3.3.2-7, "Heating Steam System – Summary of Aging Management Evaluation," cites wall thinning due to erosion for several components with a treated water (int) environment. Please discuss the basis for the statement in SLRA Table 3.3-1.	The NRC statement is seeking clarification regarding the statement in SLRA Table 3.3-1 regarding erosion not being an applicable aging effect in treated water environments in the Auxiliary Systems.
6	3.4	3.4-48	SLRA Table 3.4.2-1 does not cite wall thinning – FAC or wall thinning – erosion for the following components with a treated water (int) environment: • Carbon steel drain trap • Carbon steel flow element • Carbon steel level element • Carbon steel piping and piping components (structural integrity (attached)) • Gray cast iron pump casing • Gray cast iron strainer • Carbon steel valve body (leakage boundary (spatial)) SLRA Table 3.4.2-1 does not cite wall thinning – erosion for the following components with a treated water (int) environment: • Stainless steel drain trap	The NRC staff is seeking clarification on whether wall thinning due to FAC and/or erosion is an applicable aging effect requiring management for certain components in the main and auxiliary steam systems with a treated water (int) environment.

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
			• Stainless steel piping (leakage boundary (spatial)) • Stainless steel piping and piping components (leakage boundary (spatial)) • Copper alloy valve body (leakage boundary (spatial)) NUREG-2191, Volume 1 identifies wall thinning due to FAC for steel piping, piping components exposed to treated water. In addition, NUREG-2191, Volume 1 identifies wall thinning due to erosion for metallic piping, piping components exposed to treated water (e.g., GALL-SLR Item VIII.F.S-16, SRP Item 3.4-1, 005 and GALL-SLR Item VIII.B1.S-408, SRP Item 3.4-1, 060). Please discuss whether wall thinning due to FAC and/or erosion is an applicable aging effect requiring management for the components identified above with a treated water (int) environment.	
7	3.4	3.4-72	SLRA Table 3.4.2-2 does not cite wall thinning – FAC and/or wall thinning – erosion for the following components with a treated water (int) environment: • Carbon steel piping (Leakage boundary (spatial)) • Carbon steel piping and piping components (Pressure boundary) • Carbon steel piping and piping components (leakage boundary (spatial)) SLRA Table 3.4.2-2 does not cite wall thinning – erosion for stainless steel piping and piping components (all intended functions) with a treated water (int) environment. NUREG-2191, Volume 1 identifies wall thinning due to FAC for steel piping, piping components exposed to treated water. In addition, NUREG-2191, Volume 1 identifies wall thinning due to erosion for metallic piping, piping components exposed to treated water (e.g., GALL-SLR Item VIII.E.S-16, SRP Item 3.4-1, 005 and GALL-SLR Item VIII.D1.S-408, SRP Item 3.4-1, 060).	The NRC staff is seeking clarification on whether wall thinning due to FAC and/or erosion is an applicable aging effect requiring management for certain components in the feedwater and condensate systems with a treated water (int) environment.

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
			Please discuss whether wall thinning due to FAC and/or erosion is an applicable aging effect requiring management for the components identified above with a treated water (int) environment.	
8	3.4	3.4-4, 3.4-81	SLRA Section 3.4.2.1.3, "Auxiliary Feedwater," does not include wall thinning – FAC as an aging effect requiring management. It does include wall thinning – erosion as an aging effect requiring management for metallic components exposed to raw water. SLRA Table 3.4.2-3, "Auxiliary Feedwater – Summary of Aging Management Evaluation," cites wall thinning – FAC for carbon steel piping and piping components with a treated water (int) environment. The NRC staff notes that wall thinning – erosion is not cited for any metallic components with a steam (int) and a treated water (int) environment. The NRC staff notes that 17-0299-TR-001 states the Auxiliary Feedwater System meets exclusion criteria ET, "Excluded due to single phase fluid that is less than 200 degrees F." Please discuss whether wall thinning due to FAC and/or erosion is an applicable aging effect requiring management for metallic components in the Auxiliary Feedwater System with a steam (int) and/or a treated water (int) environment.	The NRC staff is seeking clarification on whether wall thinning due to FAC and/or erosion is an applicable aging effect requiring management for metallic components in the Auxiliary Feedwater System with a steam (int) and/or a treated water (int) environment.
9	N/A	N/A	The "scope of program" program element for AMP XI.M17 states that the program, described by NSAC-202L includes procedures and administrative controls to assure that structural integrity is maintained for piping components. Section 3.1, "Governing Document," of NSAC-202L-R4, recommends the inclusion of quality assurance requirements. IM-AA-101, "Software Quality Assurance Program," Revision 13, provides the essential elements to meet the quality assurance standards established in the Quality Assurance Topical Report. Procedure IM-AA-101 also defines four levels (A through D) of	The NRC staff is seeking clarification on the software quality assurance classification for any software products used in the PBN FAC program.

#	SLRA Section	SLRA Page	Question / Issue	Why are we asking?
			software classification based on the task for which the output is to be used. The NRC staff did not identify the quality assurance classification for the software used in the PBN FAC program. Please provide the software quality assurance classification and the bases for the classification for any software products used in the PBN FAC program.	
10	N/A	N/A	17-0299-TR-001 states that the: • Boric Acid System meets exclusion criteria EW, “Excluded because the system does not contain water or steam,” and • Reactor Coolant System meets exclusion criteria ET, “Excluded due to single phase fluid that is less than 200 degrees F.” Please discuss the basis for the exclusion criteria for the Boric Acid and Reactor Coolant Systems.	The NRC staff is seeking clarification on the basis for the exclusion criteria for the Boric Acid and Reactor Coolant Systems.
11	N/A	N/A	Section 4.10, “Operating Experience,” of FPLCORP00036-REPT-047 stated that some FAC inspections were not performed during refueling outage U1R34 (2013) because scaffolding was not in place. Please walk the NRC staff through the process for FAC inspections cannot be performed.	The NRC staff is seeking clarification on the process used when FAC inspections cannot be performed. [Note: The NRC staff noted the following two potential typographical errors in Section 4.10 of FPLCORP00036-REPT-047: (1) WO4035778 versus WO40357758, and (2) AR226209 versus AR2262609.]

GALL-SLR AMP XI.E1, Cables and Connections

1. Under “Parameters Monitored or Inspected,” NextEra stated that a sample of accessible in-scope cable and connection electrical insulation subject to an adverse localized environment are visually inspected for surface anomalies such as embrittlement,

discoloration, cracking, melting, swelling, or surface contamination." It further stated that the component sampling methodology utilizes a population that includes a representative sample of in-scope electrical cable and connection types regardless of whether or not the component was included in a previous aging management or maintenance program. The staff reviewed NP 7.7.28, "Cable Condition Monitoring Program. Section 4.5, "Considerations for Walkdown/Inspections for Accessible Non-EQ Electrical Cables and Connections," which states, in part, that a representative sample of accessible electrical cables and connections installed in adverse localized environment shall be inspected for cables and connections jacket surface anomalies, such as discoloration, swelling, cracking, or surface contamination. The corresponding program element in SLR-GALL XI.E1 states that accessible in-scope cable and connection electrical insulation subject to an adverse localized environment are visually inspected for surface anomalies. It is not clear that NextEra program element will require visual inspection of all accessible in-scope cables and connections subject to adverse localized environment. Explain how this program element is consistent with those in GALL-SLR XI.E1.

2. Under the "Parameters Monitored or Inspected," NextEra stated that to facilitate the identification of an adverse localized environment, a temperature threshold and a radiation threshold will be established for cables and connection insulation materials within the scope of this program. The corresponding program element in SLR-GALL XI.E1 states that the applicant should clearly define the most limiting temperature, radiation, and moisture environments and their basis. NextEra program element does not appear to address the basis of establishing thresholds for temperature and radiation. It also does not appear to define the most limiting moisture environment and its basis. Explain how this program element is consist with those corresponding elements in GALL-SLR XI.E1.

3. Under "Detection of Aging Effect," NextEra stated that the AMP sample consists of all accessible cables and connections in adverse localized environments. The corresponding element in GALL-SLR states that accessible electrical cables and connections are visually inspected for cable jacket and connection electrical insulation surface anomalies. The inspection of accessible cable and connection insulation material is used to evaluate the adequacy of inaccessible cable and connection electrical insulation. Explain how inaccessible cables and connection electrical insulation is evaluated.

5. Under the "Detection of Aging Effects" program element," NextEra stated that if testing is deemed necessary, a sample of 20 percent of each cable and connection type with a maximum sample size of 25 is tested. The corresponding element in GALL-SLR XI.E1 states that for a large number of cables and connections identified as potentially degraded, a sample population is tested. A sample of 20 percent of each cable and connection type with a maximum sample size of 25 is tested. It further states that the following factors are considered in the development of the cable and connection insulation test sample: environment including identified adverse localized environments (high temperature, high humidity, vibration, etc.), voltage level, circuit loading, connection type, location (high temperature, high humidity, vibration, etc.), and insulation material. Is a sample of population tested only applicable for a large number of cable and connection identified as potentially degraded? Also, describe the factors that are considered in the development of the cable and connection insulation test sample.

GALL-SLR AMPs XI.E3A, XI.E3B, and XI.E3C, Inaccessible Cables

1. Subsequent License Renewal Interim Staff Guidance (SLR-ISG)-Electrical-2021-04, Updated Aging Management Criteria for Electrical Portions of Subsequent License Renewal Guidance, states that the program applies to inaccessible or underground (e.g., installed in buried conduits, cable trenches, cable troughs, duct banks, underground vaults, or direct buried installations) cables that are within the scope of license renewal and **potentially** exposed to significant moisture. The scope of program in SLRA B.2.1.39, B.2.1.40, and B.2.1.41 does not include the term “potentially.” Explain why the term “potentially” was not included in the scope of program.

GALL-SLR AMP XI.E3A, Inaccessible medium voltage Cables

1. Under “Detection of Aging Effects,” NextEra stated that inaccessible (e.g., underground) M-V power cables within the scope of SLR exposed to significant moisture are tested to determine the age degradation of the electrical insulation. Cable testing occurs at least once every 6 years. The corresponding element in GALL-SLR states that for inaccessible medium-voltage power cable exposed to significant moisture, test frequencies are adjusted based on test results (including trending of aging degradation where applicable) and plant specific OE. Cable testing occurs at least once per every 6 years. Explain how additional tests are adjusted or determined?

GALL-SLR AMP XI.E6, Electrical Cable Connections

1. Discussion of sampling methodology and types of connections identified in the samples

2. Discussion of test methods and if any procedures have been developed

GALL-SLR AMP XI.E7, High-Voltage Insulators

1. Has an inspection frequency been decided based on operating experience?

2. Are there any nearby sources of contamination (cooling tower plume, diesel exhaust pipe)?

3. Are there any health reports covering High-Voltage Insulators?

SLRA Section 3.6.2.2.3 Transmission Conductors and Switchyard Buses and Connections

1. Are there any health reports for switchyard?

2. Are there any ARs or operating experience for switchyard transmission conductors and buses?

3. Discussion of maintenance and inspections performed in switchyard and how credit is taken for further evaluation that concludes no AMP is needed.

GALL-SLR AMP XI.E2 Sensitive Instrumentation Cables

1. General discussion of the proposed AMP and whether high-range radiation monitors are included in the scope

GALL-SLR AMP X.E1, Environmental Qualification and EQ TLAA

1. The licensee cited EPRI Report NP-1558, "A Review of Aging Theory and Technology," dated September 1980 as a reference under industry guidance. This document is generally used by industry to obtain justification/basis for activation energies for the purpose of extending the qualified life of environmentally qualified equipment. EPRI recently updated this report (July 2020) due to issues/concerns with lack of or expired technical references for certain activation energies (it's believed that between 20-30% of activation energies have been removed from the database as a result). In Section 4.4 of the SLRA the licensee stated the following:

    > "Conservatism may exist in aging evaluation parameters, such as the assumed ambient temperature of the component, unrealistically low activation energy, or in the application of a component (de-energized versus energized). The reanalysis of an aging evaluation is documented according to quality assurance program requirements, which require the verification of assumptions and conclusions."

    Clarify whether this revised document is reviewed to verify that justification/basis for activation energies remains valid for EQ components. Provide example EQ files that were updated with revised activation energy values.

2. In several areas of the SLRA, the licensee discussed operational experience related to EQ; however, they did not include the results of the recent NRC focused engineering inspection on EQ (IP 71111.21(N)).

    Can results of this inspection and how they were considered as part of their operating experience review for subsequent license renewal application be provided?

3. General discussions regarding implementation of EQ program (e.g., area temperature monitoring methods and practices, activation energy selection sources and criteria, selection of most limiting qualified components).

Fire Barrier Intended Functions

SLRA Table 3.5.2-1, “Containment Building Structure and Internal Structural Components – Summary of Aging Management Evaluation,” cites a fire barrier intended function for the following items:

- Steel, copper alloy, and elastomer air locks, equipment hatches and accessories
- Reinforced concrete walls, buttresses, dome, and ring girder (inaccessible); and internal columns, beams, slabs, and walls (accessible and inaccessible)
- Stainless steel fuel transfer tube (including penetration sleeves expansion joints, and blind flange)
- Stainless steel liners (refueling cavity) and covers (sand box, Unit 1 sump A strainer)
- Dissimilar metal welds and stainless steel penetration assemblies (electrical)
- Stainless steel radiant energy shields

SLRA Table 2.4-1, “Containment Structure and Internal Structural Components Subject to Aging Management Review,” cites a fire barrier intended function for “liner plate (containment),” however, this component type does not appear in Table 3.5.2-1.

SLRA Table 3.5.2-2, “Circulating Water Pumphouse Structure – Summary of Aging Management Evaluation,” cites a fire barrier intended function for reinforced concrete external walls and roof (accessible). In addition, SLRA Table 3.5.2-2 cites a fire barrier intended function for reinforced concrete internal columns, floors, and walls in a water – flowing environment.

SLRA Table 3.5.2-6, “Fuel Oil Pumphouse Structure,” cites a fire barrier intended function for concrete block masonry (block) walls.

SLRA Table 3.5.2-8, “Primary Auxiliary Building Structure – Summary of Aging Management Evaluation,” cites a fire barrier intended function for elastomer penetration seals.

SLRA Table 3.5.2-11, “Yard Structures – Summary of Aging Management Evaluation,” cites a fire barrier intended function for the following items:

- Earth berm
- Concrete block manholes
- Steel miscellaneous structural components

However, none of these items cite the Fire Protection program for managing applicable aging effects. Therefore, please discuss whether these items have a fire barrier intended function. If these components do have a fire barrier intended function, please discuss the basis for not citing the Fire Protection program for managing applicable aging effects.

In addition, for the following, intended functions other than fire barrier are cited but only the Fire Protection program is included as the applicable aging management program:

- SLRA Table 3.5.2-3, Fire rated doors include an intended function of Flood Barrier
- SLRA Table 3.5.2-4, Fire rated doors include an intended function of Structural Support
- SLRA Table 3.5.2-8, Fire rated doors include an intended function of Flood Barrier

Please discuss how the Fire Protection program will adequately manage alternate intended functions of “flood barrier” and “structural support.”

Point Beach SLRA – Post Breakout Session Discussion Summary and Follow-up Action Summary – Staff Technical Review Package Assignment (PWR Vessel Internals):

Applicable SLRA Sections or Tables: SLRA Section 3.1.2.2.9; SLRA AMR Items for PWR RVI components in SLRA Table 3.1.1 and 3.1.2-2; SLRA AMP Section B.2.3.7, “Reactor Vessel Internals” Program; and the gap analysis for the AMP in SLRA Appendix C (including Appendix C tables)

General Summary item and Follow-up Action: The applicant and the staff agreed to schedule a follow-up (2nd) breakout session for TRP 16. Steve Franzone (NextEra Energy) explained that, in preparation of the 2nd breakout session, Enercon staff members involved with development of the SLRA are not cleared to have written or oral access to any proprietary Westinghouse report material that may be discussed during the subsequent breakout for TRP 16.

Follow-up Action for 2nd Breakout Session on TRP 16: Billy Rogers (as the lead NRC PM for the SLRA) to schedule a follow-up breakout session for TRP-16. The 2nd breakout session may need to be broken into non-proprietary (all participants) and proprietary segments (including only those participants that are cleared for Westinghouse proprietary information) if proprietary Westinghouse material will be discussed during the breakout session.

General Summary item and Follow-up Action: During the breakout, the staff informed the applicant that the updated AMR and AMP criteria in SLR-ISG-2021-01-PWRVI (hence SLR-ISG) may be of benefit to the AMR and AMP consistency bases for PWR reactor vessel internal (RVI) components in the SLRA. The staff acknowledged that it does not have the authority to force the applicant to use the SLR-SLR as the basis for the RVI AMP, AMR, and gap analysis bases in the SLRA. Applicant stated it will look at the ISG for its impact on the AMR, AMP and AMP gap analysis bases for the RVI components in the SLRA.

Follow-up Action for 2nd Breakout Session on TRP 16: Staff and applicant to discuss whether any ISG induced changes to the SLRA’s AMR, AMP, or AMP gap analysis criteria for the PWR RVI components will be addressed through the SLRA RAI process or the applicant’s implementation of the 10 CFR 54. SLRA supplement process.

Summary for Breakout Session Topic 1 Discussions – AMR Items:

- *Plant-specific AMR items for clevis insert assembly wear surfaces and upper core plate alignment pins in SLRA Table 3.1.2-2 (pages 3.1-75 and 3.1-76)*. Staff informed the applicant that the SLR-ISG would make plant-specific items for the clevis insert wear surfaces and upper core plate alignment pins consistent with the GALL-SLR line items for these components that were updated in the SLR-ISG (this is in reference to the updated versions of GALL-SLR Items IV.B2.RP-285, IV.B2.RP-399, and IV.B2.RP-301 in Appendix B.1 of the SLR-ISG). Applicant stated it will look at SLR-ISG's impact on the plant-specific line items (Note J items) for these components in SLRA Table 3.1.2-2.

(*Summary Note/Clarification for the breakout topic on the plant-specific AMR line items*: Neither the GALL-SLR report nor the SLR-ISG include any GALL-SLR-based AMR line items for internal radial support keys, which also were the subject of plant-specific line items [Note F or Note J types] as given on page 3.1-84 of SLRA Table 3.1.2-2).

Potential Follow-up Action by Applicant: Applicant may convert the plant-specific line items for the clevis insert wear surfaces and upper core plate alignment pins to "consistent-with GALL" type line items based on the updated versions of the GALL-SLR Items for these components in the SLR-ISG. If so, the staff requests that the applicant confirm this during the subsequent (2nd) breakout session for TRP 16.

AMR Items for "Expansion" category components linked to "Primary" core barrel upper flange weld (UFW) inspections on SLRA pages 3.1-81 and 3.1-82. The staff informed the applicant that, in the MRP-227, Rev. 1-A report, the EPRI MRP redefined which Westinghouse internal components would serve as the "Expansion" category components for the "Primary" core barrel UFW inspections. The staff clarified that the EPRI removed the core barrel outlet nozzles (CBONs) as cited "Expansion" components for the UFWs and replaced them the core barrel upper flange welds (UGWs), upper axial welds (UAWs), lower flange welds (LFWs), and core support forgings or castings as the linked "Expansion category components for the UFWs. The staff informed the applicant that the 3.1.1-053b and 3.1.1-059b linked line items for the CBONs on page 3.1-81 implies that the CBONs are still "Expansion" category components for the program, and that the 3.1.1-053a linked line item on cracking of the core barrel UGWs on SLRA page 3.1-82 still implies that the UGWs welds are "Primary" category components for the program. The applicant stated that it will look at the impact of the staff's updated AMR items for the CBONs and for the core barrel "Expansion" category weld components linked to the "Primary" core barrel UFWs, as provided in Appendix B.1 of the SLR-ISG.

(*Summary Note/Clarification*: For reference, the staff deleted Items IV.B2.RP-278 and IV.B2.RP-278a for the CBONs in Appendix B.1 of the SLR-ISG, and modified AMR item IV.B2.RP-280 to cover cracking for all of the updated core barrel "Expansion" category weld types [UGWs, UAWs, and LFWs] linked to the "Primary" category core barrel UFWs in MRP-227, Rev. 1-A. However, the staff's AMR line item on loss of fracture toughness in the core barrel UGWs and in the core barrel lower girth welds [LGWs] remained unchanged for the staff's SLR-ISG review, and therefore continues to be given in AMR Item IV.B2.RP-388 of Table IV.B2 in the GALL-SLR report)

Potential Follow-up Action by Applicant: Applicant may convert the AMR line items for the CBONs and UGWs to be consistent with the updated AMR line items for these components in Appendix B.1 of the SLR-ISG. If so, the staff requests that the applicant confirm this during the subsequent (2nd) breakout session for TRP 16.

- <u>AMR Item on non-cracking effects in the bottom mounted instrument (BMI) column bodies on SLRA page 3.1-77</u>. The staff informed the applicant that, in Item W2.2 of Table 4-6 in the MRP-227, Rev. 1-A report, the EPRI MRP identified irradiation embrittlement as an applicable mechanism for the BMI column bodies, which is reflected in both the updated version of GALL-SLR Item IV.B2.RP-292 in Appendix B.1 of the SLR-ISG and in the previous version of the "RP-292" line item in Table IV.B2 of the GALL-SLR Volume 1 report . The staff informed the applicant that its 3.1.1-059b referenced line item on non-cracking effects in the BMI column bodies (SLRA page 3.1-77) does not include loss of fracture toughness (due to neutron irradiation embrittlement) as an additional non-cracking effect for the column bodies. The applicant stated that it will look at the impact of the updated version of AMR Item IV.B2.RP-292 in the SLR-ISG on the 3.1.1-059b-linked Table 2 line item that was provided for the BMI column bodies on page 3.1-77 of the SRLA.

  Potential Follow-up Action by Applicant: The applicant may convert the AMR items for the BMI column bodies on SLRA page 3.1-77 to be based on the updated AMR items for these components in the SLR-ISG (i.e., Items IV.B2.RP-292 and IV.B2.RP-293 in Appendix B.1 of the SLR-ISG). If so, the staff requests that the applicant confirm this during the subsequent (2nd) breakout session for TRP 16.

  <u>*AMR Items for control rod guide tube assembly lower flanges welds (CRGT LFWs) on SLRA pages 3.1-78 and 3.1-79*</u>. The staff informed the applicant that its 3.1.1-053 and 3.1.1-059a referenced line items for the CRGT LFWs (SLRA pages 3.1-78 and 3.1-79) imply that all of the CRGT LFWs are "Primary" category components for the program. The staff also informed the applicant that, in MRP-227, Rev. 1-A, the EPRI MPR differentiated that only the CRGT LFWs in the peripheral CRGT assemblies were "Primary" category components, and that the CRGT LFWs in the remaining CRGT assemblies (i.e., in the accessible non-peripheral assemblies) were "Expansion" category components for the Program. The staff informed the applicant that in the SLR-ISG, the staff updated its items for the CRGT LFWs (i.e., modified Items IV.B2.RP-297 and IV.B2.RP-298 for the peripheral types and developed new Items IV.B2.RP-297a and IV.B2.RP-298a items for the non-peripheral types in Appendix B.1 of the SLR-ISG) in order to be consistent with the updated criteria for the components in Line Items W2 and W2.2 of the MRP-227, Rev. 1-A report. The applicant stated that it will look at the impact that the updated versions of AMR line items for CRGT LFWs in the SLR-ISG may have on the AMR line items for the CRGT LFWs in pages 3.1-78 and 3.1-79 of the SLRA.

Potential Follow-up Action by Applicant: Applicant may convert the AMR items the CRGT LFWs to be based on the updated AMR items for these components in the SLR-ISG. If so, the staff requests that the applicant confirm this during the subsequent (2nd) breakout session for TRP 16.

- *Table 1 Item 3.1.1-028 in SLRA Table 3.1.1 (Not Used Item)*. The staff informed the applicant that its "not applicable" or "not used" basis in the line item as acceptable, but clarified that, in the updated version of SRP-SLR Table 3.1-1, Item 028 in Appendix A of the SLR-ISG, the staff allowed the item to be used even if the CRGT support pins (split pins) were replaced and made from Type 316 or 316L austenitic stainless steel and were being placed in the "No Additional Measures" category for the program. The informed the applicant that it does not need to take an action on this, but clarified that the applicant may use the updated AMR basis for Item 028 in Appendix A of the SLR-ISG (at the applicant's discretion).

- *Table 1 Item 3.1.1-087 in SLRA Table 3.1.1 (Not Used Item)*. During the breakout session, the staff explain the reason that the staff included SRP-SLR Table 3.1-1, Item 087 (on loss of material due to pitting or crevice corrosion in PWR RVI components) in Appendix A of the GALL-SLR report. The applicant stated that it will change the SLRA to adopt the 3.1.1-087 item and GALL-SLR AMR Item IV.B2.RP-382, which cite use of GALL-SLR AMP XI.M2, Water Chemistry Program, to manage loss of material due to pitting and crevice corrosion in the RVI components.

Follow-up Action by Applicant for the 2nd Breakout Session: Based on the applicant's statement during the breakout discussions for Item 087, the staff assumes that the applicant will be amending the SLRA to adopt Item 3.1.1-087 for the SLRA. Applicant to confirm whether this is a valid assumption by the staff this during the 2nd breakout session for TRP 16.

- *AMR Note E item on SLRA page 3.1-78 for managing changes in dimension and loss of fracture toughness in the BMI flux thimble tubes*. During the breakout discussion for this Note E item, the staff clarified that it did not include any AMR line items on changes in dimension due for void swelling or distortion or loss of fracture toughness due to neutron irradiation embrittlement of flux thimble tubes in either the GALL-SLR report or in the SLR-ISG. For the follow-up breakout session, the staff requested a further clarification on how the eddy current examinations credited under the PWR vessel internals program would be capable of revealing evidence of changes in dimension in the BMI flux thimble tubes. The applicant clarified that the citing of the Reactor Vessel Internals Program as the applicable program for the referenced Note E item was based on the EPRI MRP's FMECA analysis for the flux thimble tubes, which placed treatment of fracture toughness and changes in dimension effects for the thimble tubes into the "No Additional Measures" category for the program. The applicant stated that it may use the updated AMR line items for the BMI flux thimble tubes in the SLR-ISG as the updated AMR item bases for the BMI flux thimble tubes in the SLRA. The staff acknowledged that this could potentially result in the deletion of the referenced Note E item for the thimble tubes from SLRA Table 3.1.2-2.

Follow-up Action by Applicant for 2nd Breakout Session: Based on the applicant's statement during the breakout discussions for Item 087, the staff notes that the applicant may be amending the SLRA to delete the Note E item on changes of dimension and loss of fracture toughness od the BMI flux thimble tubes from the scope of SLRA Table 3.1.2-2. Applicant to confirm during the 2nd Breakout Session for TRP 16 whether or not it will be deleting the Note E item for the flux thimble tubes that is given in Table 3.1.2-2 (on page 3.1-78) of the SLRA.

<u>Summary for Breakout Topic 2 Discussion – MRP Inspection Category for the Unit 1 and 2 hold-down springs</u>: The applicant verified that (based on the gap analysis information) the hold-down springs in both unit are made from Type 403 martensitic stainless steel material. Staff acknowledged that provides a sufficient basis for placing the hold-down in the "No Additional Measures" category because the inspection bases in Item W8 of Table 4-3 in the MRP-227, Rev. 1-A report are only applicable if the hold-down springs are made from Type 304 austenitic stainless steel materials. The applicant does not need to take an action on this discussion matter and the matter is closed for the audit.

<u>Summary for Breakout Topic 3 Discussions – Application of WCAP-17096 as a supplement methodology for the Reactor Vessel Internals</u>: The applicant explained that it will be applying the latest staff-approved version of WCAP-17096, and apply any condition items (CIs) or limitations issued in the staff SE for the report. This is currently use of Westinghouse Class 3 Non-Proprietary Report No. WCAP-17096-NP, Rev. 2-A. The staff discussed how the current (and pending) parallel review of WCAP-17096, Rev. 3 may be sufficient to resolve past conditions items issued in the WCAP-17096-NP, Rev. 2-A methodology. The staff also stated, while some of the past CIs for WCAP-17096-NP, Rev. 2-A may have appeared to impose a specific type of analysis for a given type of RVI component (e.g., FMECA analysis, functionality analysis, component susceptibility analysis, fracture toughness analysis, flaw growth analysis, etc.), but clarified that topical report CIs do not normally constitute NRC requirements. As part of the breakout discussions, the staff informed the applicant that it cannot require the applicant to perform of a specific type of analysis for a given RVI component in response to a CI item unless the licensee is required to perform the specific type of analysis in accordance with an applicable NRC regulatory or licensing requirement. The staff clarified that, if a supplemental evaluation is determined to be necessary by the applicant for operability or service of a given RVI component, the type of analysis that is performed for the component is up totally up to licensee, unless a specific type of analysis is mandated as being necessary by an applicable NRC licensing or regulatory requirement .

Potential Follow-up Action by the Staff During the Second Breakout Session: The staff may decide to make the clarifications on CIs again during the subsequent breakout session for TRP 16.

Summary for Breakout Topic 4 Discussions – Neutron Fluence Methodology and Neutron Fluence Projections for the RVI Components: During the breakout discussion for this topic, the applicant confirmed that neutron fluence methodology and neutron fluence assessments for the RVI components are given in Westinghouse Class 2 Proprietary Report No. WCAP-18124-P, Rev. 0. The applicant stated that, while the applicant does not have specific 80-year fluence projections for the RVI components, the WCAP provides the basis why the 80-year neutron fluences for the specified RVI components are expected to be with the neutron fluence bands established for these components by the EPRI MRP.
The applicant informed the staff that it cannot post the WCAP-18124-P, Rev. 0 report to the audit portal due to the proprietary nature of the report. The applicant stated that, instead, it will try to have Westinghouse Electric Company (WEC) to post the proprietary report on Westinghouse portal and to have Westinghouse issue an applicable portal account, ID and password for the NRC's lead reviewer on TRP 16.

Follow-up Action by the Staff Prior to the Second Breakout Session: The staff is to review Attachment 21 in the last of enclosure of SLRA Enclosure 4 as part of his review prior to the 2nd Breakout Session for the SLRA.

Completed Action by the Applicant Prior to the 2nd Breakout Session for TRP 16: The applicant has interacted with Westinghouse so that the proprietary report can be posted on the Westinghouse Electronic Portal and have Westinghouse establish an account for the NRC's lead for TRP 16. Mr. Medoff of the staff has been given access to and an account in the Westinghouse document portal.

Summary for Breakout Discussion Topic 5 – RVI Operating Experience (OpE) Discussions:

- Generic and Plant-Specific OpE with Baffle-to-Former Bolts (BFBs). The applicant confirmed that the BFBs are treated as potentially susceptible components that are subject to UT inspections for cracking, preload, and loss of material effects. The applicant confirmed that the BFBs in Unit 1 were last inspected in 2013 and that the BFBs in Unit 2 were last inspected in 2014, with the inspections being implemented in accordance with the guidelines for BFBs at the time of the inspections (i.e., MRP-227-A). The applicant confirmed that any BFBs in which the initial inspections did not yield UT readable results were re-inspected using UT methods several years later. The applicant explained that the past UT inspections of the BFBs in Unit 1 did not result in any observable UT indications that might indicate the presence of defects in the bolts. The applicant stated that the UT inspections of the Unit 2 BFBs revealed observable indications of defects in only 12 – 15 of the bolts and that the number of bolts with detected defects were not of a sufficient number or percentage to initiate the need for "Expansion" inspections of the lower support column bolts. The applicant also stated that it evaluated the BFB UT results against the interim guidance in NSAL-16-1, Rev. 1 and stated that the results did not indicate any evidence of degraded bolt clustering, as defined in the NSAL document. The applicant stated that the past UT inspections and inspection results continue support performance of BFB re-inspections on a 10-year augmented ISI frequency per the “Primary” inspection and evaluation (I&E) criteria for the bolts in MRP-227, Rev. 1-A. The staff stated that, based on these explanations, it did not have further questions on the past inspections that had been performed on the BFBs or the applicant’s basis for setting the “Primary” category reinspection basis on a 10-Year augmented ISI interval.

Potential Follow-up Actions: None for the past UT inspections and OpE Assessment associated with the BFBs.

- Generic and Plant-Specific OpE with CRGT guide plates (guide cards). The applicant confirmed that the CRGT guide cards are treated as potentially susceptible “Primary” components for loss of material due to wear, and to a lesser extent for potential cracking that may occur in the components. The applicant clarified that it is applying the augmented inspection methods in Westinghouse Proprietary Report WCAP-17451-P, Rev. 1 as the current basis for performing the CRGT guide card inspections, including potential expansions to those components that are designated as the “Expansion” components for the inspections (Note: The identification of the “Expansion” components is identified as trade secret or proprietary information in the WCAP, and the staff is with-holding the specific formation under the with-holding requirements in 10 CFR 2.390). The applicant explained that the EPRI MRP’s basis for evaluating potential wear that may be occurring in the guide cards is based on an aggregate basis. The applicant also identified that the limiting guide cards for wear were the G11 CRGT guide card in Unit 1 and the K07 CRGT guide card in Unit 2. The applicant clarified that the past VT-3 inspections of CRGT guide cards in Units 1 and 2 did not reveal sufficient evidence of wear or cracking in aggregate levels that might induce the applicant to take further corrective actions or reduce the augmented inspection frequency for the components below a 10-Year augmented inspection frequency for the components.

  Potential *Follow-up Actions Prior to or During the 2nd Breakout Session for TRP 16*: Staff noted that the OpE assessment for the CRGT guide cards in SLRA AMP B.2.3.7 recommended reinspection or repair of the limiting CRGT guide cards during the 2020 refueling outage for the units. Therefore, for the 2nd Breakout Meeting for TRP 16, staff requests clarification from the applicant’s technical lead expert on whether CRGT guide card G11 in Unit 1 and CRGT guide card K07 in Unit 2 were re-inspected in 2020. For the staff’s confirmation, staff to re-review the Work Orders or inspection summaries on the portal for the past CRGT guide card inspections prior to commencement of the 2nd breakout session for TRP 16.

- Generic and Plant-Specific OpE with CRGT support pins (split pins). The applicant confirmed that the CRGT split pins in both units are made from Type 316 austenitic stainless steel material. Based on this clarification, the applicant stated the CRGT spilt pin may be placed in the “No Additional Measures” category for the program. Based on this clarification, the staff acknowledged the basis for placing CRGT spilt pins in the "No Additional Measures" category for the program.

Follow-up Actions Prior to 2nd Breakout Session for TRP 16: None. No further actions necessary on the EPRI MRP inspection category that is being applied to the CRGT split pins in Units 1 and 2 or the basis for including them in the "No Additional Measures" category.

- Generic and Plant-Specific OpE with the RVI clevis insert assembly components (including dowels, bolts and surfaces). During the breakout session for TRP 16, the applicant stated that it has been performing VT-3 inspections of the clevis inserts and their components (e.g., dowels and bolts) in accordance with the ASME Section XI requirements for the components. The applicant stated that the past VT-3 inspections did not reveal any evidence of cracking or loss of material in the components, and that the past inspection results continue to support re-inspections of the clevis insert assemblies at Unit 1 and Unit 2 on a 10-Year augmented ISI frequency. The applicant also confirmed that the past inspections of the clevis insert assemblies at Unit 1 and 2 did not reveal any evidence of distortion occurring in the clevis insert assemblies.

  (*Summary Note/Clarification*: The staff acknowledges that, based on the applicant's application of the generic OpE, the applicant's gap analysis results in SLRA Gap Analysis Table C.A1 has elevated the inspection category for the clevis insert assembly components from "Existing Program" components to "Primary" category components).

  Follow-up Actions Prior to or During the 2nd Breakout Session for TRP 16: None. No further actions necessary on the EPRI MRP inspection category or re-inspection frequency that the applicant will apply to the clevis insert assembly components in Units 1 and 2.

- Generic and Plant-Specific OpE with Fuel Alignment Pins in the Upper and Lower Internal Assemblies and in the Radial Support Keys. Applicant confirmed that past inspections of the fuel alignments and radial support keys did not identify any evidence of wear or cracking in the components. Applicant confirmed that the re-inspections of these components going forward will be performed on a 10-Year augmented ISI frequency based on the lack of operating experience for the components at Point Beach.

  (*Summary Note/Clarification*: The staff acknowledges that, based on the applicant's application of the generic OpE, the applicant's gap analysis results in SLRA Gap Analysis Table C.A1 has elevated the inspection category for the radial support keys to "Primary" category components; the fuel pins remain as "Existing Program" components per SRLA Gap Analysis Table C.A3).

  Follow-up Actions Prior to or During the 2nd Breakout Session for TRP 16: None. No further actions necessary on the EPRI MRP inspection categories for these components or the re-inspection frequency that the applicant is applying to the radial support keys and the upper and lower assembly fuel alignment pins in Units 1 and 2.

- Generic and Plant-Specific OpE with the Reactor Internal Thermal Shield Flexures. Applicant confirmed that re-inspections of the thermal shield flexures (SLRA Gap Analysis Table C.A1 "Primary" components) are performed on a 10-Year ISI re-frequency and that the past VT-3 inspections of the thermal shields flexures did not identify any evidence of cracking or wear in the thermal shield flexures in Units 1 and 2. Applicant clarified it is aware of the generic experience (i.e., significant amount of flaw indications detected in the flexures at another U.S. Westinghouse PWR) and that it will be re-inspecting the thermal shield flexures in Unit 1 in 2022 and the thermal shield flexures in Unit 2 in 2023.

  Follow-up Actions Prior to 2nd Breakout Session for TRP 16: None. No further actions necessary on the EPRI MRP inspection categories for thermal shield flexures or the re-inspection frequency that the applicant is applying to the thermal shield flexures in Units 1 and 2.

Summary for Breakout Topic 6 Discussions – Inspection Report Summaries: The applicant statement that the past inspections of the upper internal assembly components (as opposed to the lower internals assembly components) were performed by a vendor different from Westinghouse. The applicant explained however that the Westinghouse WestDyne report placed on the portal has all the past inspection summarizes.

Follow-up Actions Prior to or During the 2nd Breakout Session for TRP 16: None. No further actions on this matter are necessary.