BREAKOUT QUESTIONS
Aging Management Audit

Comanche Peak Nuclear Power Plant,
Units 1 and 2
License Renewal Application

December 12, 2022 – May 18, 2023

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

Fire Protection Scoping and Screening:
Section 2.3.3.7 Fire Protection System
Section 2.4.15 Fire Barrier Commodity Group
LAR Table 2.3.3-6 Equipment and Floor Drainage System Component Intended Function
LAR Table 2.3.3-7 Fire Protection System Components Subject to Aging Management Review
LAR Table 2.4-15 Fire Barrier Commodity Group Subject to Aging Management Review

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	2.3.3.7	2.3-80	The descripton of the fire protection system does not describe seismic support for fire water storage tanks, standpipes, and piping that is within the scope of license renewal in accordance with 10 CFR 54.4(a) and subject to an aging management review (AMR) in accordance with 10 CFR 54.21(a)(1).	Clarify whether there are fire protection system components that are seismically supported and 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). If the components are not within the scope of license renewal and are not subject to an AMR, provide justification for the exclusion.
2	2.3.3.7	2.3-82	Oil collection dikes and curbs are credited fire protection features. The LRA, however, only specifies the Reactor Coolant System (RCS) Oil Spillage Protection System.	Clarify whether there are other oil collection dikes and curbs such as for the diesel oil and day tanks, pumps, etc. and whether they 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). If they are not within the scope of license renewal and are not subject to an AMR, provide justification for the exclusion.

3	2.4.15	2.4-45	The fire barrier commodity group describes cable raceway fire barriers as well as insulation and wrap. It is unclear to the NRC staff what the difference is between the two.	Explain the differences between the two systems and clarify whether any other electrical raceway fire barrier systems (ERFBS) credited for cable tray, conduits, bus ducts, electrical panels or junction boxes, etc., 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). If they are not within the scope of license renewal and are not subject to an AMR, provide justification for the exclusion.

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section /XI.E2 AMP/OpE/X.E1 EQ:
TRP: 50

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	XI.E2 Basis document			In the LRA, the applicant noted that XI.E2 is consistent with all the Elements described in NUREG-1801 XI.E2, “Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits.” However, LUM00020-REPT-075 - “Insulation Material for Electrical Cables and Connections not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program Basis Document,” states the following; “For any instances where the CPNPP Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits AMP does not meet the guidance of the NUREG-1801 XI.E2 Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits AMP, additional actions which are necessary to align the CPNPP Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits AMP with the guidelines are identified, or an exception with justification is provided.” Given this above, did the applicant identify any instances where the CPNPP Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in

				Instrumentation Circuits AMP does not meet the guidance of the NUREG-1801 XI.E2 Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits AMP? If so, what actions were taken to align the CPNPP Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits AMP with the LR guidelines including any justified exceptions that may have been taken.
2	OpE	OpE		Can you explain how the condition of cables (in general) will be monitored and if there are any trigger points (e.g., cables exposed to environmental conditions that may accelerate aging) for increasing the periodicity of monitoring/testing during the extended period of operation.
3	OpE	OpE		Explain the water trend fluctuation OpE (WR#4160865,WR#4329142,WR#4540006,WR#4593784) for the Unit#2 Train B electrical manhole inspection-service water. Discuss any impact this has had on your manhole inspection process in support of your request to renew the license for the plant.
4	OpE	OpE		Explain how the operational experience in CR2015-004409 which identified water coming out of a conduit and that the conduit was rusted, was resolved and discuss any impact this has had on your proposed aging management programs for cables (specifically XI.E3).
5	X.E1 EQ			In LUM00020-REPT-081, “Comanche Peak Nuclear Power Plant Units 1 and 2 License Renewal Time-Limited Aging Analysis-Environmental Qualification of Electrical Equipment,” Rev. 1, the licensee stated “Any changes to material activation energy values as part of a reanalysis must be justified.” Per RG 1.89, Rev. 1, these changes

				would also have to be defined and documented. Confirm that this is the case for any changes to material activation energy for establishing environmental qualification of electrical equipment at Comanche Peak.
6	X.E1 EQ			With regard to CR-2017-012269, have the EQ files been updated and considered up to date?
7	X.E1 EQ			Has the licensee relied on Electric Power Research Institute (EPRI) Report NP-1558, “A Review of Aging Theory and Technology,” dated September 1980 to establish or maintain qualification of electrical equipment at Comanche Peak? If so, EPRI recently updated this report (July 2020) due to issues/concerns with lack of or expired technical references for certain activation energies. The NRC staff’s understanding is that this revision resulted in up to 30% of activation energies being removed from the database. For environmentally qualified (EQ) components that the licensee used/relied upon EPRI Report NP-1558 as the justification/basis for activation energies for extending the qualified life of EQ equipment, has the licensee reviewed this revised document to verify that their justification/basis for activation energies remains valid for EQ components for the requested period of operation?
8	Sections 2.5 and 3.6.2.3 (Fuse Holders) Table 2.5-1	Pages 2.5-2 and 3.6-10 (Fuse Holders) Page 2.5-5	ISG-5 NUREG-1800, Table 2.1-5	Section 3.6.2.3, “AMR Results Not Consistent With or Not Addressed in the GALL Report,” Consistent with NUREG-1801, XI.E5, the screening of CPNPP fuse holders (metallic clamps) applies to those that are not part of a larger (active) assembly. <u>Fuse holders inside the enclosure of an active component, such as switchgear, power supplies, power inverters, battery chargers, and circuit boards are considered piece parts of</u>

				the larger assembly. Since piece parts and subcomponents in such an enclosure are routinely inspected and regularly maintained as part of the plant's normal maintenance and surveillance activities, they are not subject to AMR. An evaluation of fuse holders at CPNPP was performed to discover the population of fuse holders that were not located in active devices, such as control panels, switchgear, MCCs and termination cabinets…. Panels, racks, and termination cabinets were also considered to be another type of active component consistent with the guidance provided in ISG-5 and were eliminated from the process. ISG-5 states, in part: The staff concludes that fuse holders are passive, long-lived electrical components within the scope of license renewal and subject to an AMR. However, fuse holders inside the enclosure of an active component, such as switchgear, power supplies, power inverters, battery chargers, and circuit boards, are considered to be piece parts of the larger assembly. For license renewal purposes, fuse holders/blocks are classified as a specialized type of terminal block because of the similarity in design and construction. Terminal blocks are passive components subject to an AMR for license renewal. However, like fuses, terminal blocks located inside the enclosure of an active component are considered to be piece parts of the

				<u>larger assembly</u> and, thus, are outside the scope of license renewal. The staff notes that, based on the guidance in ISG-5, <u>panels, racks, and termination cabinets</u> would be considered active components if they included active electrical/electronic components. This is consistent with the guidance in NUREG-1800, Table 2.1-5, “Typical Structures, Components, and Commodity Groups, and 10 CFR 54.21(a)(1)(i) Determinations for Integrated Plant Assessment” which provides switchgear, load centers, motor control centers (MCCs), distribution panel, and internal component assemblies as active electrical and I&C component commodity group and not meeting the criteria of 10 CFR 54.21(a)(1)(i). In addition, the staff notes that the applicant did not list <u>racks and termination cabinets</u> as electrical and I&C component commodity group in the LRA, Table 2.5-1, “Typical Structures, Components, and Commodity Groups, and 10 CFR 54.21(a)(1)(i) Determinations for Integrated Plant Assessment.” <u>Question</u>: Please clarify if the above-mentioned panels, racks, and termination cabinets include active electrical/electronic components. Also, explain why the racks and termination cabinets are not listed in the electrical and I&C component commodity groups of Table 2.5-1 of the LRA.
9		Pages 2.5-2 and 3.6-10 (Fuse Holders)	FSAR, page 8.3-40	In the LRA section 3.6.2.3 and section 2.5.1.3, “Elimination of Electrical and I&C Commodity Groups Not Applicable to CPNPP,” the applicant stated that the fuse holders (metallic clamps) at CPNPP are considered piece parts of a larger (active) assembly, and therefore they do are not subject to an aging management review for LR.

				The staff is unable to confirm the applicant's above-mentioned statement (i.e., CPNPP fuse holders, are pieces part of larger active assembly) for some fuses based on the FSAR descriptions of the fuses. Specifically, on page 8.3-40 of the FSAR (page 3152 of the pdf), the applicant states: The Class 1E Electronics Boxes (X-LY-4849A-1, X-LY-4849A-2, X-LY-4849B-1 and X-LY-4849B-2) are connected to the Non-Class 1E annunciator in the Spent Fuel Pool Panel CPX-EIPRLV-06. At the Electronics Box, the Class 1E Cables from the level switches and 120V AC power supply cable enter the box from the top where as the Non-Class 1E cables from the annunciator enter the box from the bottom. The cables from the Class 1E Electronics Boxes to the plant computer are Non-Class 1E. The conductors originate at a Class 1E I/O board and is routed through an isolation device to the terminal block within the Electronics Box. <u>To ensure the isolation device is protected from hot shorts, each conductor is independently fused between the isolation device and the terminal block. Two 1/4 amp, 250 VAC Class 1E fuses on the "+" and "-" of the isolator output are used for protection. These fuses will open under an abnormal faulted circuit condition to prevent damage to the isolator. The circuit is considered to be Non-Class 1E after the Class 1E fuses.</u> The conductors are routed away from any Class 1E device inside the Electronics Boxes, and are landed below the annunciator circuits on the terminal block. However, inside the Electronics Box the Class 1E and Non-Class 1E Cables both terminate on the same terminal block. The terminal

				block is heavy duty, barrier type, rated 600 Volts and 75 Amps AC, with breakdown voltage 13,000 V RMS line to line. The line to line spacing between the terminals is 0.66 inches. Non-Class 1E Cables are rated at 600 V AC and are fire retardant…. The Non-Class 1E Cables that originate from the Spent Fuel Panel CPX-EIPRLV-06 annunciator are wired to the dry contacts of Class 1E relays installed in the Electronics Box. It's not clear to the staff if the above-mentioned Class 1E fuses are located inside an active electrical equipment/enclosure/electronic box. <ins>Question</ins>: Please clarify where the above-mentioned Class 1E fuses are in terms of the active component in which they are located.
10		Page 2.1-25 (SBO)	FSAR pages 8B-1 and 7.6-1	In the LRA, the applicant stated that all electrical and I&C systems (except meteorological instrumentation and security systems) are included within the scope LR. Based on the applicant's scoping methodology, the staff understands that the electrical power systems required to cope with the SBO rule (10 CFR 50.63) are included in the scope of LR. For this reason, it is not clear to the staff why the applicant stated in FSAR section 8B, "Station Blackout," that "<ins>no power source is required to support the SBO unit,</ins>" which is part of the below sentence: In the event that a single EDG is not available in the Non-SBO Unit, such as during a two train EDG outage, <ins>no power is required to support the SBO</ins>

				Unit, since no common equipment is required in support of the SBO Unit. Question: Please clarify if any power source (e.g., DC power source) is required to support the systems required to cope with an SBO event in one unit in case both EDGs in the non-SBO unit are unavailable.

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License Renewal Application (LRA) Breakout Audit Questions

LRA Section 4.2.1
TRP: 59.1

Question Number	LRA Section	LRA Page	Background / Issue	Discussion Question / Request
	4.2.1	4.2-3	• The LAR states that fluence projections were performed with WCAP-18124-A • I was not able to find a previous CPNPP licensing action that cited WCAP-18124-A.	Discussion of whether addition of WCAP-18124-A to the licensing basis is being requested for approval in the current LRA or when WCAP-18124-A was added to the licensing basis
	4.2.1	4.2-3	• WCAP-18124-A, Rev. 0 contains two limitations and conditions. • L&C 1 requires additional justification for application of the method outside the active height of the core. • This L&C can be met by referencing both WCAP-18124-NP, Rev. 0 and Supplement 1 to WCAP-18124-NP, Rev. 0 in the licensing request, but only WCAP-18124-NP, Rev. 0 is referenced. • L&C 2 places additional conditions on use of least-squares adjustment. • WCAP-18630-NP, Rev. 0, Section 2.3 states that comparison with surveillance CPNPP surveillance capsules were not used to modify calculated surveillance capsule and pressure vessel neutron exposures.	Discussion regarding disposition of WCAP-18124-A, Rev. 0 limitations and conditions.

	4.2.1	4.2-3	• The LRA does not contain details of the neutron fluence calculation.	Provide details of the neutron fluence calculation on the docket.
	Appendix A to WCAP-18630-NP (not presently docketed)		• Appendix A to WCAP-18630-NP provides EVND results. • EVND results for Unit 1 at the core midplane, core top, core bottom, RPV supports, and bottom head, are presented. • EVND results for Unit 2 at the core midplane, core top, and core bottom. • M/C differences for Unit 2, at core bottom, are comparable to M/C differences for Unit 1 , at RPV supports.	Discussion on whether EVND results presented in WCAP-18630 represent all capsules irradiated at CPNPP, and how determinations are made regarding EVND insertion, withdrawal, and analysis schedules.

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section 4.2.2 – 4.2.5 /TLAA:

Question Number	LRA/SLRA Section	LRA/SLRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	LRA Section 4.2 – Tables with material property values for Cu and Ni		• Copper and nickel for the "BELTLINE" material is slightly different that the values in FSAR Chapter 5 • I understand that the values in LRA match the information provided in the SPU LAR in 2008/2009 timeframe – but there wasn't a discussion that I was able to find that explained the slight variations in these material property values from FSAR and SPU/LRA. • Example for Unit 1 – not all inclusive • Materials R1107-1, -2 and -3 • FSAR Cu - 0.06, 0 .06, 0.05, respectively • LRA Cu - 0.07, 0.07, 0.06, respectively	Identify any additional discrepancies for Cu, Ni and RTndt (U) – if any Can the licensee discuss the basis for these slight changes/variations between the FSAR and LRA What was the basis for the revision in values??
2			• Initial USE - LRA Table 4.2.3-1	Are there any other discrepancies?

			EXAMPLES (not all inclusive) • Unit 1 - Beltline region weld metal (heat # 88112) – “footnote a” cited FSAR table 5.3-2A for initial values • Initial USE in LRA -133 ft-lbs • FSAR Table – 150 ft-lbs • Unit 1 - Lower Shell Plate 1108-1 – footnote a cited FSAR table 5.3-2A for initial values • Initial USE in LRA - 85 ft-lbs • FSAR Table – 84.5 ft-lbs • Unit 1 – Intermediate Shell R1107-1 • Initial USE in LRA - 94 ft-lbs • FSAR Table – 93.5 ft-lbs There are variations in values reported in LRA and FSAR. I understand that “rounding” may be the reasoning – but in at least one instance initial USE values in LRA were reported to first decimal place – So not sure if initial values actually were changed or in some instances there was rounding.	Discuss why there is a discrepancy between the LRA and FSAR – If this is related to Question #1 – please address together.

3			LRA identifies upper shell plate as R-3806-1, -2 and -3 UFSAR identifies is what appears to be the same materials with “A” designation.	It looks like they are the same material – if so, why the difference in naming/identification?
4	WCAP 18630-NP		Foot note d – copper and nickel from CE-NPSD-1119- table 5 - • Intermediate Shell longitudinal weld seams 101-124A, B and C (heat# 89833) • Lower shell long weld seams 101-142 A, B and C (heat# 89833) • Intermediate to Lower shell girth weld seam 101-171 (heat # 89833) • Upper Shell to Intermediate shell girth weld seam 103-121 (heat# 3P7317) CE Report LD-79-036 - Report was cited as the basis for Copper and Nickel for Upper Shell Longitudinal Weld Seams 101-122 A, B and C (heat# 89827) CE-NSPD-1119 is over 20 years old and CE Report LD-79-036 is over 40 years old	Discussion as to how these initial values were determined for materials that cited CE Report LD-79-036 Provide CE Report LD-79-036 Since the cited reports are relatively old, has any effort been made to see if there is any recent information that is applicable to these materials? If not, does there need to be ? ***Question 4 and 5 can be addressed together***

5			WCAP 18630-NP - Table 3-1 and 3-2 – both materials for BOTH units– Related to initial USE values • Upper Shell to Intermediate Shell Girth Weld Seam 103-121 • Upper Shell Longitudinal Weld Seam 101-122 A, B and C • Cites CE Report LD-79-036 – cited as the basis initial USE values Unit 1 • Upper Shell Plate to Intermediate Shell girth weld seam 103-121 (HT# 90149) • Basis for initial value of 144 ft-lbs • Upper Shell Longitudinal Wels Seams 101-122A, B and C • Basis for initial value of 197 ft-lbs Unit 2	Discussion as to how these initial values were determined for materials that cited CE Report LD-79-036 Provide CE Report LD-79-036 Since the cited report is relatively old, has any effort been made to see if there is any recent information that is applicable to these materials? If not, does there need to be ? ***<u>Question 4 and 5 can be addressed together</u>***

			• Upper Shell to Intermediate Shell Girth Weld 103-121 (HT# 3P7317) • Basis for initial value of 99 ft-lbs • Upper Shell Longitudinal Weld Seams 101-122A, B and C (HT# 89827) • Basis for initial value of 142 ft-lbs	
6			Unit 1 – Upper Shell Longitudinal Weld Seams 101-122 A, B and C (heat #4P6052) LRA Table 4.2.2-1 - Footnote f – does not discuss/reference the credibility evaluation for this material WCAP-18630 – NP – Section B.2 indicates that this material is the same from Seabrook, Unit 1, and Millstone, Unit 3 WCAP-18607 – NP was cited as having the credibility evaluation for this material	Does WCAP-18607 have the credibility evaluation for both Millstone and Seabrook surveillance data? Provide a quick summary of the credibility evaluation in WCAP-18607 Please provide WCAP-18607 – NP and/or ***<u>any other document</u>*** that contains the credibility evaluation for this material from Seabrook, Unit 1, and Millstone, Unit 3 Provide a quick summary of the credibility evaluation in WCAP-18607 or whichever document contains the evaluation.

7			Unit 2 – Upper Shell to Intermediate Shell Girth Weld 103-121 (HT# 3P7317) LRA Table 4.2.2-2 - Footnote f – does not discuss/reference the credibility evaluation for this material WCAP-18630 – NP – Section B.2 indicates that this material is the same from Palo Verde unit 2 WCAP-16524 – NP was cited as having the credibility evaluation for this material Table 5-5 of WCAP-18630 identifies that the surveillance weld is from Millstone 3 – this is different from the Table 5-3 and Appendix B of WCAP-18630 (i.e., surveillance weld is from Palo Verde)	Which plant is the surveillance weld from ? Palo Verde? Or Millstone? Please provide WCAP-16524 – NP and/or any other document that contains the credibility evaluation for this material from Palo Verde Unit 2 and/r Millstone Unit 3 – which ever plant(s) the surveillance weld is from Provide a quick summary of the credibility evaluation in WCAP-16524 or whichever document contains the evaluation.
8	LRA Section 4.2.5 – PT limits		Dispositioned in accordance with (iii) – LRA states “The effects of aging on the intended function(s) of the reactor vessels will be adequately managed for the PEO. The Reactor Vessel Surveillance AMP described in B.2.3.18 will ensure that updated P-T limits based upon updated ART values will be submitted to the ***<u>NRC for approval</u>*** prior to exceeding the current terms of applicability of CPNPP Units 1 and 2.	It appears that Comanche Peak has been approved for PTLR - If this is true, why would ART values need to be submitted for NRC review and approval? Is there something unique for Comanche with the PTLR approval ? With a PTLR - Why would PT limits need to be

				submitted to the NRC for review and approval? Didn’t see anything in PTLR approval in 2007 and TS 5.6.6

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section /AMP/B2.3.7 PWR Vessel Internals

Question Number	LRA Section	LRA Page	Background / Issue	Discussion Question / Request
1	AMP B2.3.7	LRA Page B-57 and B-58	Page B-57 of the AMP states PWR Owners Group (PWROG) issued guidance OG-21-160 to address clevis insert degradations. The PWROG-referenced guidance is new guidance of how to perform ultrasonic test (UT) inspections of the clevis insert bolts. The current aging management basis for clevis insert bolts and wear surfaces in Item W14 in Table 4-9 of MRP-227, Rev. 1-A only calls for ASME Section XI VT-3 visual inspections of the clevis insert bolts and wear surfaces, as modified by the VT-3 criteria in Westinghouse Bulletin No. TB-14-5. Item W14 in Table 4-9 of MRP-277, Rev. 1-A does not include or reference use of PWROG guidance OG-21-160 (although the staff acknowledges that use of the OG-21-160 guidance was added to Item W14 in Table 4-9 of MRP-227, Rev. 2, which is under the staff review at present).	(1)Discuss how Item W14 in Table 4-9 of MRP-227, Rev. 1-A will be implemented during the period of extended operation for the CPNPP units and the scope of supplemental Westinghouse or PWROG guidance that will be implemented under the inspection and evaluation (I&E) criteria of Item W14 in Table 4-9 of MRP-227, Rev 1-A. (2) If the guidance in OG-21-160 will be implemented in addition to Westinghouse Bulletin No. TB-14-5, discuss whether OG-21-160 will be identified as an Enhancement of Item W14 in Table 4-9 of MRP-227, Rev. 1-A and an Enhancement as part of AMP B2.3.7.
2	AMP B2.3.7	LRA Pages B-58 to B-59	In the Operating Experience section of AMP B 2.3.7, the applicant discussed industry operating experience of upper and lower assembly fuel alignment pins. The applicable generic operating experience is associated with U.S. reports of detected surface breaking indications induced by wear in the pins of another nuclear plant. The applicant states that recommendations for managing wear in the pins is given	(1)Discuss whether the fuel alignment pins in the upper and lower reactor internal assemblies at CPNPP units are designed as ASME Section XI Examination Category B-N-3 components. If so, discuss whether the ASME Section XI inspection will be performed on the alignment pins regardless of the fact that the alignment pins do not include a malcomized surface that would

			in Westinghouse Bulletin TB-14-6, but clarifies that the guidance only applies to the fuel alignment pins made from Type 304 stainless steel with a malcomized surface treatment. The applicant states that this operating experience is not applicable to CPNPP because the fuel alignment pins at CPNPP are made of cold-worked Type 316 stainless steel (316 SS). The staff notes that MRP-227, Rev. 1-A is not to be used as a replacement for implementing ASME Section XI inspections of reactor vessel internal components that are designated as ASME Code Class Internal components in the current licensing basis (CLB).	make them susceptible to wear. (2) Discuss whether the design of the alignment pins at CPNPP are susceptible to cracking based on the cold working of the 316 SS materials that were used in fabrications of the alignment pins.
3	AMP B2.3.7	LRA Page B-60	Inspections of Westinghouse-design control rod guide tube (CRGT) guide plates (guide cards) are covered by Item W1 in Tables 4-3 and 5-3 of MRP-227, Rev. 1-A. The fourth paragraph on LRA page B-60 states that during the Spring 2022 outage, all 53 Unit 1 guide tubes and associated guide cards were inspected and measured for wear based on guidance in WCAP-17451, Revision 1. The applicant stated that the low wear levels were observed at Unit 1. The applicant further stated that it will inspect Unit 2 guide cards in Spring 2023. Item W1 in Tables 4-3 and 5-3 of MRP-227, Rev. 1-A includes the CRGT guide cards as Primary category components that are subject to VT-3 visual inspections for wear-type flaw indications, without any designated or linked "Expansion" category components. Item W1 indicates that the inspections are done per the guidance in Proprietary Report WCAP-17451, Rev. 1. However, contrary to Item W1 in Table 4-3 of MRP-227, Rev. 1-A, the proprietary guidance in WCAP-17451, Rev. 1 indicates that there are two types of Westinghouse CRGT assembly components that serve as "Expansion,"	Discuss CRGT guide card inspections and the results of past inspections performed on the Unit 1 CRGT guide cards, including the following questions: 1) For CPNPP units, are there applicable "Expansion" components for the CRGT guide cards in the AMP? If so, identify the linked Expansion-category components for the Primary category CRGT guide cards in the AMP. 2) If there are applicable Expansion category components for the CRGT guide cards, was the degree of wear detected from the Unit 1 inspections sufficient to call for sample-*expansion* to the lined CRGT Expansion-category component types?

			instead of being Primary, components for Westinghouse-design CRGT guide cards.	
4	AMP B2.3.7	LRA Page B-61	Page B-61 discusses operating experience with PWR RPV nozzle thermal sleeves at the flange locations per NSAL-18-01, NSAL-20-01, Rev. 1, and MRP-2018-027 guidelines. The third paragraph on page B-61 of the CPNPP LRA states that "…At CPNPP, baseline inspections were completed during the Fall 2021 outage for Unit 2. Normal wear was found during visual inspections with a recommendation to re-inspect within 6 cycles. Baseline inspections for Unit 1 were completed during the Spring 2022 outage, also showing normal wear with a recommendation to re-inspect in 12 cycles…"	The re-inspection for the thermal sleeve flange wear at Unit 2 is 6 cycles but for Unit 1 the reinspection is 12 cycles. Confirm whether Unit 1 has a longer re-inspection interval because Unit 1 has a new RPV head which has not had significant wear at the thermal sleeve flange.
5	AMP B2.3.7	LRA Pages B-63 and top B-64	In MRP-277, Rev. 1-A, Table 4-6, the core barrel (CB) assembly upper girth weld (UGW) is established as an Expansion-category Item W3.1 weld for the linked Primary-category Item W3 upper flange weld (UFW) in Table 4-3 of MRP-227, Rev. 1-A. The NRC staff notes that, recently, the licensee for a U.S. Westinghouse-design nuclear plant detected several circumferential flaws in the UGW of the unit's CB assembly during the fall 2022 refueling outage. Some of the flaws were reported to have significant depths with respect to the weld wall thickness.	(1)Considering recent operating experience of the degraded UGW in the CB in a PWR plant, discuss whether CPNPP will inspect the CB UGWs in the near future, as the CB UGWs are currently designated as an Expansion-category welds for the CPNPP units. (2) Discuss whether the CB UFWs and UGWs are designated as ASME Section XI Examination Category B-N-3 weld for the CB assemblies. If so, discuss whether past ASME Section XI VT-3 inspections of the welds have identified any relevant flaws in the UFWs and UGWs.
6	AMP B2.3.7	LRA Page B-57	The applicant stated that "…The PWR Vessel Internals AMP will be consistent with NUREG-1801, Section XI.M16A, "PWR Vessel Internals", as superseded by SLR-ISG-2021-01-PWRVI… ". The NRC staff notes that NUREG-1801 is not superseded by SLR-ISG-2021-01-PWRVI, rather SLR-ISG-2021-01-PWRVI provides update to NUREG-1801. It is not clear whether the AMP will be consistent with SLR-ISG-2021-01-PWRVI. In	(1)Confirm that AMP B2.3.7 follows the version of AMP XI.M16A in the SLR-ISG-2021-01-PWRVI, without the need of an RVI gap analysis. (2) Confirm that Revision 2 of NUREG-1801 will be used in the AMP.

			addition, NUREG-1801 has two revisions, Revisions 1 and 2. It is not clear which revision will be used.	

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section B.2.3.3 “Reactor Head Closure Stud Bolting”

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	4.7.3	4.7-4 through -6	Section 4.7.3 describes a series of NRC-approved topic reports that support a 20-year inspection interval for the RCP flywheel. The appropriate staff SE (ML19198A056) states that applicants implementing this methodology to justify a 20-year inspection interval should confirm that several aspects of the underlying analysis are applicable to the applicant.	Confirm the following, based upon the referenced SE: • the normal operating speed for the RCP flywheels is 900 rpm • the design limiting speed for the RCP flywheels is 1125 rpm • the maximum overspeed following a design basis LOCA is 1368 rpm • it is appropriate to use 70°F as the medium temperature for design limiting event (Table 3-2) in the PFM analysis • 6000 start/stop cycles is bounding

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions
LRA Section/AMP: XI.M37 Flux Thimble Tube Inspection

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	Program Basis document – Section 4.1 “scope of program” element	Page 10 of 21	Scope of program element provides references to the applicant response to NRC Bulletin 88-09. Reference 9.9 provides the NRC response to Unit 1. For unit 1 – it looks like the NRC letter was issued 2 months after the initial inspection results were submitted to the NRC (letter dated Dec 20, 1991) Log # TXX-91455 File # 10119 904.4 Ref. # NRCB 88-09 December 20, 1991	Was there an NRC response for Unit 2 actions related to NRC Bulletin 88-09?

			Unit 2 initial inspection results submitted November 7 1994 - ML20078E542	
2	Program Basis document – Section 4.5 “monitoring and trending” element	Page 12 of 21	Monitoring and trending element discusses the use of WCAP 12866. It also discusses that this methodology has “sufficient conservatism.” WCAP-12866 has been not approved by the NRC generically	• Provide WCAP-12866 for review during audit. • Provide discussion of the projection methodology in WCAP-12688 and its use in CPNPP – including discussion of the “sufficient conservatism” mentioned in the program basis document. • How long has the methodology in WCAP-12866 been used at CPNPP? • Has any other projection methodology been used at CPNPP? If so, please describe and why the switch to WCAP-12866 • How well has WCAP-12866 projected wear to-date? Has there

				been any issues/anomalies with the projections where the predictions were non-conservative? • Portal contains inspection results for last two inspections at each units – Can results from the two prior inspections from each unit also be provided.
3	LRA Section B.2.3.23 – Operating Experience Section & Program basis document – Section 4.4 "Detection of Aging	LRA Page B-148 & Page 11 of 21	LRA states "Due to the potential thinning of flux thimble tubes as reported in NRC IEB 88-09 CPNPP commenced examinations of the flux thimble tubes at CPNPP Unit 1 and Unit 2 using eddy current examination in 1991. Flux thimble tubes are presently ***examined each refueling outage***" and "Eddy current inspection results of flux thimble tubes for the ***last two outages of each*** unit were reviewed." Program Basis document states "Flux thimble tubes are presently examined every other refueling outage (every 3 years)" – ^^this is consistent with the PM349506 and PM349507"	• Discuss the discrepancy between documents and the inspection frequency. • So currently inspections at each unit are done each outage or every other outage regardless of the projected wear (except if the thimble tube won't make it to next inspection)? (i.e.,

	Effects" element			inspection frequency/intervals are not extended for longer than the set periodicity?)
4	Program basis document – Discussion of RAIs from other LRA/SLRA	Page 9 of 21	Program basis document states that CPNPP "is participating in a PWR Owners Group project to examine the historical eddy current results with hopes to extend the interval between inspections"	• Provide description of this effort – what does it entail, status, results, etc? • Is this going to only look at plant-specific results to extend the inspection intervals? If not, additional discussion of this effort would be helpful. • Currently, is the extension of inspection intervals based on plant-specific results prohibited?
5	Program basis document – Section 4.6 – "Acceptance	Page 12 of 21	Program element 6 of XI.M37 states: Acceptance criteria different from those previously documented in the applicant's response to IE Bulletin 88-09 and amendments thereto, as accepted by the NRC, should be justified. Program basis document states – "wear acceptance criteria have been established, using the guidance of WCAP-12866…"	• Was the use of WCAP-12866 at CPNPP previously approved as part of the response to NRC Bulletin 88-09? (the CPNPP response to

	Criteria" element		Program basis documents also discusses action levels started at 20% through-wall wear until 80% through-wall wear – and references MRS-GEN-1180 (levels mentioned as part of calibration/standard in Section 7). This document did not appear to discuss the actions taken for each of these levels	Bulletin 88-09 in ML19327B845 doesn't provide the details)? • Provide discussion of the different actions taken based on the action levels discussed in the Program Basis document (i.e., 10%, 20% all the way through 80%). • Do these action levels also equate to when thimble tubes are capped, move/repositioned, removed/replaced, etc?? - If not how/when is this decision made? • Where is this currently documented on what actions are going to be taken based on through wall wear percent? - Didn't see it in the PM 349506, 349507,

				TPVEN_MRS-GEN 1180 and 1304

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section B.2.3.3 "Reactor Head Closure Stud Bolting"

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	B.2.3.3	See cited CR/EV-CR	CR-2019-003497/EV- CR-2019-003497-1 discusses the presence of demineralized water (for use as UT couplant) in stud holes. This CR/CR-EV fully described the situation and stated that the water will evaporate.	Have you checked in subsequent outages that there is no more water? Has there been any occurrences of this since 2019 (or was this a one-off situation)?
2	B.2.3.3	B-40	Regarding: ISI exam reports 1RF20 ISI Report.pdf includes: ISI exam sheet for B-G-1 B6.40 (threads in flange) states that "Geometric indication (sleeve) seen 360° around stud hole #6. The indication exceeded 20% DAC but did not exceed 50% DAC."	Clarify "Geometric indication (sleeve) seen 360° around stud hole #6. The indication exceeded 20% DAC but did not exceed 50% DAC."
3	B.2.3.3	B-40	The last operating experience in LRA Section B.2.3.3 states that outage delays occurred due to HydraNuts. CR-2013-000912 was created to indicate that HydraNuts have been replaced with original nuts.	What was the issue with the HydraNuts?
4	B.2.3.3	B-40	CR-2017-004534 says that in 2RF16, Stud #32 of Unit 2 was determined that it could not be removed, i.e., it was stuck.	Neither CR-2017-004534 nor the LRA provide information regarding the resolution of stuck Stud #32 in 2RF16. Can you describe the resolution to the stuck Stud #32 in 2RF16?

5	B.2.3.3	B-40	Questions #5 through #10 pertain to the damaged threads described as one of the operating experiences in LRA Section B.2.3.3 that occurred in July 2014. The staff would like to understand contributors to this particular OE and ensure that it is being adequately managed.	a. Is the damage just in the stud hole or also in the studs themselves? b. With respect to the flange upper surface, where is the approximate location of the damaged threads?
6	B.2.3.3	B-40	See #5 above	a. What were the factors that contributed to the thread damage and applicant's corrective actions, per 10 CFR 50 Appendix B via Element 7 of XI.M3 AMP, to prevent such damage in the future? b. If due to high tensioning loads, is there a limit placed on tensioning loads in the procedures (MSM-C1-9901 and MSM-C2-9901)?
7	B.2.3.3	B-40	See #5 above	How many studs are used to tension per unit and how many studs/holes were damaged at each unit?
8	B.2.3.3	B-40	See #5 above	a. Were the damaged stud holes repaired and/or damaged studs replaced? b. The LRA states that the ISI program will continue to monitor these locations; has the number of damaged threads increased for any stud hole with damaged threads, especially the one stud hole with 13.75 damaged threads since 2014? c. Has there been any **new** thread damage since 2014?
9	B.2.3.3	B-40	See #5 above	a. Is thread damage related to stuck studs, ie, any evidence that damage and being stuck are correlated?

				b. Is there a potential corrosion effect on the stuck/damaged studs especially during refueling outages?
10	B.2.3.3	B-40	See #5 above	a. Is 13.75 threads the highest number of damaged threads from both units? b. Regarding the allowable damaged threads of 17.22: is there a limit on tension/detension cycles?

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section /AMR/PWR Vessel Internals

Question Number	LRA Section	LRA Page	Background / Issue	Discussion Question / Request
1	Table 3.1.2-2	LRA page 3.1-83 (pdf page 410)	Items IV.B2.RP-345a and IV.B2.R-424 in Table 3.1.2-2 on page 3.1-83 are identified as bottom mounted instrumentation (BMI) flux thimble tubes with Notes C, D, and E. These two items are correlated to aging effects of SLR-ISG-2021-01-PWRVI Item IV.B2.RP-345a, GALL-SLR Items IV.B2.R-345 and R-424 such as loss of fracture toughness due to irradiation embrittlement (IE), loss of material due to wear, change of dimensions to void swelling (VS) or distortion, and loss of preload due to irradiation-assisted stress relaxation or creep (ISR/IC). These line items apply to the reactor vessel internal portion of the thimble tubes which are not reactor coolant pressure boundary. However, the BMI flux thimble tubes extend to the outside of the reactor vessel which become part of pressure boundary and are identified with items IV.A2.RP-28, -154 and -284 in Table 3.1.2-1 (page 3.1-62). The aging effects for thimble tubes outside of the RPV are different from the thimble tube that are inside of the RPV. LRA Section 3.1.2.2.9, page 3.1-13, indicates that the internal, non-reactor coolant pressure boundary portions of the flux thimble tubes are “No Additional Measures” components per MRP-227, Rev. 1-A as linked to SRP-SLR Table 3.1-1, Item 3.1-1, 055c and GALL-SLR Item IV.B2.RP-265. Thus, it appears that a discrepancy exists regarding (1) which line item or items should be used for the PWR vessel internals portions of the flux thimble per LRA Section 3.1.2.2.9 and	Discuss the discrepancy on AMR items and AMP bases for the non-pressure boundary portions of the flux thimble tubes that are located inside the RPV (i.e., noting that the pressure boundary portions of the thimble tubes are adequately covered by applicable line items in LRA Table 3.1.2-1 on page 3.1-62.

			those cited for the flux thimble tube non-pressure boundary portions in Table 3.1.2-2, and (2) whether the PWR vessel internal, non-pressure boundary portions of the BMI flux thimble tubes are being subject to MRP-227 aging management activities (i.e., condition monitoring inspections).	
2	Table 3.1.2-2	LRA page 3.1-83 (pdf page 410)	Table 3.1.2-2, page 3.1-96, includes line items, IV.B2.RP-289 and RP-345 on cracking of the core barrel flanges that align to GALL-SLR Item IV.B2.RP-289 using either Note C or D. However, the applicable and corresponding Item IV.B2.RP-289 in SLR-ISG-2021-01-PWRVI applies specifically to MRP-227 Existing Program bases for Westinghouse-design lower core plates or extra-long (XL) lower core plates, where the plates are subject to ASME Code Section XI VT-3 visual inspections for cracking per Item W12.a in Table 4-9 of MRP-227, Rev. 1-A and for loss of material due to wear per Item W12.b in Table 4-9 of MRP-227, Rev.1-A. The GALL-SLR report (as updated inclusive of changes made in SLR-ISG-2021-01-PWRVI) does not include an applicable AMR item for cracking of core barrel flanges.	Confirm that the aging effect of cracking on core barrel flanges associated with line item IV.B2.RP-289 is included in Table 3.1.2-2 on page 3.1-83 such that the item is aligned with Item IV.B2.RP-289 in SLR-ISG-2021-01-PWRVI for the ASME Section XI, VT-3 visual inspections that will be applied to the "Existing Program" category CPNPP core barrel flanges per Item W10 in MRP-227, Rev. 1-A.
3	Table 3.1.2-2	LRA page 3.1-96 (pdf page 423)	Table 3.1.2-2, page 3.1-96 includes line items for neutron shield panel support pins that align to GALL-SLR Items IV.B2.RP-271 and RP-272 using Note C or D. These line items are included in Table 3.1.2-2 to manage irradiation-assisted stress corrosion cracking (IASCC) and fatigue induced cracking, loss of material due to wear, loss of preload due to irradiation-assisted stress relaxation or creep (ISR/IC), and loss of fracture toughness due to irradiation embrittlement (IE) in the pins. The referenced GALL-SLR IV.B2.RP-271 and IV.B2.RP-272 items apply to management of applicable cracking, loss of preload, loss of fracture, changes in dimensions, and loss of material mechanism in Primary category Westinghouse-design baffle-to-former (BF) bolts using GALL/GALL-SLR AMP	Discuss the discrepancies on the AMR items provided for neutron shield panel supports pins on LRA page 3.1-96. Specifically, discuss the following questions related to the AMR line items for the neutron shield panel support pins: 1) Are the referenced AMR line items for the neutron shield panel support pins will be subject to augmented, Primary category inspections per a plant-specific adjustment of the MRP-227, Rev. 1-A protocols?

			XI.M16A, where the bolts are subject to augmented aging management inspections per Item W9 in Table 4-3 of the MRP-227, Rev. 1-A report. The LRA's adoption of the referenced "RP-271" and "RP-272" items under Note C or D bases could only be applied to the neutron shield panel support pins if the pins were subject to augmented Primary item inspections in the manner that Primary category ultrasonic test (UT) inspections are applied to and performed on Westinghouse-design baffle former bolts per Item W9 in Table 4-3 of MRP-227, Rev. 1-A. Instead, in MRP-227, Rev. 1-A, all neutron shield panel components are placed in the "No Additional Measures" of the program because the pins do not screen-in for any aging effects per MRP-227, Rev. 1-A.	2) If the neutron shield panel support pins are justified for placement in the "No Additional Measures" category of AMP B2.3.7, PWR Vessel Internals Program, discuss why isn't Table 3.1.2-2 giving a single line item for the neutron shield panel support pins that aligns to SRP-SLR Table 3.1-1, Item 3.1-1, 055c and GALL-SLR Item IV.B2.RP-265?
4	Table 3.1.2-2	LRA page 3.1-97 (pdf page 424)	Table 3.1.2-2 includes line items for radial support keys and associated bolts that align to either GALL-SLR Items IV.B2.RP-285 or RP-399 using Note C or D. These line items are included in Table 3.1.2-2 to manage IASCC and fatigue induced cracking, changes in dimension due to void swelling (VS) or distortion, loss of material due to wear, loss of preload due to irradiation-assisted stress relaxation or creep (ISR/IC), and loss of fracture toughness due to irradiation embrittlement (IE) in the radial support key components. The cross-referenced GALL-SLR items IV.B2.RP-285 and RP-399 items apply to aging management of cracking due to stress corrosion cracking or fatigue, loss of material due to wear, and loss of preload due to ISR/IC (bolts only) in reactor pressure vessel interfacing clevis insert assembly components, including bolts or screws, dowels, and clevis insert surfaces, where the GALL-SLR based line item calls out the ASME Code, Section XI inspection-based Existing Program criteria in Item W14 of Table 4-9 in MRP-227, Rev. 1-A for the clevis component types.	Discuss Note C or D-related AMR line items for radial support keys and associated bolts in Table 3.1.2-2, page 3.1-97. Specifically, confirm whether the proposed AMR line items are conservatively being included in Table 3.1.2-2. page 3.1-97, to add in the radial support keys and associated bolts as ASME Code Section XI-based Existing Program components for the AMP B2.3.7, PWR Vessel Internals Program and the referenced aging effects, such that Table 3.1.2-2 is consistent with Item W20 of Table 4-9 in MRP-227, Rev. 2.

			The GALL Rev. 2 and GALL-SLR reports do not include a similar Existing Program-based AMR line items for the radial support key components. However, the EPRI MRP added the radial support keys in as new Existing Program components in the newly submitted MRP-227, Rev. 2 report through inclusion of the component type in newly proposed Item W20 of Table 4-9 in MRP-227, Rev. 2.	
5	Table 3.1.2-2	LRA page 3.1-98 (pdf page 425)	Table 3.1.2-2, page 3.1-98, includes two line items for the upper internals assembly (UIA) support ring that aligns to Item IV.B2.RP-288 using Note C. These line items are included in Table 3.1.2-2 to manage loss of fracture toughness due to irradiation embrittlement (IE) and loss of material due to wear in the UIA support rings. The cross-referenced item IV.B2.RP-288 in SLR-ISG-2021-01-PWRVI applies to aging management of loss of material due to wear, and loss of fracture toughness due to IE, and changes in dimension due to void swelling (VS) or distortion in Westinghouse-design lower core plates or extra-long (XL) lower core plates.	(1)Discuss Note C or D-related AMR line items for the UIA support rings on page 3.1-98. Specifically, confirm whether the proposed AMR line items are conservatively being included in Table 3.1-2, page 3.1-98, to add loss of fracture toughness and loss of material due to wear as aging effects (i.e, in addition to cracking) for the ASME Section XI-based Existing Program criteria for the UIA support rings that are aligned to Item W11 of Table 4-9 in MRP-227, Rev. 1-A. (2) Discuss whether the RP-288 based line items for UIA support ring in Table 3.1.2-2 need to include an associated AMR line item on aging management of changes in dimension due to void swelling or distortion in the UIA support rings to be consistent with item IV.B2.RP-288 in SLR-ISG-2021-01-PWRVI .
6	Table 3.1.2-2	LRA pages 3.1-91 - 3.1-96	Table 3.1.2-2, pages 3.1-91 to 3.1-96, includes a number of AMR line items for lower support columns and associated column components that align to GALL-SLR Item IV.B2.RP-290 (for non-cracking effects) or IV.B2,RP-291 (for cracking) using Note A or B. These items correspond to Expansion Item W4.4 in Table 4-6 of MRP-227, Rev 1-A.	(1)Discuss any unit-specific differences between the design of the lower support columns and associated components in Unit 1 from those that are included in Unit 2. (2) If applicable, discuss whether those design differences will cause any adjustments of the MRP-227, Rev. 1-A inspection and

		(pdf pages 418 – 423)	However, the line item entries in Table 3.1.2-2 differentiate between the lower support column components in Unit 1 and Unit 2.	evaluation protocols for Expansion Category Item W4.4 lower support columns (independent of whether the columns are made from either cast austenitic stainless steel materials or wrought stainless steel materials).
7	Table 3.1.2-2	LRA Pages 3.1-85 To 3.1-86	AMR line item IV.B2.RP-386 in GALL, Rev. 2 (pdf page 249) covers management of loss of material due to wear in Westinghouse-design control rod guide tube (CRGT) C-tubes and sheaths. Although Table 3.1.2-2, pages 3.1-85 to 3.1-86 includes corresponding line items for aging management of loss of material due to wear of CRGT that corresponds to Primary Item W1 CRGT guide plates (i.e., CRGT guide cards), Table 3.1.2-2 does not include corresponding AMR items for the CRGT C-tubes and sheaths.	Discuss why Table 3.1.2-2 does not include any AMR item or items for managing loss of material due to wear in the CRGT C-tubes and sheaths. Specifically, discuss why the CRGT C-tubes and sheaths are not included within the scope of the line item on loss of material due to wear for the CRGT guide cards that is aligned to Item IV.B2.RP-296 in Table 3.1.2-2 on page 3.1-86.
8	Table 3.1.2-2	N/A	Table 3.1.2-2 does not include any component-specific AMR line items for control rod guide tube (CRGT) support pins (spilt pins) that are included in the CPNPP unit-specific CRGT assembly designs. In Table 3.1-1, Item 028 of SLR-ISG-2021-01-PWRVI, the basis for managing Westinghouse-design CRGT spilt pins was dependent on whether the pins were: (1) made from X-750 nickel-based alloy materials (versus Type 316 stainless steel materials), and whether the spilt pins were Examination Category B-N-3 components of Table IWB-2500-1 of the ASME Code, Section XI where the ASME Section XI inspections would be credited for aging management of cracking in the pins. In Table 3.1-1 of SLR-ISG-2021-01-PWRVI, Item 3.1-1, 028, replacement CRGT split pins made from type 316 stainless pins (and not subject to ASME Code Class inspections credited for aging management) were permitted to be placed in the “No Additional Measures” category consistent with the	Discuss how the CRGT split pins are being managed for “cracking” consistent with criteria defined in the “GALL-SLR Item” column entry of Table 3.1-1, Item 028, of SLR-ISG-2021-01-PWRVI. Discuss the following design basis for the CRGT spilt pins: (1) material of fabrication for the pins in both units, (2) whether the CRGT spilt pins in both units are original pins or replacement pins, (3) whether the CRGT spilt pins are defined in the CPNPP design basis that required to be inspected periodically under Examination Category B-N-3 of Table IWB-2500-1 of the ASME Code, Section XI, and (4) if the pins are ASME Code Class components, discuss whether the ASME Section XI inspections

			criteria established for CRGT spilt pins in MRP-227, Rev. 1-A. The staff notes that this is predicated on the condition that, if CRGT spilt pins are ASME Code Class pins, the ASME Section XI inspections of the pins are not credited for Existing Program credit in MRP-227, Rev.1-A such that the ASME inspections of the pins would be credited under the applicant's AMP B.2.3.1 ISI Program per GALL Item IV.E.R-444 in the SLR-ISG-2021-01-PWRVI.	scheduled for the CRGT spilt pins are being credited for aging management of cracking in the pins per the AMP B.2.3.1, ASME Section XI ISI Program versus ASME-based Existing Program basis in the AMP B.2.3.7, PWR Vessel Internals Program.

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License Renewal Application (LRA) Breakout Audit Questions

LRA Section B.2.3.18, Reactor Surveillance AMP

Question Number	LRA/SLRA Section	LRA/SLRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	LRA Section B.2.3.18 – Program Description	LRA Page B-126-127	• Unit 1 – Capsule Z reinserted prior to 36EFPY – need additional 9EFPY – to get 80 EFPY fluence. • Capsule W and V – can also be used – 13 additional EPFY • Unit 2 - Capsule Z reinserted prior to 36EFPY – need additional 8EFPY – to get 80 EFPY fluence. • Capsule Y and V – can also be used – 14 additional EPFY • Enhancement to Element 4 states: A capsule in each unit will be reinserted prior to 36 EFPY in order to achieve at least a vessel equivalent fluence of 80 EFPY. • The Enhancement and UFSAR Supplement (A.2.2.18) is very open ended – there needs to be some level of regulatory certainty regarding what/when is being credited for aging management (i.e., which capsule at what time)	• What is the current best-estimate approximate calendar year that the capsule for each unit will be inserted back into the reactor vessel at 36 EFPY – for both units • What is the current best-estimate approximate calendar year that the capsules for each unit will be withdrawn after the needed exposure in the reactor (Unit 1 – Capsule Z, W and V) and (Unit 2 – Capsule Z, Y and V) • Given responses to above questions – provide discussion regarding lack of detail in LRA (Appendix A and B) for enhancement to element 4, “Detection of Aging Effects” for the capsule withdrawal schedule

			• Additionally, this AMP is tied with Appendix H to 10 CFR Part 50 – Section III.B.3 states "A proposed withdrawal schedule must be submitted with a technical justification as specified in § 50.4. The proposed schedule must be approved prior to implementation."	
2	LRA Section B.2.3.18 – Enhancements	LRA Page B-127-128	• Enhancement to Element 4 and 7 states: "The capsule withdrawal schedule will be documented in the PTLR and note that changes require NRC approval per 10 CFR 50, Appendix H." • FSAR section 5.3.1.6.1 states "The schedule for removal of the capsules for post-irradiation testing is found in Section 2.4 of PTLR for the respective units." • PTLR (ML21075A112) Section 2.4 states "A withdrawal schedule for Units 1 and 2 are not necessary, because all Units 1 and 2 surveillance capsules have been withdrawn from the reactor vessel."	• Where is the latest information for capsule schedule/fluence/EFPY/lead factor\etc, info be found? – • Provide document on portal – is this document publicly available (i.e., been submitted to the NRC and in ADAMs) – latest document I was able to find were capsule reports when they were submitted per Appendix H in the 2000 timeframe. • Discuss the thinking/rationale for needing this enhancement? Seems like the FSAR/PTLR has provisions related to a capsule withdrawal schedule.
3	LRA Section B.2.3.18 – Operating Experience	LRA Page B-128-129	• Operating experience example for July 2019 – indicates that PTLR needed to be updated to include data from latest capsule withdrawal (unit 1 – 2005 and Unit 2 -2009) • LRA Section B.2.3.18 states - Therefore, this is considered administrative for the purpose of	• Can you explain the situation – from LRA it looks like capsule data was never incorporated/assessed into the PTLR until 2020/2021. • It's not clear how this situation was administrative – when Appendix H has reporting requirements related to whether or not PT limits need to be

			tracking the revision of the PTLR to incorporate the results from the Capsule X (Unit 1) and Capsule W (Unit 2) analysis.	revised based on capsule testing – See App H - Section IV.C • So, the PT limits were not assessed to see if the they were impacted by the capsule data for 16 and 12 years for unit 1 and 2, respectively? Provide discussion of circumstances/situation related to OPEX example.
4	LRA Section B.2.3.18 – Enhancements	LRA Page B-127-128	• Enhancement to Element 4 and 5 states "The AMP documents will be modified to establish operating restrictions to ensure that the plant is operated within the material aging OE, i.e., the cold leg operating temperature during normal operation will be limited to 525°F (minimum) to 590°F (maximum)."	• The temperature range being selected – is it from Section 1.3, "Limitations" of RG 1.99, Rev 2? • Considering that this is an enhancement (i.e., existing program isn't currently doing it) What is the current operating window CPNPP is operating within for the cold leg operating temperature during normal operation • What is expected to actually be updated to implement this enhancement – something in the PTLR, Tech specs, internal operating procedure, something else

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LRA Section B.2.2.1, "Fatigue Monitoring"

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	B.2.2.1	B-21	LRA Section B.2.2.1 addresses Enhancement 1 regarding the "scope of the program" and "preventive actions" program elements of the Fatigue Monitoring Aging Management Program (AMP). The enhancement states that the program will be modified to include environmentally assisted fatigue (EAF) analyses for locations that have been determined to be sentinel locations through the EAF screening evaluation in addition to those listed in NUREG/CR-6260. The sentinel locations are also called limiting EAF locations. In comparison, LRA Section 4.3.4 addresses the EAF analyses. Specifically, LRA Tables 4.3.4-1 and 4.3.4-2 describe the equipment (component) sentinel locations and piping sentinel locations, respectively. These sentinel locations include those identified for newer vintage Westinghouse plants in NUREG/CR-6260, which are applicable to the Comanche Peak plant, and other sentinel locations (plant-specific sentinel locations) that may be more limiting than the generic locations identified in NUREG/CR-6260. With respect to the sentinel locations identified in NUREG/CR-6260, LRA Tables 4.3.4-1 and 4.3.4-2 provide the 60-year projected environmental cumulative usage factor (CUF_{en}) values. In contrast, Tables 4.3.4-1 and 4.3.4-2 do not provide CUF_{en} values for the sentinel locations other than those identified in NUREG/CR-6260 in relation to	1. Clarify whether the applicant will submit for the NRC staff's review and approval the 60-year projected CUF_{en} values for the plant-specific sentinel locations (not listed in NUREG/CR-6260) at least a certain time period (e.g., one year) before entering the period of extended operation to demonstrate the effectiveness of the Fatigue Monitoring AMP with no need for additional activities (e.g., flaw tolerance analysis). If so, discuss a potential need to revise Enhancement 1 of the program accordingly. If not, provide justification for why such a submittal is not necessary. 2. Clarify whether this enhancement also include the monitoring of environmental cumulative usage factor (CUF_{en}) values to ensure that the CUF_{en} values do not exceed the design limit (1.0). If not, provide justification for why such monitoring is not necessary as part of Enhancement 1.

			Enhancement 1 of the Fatigue Monitoring AMP (i.e., the enhancement to the program to estimate the CUF_{en} values for these non-NUREG/CR-6260 sentinel locations for the extended period of operation). However, the staff requests clarification regarding whether the applicant will submit for NRC staff's review and approval the 60-year projected CUF_{en} values for the plant-specific sentinel locations (not listed in NUREG/CR-6260) at least a certain time period (e.g., one year) before entering the period of extended operation to demonstrate the effectiveness of the Fatigue Monitoring AMP with no need for additional activities (e.g., flaw tolerance analysis). In addition, the staff requests clarification regarding whether this enhancement include the monitoring of CUF_{en} values to ensure the CUF_{en} does not exceed the design limit (1.0).	
2	B.2.2.1	B-21	LRA Section B.2.2.1 addresses Enhancement 2 regarding the "preventive actions" program element of the Fatigue Monitoring aging management program (AMP). The enhancement states that the program will be modified, as needed, to monitor the environmental effects at the sentinel locations. It is not clear to the staff what parameters will be monitored for environmental effects in the enhancement and what AMP will be credited for the monitoring of environmental effects (e.g., specific water chemistry parameters). The staff also requests clarification regarding the context of the phrase, "as needed" in the enhancement (e.g., if a need for this enhancement has not been determined, why it has not been determined yet). The staff further requests clarification regarding why the environmental effects are not monitored for the EAF locations other than the sentinel locations even though there is a need to ensure that the assumptions used in the	1. Clarify what parameters will be monitored for environmental effects in the enhancement and what AMP will be credited for the monitoring of environmental effects (e.g., water chemistry parameters such as dissolved oxygen). In addition, clarify whether this enhancement also monitors the CUF_{en} values to ensure that they do not exceed the design limit (1.0). 2. Explain the context of the phrase "as needed" in the enhancement (e.g., if a need for the enhancement has not been determined, why it has not been determined yet). 3. Discuss why the environmental effects (e.g., water chemistry parameters such as dissolved oxygen) are not monitored

			EAF analysis are valid for both sentinel and non-sentinel locations for the extended period of operation. In addition, the staff requests clarification regarding whether this enhancement include the monitoring of CUF_{en} values to ensure that the CUF_{en} values do not exceed the design limit (1.0).	for the EAF locations other than the sentinel locations even though there is a need to ensure that the assumptions for environmental conditions used in the EAF analysis are valid for both sentinel and non-sentinel locations. Alternatively, discuss a potential need to revise the enhancement in order to include the monitoring of environmental effects for EAF locations other than the sentinel locations.
3	B.2.2.1	B-21	LRA Section B.2.2.1 addresses Enhancement 3 regarding the “parameters monitored or inspected” program element of the Fatigue Monitoring AMP. The enhancement states that the program will be revised to account for additional critical thermal and pressure transients for components that have been identified to have a fatigue time-limited aging analysis (TLAA). In comparison, LRA Section 4.3.1 addresses the design transients and their 60-year cycle projections that are used in the fatigue TLAAs. Specifically, LRA Tables 4.3.1-2, 4.3.1-3 and 4.3.1-4 describe the reactor coolant system (RCS) transients, normal condition auxiliary system transients and auxiliary system transients with applicable components, respectively. The staff requests clarification regarding whether LRA Tables 4.3.1-2, 4.3.1-3 and 4.3.1-4 include the thermal and pressure transients that will be added as part of Enhancement 3 of the Fatigue Monitoring AMP. If not, the staff requests clarification regarding why the transients to be added in the enhancement are not evaluated in these tables in LRA Section 4.3.1. In addition, the staff requests clarification regarding the identification of which transients will be added as part of the	1. Clarify whether LRA Tables 4.3.1-2, 4.3.1-3 and 4.3.1-4 include the thermal and pressure transients that will be added as part of Enhancement 3 of the Fatigue Monitoring AMP. If not, explain why the transients to be added as part of the enhancement are not evaluated in these LRA tables (e.g., evaluation to confirm that the 60-year projected cycles do not exceed the design transient cycles). 2. Describe what transients will be added as part of Enhancement 3 and the basis why these transients are not included in the existing fatigue monitoring.

			enhancement and the basis why these transients are not included in the existing fatigue monitoring.	
4	B.2.2.1	B-21	LRA Section B.2.2.1 addresses Enhancement 4 regarding the “acceptance criteria” program element of the Fatigue Monitoring AMP. The enhancement states that the program will be modified to include acceptance criteria based on the 60-year cycle projections used in the supporting analyses. The staff noted that the enhancement does not clearly discuss whether the acceptance criteria for transient cycles will be established in such a manner to ensure that the fatigue design limit is not exceeded (e.g., CUF and CUF_{en} do not exceed 1.0). The staff also requests clarification regarding whether the acceptance criteria addressed in the enhancement include the criteria for parameters other than transient cycles.	1. Clarify whether the acceptance criteria related to transient cycles will be established in such as manner to ensure that the fatigue design limit is not exceeded (e.g., CUF and CUF_{en} do not exceed 1.0). 2. Clarify whether the acceptance criteria addressed in the enhancement include the criteria for parameters other than transient cycles. If so, clarify those parameters and their acceptance criteria that will be addressed in the enhancement.
5	B.2.2.1	B-25	LRA Section B.2.2.1 addresses Enhancement 5 regarding the “corrective actions” program element of the Fatigue Monitoring AMP. The enhancement states that the program will be modified to provide clarity on when to initiate corrective action. The staff noted that the enhancement does not clearly explain what clarity will be specifically provided regarding when to initiate corrective action. The staff also requests clarification regarding whether this enhancement include a timely initiation of corrective actions to address the CUF values that approach the screening criteria (0.1) for high energy line break (HELB) location postulation described in LRA Section 4.3.5, “High-Energy Line Break Analyses.” In addition, the staff requests clarification regarding whether the corrective actions for new additional HELB locations are also part of this enhancement.	1. Explain the specific clarity that will be provided regarding when to initiate corrective action in the enhancement. As part of the response, describe whether the enhancement includes a timely initiation of corrective actions to address the CUF values that approach the screening criteria (0.1) for HELB location postulation. 2. Clarify whether the corrective actions for new additional HELB locations are also part of the enhancement (e.g., additional HELB analysis for newly identified break locations and their effects during the period of extended operation). If so, revise the enhancement accordingly. If not, confirm that the existing program

				includes such corrective actions for newly identified, additional HELB locations and the associated HELB analyses.
6	B.2.2.1	B-21	Regulatory Issue Summary (RIS) 2008-30, “Fatigue Analysis of Nuclear Power Plant Components,” addresses fatigue usage calculations that consider component’s response to a step change in temperature. The fatigue analysis discussed in the RIS pertains to a detailed stress analysis of the component with the use of the Green’s (or influence) function. The RIS indicates that the concern involves an input in which only one value of stress is used for the evaluation of the actual plant transients in the detailed stress analysis for the component. The RIS also discusses that the detailed stress analysis requires consideration of six stress components, as discussed in ASME Code, Section III, Subsection NB, Subarticle NB-3200. In comparison, LRA Section B.2.2.1 evaluates the operating experience related to the Fatigue Monitoring AMP. However, the operating experience discussion in LRA Section B.2.2.1 does not include the evaluation regarding RIS 2008-30. The staff requests clarification regarding whether the Fatigue Monitoring AMP does not have the concern addressed in RIS 2008-30.	1. Provide the operating experience evaluation regarding the potential concern addressed in RIS 2008-30 to confirm that the Fatigue Monitoring AMP does not have such a concern.

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section B.2.3.6 Thermal Aging Embrittlement of Cast Austenitic Stainless Steel

No.	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	Section B.2.3.6	B-52	LRA section refers to an outdated NUREG/CR-4513. It also states ferrite contents were calculated based on its guidance. The staff notes that NUREG/CR-4513, Revision 2 w/ Errata was published in March of 2021. Flaw Tolerance Evaluation performed by Westinghouse, “Flaw Tolerance Evaluation for Susceptible Reactor Coolant Loop Cast Austenitic Stainless Steel Piping Components in Comanche Peak Units 1 and 2 for 60-Year License Renewal,” refers to and uses NUREG/CR-4513, Revision 2 w/ Errata.	Is the program owner aware of NUREG/CR-4513, Revision 2 w/ Errata? Discuss whether the proposed program is also consistent with the guidance of NUREG/CR-4513, Revision 2 w/Errata.
2	Section B.2.3.6	B-52	Page17 of document LTA-SDA-20-087-NP, Revision 0, “Flaw Tolerance Evaluation for Susceptible Reactor Coolant Loop Cast Austenitic Stainless Steel Piping Components in Comanche Peak Units 1 and 2 for 60-Year License Renewal,” states the subject components (CF8A materials) contain less than 25% ferrite. The evaluation procedures used were based on Appendix C of ASME Section Xi,2007 edition with 2008 addenda which has a restriction of 20% ferrite content. In addition, two modes of failures were postulated – plastic collapse and unstable ductile tearing.	Discuss how acceptance criteria of different ferrite contents were reconciled. Discuss if the assumed mode of failure is conservative in the flaw tolerance evaluation.

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA AMR Items Cracking Due to Stress Corrosion Cracking

No.	LRA Section	LRA Page	Question / Issue	Why are we asking?
1	Section 3 Table 3.2-1 Item 3.2-1, 021	3.2-20	The applicant claimed that Generic Aging Lessons Learned (GALL) Report item 3.2-1, 021 is not applicable because the Comanche Peak safety injection (SI) accumulators are maintained at containment ambient conditions (<140 °F). However, the Updated Final Safety Analysis Report states that the accumulators are located inside containment, and are operated at temperature between 70 °F to 150 °F. Clarify inconsistency between the license renewal application and GALL Report.	According to 10 CFR 54.21 and GALL Report, accumulators exposed to treated water (borated) >140 °F is within scope of license renewal and subject to aging management review.

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA/SLRA TLAA Section 4.7.1 – Leak-Before-Break

Question Number	LRA/SLRA Section	LRA/SLRA Page	Background/Issue	Discussion Question/Request
1	4.7.1	4.711	NUREG-0797, Supplement 23, "Safety Evaluation Report related to the operation of Comanche Peak Steam Electric Station, Units 1 and 2"	In NUREG-0797, Supplement 23, "Safety Evaluation Report related to the operation of Comanche Peak Steam Electric Station, Units 1 and 2," the staff identified a concern related to potential thermal stratification in the pressurizer surge line during forced cooldown of the reactor coolant system (RCS) to effect repairs of leakage discovered in the surge line during normal plant operations. The staff's concern is that the combined thermal stresses associated with the forced cooldown process and the thermal stratification phenomenon could overstress the potentially weakened surge line implied by the presence of leakage in the line. In response to the staff's concerns, the applicant committed to revise the plant operating procedures to provide prompt depressurization in the event of a pressurizer surge line leak. Please provide verification that the plant operating procedures were revised and please provide any recent additional modifications to the plant procedures which address depressurization in the event of a pressurizer surge line leak.

2	4.7.1	4.7-1	NUREG-0797, Supplement 23, “Safety Evaluation Report related to the operation of Comanche Peak Steam Electric Station, Units 1 and 2”	In NUREG-0797, Supplement 23, “Safety Evaluation Report related to the operation of Comanche Peak Steam Electric Station, Units 1 and 2,” the applicant committed to monitor the temperature of the residual heat removal (RHR) suction lines for the potential for thermal fatigue. The applicant stated that, should thermal fatigue develop, the applicant would be alerted by fluctuating temperatures on the RHR lines. The staff concluded that the monitoring program will be adequate to confirm that the RHR suction lines will not fail as a result of thermal fatigue. The staff stated that the monitoring program is to be continued until sufficient data are developed, either through industry generic or plant-specific programs. In a letter dated August 9, 1989, the applicant agreed to obtain staff approval before terminating the monitoring program. Provide information whether the monitoring program is still being utilized, and if not, what other programs are being utilized to monitor the RHR suction lines for the potential for thermal fatigue.
3	4.7.1	4.7-1	Alloy 182/82 welds are susceptible to PWSCC.	In the time limited aging analysis (TLAA), it states Comanche Peak, Unit 2 has completed the partial implementation of a mechanical stress improvement process (MSIP), where the loop 4 cold leg and loop 2 hot leg nozzles have achieved the required compressive residual stresses on the inner surface to mitigate primary water stress corrosion cracking (PWSCC) concerns for Alloy 82/182 welds. Please

				provide a schedule for the completion of the MSIP for the remaining nozzle locations for Unit 2.
4	4.7.1	4.7-1	The applicant states that only a partial of the Alloy 182/82 welds have undergone a mitigation.	Please provide how the applicant is monitoring the unmitigated Alloy 82/182 welds via an aging management program (AMP), inservice inspection (ISI), walkdowns, etc.
5	4.7.1	4.7-2	The applicant states that DMW locations at the RPV nozzles were evaluated for LBB.	In the TLAA, the applicant states that dissimilar metal welds (DMW) locations at reactor pressure vessel (RPV) nozzles that have Alloy 82/182 weld materials were evaluated for Leak-Before-Break (LBB). Please provide a list of the DMW locations.
6	4.7.1	4.7-2, 4.7-3	Clarification is needed for the dispositions between the reactor coolant loop piping and the accumulator injection, RHR and pressurizer surge lines.	Please clarify why the dispositions for the reactor coolant primary loop piping LBB analysis is 10 CFR 54.21(c)(1)(ii) and the disposition for the accumulator injection lines, RHR lines and the pressurizer surge line piping is 10 CFR 54.21(c)(1)(i). The reviews which came to the conclusions both appear to be based on the projected 60-year cycles vs. the 40-year design cycles.

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section 4.3.6, "High-Energy Line Break Analyses"

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	4.3.6	4.3-22 4.3-23	LRA Section 4.3.6 addresses the high energy line break (HELB) analyses. The LRA section indicates that, in the HELB analyses, the time limited portion of the analysis is related to the screening criterion of a CUF value 0.1 for break location postulation. In comparison, Final Safety analysis Report (FSAR), Section 3.6B.2 describes the current licensing basis (CLB) screening criteria that are used to determine the intermediate locations of postulated breaks for the HELB analyses. Specifically, FSAR Section 3.6B2.1.2 indicates that the CUF value of 0.1 is included in the screening criteria for HELB location postulation for ASME Code Section III Class 1 piping. FSAR Section 3.6.B2.1.2 also indicates that the postulation of HELB locations for ASME Code non-Class 1 piping is, in part, based on the allowable stress range for expansion stress (S_A), consistent with Branch Technical Position MEB 3-1 (ADAMS Accession No. ML052340555). S_A may need to be adjusted by a stress range reduction factor that is determined by the number of thermal cycles, as addressed in the implicit fatigue analysis in LRA Section 4.3.3. However, LRA Section 4.3.6 does not clearly discuss whether the HELB location postulation for ASME Code non-Class 1 piping, which involves S_A, is a basis for identifying the HELB analysis as a time-limited aging analysis (TLAA).	1. Clarify whether the HELB location postulation for ASME Code non-Class 1 piping, which involves S_A, is one of the bases for identifying the HELB analysis as a TLAA. If not, explain why the HELB location postulation for ASME Code non-Class 1 piping is not a basis for identifying the HELB analysis as a TLAA. 2. Clarify whether additional break locations and their effects will be evaluated in the HELB analysis as part of corrective actions under the Fatigue Monitoring aging management program if new additional piping break locations are identified based on (1) the CUF threshold of 0.1 or (2) the reduction in S_A. If not, explain why such additional HELB locations do not need to be evaluated in the HELB analysis.

			In addition, LRA Section 4.3.6 does not address what corrective action will be taken if additional break locations are identified due to the increased fatigue cycles for ASME Code Class 1 and non-Class 1 piping in the HELB analysis during the period of extended of operation.	

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section 4.3.5, “Reactor Vessel Internals Fatigue Analyses”

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	4.3.5	4.3-21	LRA Section 4.3.5 addresses the fatigue time-limited aging analyses (TLAAs) for reactor vessel internal (RVI) components. The applicant indicated that the following reference for CPNPP stretch power uprate (SPU) includes the most recent fatigue evaluations in the current licensing basis (CLB) for RVI components (Reference: WCAP-16840-NP, Comanche Peak Nuclear Power Plant Stretch Power Uprate Licensing Report, Rev. 0, ADAMS Accession Nos. ML072490310 and ML072490358). The applicant also explained that the fatigue evaluation in the reference determined that the SPU did not affect the bounding cumulative usage factors (CUFs) for RVI components and, therefore, the CUF values continue to meet the design limit (not to exceed 1.0). In addition, the applicant indicated that the analyses performed for the RVI components are based upon the subset of the reactor coolant system design transients used in the fatigue analyses for the reactor vessel, which are shown in LRA Table 4.3.1-2. In comparison, the design transients used in the CLB CUF evaluations for RVI components are included in WCAP-16840-NP report, Table 2.2.6-1. In its review, the staff noted the potential inconsistency in the design transients between LRA Table 4.3.1-2 and WCAP-16840-NP, Table 2.2.6-1. Specifically, LRA Table 4.3.1-2 include the “bypass line tempering valve” transient, which is only applicable	1. Clarify whether the “bypass line tempering valve” transient of Unit 2 in LRA Table 4.3.1-2 is identical to the “split flow bypass valve” transient of Unit 2 in WCAP-16840-NP, Table 2.2.6-1. If so, explain why the design cycles for these transients are different (i.e., 20 cycles versus 40 cycles). 2. If the “bypass line tempering valve” transient of Unit 2 in LRA Table 4.3.1-2 is not identical to the “split flow bypass valve” transient of Unit 2 in WCAP-16840-NP, explain why this inconsistency in the design transients are acceptable for the fatigue TLAAs on ASME Code Class 1 piping systems, non-Class 1 piping systems and RVI components. As part of the response, clarify whether these valve transients are the transients that should be included in the fatigue analyses for RVI components.

			to CPNPP Unit 2 and has 20 design cycles. However, this transient is not included in WCAP-16840-NP, Table 2.2.6-1. In addition, WCAP-16840-NP, Table 2.2.6-1 includes the "split flow bypass valve" transient, which is only applicable to CPNPP, Unit 2 and has 40 design transients. However, this transient is not included in LRA Table 4.3.1-2. Therefore, the staff found a need to clarify the following items: (1) whether the "bypass line tempering valve" transient LRA Table 4.3.1-2 is identical to the "split flow bypass valve" transient in WCAP-16840-NP, Table 2.2.6-1 and, if so, why the design cycles for these transients are different (i.e., 20 cycles in LRA 4.3.1-2 and 40 cycles in WCAP-16840-NP, Table 2.2.6-1); and (2) if these transients are not identical, why the design transients in LRA Table 4.3.1-2 are not consistent with the existing design transients in WCAP-16840-NP.	
2	4.3.5	4.3-21	LRA Section 4.3.5 addresses the fatigue time-limited aging analyses (TLAAs) for reactor vessel internal (RVI) components. The applicant indicated that the following reference for CPNPP stretch power uprate includes the most recent fatigue evaluations in the current licensing basis (CLB) for RVI components (Reference: WCAP-16840-NP, Comanche Peak Nuclear Power Plant Stretch Power Uprate Licensing Report, Rev. 0, ADAMS Accession Nos. ML072490310 and ML072490358). Note (2) of WCAP-16840-NP, Table 2.2.3-6 indicates that the basis of the baffle-former bolt qualification is a fatigue test and the evaluation of the revised loads consisted of demonstrating that the loads associated with stretch power uprate are acceptable for the plant design life (i.e., 40 years). However, LRA Section 4.3.6 does not clearly discuss whether the baffle former bolts are also qualified for the extended period of operation (up to 60 years of operation). Therefore, the staff found a need to clarify whether the applicant has adequately evaluated	1. Clarify whether a time-limited assumption is involved in the fatigue evaluation for the RVI baffle former bolts that are related to fatigue test results. If so, discuss the fatigue TLAA on the baffle former bolts for the extended period of operation and its disposition in accordance with 10 CFR 54.21(c)(1)(i), (ii) or (iii).

			the potential time-limited assumption in the fatigue evaluation for the baffle former bolts.	

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section 4.3.4, "Environmentally Assisted Fatigue"

Question Number	SLRA Section	SLRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	4.3.4	4.3-14	LRA Section 4.3.4 addresses the environmentally assisted fatigue (EAF) analysis for ASME Code Section III Class 1 piping and components. LRA Section 4.3.4 indicates that, during the screening evaluation of the EAF analysis, the applicant adjusted the cumulative usage factor (CUF) values by any applicable factors to correct for differences between the fatigue curves used in the existing fatigue evaluation (e.g., Section III Appendix I of the ASME Code) and the fatigue curves applicable to the industry document, as required for environmental cumulative usage factor (CUF_{en}) calculations for license renewal. The existing fatigue analysis is also called source fatigue analysis. The staff found a need to clarify whether the more recent industry guidance for fatigue design curves is consistent with RG 1.207 that approves the use of NUREG/CR-6909, Revision 1 for EAF analysis. The staff also needs to confirm the adequacy of the adjustment factor used to adjust the existing fatigue design curve in consideration of the more recent guidance on fatigue design curves for each material (i.e., each of carbon steel, low-alloy steel, stainless steel and nickel alloy).	1. Clarify whether the more recent industry guidance for fatigue design curves mentioned as part of the screening evaluation is consistent with RG 1.207 that approves the use of NUREG/CR-6909, Revision 1 for EAF analysis. 2. Describe the adjustment factor that was used to adjust the existing fatigue design curve in consideration of the more recent guidance on fatigue design curves for each material (i.e., each of carbon steel, low-alloy steel, stainless steel and nickel alloy). In addition, explain the technical bases of the adjustment factors for the fatigue design curves and CUF_{en} calculations.

2	4.3.4	4.3-15	LRA Section 4.3.4 addresses the environmentally assisted fatigue (EAF) analysis for ASME Code Section III Class 1 piping and components. LRA Section 4.3.4 indicates that in the EAF screening evaluation the applicant used the maximum environmental fatigue correction factor (F_{en}). However, the section does not clearly describe how the applicant determined the maximum F_{en} values.	1. Explain how the applicant determined the maximum F_{en} values for the materials in the EAF screening evaluation. As part of the discussion, clarify whether the calculations of the maximum F_{en} values are consistent with the guidance in NUREG/CR-6909, Revision 1.
3	4.3.4	4.3-15 4.3-16	LRA Section 4.3.4 addresses the environmentally assisted fatigue (EAF) analysis for ASME Code Section III Class 1 piping and components. LRA Section 4.3.4 indicates that, in the further evaluation of EAF for sentinel (limiting) location identification, the applicant considered the technical rigor of different stress analysis methods and the level of conservatism associated with the stress analysis methods. The applicant explained that the results of determining the technical rigor and the associated conservatism are the stress analysis method rankings for EAF locations (also called stress basis comparison rankings). The applicant indicated that the lowest stress analysis method ranking involves the highest conservatism in the calculations of screening environmental cumulative usage factor (screening CUF_{en}) values among the stress analysis methods. The applicant also explained that EFA locations with the lower screening CUF_{en} values and lower rankings may be removed from the sentinel location list in comparison with the other EAF locations. However, the applicant did not clearly discuss the stress analysis method rankings and their technical bases. In addition, the staff found a need to clarify whether EAF locations are removed from the sentinel location list only if both the screening CUF_{en} value and stress analysis method	1. Explain the stress analysis method rankings and their technical bases for the sentinel location identification. 2. Clarify whether EAF locations, which were identified from the screening evaluation, are removed from the sentinel location list only if both the screening CUF_{en} value and stress analysis method ranking are lower than those of a more limiting location, respectively. If not, provide justification for why the EAF locations with the higher screening CUF_{en} or higher stress analysis method ranking are removed from the sentinel location list.

			ranking are lower than those of a more limiting location, respectively.	
4	4.3.4	4.3-16	LRA Section 4.3.4 addresses the environmentally assisted fatigue (EAF) analysis for ASME Code Section III Class 1 piping and components. LRA Section 4.3.4 indicates that, in the further evaluation of EAF for sentinel (limiting) location identification, the applicant compared sentinel locations of different transient sections within common systems or equipment to determine one or two sentinel locations per system or equipment. The applicant defined a transient section as a group of sub-components and locations that experience the same transients (i.e., thermal, and related loadings). The staff found a need to clarify the following items related to the sentinel location identification: (1) which transient sections do not identify a sentinel location; (2) which transition sections are bounding for the transition sections that do not identify a sentinel location; and (3) how the applicant determined that the other transition sections are sufficiently bounding for the transition sections that do not identify a sentinel location given that different transition sections experience different transients.	1. Clarify the following items related to the sentinel location identification: (1) which transient sections do not identify a sentinel location; (2) which transition sections are bounding for the transition sections that do not identify a sentinel location; and (3) how the applicant determined that the other transition sections and their sentinel locations are sufficiently bounding for the transition sections that do not identify a sentinel location and their maximum screening CUFen. 2. In addition, explain how other transition sections and their sentinel locations can be bounding for certain transition sections and their leading EAF locations even though different transition sections experience different transients. As part of the discussion, clarify whether the fatigue monitoring activities provide reasonable assurance that the CUF_{en} values of a transition section bounded by another transition section do not exceed the design limit (1.0).

5	4.3.4	4.3-17	LRA Section 4.3.4 addresses the environmentally assisted fatigue (EAF) analysis for ASME Code Section III Class 1 piping and components. LRA Section 4.3.4 describes the EAF evaluation of the sentinel locations (limiting locations) that were determined in the screening evaluation. The applicant explained that, for some EAF locations, minor conservatisms in the existing fatigue analysis were removed through approaches such as stress algorithm refinements and use of more appropriate stress concentration factors. This group of EAF locations, which reduced minor conservatism, is also called the simplified evaluation group in the LRA section. However, LRA Section 4.3.4 does not clearly describe the stress algorithm refinements and the more appropriate stress concentration factors, which were used to reduce the conservatisms in the existing CUF calculations.	1. Discuss in more detail the stress algorithm refinements and the more appropriate stress concentration factors, which were used to reduce the conservatisms in the existing CUF calculations. As part of the responses, describe the Code provisions or guidance documents used in the refinements of CUF values for license renewal.

6	4.3.4	4.3-19 4.3-20	LRA Section 4.3.4 addresses the environmentally assisted fatigue (EAF) analysis for ASME Code Section III Class 1 piping and components. LRA Tables 4.3.4-1 and 4.3.4-2 describe the sentinel locations (limiting locations) for the Class 1 components and piping lines, respectively, based on EAF screening evaluation. In addition to the screening evaluation results, these LRA tables provide the 60-year projected environmental cumulative usage factor (CUF_{en}) values for the generic sentinel (limiting) locations described in NUREG/CR-6260, which are applicable to the Comanche Peak Nuclear Power Plant. However, LRA Tables 4.3.4-1 and 4.3.4-2 do not provide 60-year projected CUF_{en} values for the plant-specific sentinel locations determined in the screening evaluation (e.g., control rod drive mechanism housings and reactor vessel bottom mounted instrumentation tubes). These plant-specific locations are additional sentinel locations that may be more limiting than the generic sentinel locations described in NUREG/CR-6260. LRA Section 4.3.4 indicates that the detailed calculations of CUF_{en} values for these plant-specific sentinel locations will be performed as part of the Fatigue Monitoring aging management program (AMP). Specifically, LRA Tables 4.3.4-1 and 4.3.4-2 explain that, for the plant-specific sentinel locations associated with note (b) of the tables, the CUF_{en} values evaluation will be calculated as part of the Fatigue Monitoring AMP (LRA Section B.2.2.1). However, LRA Section 4.3.4 does not clearly discuss whether the applicant will submit the 60-year projected CUF_{en} values of the sentinel locations, which will be calculated as part of the Fatigue Monitoring AMP, for staff's review and approval a certain time period prior to entering	1. Clarify whether the applicant will submit the 60-year projected CUF_{en} values of the plant-specific sentinel locations, which will be calculated as part of the Fatigue Monitoring AMP, for staff's review and approval a certain time period prior to entering the period of extended operation (e.g., at least one year prior to entering the period of extended operation). If so, revise the LRA accordingly. If not, provide justification for why such a submittal is not necessary to confirm that the Fatigue Monitoring AMP and the associated CUF_{en} calculations will effectively manage the aging effect of EAF for the plant-specific sentinel locations without additional aging management activities (e.g., flaw tolerance evaluations). 2. Clarify how the Steam Generators AMP will manage the aging effect of EAF for the steam generator components other than steam generator tubes, which are listed in LRA 4.3.4-1. As part of the response, describe the aging management activities (e.g., inspection activities) for the non-tube components of steam generators and why these activities are sufficient for the aging management of the steam generator components other than tubes without CUF_{en} calculations.

			the period of extended operation (e.g., at least one year prior to entering the period of extended operation). The staff views that such a submittal for staff's review is necessary to confirm that the Fatigue Monitoring AMP and the associated CUF_{en} calculations will effectively manage the aging effect of EAF for the plant-specific sentinel locations without additional aging management activities (e.g., flaw tolerance evaluations). In addition, LRA Table 4.3.4-1 identifies plant-specific sentinel locations of steam generator components, which are associated with note (c) of the table (e.g., steam generator tubes, tube plate/lower shell location (Unit 1 only) and tubesheet primary side junction of lower shell channel head (Unit 2 only)). Note (c) of LRA Table 4.3.4-1 indicates that the applicant will use the Steam Generators AMP to manage the aging effect of EAF for these steam generator components. The staff found a need to clarify how the Steam Generators AMP will manage the aging effect of EAF for the steam generator components other than steam generator tubes, which are listed in LRA 4.3.4-1.	

7	4.3.4	4.3-20	LRA Table 4.3.4-2 describes the sentinel (limiting) locations of EAF for piping lines and the associated 60-year projected CUF_{en} values. The table identifies only stainless steel as a fabrication material of piping lines and does not include carbon steel, low alloy steel or nickel alloy. In comparison, LRA 4.3.4 indicates that the sentinel location is identified for each material type in a given transient section, which is a group of sub-components and locations that experience the same transients (thermal and related loadings). Therefore, the staff found a need to clarify why LRA Table 4.3.4-2 does not include a sentinel location of piping fabricated with carbon steel, low alloy steel or nickel alloy.	1. Explain why LRA Table 4.3.4-2 includes sentinel locations of piping only fabricated with stainless steel. If the applicant determined that the stainless steel piping locations bound the piping locations fabricated with low alloy steel, carbon steel or nickel alloy in a transition section, provide the basis of the applicant's determination on the bounding nature of the stainless steel locations (e.g., comparisons of F_{en} and CUF_{en} values between the different materials to determine the bounding location and material).

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section 4.3.3, "ASME Section III, Class 2 and 3 Allowable Stress Analyses"

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	4.3.3	4.3-10 4.3-8	LRA Section 4.3.3 addresses the allowable stress analyses for the non-Class 1 piping systems that were designed in accordance with the requirements of ASME Code, Section III (Class 2 and 3) and ANSI B31.1 Code. LRA Section 4.3.3 indicates that LRA Table 4.3.2-4 demonstrates that the 60-year projected transient cycles for these piping systems do not exceed 7000 cycles specified in ASME Code, Section III and, therefore, a stress range reduction factor of 1 (i.e., no reduction) is applied to the allowable stresses in the stress analyses. However, LRA Table 4.3.2-4 does not clearly describe the following items related to the total cycle projections for each piping system or line: (1) which specific design transients were considered in the total cycle projections; (2) 60-year projected cycles of the specific transients; and (3) technical bases for how the projected cycles were determined (e.g., piping system design information, plant operation procedures, test requirements, UFSAR information and specific system-level knowledge). LRA Section 4.3.3 also describes the total 60-year projected transient cycles for the reactor coolant system of Comanche Peak Nuclear Power Plant (CPNPP) Units 1 and 2 (i.e., 3926 and 4081 cycles, respectively). However the LRA does not clearly describe how these total cycles were determined. In addition, the staff found a need to clarify which piping systems	1. Provide the following information related to the total cycle projections for each piping system or line described in LRA Table 4.3.2-4: (1) which specific design transients were considered in the total cycle projections; (2) 60-year projected cycles of the specific design transients; and (3) technical bases for how the projected cycles were determined (e.g., piping system design information, plant operation procedures, test requirements, UFSAR information and specific system-level knowledge). 2. Describe how the applicant estimated the total 60-year projected transient cycles for the reactor coolant system of CPNPP Units 1 and 2 (i.e., 3926 and 4081 cycles, respectively). In addition, clarify which piping systems and lines in LRA Table 4.3.2-4 are subject to these RCS transient cycles.

			and lines in LRA Table 4.3.2-4 are subject to these reactor coolant system (RCS) transient cycles.	

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section 4.3.2, “Metal Fatigue of Class 1 Components”

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	4.3.2	4.3-9	LRA Section 4.3.2 addresses the fatigue time-limited aging analysis (TLAA) for ASME Code Section III, Class 1 piping and components. The section indicates that the reactor coolant pumps conform to the waiver of fatigue requirements of ASME Code, Section III and therefore do not require a detailed fatigue evaluation. However, the LRA does not describe specific code provisions that the applicant used in the fatigue waiver evaluation and the reference document that contains the fatigue wavier evaluation.	1. Discuss the specific code provisions that the applicant used in the fatigue waiver evaluation and the reference document that contains the fatigue wavier evaluation.

2	4.3.2	4.3-9	LRA Section 4.3.2 addresses the fatigue time-limited aging analysis (TLAA) for ASME Code Section III, Class 1 piping and components. During its review of operating experience related to fatigue analyses, the staff noted that the following reference indicates that some of the reactor vessel closure stud and hole threads were damaged during the plant operation (Reference: CR-2014-008181, "Several Reactor Vessel Closure Stud/hole Threads Have Incurred Damage Over the Life of the Plant Resulting in Stud/hole Thread Loss," July 16, 2014). Specifically, the reference indicates that the most limiting thread loss occurred in the Unit 1 reactor vessel closure hole number 25. The reference also identifies a need to evaluate the fatigue life of the most limiting stud/hole if the fatigue life is affected. However, LRA Section 4.3.2 does not discuss the effect of the thread loss on the existing fatigue analysis of the Unit 1 reactor vessel closure hole number 25 (limiting hole) and the associated stud.	1. Discuss the effect of the thread loss on the existing fatigue analysis of the most limiting stud and hole (i.e., Unit 1 reactor vessel closure stud and hole number 25). As part of the discussion, clarify whether the 60-year projected cumulative usage factor (CUF) for the most limiting stud and hole is less than the design limit (1.0). If not, discuss how the applicant will manage the effect of fatigue for the studs and holes that may have a 60-year projected CUF value exceeding the design limit.

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section 4.3.1, "Transient Cycle Projections for 60 Years"

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	4.3.1	4.3-8	LRA Section 4.3.1 addresses the design transients and their 60-year cycle projections. Specifically, LRA Table 4.3.1-4, describes the projected cycles related to the auxiliary piping systems (e.g., residual heat removal and accumulator piping systems) connected to the reactor coolant system. In addition, the table addresses the transient cycles for other ASME Code Section III non-Class 1 and ANSI B31.1 piping systems (e.g., process sampling and liquid waste processing piping systems that are not associated with the reactor coolant system or auxiliar piping lines connected to the reactor coolant system). However, the title of LRA Table 4.3.1-4, "CPNPP 60-year Projected Transient Cycles For Auxiliary System Transients and Applicable Components," does not clearly reflect that the table includes the transient cycle projections for ASME Code non-Class 1 and ANSI B31.1 piping systems other than the auxiliary piping systems. Therefore, the staff found a need to clarify whether LRA Table 4.3.1-4 describes the design transients and their 60-year projected cycles for the auxiliary piping systems and the other ASME Code Section III non-	1. Clarify whether LRA Table 4.3.1-4 describes the design transient cycles and their 60-year projected cycles for the auxiliary piping systems and the other ASME Code Section III non-Class 1 and ANSI B31.1 piping systems, which are not included in Class 1 piping systems. If not, provide justification for why the table does not fully address the cycle projections for ASME Code Section III non-Class 1 and ANSI B31.1 piping systems other than the auxiliary piping systems. 2. Clarify whether LRA Table 4.3.1-4 describes the non-Class 1 piping systems/lines and the total transient cycles for each of the non-Class 1 piping systems/lines. As part of the discussion, clarify the relationship between LRA Tables 4.3.1-3 and 4.3.1-4.

			Class 1 and ANSI B31.1 piping systems, which are not included in the Class 1 piping systems. In addition, it appears that LRA Table 4.3.1-4 mainly describes the non-Class 1 piping systems/lines and the total combined projected transient cycles for each of the non-Class 1 piping systems/lines (compared to a total of 7000 cycle limit). This aspect is not clearly discussed in LRA Section 4.3.1 and the first column of LRA Table 4.3.1-4 has a column description of "Transient" rather than "Systems/Lines." The staff found a need to clarify this aspect of LRA Table 4.3.1-4.	
2	4.3.1	4.3-7	LRA Section 4.3.1 addresses the design transients and their 60-year cycle projections. Specifically, the title of LRA Table 4.3.1-3, "CPNPP 60-year Transients Normal Condition Auxiliary System Transient Events," indicates that the table describes the normal condition transients and their 60-year cycle projections. Accordingly, the title of LRA Table 4.3.1-3 suggests that the table does not address the 60-year transient cycles projections for the transients of the upset or test conditions. Therefore, the staff found a need to clarify whether LRA Table 4.3.1-3 includes the design transients of the upset and test conditions and their 60-year cycle projections. In addition, the staff need to clarify whether LRA Table 4.3.1-3 includes the design transients for the ASME Code Section III non-Class 1 piping and ANSI B31.1 piping other than the auxiliary piping connected to the reactor coolant system (e.g., process sampling related transient).	1. Clarify whether LRA Table 4.3.1-3 includes the design transients of the upset and test conditions and their 60-year cycle projections. If not, explain why the table does not address the design transients of the upset and test conditions and their 60-year cycle projections. 2. Clarify whether LRA Table 4.3.1-3 includes the design transients for the ASME Code Section III non-Class 1 piping and ANSI B31.1 piping other than the auxiliary piping connected to the reactor coolant system (e.g., process sampling related transient). If not, explain why the table does not address the design transients for the non-Class 1 piping and ANSI B31.1 piping other than the auxiliary piping connected to the reactor coolant system.

3	4.3.1	4.3-7	LRA Section 4.3.1 addresses the design transients and their 60-year cycle projections. Specifically, LRA Table 4.3.1-3 addresses the design transients and their cycle projections for non-Class 1 piping systems. The staff also noted that the following reference describes the 60-year projected cycles for non-Class 1 piping systems (Reference: LTR-SDA-II-21-32-P, Comanche Peak Unit 1 and 2 License Renewal: Class 2 and 3 Piping Fatigue Evaluation). The staff further noted that, for the following transients, the 60-year projected cycles are inconsistent between the reference above and LRA Table 4.3.1-3: (1) “charging flow step decrease and return to normal” transient; (2) “charging flow step increase and return to normal” transient; (3) “letdown flow step decrease and return to normal” transient; and (4) “letdown flow step increase and return to normal” transient. For example, LRA Table 4.3.1-3 estimates a 60-year projected cycle number of 4373 for the “charging flow step decrease and return to normal” transient, which is significantly higher than that estimated in the LTR-SDA-II-21-32-P report. In addition, the LTR-SDA-II-21-32-P report does not identify “charging/letdown cooldown” and “charging/letdown heatup” transients as design transients in contrast with LRA Table 4.3.1-3 that identifies those transients as design transient and describes their 60-year projected cycles. The staff found a need to resolve the potential inconsistencies discussed above regarding the projected cycles numbers and the list of design transients for non-Class 1 piping systems.	1. Explain why the 60-year projected cycles are inconsistent between the LTR-SDA-II-21-32-P report and LRA Table 4.3.1-3 for the following transients: 1) “charging flow step decrease and return to normal” transient; (2) “charging flow step increase and return to normal” transient; (3) “letdown flow step decrease and return to normal” transient; and (4) “letdown flow step increase and return to normal” transient. As part of the discussion, clarify whether LRA Table 4.3.1-3 provides the adequate projected cycles for these transients. 2. For the “charging/letdown cooldown” and “charging/letdown heatup” transients, describe (a) the technical basis for identifying these transients as design transients for non-Class 1 piping systems in LRA Table 4.3.1-3 and (b) which system/line transients in LRA Table 4.3.1-4 include these charging/letdown design transients.

4	4.3.1	4.3-7 4.3-8	LRA Section 4.3.1 addresses the design transients and their 60-year cycle projections. Specifically, LRA Table 4.3.1-2 and 4.3.1-3 describe the design transients and their cycle projections for the reactor coolant system and non-class 1 piping systems, respectively. In addition, LRA Table 4.3.1-4 addresses the piping system or line transients and their 60-year projected cycles for the non-Class 1 piping. The projected cycle number of a piping system/line transient may be determined by combining the cycles of specific design transients that are applicable to the piping system or line. However, LRA Table 4.3.1-4 does not clearly describe which specific design transients and their projected cycles, including the design transients and cycles in LRA Tables 4.3.1-2 and 4.3.1-3, are considered in the determination of the total combined projected cycles for each piping system or line. In addition, LRA Section 4.3.1 indicates that cycle counting is not performed on some design transients in the Fatigue Monitoring aging management program (AMP). However, LRA Section 4.3.1, including LRA Tables 4.3.1-2, 4.3.1-3 and 4.3.1-4, does not clearly describe which design transients are not counted in the Fatigue Monitoring AMP and the basis for the absence of cycle counting (e.g., the design cycle number has a significant cycle margin compared to the 60-year projected cycles or the transient causes a negligible effect on fatigue).	1. For each piping system or line listed in LRA Table 4.3.1-4, clarify which specific design transients and their 60-year projected cycles, including the design transients and projected cycles in LRA Table 4.3.1-3, are considered in the determination of the total projected cycles for the piping system or line. 2. Describe which design transients in LRA Section 4.3.1, including LRA Tables 4.3.1-2, 4.3.1-3 and 4.3.1-4, are not counted in the Fatigue Monitoring AMP and the basis for the absence of cycle counting (e.g., the design cycle number has a significant cycle margin compared to the 60-year projected cycles or the transient does not cause an effect on fatigue).

5	4.3.1	4.3-6	LRA Table 4.3.1-2 indicates that the 60-year projected cycles for the following test transients are determined as the same with the current licensing basis design cycles: (1) primary side hydrotest transient; (2) secondary side hydrotest transient; (3) primary side leak test transient; (4) secondary side leak test transient; and (5) boron injection tank (BIT) check valve test transient (applicable only to Unit 2). Note (d) of LRA Table 4.3.1-2 also indicates that the design cycles of these test condition transients are used as the 60- year projected cycles without adjustment. However, the staff found a need to further clarify the basis for the approach used in note (d) of LRA Table 4.3.1-2. Specifically, the staff found a need to clarify whether the actual cycles of these test transients are significantly less than the 60-year projected cycles.	1. Describe the basis for the approach used in note (d) of LRA Table 4.3.1-2 (i.e., the design cycles of the test transients can be used as the 60-year projected cycles without adjustment). As part of the response, clarify whether the actual cycles of these test transients are significantly less than the 60-year projected cycles.
6	4.3.1	4.3-3	LRA Section 4.3.1 addresses the design transients and their 60-year cycle projections. The LRA section explains that the 40-year design transients bound the number of cycles projected to occur during 60 years of operation except for Unit 1 “letdown flow shutoff with prompt return to service” transient. LRA Table 4.3.1-3 describes a 60-year projected cycle number of 233, which is greater than a design cycle number of 200 for the transient. LRA Section 4.3.1 also explains that the Unit 1 “letdown flow shutoff with prompt return to service” transient was evaluated in the environmentally assisted fatigue (EAF) analysis related to the limiting (sentinel) locations defined in NUREG/CR-6260. The staff noted that the EAF analysis in LRA Section 4.3.4 and Table 4.3.4-2 indicate that the charging nozzle is the limiting EAF location for the Unit 1 charging system and that the 60-year projected	1. Clarify why the Unit 1 “letdown flow shutoff with prompt return to service” transient, which is identified as a design transient for non-Class 1 piping systems in LRA Table 4.3.1-3, and its cycles are used as input to the EAF analysis that mainly addresses the environmental effect on the metal fatigue in Class 1 piping systems and components. 2. Considering that the number of 60-year projected cycles for the Unit 1 “letdown flow shutoff with prompt return to service” transient is greater than the number of design cycles for the transient, clarify the following: (1) whether the Unit 1 charging nozzle is the most limiting EAF location among the Unit 1 piping, component and support locations that are subject to the transient; and (2) whether the 60-year projected cumulative usage factor

			environmental cumulative usage factor (CUF_{en}) for the location is less than the design limit (1.0). However, it is not clear to the staff why the Unit 1 “letdown flow shutoff with prompt return to service” transient, which is identified as a design transient for non-Class 1 piping systems in LRA Table 4.3.1-3, and its cycles are used as input to the EAF analysis that mainly addresses the environmental effect on the metal fatigue in Class 1 piping systems (i.e., reactor coolant pressure boundary piping and components). In addition, the staff found a need to clarify the following: (1) whether the Unit 1 charging nozzle is the most limiting EAF location among the Unit 1 piping and component locations that are subject to the Unit 1 “letdown flow shutoff with prompt return to service” transient; and (2) whether the 60-year projected cumulative usage factor (CUF) values of the Unit 1 piping, component and support locations subject to the Unit 1 “letdown flow shutoff with prompt return to service” transient do not exceed the design limit.	(CUF) values of the Unit 1 piping, component and support locations (e.g., pressurizer support) subject to the transient do not exceed the design limit (1.0). 3. In addition, clarify whether the cycles of the Unit 1 “letdown flow shutoff with prompt return to service” transient will be monitored to ensure that the CUF and CUF_{en} values associated with this transient do not exceed the design limit (1.0) for the period of extended operation.
7	4.3.1	4.3-5	LRA Table 4.3.1-2 include design transients related to plant loading and unloading (e.g., “plant loading between 0 and 15 percent of full power” and “plant unloading between 0 and 15 percent of full power” transients).	1. As baseline information on operating conditions related to fatigue analyses, discuss the following items: (1) whether CPNPP, Units 1 and 2 have been always operated a base load unit and (2) whether the applicant has a plan to operate its reactor units as a load-following unit.

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components

Question Number	LRA/SLRA Section	LRA/SLRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	XI.M11B 4.3.2 – 4.5.2, 4.10.2, Appendix B	9-12, 15-17, 30-33	The requirements to impose N-729, N-722, and N-770 as cited in the Aging Management Program (AMP) are under 10 CFR 50.55a(g)(6)(ii)(D), (E), and (F) and its conditions, respectively.	Confirm AMP cites the regulations under 10 CFR50.55a(g)(6)(ii)(D), (E) and (F) with the required conditions at each time the applicant cites N-729, N-722, and N-770 in the current version of the AMP.
2	XI.M11B 4.10	10 of 28	Regulatory Issue Summary (RIS) 2018-06 was issued on December 10, 2018, which clarifies the inspection requirements for reactor pressure vessel upper head bare metal visual examinations	Confirm the review of operating experience for RIS 2018-06.
3	XI.M11B 4.10	10 of 28	Materials Reliability Program (MRP) 384, “Guideline for Nondestructive Examination of Reactor Vessel Upper Head Penetrations”, provides nuclear power plant owners with guidance for planning and executing reactor vessel upper head (RVUH) penetration examinations in a manner that will minimize the likelihood of human errors and maximize the probability of success. MRP-384 stated the following items under Table 7.1:	Confirm the review of operating experience for MRP-384.

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Table 7-1
Recommendations
<table>
<tr><th>Item</th><th>Implementation Category</th><th>NEI 03-08 Requirement</th></tr>
<tr><td>1</td><td>Needed</td><td>The utility shall develop and implement
RVUH examination oversight process in
accordance with the requirements in Se</td></tr>
<tr><td>2</td><td>Needed</td><td>Oversight personnel and examination w
data analysts shall receive training in
accordance with the requirements of Se
5.1 and 5.2.</td></tr>
<tr><td>3</td><td>Good Practice</td><td>The utility should develop a process tha
provides a work environment that minim
distractions for the data acquisition and
personnel in accordance with Section 6</td></tr>
<tr><td>4</td><td>Needed</td><td>The utility shall require independent and
the NDE data by at least two data anal
working separately and independently
accordance with the requirements of Se
6.2.2.</td></tr>
<tr><td>5</td><td>Needed</td><td>RVUH examination data shall be evalua
comparing the current outage examinat
to past examination data in accordance
the requirements of Section 6.2.3.</td></tr>
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<td>4</td>
<td>XI.M11B

4.4.2</td>
<td>8 of 28</td>
<td>In the St. Lucie Extended Period of Operation (EPO) request, the applicant identified full structural dissimilar metal butt welds required to be inspected under 10 CFR 50.55a(g)(6)(ii)(F), which required a time limiting aging analysis (TLAA 4-7-8) due to potentially triggering additional inspections in accordance with Note 10 of Table 1 of N-770-5, “Those welds not included in the 25% sample shall be</td>
<td>Confirm that a TLAA is not necessary for your full structural overlayed dissimilar metal butt welds.</td>
</tr>
</table>

			examined prior to the end of the mitigation evaluation period if the plant is to be operated beyond that time.”	
5	XI.M11B 4.2.2	8 of 28	The licensee identified a Mechanical Stress improvement Process (MSIP™) equipment failure which caused a the MSIP™ for the Unit 2 nozzle hot and cold leg dissimilar metal welds only being partially completed in 2RF19.	Confirm that a TLAA is not necessary to address this MSIP issue.
6	XI.M11B 4.4.2	8 of 28	In response to item 4 above, the applicant noted that LRA Table 3.1.2-3 includes a line item for cumulative fatigue damage of the “Pressurizer DMW (SWOL)” which credits TLAA discussed in LRA Section 4.3. Further the applicant notes that the LBB calculational methodology is applicable for the CPNPP and has been utilized for the LBB analysis for Alloy 82/182 welds with SWOL for the pressurizer surge line nozzle. Most significantly, the applicant explains that the acceptance criterion for the pressurizer SWOL is to verify that an unidentified crack will not propagate to the SWOL interface during a 10-year ISI interval. Since the crack is not qualified for the life of the plant, but only the inspection interval, the fatigue crack growth analysis is not a TLAA. Therefore, a TLAA pursuant to N-770-5 Note 10 is not applicable for CPNPP. The NRC staff have reviewed the licensee’s response and discussion during the 3/16 breakout session. Under Section 4.0 Results, of LTR-SDA-11-20-19, a similar discussion to the position above is provided as the applicant’s review of a need for the TLAA for SWOL welds. However, this review is limited only to the fatigue growth mechanism. As noted in Table 3-2 of LTR-SDA-11-20-19, “Flaw growth due to stress corrosion cracking” also potentially applies to CPNPP. Note 10 of Table 1 of N-770-5 establishes an inspection requirement as an augmented examination after the application of the SWOL at CPNPP. This requirement is	Given the inservice inspection area of the SWOL is limited to the outer 25% of the susceptible material of the original weld, and the original weld remains susceptible to PWSCC initiation and growth, is the applicant’s AMP, and current methodology to address the inspection frequency requirement of Note 10 of ASME Code Case N-770-5, adequate for the period of extended operation?

			based on a calculation to determine the mitigation evaluation period to determine if the plant is to be operated beyond that time. This calculation includes the degradation mechanism of primary water stress corrosion cracking (PWSCC). Additionally in Section 5.0, "Conclusion" of the NRC safety evaluation authorizing the applicant's request for the installation of the SWOL authorized the proposed modifications for the remaining service life of the subject welds. (ADAMS Accession No. ML072270704).	

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

One -Time Inspection of ASME Code Class 1 Small-Bore Piping Management Program (AMP)

Question Number	LRA Section	LRA Pages	Background / Issue	Discussion Question / Request
1	B.2.3.31	B-137 & B-139	LRA states that CPNPP 1 and 2 have not experienced a failure in its ASME Code Class 1 Piping in 32 and 29 years of operation. Therefore, the one-time program will inspect a sample of a minimum of 3% of the weld population or a maximum of 10 welds for each type for each unit. LRA further states, "Review of plant-specific OE indicates that cracking of ASME Code Class 1 piping has not occurred at CPNPP." A similar statement is made in the program basis document LUM00020-REPT-058. Additionally, staff's review of plant-specific OE confirmed that there do not appear to be any instances when cracking was identified on ASME Code Class 1 piping for either of the CPNPP 1 and 2.	Staff's understanding is that the statements in the LRA and the program basis document are applicable to all ASME Code Class 1 piping at CPNPP 1 and 2, and not just the ASME Class 1 small-bore piping. Please confirm if the above is applicable to all ASME Code Class 1 piping.
2			NRC issued Information Notice (IN) 2007-21, Supplement 1 (ADAMS Accession ML20225A204), "Pipe Wear due to Interaction of Flow-Induced Vibration and Reflective Metal Insulation," to alert licensees to recent industry operating experience regarding potential abrasive wear of ASME Code Class 1 and 2 pipes caused by flow-induced vibration and reflective metal insulation conditions. It appears that at least one of the instances of pipe wear from IN 2007-21, Supplement 1 was captured in "Signed 1RF21 EOC21 Report." Specifically, external OE-466059 (page 7) discussed degraded reactor coolant system piping due to mirror insulation fretting and relayed the information to be shared by the site's	If this IN has not been reviewed for applicability to CPNPP, confirm when and what actions the applicant is planning to take to ensure this aging effect will be managed.

			insulators that handle removal of insulation for flow accelerated corrosion-related inspections.	

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section AMP B.2.3.27: Buried and Underground Piping and Tanks

Question Number	**LRA Section**	**LRA Page**	**Background / Issue** **(As applicable/needed)**	**Discussion Question / Request**
1	B.2.3.27	B-158	LRA Section B.2.3.27, "Buried and Underground Piping and Tanks," states "[p]eriodic inspections will also be performed based on plant OE [operating experience] and the performance of the plant cathodic protection system."	GALL Report Table XI.M412, "Inspection of Buried and Underground Piping and Tanks," also considers coatings, backfill, and soil corrosivity when determining the number of periodic inspections. It is unclear to the staff if these factors will also be considered (in addition to OE and cathodic protection system performance).
2	B.2.3.27	B-158	LRA Section B.2.3.27 states "[f]or steel components, the acceptance criteria for the effectiveness of the cathodic protection is less than or equal to -850 mV." A similar statement is also included in the FSAR supplement. The staff reviewed MSE-P0-1328, "Cathodic Protection Annual Survey," and noted coppercopper sulfate reference electrode voltage should be less than or equal to -0.85 volts instant off (for 1E piping) or -0.85 volts on (for non-1E piping). The staff reviewed EPG-9.03, "Underground Pipe and Tank Program," and noted the cathodic protection systems acceptance criteria is -850 mV instant-off or the 100 mV shift.	The staff requests a discussion with respect to cathodic protection acceptance criteria. GALL Report AMP XI.M41 recommends -850 mV instantoff. The staff does not agree with the use of a potential of –850 mV instant-on without measurement or calculation of voltage drops. In addition, the staff disagrees with the use of the 100 mV cathodic polarization acceptance criterion (in the mixed metal environment) without confirmatory testing to verify that all metals are adequately protected.
3	N/A	N/A	GALL Report AMP XI.M41 recommends a cathodic protection critical potential of 1,200 mV to prevent damage to coatings or base metals.	The staff requests a discussion with respect to cathodic protection critical potentials. The staff could not identify this recommendation in MSE-

				P0-1328, “Cathodic Protection Annual Survey,” or in any of the enhancements.
4	B.2.3.27	B-159	Exception No. 1 states the following (in part): • The DGFOSTs [diesel generator fuel oil storage tanks] are inspected internally every 20 years through visual inspection and ultrasonic thickness measurements. The current wall thickness is evaluated for acceptability of the expected wall thickness at the next scheduled inspection based upon historical corrosion rates. • The exteriors of the tanks are coated. • Recent cathodic protection performance (within the 10-yr period prior to the PEO) was reviewed based on pipe to soil potentials taken during annual surveys, which indicates the FOSTs are being satisfactorily protected by the system. • Soil corrosivity samples around the site were taken in 2010. The soil sample analysis indicated the corrosion potential of buried systems was mitigated to minimal levels by sufficient cathodic protection. • Prior to being placed in service, each EDG fuel oil storage tank was inspected, and UT readings were taken. The UT readings were taken using a 42-point gridded inspection plan. Re-inspections, on a 10-year interval after being place in service, found the tanks with satisfactory results.	The staff requests a discussion on the exception with a focus on the following areas: • Type(s) of external coatings used for the tanks. • Results of soil corrosivity testing conducted in 2010. • Whether the proposed inspection frequency would be dependent on the future performance of the cathodic protection system. • UT inspection results.
5	B.2.3.27	B-164	OE Example No. 2 states the following (in part): “In April of 2015, an opportunistic inspection was performed during the excavation process for a potable water leak which exposed Fire Protection piping. The inspection noted coating (coal tar wrap) damage. The inspection noted	The staff requests a discussion with respect to if other inspections of inscope buried piping have noted external coating damage (or if this is an isolated issue).

			construction dunnage still in place under the pipe, which is not in compliance with the requirements of 2323-SS-008, CPSES Excavation and Backfill Specification."	
6	B.2.3.27	B-164	OE Example No. 3 states the following (in part): "The 2018 and 2019 health reports identified some issues with potable water leaks."	The staff requests a discussion with respect to (a) if the leaks were due to age-related degradation; and (b) if piping materials used in the potable water system are representative of in-scope buried piping.
7	B.2.3.27	B-160 B-161	LRA Section B.2.3.27 includes the following enhancements related to monitoring fire pump activity. • Revise procedures to trend the fire pump activity (or similar parameter) to identify concerns with buried fire water yard loop header leakage. • Revise acceptance criteria to ensure there is no evidence that backfill caused damage to the respective component coatings or the surface of the component (if not coated), and changes in fire pump activity (or similar parameter) that cannot be attributed to causes other than leakage from buried piping are not occurring. • Revise procedure to state when using the option of monitoring the activity of a fire pump instead of inspecting buried fire water system piping, a flow test or system leak rate test is conducted by the end of the next refueling outage or as directed by the current licensing basis, whichever is shorter, when unexplained changes in fire pump activity (or equivalent equipment or parameter) are observed. LRA Section 2.3.3.7, "Fire Protection System," states "[w]ater is supplied to the underground fire loop by the lead pump, the electric motor-driven pump, when the	The staff requests a discussion with respect to which fire pumps the enhancements are referring to. GALL Report AMP XI.M41 recommends monitoring jockey pump activity.

			jockey pump cannot maintain the system pressure above a predetermined set point.”	
8	N/A	N/A	The staff reviewed CR-2015-010120 and noted the piping vault for the B train station service water system was inspected and the general condition of the area consisted of a confined space with approximately 0.5 to 0.75 inches of water on the floor. Comanche Peak Nuclear Power Plant - NRC Integrated Inspection Report 5000445/2015004 and 05000446/2015004 (ML16035A494) notes “[o]n October 17, 2015, the licensee identified corrosion on unit 2 service water piping, in an infrequently entered tunnel...” The GALL Report states “[u]nderground piping and tanks are 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 restricted.” Based on the staff’s review of the LRA, there are no in-scope components exposed to an underground environment.	Based on its review of the subject OE, the staff requests a discussion with respect to if there is inscope underground piping at Comanche Peak.
9	N/A	N/A	The staff reviewed CR-2020-006151 and noted based on biocide results the leak was determined to be fire protection water.	The staff requests a discussion with respect to the cause of the subject leak.

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA/SLRA Section /TLAA/AMP/Scoping and Screening:

Question Number	LRA/SLRA Section	LRA/SLRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
# 1	Item Number 3.2-1, 067 (Comanche Peak Nuclear Power Plant Units 1 and 2 License Renewal Application) From Item 3.2-1, 004, (Comanche Peak Nuclear Power Plant Units 1 and 2 License Renewal Application) From Item 3.2-1, 063 (Comanche Peak Nuclear Power Plant Units 1 and 2 License Renewal Application)	514/2289 488/2289 512/2289	Chapter XI.M29, “Aboveground Metallic Tanks” is not applicable because there are no stainless steel or aluminum tanks (within the scope of Chapter XI.M29, “Aboveground Metallic Tanks”) exposed to soil or concrete in the Engineered Safety Feature (ESF) Systems. However, the aboveground outdoor tanks in the ESF Systems include the missile-protected, stainless steel-lined concrete Refueling Water Storage Tank (RWST) in the CSS (Core Spray System.) The RWST lining exposed to air -outdoor (above water line through vents in the concrete tank) and concrete is included in items 3.2-1, 004, and 3.2-1, 063. The component Stainless steel Piping, piping components, and piping elements; tanks exposed to Air – outdoor uses AMP XI.M36, "External Surfaces Monitoring of Mechanical Components." The Stainless-steel Piping, piping components, and piping elements exposed to Air – indoor, uncontrolled (External), Air with borated water leakage, Concrete, Gas, Air – indoor, uncontrolled (Internal.) Includes the RWST liner that is encased in the concrete of the tank. Does not mention any AMP.	During my review of AMP XI.M36, the AMP does not mention the use (and inspection) of sealant or caulking 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 as required by NUREG-2191 AMP XI.M29. (Page 267/505) (GALL-SLR Report, Vol. 2) Are you planning to enhance the AMP Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components." to meet the minimal requirements of NUREG-1801 AMP Chapter XI.M29 “Aboveground Metallic Tanks”? Please submit updated drawings of the Refueling Water Storage Tank (RWST) in the CSS (Core Spray System)?

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section: B.2.3.15

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	2.3.3.9	2.3-107	LRA Section 2.3.3.9, "Potable and Sanitary Water System," states, "the CLS process tubing that enters the SWIS is encased in either PVC or metallic piping." This section also states, the PVC piping "prevents the process tubing from leakage or spraying onto nuclear safety related components," and "The metallic piping encasing the process tubing is a penetration sleeve which acts as a fire barrier." LRA Table 3.5.2-15 includes carbon steel and stainless steel penetration sleeves with a fire barrier intended function. The identified environments are indoor uncontrolled air, outdoor air, and air with borated water leakage. LRA Table 3.3.2-9a includes external and internal indoor uncontrolled air for the PVC piping. The staff notes that plant-specific note 1 states that "during normal operation the process tubing is not expected to be ruptured, and therefore the PVC piping is exposed to an internal air – indoor (uncontrolled) environment." It is unclear from the LRA what material the process tubing is and whether it may leak prior to rupture. If it may leak prior to rupture, then the PVC piping and metallic penetration sleeve could be exposed to sodium hypochlorite or sodium bromide.	Please discuss what material the Chlorination System process tubing is and whether it could leak prior to rupture, resulting in the PVC piping and metallic penetration sleeve being exposed to sodium hypochlorite or sodium bromide.
2	2.4	2.4-47	LRA Section 2.4.15 and Table 2.4-15 includes carbon steel commodity "fire protection hose stations (racks, reels, and supports)" managed by the Fire Protection program. However, the intended function for this commodity is	Please discuss whether the intent was to manage the carbon steel commodity "fire protection hose stations (racks, reels, and supports)" with the Fire Protection

			Structural Support. It appears that this commodity was included in the Fire Barrier Commodity Group based on the statement in LRA Section 2.4.15 that states, “That is, serve as a rated fire barrier or provide structural support for components used for manual firefighting.” AMR Item 3.3-1, 059 in Table 3.3-1 is cited for managing fire protection hose stations (racks, reels, and supports) by the Fire Protection program. This AMR item is cited in LRA Table 3.5.2-15. LRA Sections A.2.2.15 and B.2.3.15 do not address fire protection hose stations (rack, reel, and support). In addition, LUM00020-REPT-052 does not address this commodity. LRA Section 2.3.3.7 and Table 2.3.3-7 include hose station with a Structural Support intended function. LRA Table 3.3.2-7 cites AMR Item 3.3-1, 078 to manage loss of material of carbon steel hose stations exposed externally to indoor uncontrolled air by the External Surfaces Monitoring of Mechanical Components program. Section 4.1.2 of LUM00020-REPT-053 states, “Fire hose stations and stand pipes are considered piping and are managed under this program [Fire Water System program].” There are several carbon steel piping items in LRA Table 3.3.2-7 that could be associated with hose stations, but it is unclear. Section 8.6 of Procedure FIR-309 addresses hose station racks.	program. If so, please discuss how the Fire Protection program will manage loss of material of the carbon steel fire protection hose stations (rack, reel, and support) to ensure the Structural Support intended function will be maintained. Please also discuss whether any of the carbon steel piping items in LRA Table 3.3.2-7 are associated with hose stations.
3	3.5	3.5-185	LRA Table 3.5.2-15 includes masonry block commodity “concrete block (removable) for opening” with a fire barrier intended function managed by the Masonry Walls and Structures Monitoring programs. The Fire Protection program is not credited for managing applicable aging effects for this masonry block commodity.	Please discuss whether both the Masonry Walls and Fire Protection programs will manage the masonry block commodity “concrete block (removable) for opening.” If the intent is for only the Masonry Walls program to manage this masonry block

			Plant-specific note 1 in LRA Table 3.5.2-15 for this masonry block commodity states, "Furthermore, the Masonry Walls (B.2.3.33) AMP and Fire Protection (B.2.3.15) AMP credit and communicate with each other." A similar statement is made in LRA Table 3.5-1 for AMR item 3.5-1, 070. LRA Section B.2.3.33 states, "Masonry walls that perform a fire barrier intended function are also managed by the Fire Protection (B.2.3.15) AMP." The removable concrete block openings are in a fire wall. Therefore, it appears that both the Masonry Walls and Fire Protection programs should be credited. The staff notes that LRA Section B.2.3.15 does not cite the Masonry Walls program. In addition, the staff did not identify where the Masonry Walls program was cited in LUM00020-REPT-052 as a program that credits or is credited by the Fire Protection program. FIR-311 which includes visual inspection of walls, floors, and ceilings does not reference the Masonry Walls or Structures Monitoring programs. LUM00020-REPT-052 states that the Masonry Walls program is credited by the Fire Protection program. Section 4.4.2 of this document states, "Masonry walls that are fire barriers are visually inspected in accordance with the CPNPP Fire Protection AMP [Ref. 9.16]." LUM00020-REPT-071 states that the Structures Monitoring program credits and is credited by the Fire Protection program. Plant-specific note 3 in LRA Table 3.5.2-15 for this masonry block commodity states, "The Masonry Walls (B.2.3.33) AMP and Fire Protection (B.2.3.15) AMP credit and communicate with each other." However, the Structures Monitoring program is credited for this line item.	commodity, then discuss how the program descriptions and procedures for the Masonry Walls and Fire Protection programs reflect this. Finally, discuss whether plant-specific Note 3 was supposed to cite the Structures Monitoring program rather than the Masonry Walls program.
4	3.5	3.5-185	LRA Table 3.5.2-15 includes galvanized steel commodity "damper housing. AMR Items 3.5-1, 095 (None) and 089 (Boric Acid Corrosion Program) are cited for this commodity. The Fire Protection program is not cited for managing aging	Please discuss why the Fire Protection program is not cited for managing aging effects for the fire damper housing and why the LRA does not address fire dampers in the

			effects. LRA Section A.2.2.15 do not address fire dampers and LRA Section B.2.3.15 only includes OE on a degraded gravity damper in the fire water pump house. Section 3.1 of LUM00020-REPT-052 states a principal objective of the Fire Protection program is “to conduct periodic visual inspection of fire dampers.” Fire dampers are further addressed in several program element sections in this document (Sections 4.1.2, 4.3.2, 4.4.2, 4.5.2, and 4.6.2). In addition, Procedure No. MSM-P0-0705 is for fire damper inspection and cleaning. This procedure does not include fire damper functional testing. The staff notes that NUREG-2191, “Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report,” addresses fire dampers (AMR items, program description, and FSAR supplement). LRA Section B.2.3.15 and LUM00020-REPT-052 includes operating experience related to a degraded gravity damper. CR-2012-006446 states that the suspected cause was age and wear, and that the damper was worn, however, it does not state where the damper was worn (e.g., damper housing).	program elements like that in LUM00020-REPT-052. Is there a procedure for fire damper functional testing? Please discuss whether the gravity damper is installed for fire protection and provide information on where the damper was worn.
5	3.5	3.5-185	LRA Table 3.5.2-15 includes galvanized steel commodity “door” and cites AMR Items 3.5-1, 095 (indoor uncontrolled air, non) and 3.3-1, 059 (outdoor air, loss of material, Fire Protection program), and 3.5-1, 089 (air with borated water leakage, loss of material, Boric Acid Corrosion program).	Given that AMR Item 3.3-1, 059 is for managing loss of material due to wear of steel fire rated doors exposed to indoor uncontrolled air and outdoor air, please discuss use of AMR Item 3.5-1, 095 for galvanized steel commodity “doors” exposed to indoor uncontrolled air.
6	3.5	3.5-187	LRA Table 3.5.2-15 includes silicate radiant energy shield, subliming compound, ceramic fiber/blanket, and stainless steel insulation and wrap exposed to outdoor air. This table also includes elastomer penetration seals and carbon steel	Please discuss where these insulations and wraps, penetration seals, and penetration sleeves are located to be exposed to outdoor air. If these fire barriers are not protected from

			and stainless steel penetration sleeves exposed to outdoor air. Loss of material of the stainless steel insulation and wrap and penetration sleeve exposed to outdoor air in LRA Table 3.5.2-15 are managed by the Structures Monitoring program (AMR Item 3.5-1, 093) and cites plant-specific Note 5, which states, “Relative to stainless-steel components located outdoors, the Structures Monitoring (B.2.3.34) AMP is focused on areas with potential for frequent or prolonged water pooling and communicates with the Fire Protection (B.2.3.15) AMP as warranted.” This plant-specific note does not provide details on how the Structures Monitoring program is adequate to manage the aging effects for the stainless steel insulation and wrap and penetration sleeve to ensure the Fire Barrier intended function is maintained. The staff notes that the discussion of AMR Item 3.5-1, 093 states, “For stainless steel and aluminum, the focus is on areas where water could pool or get within insulation jacketing.” The staff did not find information related to these stainless steel insulation and wrap and penetration sleeve in Structures Monitoring or Fire Protection program descriptions or documents on the portal.	weather, please discuss any impacts on the aging effects to be managed. Please discuss what the specific ceramic fiber/blanket materials are. Please discuss how the Structures Monitoring program is adequate to manage the aging effects for the stainless steel insulation and wrap and penetration sleeve to ensure the Fire Barrier intended function is maintained. Please discuss how the program descriptions and procedures for the Structures Monitoring and Fire Protection programs reflect this.
7	3.5	3.5-188	LRA Table 3.5.2-15 includes masonry block commodity “wall, floor, and ceiling” with a fire barrier intended function managed by the Masonry Walls program. Like Question Number 3, the Fire Protection program was not cited for this masonry block commodity. The staff notes that LRA Section 3.5.2.2.2.2 cites only the Masonry Walls program for managing the fire bricks and mortar.	Please discuss whether both the Masonry Walls and Fire Protection programs will manage the masonry block commodity “wall, floor, and ceiling.”

8	3.5	3.5-189	LRA Table 3.5.2-15 includes gypsum fire barrier walls, floors, and ceilings exposed to indoor uncontrolled air and cites the Fire Protection program for managing cracking, loss of bond, and loss of material. The table includes no NURE-1801, Revision 2 or Table 1 Items and includes generic note F, "Material not in NUREG-1801 for this component," and plant-specific note 4, "Gypsum drywall is utilized throughout the plant to provide a fire barrier which is lightweight and where unit masonry or concrete is not feasible. This lightweight fire barrier material is not addressed in NUREG-1801; however, aging is managed by the Fire Protection (B.2.3.15) AMP." This plant-specific note does not provide why the Fire Protection program is adequate to manage the aging effects for the gypsum drywall.	Please discuss how the Fire Protection program is adequate for managing the aging effects for the rated gypsum drywall to ensure the Fire Barrier intended function is maintained. Provide rating (hours) of gypsum drywall fire barrier.
9	A.2.2.15 and B.2.3.15	A-18 and B-103	LRA Sections A.2.2.15 and B.2.3.15 state that the Fire Protection program "includes visual inspections of 100 percent of each type of penetration fire seal every 15 years." LRA Section A.2.2.15 goes on to state that the visual inspections of penetration fire seals are "in accordance with the plant's NRC-approved fire protection program." The Parameters Monitored/Inspected and Detection of Aging Effects program elements in Rev. 2 of NUREG-1801 state that not less than 10% of each type of seal are to be visually inspected at a frequency in accordance with an NRC-approved fire protection program or at least once every refueling outage.	Please verify that 100 percent of each type of penetration fire seal every 15 years also meets not less than 10 percent of each type of penetration fire seal. Please identify the reference that states visual inspections of 100 percent of each type of penetration fire seal every 15 years is in accordance with the plant's NRC-approved fire protection program. Does this document also have a similar statement related to the frequency of visual inspections of fire door surfaces and function testing of fire doors (see LRA Section A.2.2.15)?
10	N/A	N/A	Section 3.3 in Revision 1 of LUM00020-REPT-052 states that Structures Monitoring AMP will manage drip shields and references Revision 1 of LUM00020-REPT-071. However, Revision 1 of LUM00020-REPT-071 does not address drip shields. The LRA does not explicitly address "drip shields." However, the LRA does include drip pans for the RCS oil spillage collection (Fire Protection System) and drip pans for	Please discuss whether there are drip shields like those at Turkey Point at Comanche Peak or other drip shields that are to be managed by the Structures Monitoring AMP. If there are, please discuss where the drip shields are addressed in the LRA.

			certain ventilation systems. The staff notes that none of the drip pans addressed in the LRA are managed by the Structures Monitoring AMP.	
11	N/A	N/A	Section 6.1 of EPRI 3002013084, "Long-Term Operations: Subsequent License Renewal Aging Effects for Structures and Structural Components (Structural Tools)," November 2018, states, in part, "wire and other appurtenances used to secure fire wrap to the item being protected – is considered to be part of the fire wrap itself." It is unclear from the staff's review of the available information whether there are materials used for securing fire wraps.	Please discuss whether materials are used to secure fire wraps. If so, please discuss where they are addressed in the LRA, including AMR items for managing applicable aging effects.
12	N/A	N/A	Procedure FIR-311 is related visual inspection of fire rated assembly (Thermo-lag, radiant energy shield, and walls, floors, and ceilings). Sections 8.2.3, 8.3, and 8.4.3 state to verify no signs of degradation or damage and provides signs of degradation or damage to look for. However, other than cracks for Thermo-lag, the aging effects in the LRA for these fire barriers, are not provided as signs of degradation or damage to look for.	Please discuss whether the procedure will be updated to reflect the aging effects identified in the LRA.
13	N/A	N/A	Sections 4.3.2, 4.5.2, and 4.6.2 of LUM00020-REPT-052 use the phrase "abnormal degradation."	Please discuss what is meant by "abnormal degradation."
14	N/A	N/A	The Monitoring and Trending program element for the Fire Protection program in NUREG-1801, Revision 2 states that the inspection results for penetration seals, fire barriers, and doors are trended for future actions and periodic tests of the halon fire suppression system provides data for trending. LRA Sections A.2.2.15 and B.2.3.15 do not address trending of inspection results. Section 4.5.2 of LUM00020-REPT-052 states, "The halon system is evaluated and trended by the Fire Protection Strategic Engineering Engineer [Ref. 9.2 Section 6.4]." However, for fire barriers, it only addresses	Please discuss how the inspection results for fire barriers will be trended for future actions.

			documenting the discovered condition, nothing on trending the results for future actions. In addition, FIR-310 and FIR-311 do not address trending the results for future actions.	
15	N/A	N/A	FIR-310 Sections 8.2.4.2 and 8.2.5.2 states to document penetration seals that are inaccessible and that inaccessible penetration seals are not required to be visually inspected. It is not clear if procedures include looking at similar penetration seals that would be representative of the inaccessible seals to determine if additional action should be taken with regard to the inaccessible seals.	Please discuss whether inspection results of similar penetration seals are used to gain inform the possible condition of the inaccessible seals.
16	N/A	N/A	FIR-310 includes width, depth, and length limits for gaps, cuts, rips, gouges, tears, cracks in Type 1, Type 5, and Type 9 seals. In addition, MSG-1018 includes depth limits on cuts when installing seals. However, these documents do not appear to include the basis or the reference containing the basis for the length and width limits.	Please discuss the basis for the width, depth, and length limits for penetration seals.
17	N/A	N/A	The NOTE for Section 8.10 in MSG-1018 states when repairing seal damage or rework existing penetrations, the penetration is to be returned to the “previously acceptable installed condition.”	Please clarify whether “previously acceptable installed condition” means the original installed condition that was analyzed for the specific application?

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

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

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	B.2.3.28	B-167	Exception No. 2 states the following (in part): • The SI [safety injection] Pump Lubricating Oil Reservoirs are internally lined. CPNPP samples the lubricating oil quarterly, and the system includes an oil filter to remove debris and particulates prior to oil reaching the bearings. The oil filter is cleaned quarterly as part of the lubricating oil sampling activities. • This alternate approach has been identified and accepted in an early LRA where that facility had a similar arrangement for SI pump lube oil reservoirs. (Reference ML1582A051).	The staff requests a generic discussion on the following exception, with a focus on the following areas: • Type of lining used in the SI pump lubricating oil reservoirs. • Specifications related to the oil filter (e.g., mesh size). • Discussion of NRC precedent (the ML number referenced in the LRA does not produce any hits in ADAMS).
2	A.2.2.28	A-24	LRA Section A.2.2.28, “Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks,” states “[f]or coated/lined surfaces determined to not meet the acceptance criteria, physical testing ***may be*** performed in conjunction with repair or replacement of the coating/lining.” SRP-SLR Table 3.0-1, “FSAR Supplement for Aging Management of Applicable Systems,” states the following: • “[f]or coated/lined surfaces determined to not meet the acceptance criteria, physical testing ***is*** performed where physically possible (i.e., sufficient room to	1. The staff’s understanding is that the use of the phrase “may be” (in bold and italicized to the left) is linked to language in Exception No. 4. The staff requests a discussion to confirm. 2. The staff requests a discussion with respect to why the training and qualification of individuals involved in cementitious linings inspections is not addressed in LRA Section A.2.2.28 (given that the program scope

			conduct testing) in conjunction with repair or replacement of the coating/lining." • "[f]or cementitious coatings, training and qualifications are based on an appropriate combination of education and experience related to inspecting concrete surfaces."	includes cementlined ductile iron piping).
3	2.3.4.4	2.3-153	FSAR Section 10.4.5, "Circulating Water System," states "[t]he circulating water system is composed of stainless steel, plastic, and carbon steel piping. Piping which is stainless steel, plastic, or carbon steel less than or equal to 2" in diameter is unlined. Carbon steel piping 2½" in diameter and larger is epoxylined. The main condenser water boxes are epoxylined." LRA Section 2.3.4.4, "Main Turbine and Auxiliaries System," states "[t]here are no CW [circulating water] system mechanical components subject to AMR."	The staff request a discussion with respect to if the subject components referenced in the FSAR are in-scope for license renewal.
4	N/A	N/A	GALL Report AMP XI.M42 states for piping, either inspect a representative sample of 73 1-foot axial length circumferential segments of piping or 50 percent of the total length of each coating/lining material and environment combination, whichever is less.	The staff could not identify where sample size is addressed in either the LRA or program basis document on the ePortal.

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section AMP B.2.3.20: Selective Leaching

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	N/A	N/A	N/A	Operating experience (OE) keyword list includes "degraph." In addition, the staff requests "graphiti". This would capture graphitic corrosion and graphitization.
2	N/A	N/A	LUM00020-REPT-057 (program basis document for the Selective Leaching program on the ePortal) allows for elimination of selective leaching inspections for gray cast iron components that have been effectively protected by cathodic protection. Buried components within the scope of the Selective Leaching program include the following: (a) gray cast iron valve bodies and fire hydrants; and (c) cementlined ductile iron piping.	The staff request a discussion with respect to the scope of cathodically protected buried components susceptible to selective leaching.
3	N/A	N/A	LUM00020-REPT-057 allows for elimination of selective leaching inspections for buried components where visual examinations of inscope buried piping have not revealed any coating damage. OE Example No. 2 (for the Buried and Underground Piping and Tanks AMP) states "[i]n April of 2015, an opportunistic inspection was performed during the excavation process for a potable water leak which exposed Fire Protection piping. The inspection noted coating (coal tar wrap) damage. The inspection noted construction dunnage still in place under the pipe, which is not in compliance with the requirements	GALL Report AMP XI.M41 allows for the elimination of selective leaching inspections for buried components where visual examinations of in-scope buried piping have not revealed any coating damage. Given the subject OE, the staff request a clarifying discussion with respect to why this exclusion is applicable at Comanche Peak.

			of 2323-SS-008, CPSES Excavation and Backfill Specification.”	
4	N/A	N/A	FSAR Section 4.5, “Reactor Materials,” states “[t]he coil housings require a magnetic material. Both low carbon cast steel and ductile iron have been successfully tested for this application. The choice, made on the basis of cost, indicates that ductile iron will be specified on the control rod drive mechanism (CRDM).”	The staff request a clarifying discussion with respect to if the ductile iron coil housings referenced in the FSAR are inscope for license renewal.

<u>**Comanche Peak Nuclear Power Plant, Units 1 and 2**</u>
<u>**License Renewal Application (LRA) Breakout Audit Questions**</u>

LRA Section AMP B.2.3.24: Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	B.2.3.24 A.2.2.24	B-149 A-22	SLRA Section B.2.3.24, “Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components,” states “[t]his AMP will also be used to manage cracking due to SCC in stainless steel components exposed to aqueous solutions. Periodic visual inspections or surface examinations may be conducted to manage cracking every 10 years during the PEO. Visual inspections will be conducted in lieu of surface examinations only when the visual inspection methods have been shown to be capable of detecting cracking.” SLRA Section A.2.2.24, “Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components,” states “[s]urface examinations or visual inspections methods shown to be capable of detecting cracking, such as VT-1 examinations, will be conducted to detect cracking of stainless steel components.”	The subject recommendations come from guidance in GALLSLR; however, surface examinations or VT-1 examinations are specified. The staff request a discussion with respect to why VT-1 examinations are not specified.
2	B.2.3.24	B-150	Operating Experience (OE) Example No. 1 states “[t]he fourth quarter of 2020 Vents & Drains system health report identified a material condition within the system. A pipe segment located within the AB was identified with multiple pin hole leaks in various locations.” The staff noted several instances of throughwall leaks in vents and drains system piping during its audit.	The staff requests a discussion on the subject OE and whether it is representative of the condition of inscope piping at Comanche Peak.

3	Table 3.3.2-5	3.3-178	LRA Table 3.3.2-5 states loss of material for aluminum pump casings exposed to waste water will be managed by the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program. The AMR item cites generic note G.	Based on SLR guidance documents, cracking may be an aging effect requiring management (see SRPSLR Section 3.3.2.2.8, "Cracking Due to Stress Corrosion Cracking in Aluminum Alloys.") The staff requests a discussion on whether cracking should also be cited for this material and environment combination.
4	Table 3.3.2-7	3.3-197	LRA Table 3.3.2-7 states hardening, loss of strength, and loss of material for elastomer flexible hoses exposed to waste water will be managed by the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program. The AMR items cite generic note G.	Based on its review of SLR guidance documents, the staff requests a discussion on whether flow blockage due to fouling should also be cited for this material and environment combination.
5	Table 3.3.2-12	3.3-282	LRA Table 3.3.2-12 states loss of material for PVC piping exposed to raw water will be managed by the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program. The AMR item cites generic note G.	Based on its review of SLR guidance documents, the staff requests a discussion on whether flow blockage due to fouling should also be cited for this material and environment combination.

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section N/A: Aluminum

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	Table 3.3.2-8c	Page 3.3-240	LRA Table 3.3.2-8c, "Miscellaneous Ventilation Systems – Summary of Aging Management Evaluation," states that aging effects for aluminum fan housings exposed to outdoor air are not applicable and no AMP is proposed. The AMR items cite generic note I.	This material and environment combination is not addressed in LR guidance documents; however, it is addressed in SLR guidance documents. Guidance with respect to whether cracking and loss of material are applicable aging effects for aluminum are contained in SRP-SLR Sections 3.3.2.2.8, "Cracking Due to Stress Corrosion Cracking in Aluminum Alloys," and 3.3.2.2.10, "Loss of Material Due to Pitting and Crevice Corrosion in Aluminum Alloys." The staff requests a discussion with respect to why aging effects are not applicable for the subject components.

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LRA Section N/A: Titanium

Question Number	**LRA Section**	**LRA Page**	**Background / Issue** **(As applicable/needed)**	**Discussion Question / Request**
1	N/A	N/A	FSAR Section 10.4.1.1.2, “System Description,” states “[t]he titanium used in the condenser tubes has good erosion and steam impingement resistance.” FSAR Section 10.4.1.1.3, “Safety Evaluation,” states “[t]he main condenser tube bundles and tube sheet assemblies are constructed of titanium and titanium clad carbon steel respectively.” FSAR Section 10.4.5.2, “System Description,” states “[t]itanium tubes are used throughout the condenser tube bundle assemblies in both the air removal and condensing sections.”	FSAR Chapter 10, “Steam and Power Conversion System,” includes several references to titanium components. Based on the staff’s review of the LRA, there are no inscope titanium components at Comanche Peak. Staff request a clarifying discussion with respect to if there are inscope titanium components at Comanche Peak.

Comanche Peak Nuclear Power Plant, Units 1 and 2
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LRA/SLRA Section /TLAA/AMP/Scoping and Screening:

Question Number	LRA/SLRA Section	LRA/SLRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
# 1	Report Number LUM00020-REPT-055 XI.M30 Fuel Oil Chemistry (Comanche Peak Nuclear Power Plant Units 1 and 2 License Renewal Application Basis Document)	11/38	Exception 1: The EDG DFOSTs are drained, cleaned, visually inspected, and ultrasonically inspected on a 20-year frequency per Preventive Maintenance (PMs). NUREG-1801, Rev. 2 guidance recommends draining, cleaning and visually inspecting each diesel fuel tank at least once during the 10-year period prior to the PEO, and on a 10-year frequency during the PEO.	Under NUREG-1801, Rev 2 the 10-year interval was considered an acceptable duration such that any unforeseen degradation mechanism would be identified prior to challenging tank integrity. While previous inspections have not identified historical corrosion or wall thinning. Can you please explain to the NRC staff in detail including drawings (and/or graphics, presentation) why this does not preclude a new or more aggressive degradation mechanism from occurring during the period of extended operation? Also, based on the description of exception #2, there could be 5 or more inches of standing water in the bottom of the tank, and it could exist for long periods of time without being identified. Do you think this could be one potential source of degradation that would only be identified by a full draining and examination of the tank?
# 2	Report Number LUM00020-REPT-055	12/38	Exception 2: The CPNPP Fuel Oil Chemistry AMP collects fuel oil samples from the lower portion (6 inches from the bottom) of the EDG DFOSTs on a 31-day frequency. NUREG-1801 Rev. 2 guidance	Under NUREG 1801, Vol. 2, the alternative sampling method allowed by GALL calls for a sample from the lowest point in the tank. The sampling point proposed by Comanche Peak is

	XI.M30 Fuel Oil Chemistry (Comanche Peak Nuclear Power Plant Units 1 and 2 License Renewal Application Basis Document)		recommends periodic multilevel sampling to provide assurance that fuel oil contaminants are below unacceptable levels. If tank design features do not allow for multilevel sampling, a sampling methodology that includes a representative sample from the lowest point in the tank is allowed. Previously, the bottom sample methods not at the lowest portion of the tank were approved, if the applicant was able to show that there was a set of sampling points (pipe and valves) and this was the only way to access a sample. In addition, the applicant was able to show that significant flow and tank turnover occurred such that any water would be mixed into the sample taken from a location several inches away from the lowest point in the tank.	6 inches above the tank bottom. Please explain clearly to the NRC staff how this sample would identify a layer of water, microorganism, and/or contamination on the tank bottom? The sample at Comanche Peak will being taken by lowering a thief into the tank. Why is 6 inches as close to the bottom as you can get? Given the size of the tank Please, explain in detail (using drawings and/or graphics, presentation). Do you think it would be possible whether any flow would occur prior to sampling that would mix a water layer into the tank level where a sample is taken?

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section AMP B.2.3.11: Open Cycle Cooling Water System

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	B.2.3.11 3.3.2.2.8	B-89 3.3-25	SLRA Section B.2.3.11, "Open Cycle Cooling Water" (OCCW) AMP states that recurring internal corrosion (RIC) has not been identified for components within the scope of the OCCW AMP. This section further states that CPNPP's response to IER L3 1453 was to review the set of plant-specific OE for ten years prior for evidence of thru-wall pipe leaks in class 3 SSW piping and no leaks were identified during that time period. • TR-2021-005356 – leak at circulating water lube pumps leaking at several gallons per minute (gpm). • TR-2020-9117 – the end bell of the 2-02 component cooling water heat exchanger (HX) was leaking on the station service water side at approximately 5 gpm. • CR-2019-002492 – A pencil-sized stream of leakage observed from the 2-01 component cooling water HX while 2SW-0023 was throttled to obtain test pressure. • CR-2016-4868 – After completion of maintenance to clean CCP 101 lube oil cooler, a service water leak was discovered coming from significant pitting of the flange sealing surface of the cooler end bell. This pitting also documented previously in CR-2014-1804. • More flange pitting documented in CR2020-7876, CR20207762, & CR201112880. SLRA Section 3.3.2.2.8, "Loss of Material due to Recurring Internal Corrosion (Per LR-ISG-2012-02)" states, "Corrosion in the	The listed TRs and CRs appear to contradict the statement that recurring internal corrosion is not occurring at CPNPP. The staff requests discussion on the cited TRs and CRs as to why they do not provide evidence that RIC is occurring at CPNPP.

			SSW meets the criteria to be considered RIC based solely on number of instances. However, no minimum wall thickness criteria were exceeded. Reduction in wall thickness has been monitored at a frequency to sufficiently identify any issues. Based on the monitoring, the CPNPP SSW does not meet criteria for extent of degradation." • CR-2016-4786 – While performing work order 5042345, it was noted that five nodule corrosion areas were below the minimum acceptable pipe wall thickness.	
2	2.4.8	2.4-22	SLRA Section 2.4.8, "Service Water Intake Structure" states, "Traveling screens perform their function with moving parts, are "active" and are not subject to AMR." The traveling screens provide filtration, which is defined as a passive function in Table 2.1-4(b) of the SRP. Traveling screens are typically monitored for high differential pressure, but they typically do not have a low differential pressure indication, which is what would be needed to accomplish an inspection function to verify there was not a large hole in the screens that would allow debris to pass through the screen.	The staff would like to discuss documenting the current inspections that are performed during "operator rounds" and how those inspections could be used as an aging management inspection for loss of material in the traveling screens.

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LRA Section: B.2.3.16

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	2.3	2.3-81	LRA Section 2.3.3.7 states, “The diesel engines for fire pumps and the engine mounted components are part of the active engine assembly. These active components are not subject to AMR. The exhaust piping, exhaust silencer, flame arrestor and heat exchanger are subject to AMR [emphasis added].” Drawing DDMEC_MX-0225_009_CP11-LR shows a water jacket cooler and flex connection highlighted in green which means these components are subject to AMR. However, the drawing also has a LR Note that states, “Diesel Engines and Water Jacket Cooler are Complex Assemblies Per NEI 95-10 and are not Subject to AMR.” The staff noes that NEI 95-10, Reference 12 states: “if a nonsafety-related diesel generator is required for safe shutdown under the fire protection plan, the diesel generator and all SSCs specifically required for that diesel to comply with and operate within the Commission’s regulations based on the applicant’s design specifications for that diesel shall be included within the scope of license renewal under 10 CFR 54.4(a)(3). This may include, but should not be limited to the cooling water system or systems required for operability,” LRA Table 3.3.2-7 includes several components associated with the fire pump diesel engines. However, it does not	Please discuss revising the LRA (including drawing DDMEC_MX-0225_009_CP11-LR) to include the heat exchangers, water jacket coolers, and flex connections associated with the fire pump diesel engines in accordance with industry guidance.

			appear to include heat exchangers, water jacket coolers, or flex connections.	
2	3.3	3.3-206	LRA Table 3.3.2-7 includes component type “Strainer (Electric motor driven pump suction).” However, LRA Sections A.2.2.16 and B.2.3.16 state, in part, “There are no fire pump suction strainers for the main fire protection pumps…within the scope of LR.” The staff notes that the “Fire Water System Inspections and Tests” table in LRA Section B.2.3.16 states, “The Fire Pumps receive treated water and do not have suction screens.”	Please clarify the information presented in the LRA with regards to fire pump suction strainers and screens.
3	2.4.11	2.4-28	LRA Section 2.4.11 states, “The tanks are provided with a reinforced concrete ring wall type foundation below the tank walls.” The Discussion of AMR Item 3.3-1, 128 in LRA Table 3.3-1 states, “In addition, the Fire Water System (B.2.3.16) AMP includes enhancement to address the steel/concrete interface.” LRA Table 3.3.2-7 does not address Tank (Fire Water Storage) exposed externally to concrete. The concrete ring wall type foundation typically has sand or soil inside the ring that provides support to the tank bottom, however, LRA Table 3.3.2-7 does not address Tank (Fire Water Storage) exposed externally to sand or soil. In addition, the LRA does not appear to address the use (and inspection) of sealant or caulking at the steel/concrete interface to mitigate tank corrosion. LRA Table 3.3.2-7 cites AMR Item 3.3-1, 136 with a Generic Note B for managing loss of material of the carbon steel with internal coating/lining tank (Fire Water Storage) exposed internally to treater water. However, AMR Item 3.3-	Please discuss why LRA Table 3.3.2-7 does not include concrete and sand or soil external environments for the fire water storage tank. Please discuss where the LRA addresses use (and inspection) of sealant or caulking at the steel/concrete interface. Please discuss the use of AMR Item 3.3-1, 136 for internally coated/lined fire water storage tanks. Please discuss the basis for only performing bottom-thickness measurements on each tank during the first 10-year period of the period of extended operation and when there is evidence of pitting or corrosion.

			1, 136 does not address internally coated/lined fire water storage tanks. The staff notes that "Fire Water System Inspections and Tests" table in LRA Section B.2.3.16 states, "In addition, the new plant implementing document will require bottom-thickness measurements on each tank during the first 10-year period of the period of extended operation. Tank bottoms will be tested for metal loss and/or rust on the underside by use of ultrasonic testing where there is evidence of pitting or corrosion. Removal, visual inspection, and replacement of random floor coupons are an acceptable alternative to ultrasonic testing." The staff notes that XI.M29 in NUREG-1801 addresses tank bottoms exposed to concrete. Specifically, to ensure loss of material is not occurring at inaccessible locations, ultrasonic testing thickness measurements are taken whenever the tank is drained and at least once within 5 years of entering the period of extended operation. The staff notes that NUREG-2191 also points to XI.M29 for tank bottoms exposed to concrete. Table XI.M29-1 states that loss of material is managed for steel external surfaces exposed to soil or concrete by performing volumetric inspections from the inside surface each 10 year period starting 10 years before the subsequent period of extended operation. It is unclear to the staff how performing the bottom-thickness measurements during the first 10-year period of extended operation and when there is evidence of pitting or corrosion meets the guidance in NUREG-1801 and NUREG-2191 related to fire water storage tank bottoms exposed to concrete.	
4	3.2.2.2.9	3.2-11	LRA Section 3.2.2.2.9 states, "As such, credited AMPs, such as Fire Water, Inspection of internal Surfaces in Miscellaneous Piping and Ducting Components, and Open cycle Cooling Water, do not require enhancement to	Please discuss whether the Fire Water System should have been referenced in LRA Section 3.2.2.2.9.

			address RIC in ESF systems." However, LRA Tables 3.2.2-1 through 3.2.2-5 do not credit the Fire Water System program with managing components in any ESF systems.	
5	A.2.2.16	A-18	The Scope of Program program element of LR-ISG-2013-01 states, "However, where the fire water storage tank internals are coated, the Fire Water System Program and FSAR Summary Description of the Program should be enhanced to include the recommendations associated with training and qualification of personnel <u>and the "corrective actions" program element</u> [emphasis added]." The Detection of Aging Effects program element of LR-ISG-2013-01 states, "The training and qualification of individuals involved in coating/lining inspections <u>and evaluating</u> [emphasis added] degraded conditions is conducted in accordance with an ASTM International standard endorsed in RG 1.54 including staff limitations associated with a particular standard, except for cementitious materials." LRA Section A.2.2.16 does not include statements regarding the training and qualification of personnel evaluating degraded conditions and it does not include statements regarding the corrective actions recommendations.	Please discuss the omission of statements regarding personnel evaluating degradation and corrective actions from LRA Section A.2.2.16.
6	A.2.2.16 and B.2.3.16	A-18 and B-106	LRA Sections A.2.2.16 and B.2.3.16 state, "The augmented examinations for the portions of normally dry piping that are periodically wetted <u>or experiencing recurring internal corrosion</u> [emphasis added] include (a) periodic full flow tests at the design pressure and flow rate, or internal inspections, and (b) volumetric wall thickness evaluations." LRA Section B.2.3.16 also states, "The review did not identify instances of recurring internal corrosion within the fire protection systems." However, LRA Section 3.3.2.2.8 states that recurring internal corrosion is not applicable for Auxiliary Systems, including the Fire Water System. The Discussion for AMR Item 3.3-1, 127 in LRA Table 3.3-1	Please discuss whether recurring internal corrosion is occurring in the Fire Protection System. In addition, please discuss each identified CR and TR to demonstrate whether it is or is not pressure boundary internal corrosion.

			states, “Not applicable. Loss of material due to recurring internal corrosion has not been identified in metallic piping, piping components, and tanks exposed to raw water or waste water in the auxiliary systems.” Finally, LRA Table 3.3.2-7 does not cite AMR Item 3.3-1, 127 for any components in the Fire Protection System. Operating experience reviews identified the following reports of leakage, but it is not clear whether these are pressure boundary leaks caused by internal corrosion: • CR-2021-008314, Dec-08-2021, 6-inch FP pipe has through-wall leak • TR-2021-007722, Nov-12-2021, Leakage rate for 2FP-0493C has risen to 0.12 gpm • TR-2021-007069, Oct-23-2021, Leaking FW riser • TR-2021-004976, Jul-28-2021, X-HV-4107D leaking from Retard Chamber • TR-2021-004964, Jul-28-2021, XFP-0595 leak rate of 5-6 dpm • CR-2021-001386, Feb-23-2021, 5dpm leak on FP line • CR-2021-001377, Feb-23-2021, FP leak in Admin Bldg • TR-2021-001032, Feb-09-2021, Pin hole leak in 2.5-inch sprinkler pipe • CR-2017-005038, Apr-15-2017, Fire suppression pipe has a slow leak • CR-2016-006407, Jul-03-2016, Small through-wall leak on FP piping	

			• CR-2014-000046, Jan-3-2014, Leak on side of elbow downstream of 1FP-0093E • CR-2013-009025, Aug-29-2013, Through-wall leak in FP piping • CR-2012-005114, May-18-2012, Water dripping from strainer Therefore, it is unclear to the staff whether recurring internal corrosion is an applicable aging effect for the Fire Protection System.	
7	B.2.3.16	B-106	Appendix L of LR-ISG-2012-02 states that portions of water-based fire protection system components that have been wetted but are normally dry are subject to augmented tests or inspections. In addition, LRA Section B.2.3.16 includes an enhancement to perform augmented tests and inspections on piping segments that cannot be drained or collect water. However, the staff did not find where the portions subject to augmented testing or inspection were identified.	Please identify which portions of the fire protection system are subject to augmented testing or inspection because they are normally dry but periodically subjected to flow and cannot be drained or allow water to collect.
8	B.2.3.16	B-114	Appendix L of LR-ISG-2012-02 recommends that main drain tests follow Section 13.2.5 of NFPA 25. Section 13.2.5 of NFPA 25 requires "main drain tests to be conducted annually at each water-based fire protection system riser to determine whether there has been a change in the condition of the water supply piping and control valves." It also states, "When there is a 10 percent reduction in full flow pressure when compared to the <u>original acceptance test or previously performed tests</u> [emphasis added], the cause of the reduction shall be identified and corrected if necessary." "Fire Water System Inspections and Tests" table in LRA Section B.2.3.16 states, "Test results will be compared to <u>previous results</u> [emphasis added] to determine if there has	To identify whether significant degradation of the fire water system supply has been occurring over several years, please discuss whether test results will also be compared to the original acceptance test (or baseline). If a baseline, please provide discussion of how that baseline was determined.

			been a 10% or greater reduction in full flow pressure, and if there is, the issue will be entered into the CAP and the cause of the reduction will be identified and corrected, as necessary." The staff notes that if the test-to-test pressure monitoring only uses the immediately prior test result, significant degradation of the fire water system supply over several years may not be identified while still being less than a 10 percent reduction from the previous test.	
9	B.2.3.16	B-106	The Acceptance Criteria program element in NUREG-1801 states, in part, "…(b) no unacceptance signs of degradation are observed during non-intrusive or visual inspection of components, (c) minimum design pipe wall thickness is maintained, (d) no biofouling exists in the sprinkler systems that could cause corrosion in the sprinklers." LRA Section B.2.3.16 states, in part, "…(b) no unacceptable signs of degradation <u>or fouling</u> are observed during nonintrusive or visual inspections, and (c) <u>in the event surface irregularities are identified, testing is performed to ensure</u> minimum design pipe wall thickness is maintained." The guidance does not include fouling in Acceptance Criteria (b) but in Acceptance Criteria (d), which is not included in LRA Section B.2.3.16. In addition, Acceptance Criteria (c) in LRA Section B.2.3.16 does not appear to take into account thickness measurements performed on periodically wetted but normally dry pip and fire water storage tank bottoms.	Please discuss the differences in the Acceptance Criteria in LRA Section B.2.3.16 from NUREG-1801.
10	3.3	3.3-199	LRA Table 3.3.2-7 cites AMR Item 3.3-1, 091 for managing loss of material for carbon steel piping exposed internally to waste water. While AMR Item 3.3-1, 091 does not include flow blockage in NUREG-1800, Rev. 2, flow blockage is included in this AMR item in NUREG-2191.	Please discuss whether flow blockage is an applicable aging effect requiring management for carbon steel piping exposed internally to waste water.

11	3.3	3.3-194	LRA Table 3.3.2-7 does not identify erosion as an applicable aging effect for any Fire Protection system components.	Please discuss whether any operational experience related to erosion mechanisms, including cavitation, flashing, droplet impingement, or solid particle impingement have been identified.
12	N/A	N/A	Section 4.4.2 of LUM00020-REPT-053 states, "If the environmental conditions (e.g., type of water, flowrate, temperature) and material that exist on the interior surface of the underground and buried fire protection piping are similar to the conditions that exist within the above grade fire protection piping, the results of the inspections of the above grade fire protection piping can be extrapolated to evaluate the condition of buried and underground fire protection piping for the purpose of identifying inside diameter loss of material." This section further states, "With respect to buried cement lined ductile iron piping, the environmental and material conditions are similar to those above grade, with the exception that the ductile iron piping is cement lined."	Is the intent to assume what is occurring with the above grade fire protection piping that is not cement lined is occurring with the buried cement lined ductile iron fire protection piping?
13	N/A	N/A	Section 3.3 of LUM00020-REPT-053 states that the Turkey Point RAI related to trending sprinkler system inspection and tests results is applicable however, LR-ISG-2002-02 does not discuss trending of deposits. This section goes on to state that buried and underground flow tests are monitored and trended, and that the Fire Water System program will be enhanced to monitor and trend standpipe and hose system and main drain flow test results. It does not discuss whether monitoring and trending of sprinkler system flow tests are performed. It is unclear whether deposits are monitored and trended for any flow tests.	Please discuss whether sprinkler system flow tests are monitored and trended. Please discuss what parameters are/will be monitored and trended for flow tests.

<u>Comanche Peak Nuclear Power Plant, Units 1 and 2</u>
<u>License Renewal Application (LRA) Breakout Audit Questions</u>

LRA Sections 3.2.2.2.3.2, 3.2.2.2.6, 3.3.2.2.3, 3.3.2.2.5, 3.4.2.2.2, 3.4.2.2.3

Question Number	LRA/SLRA Section	LRA/SLRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	3.2.2.2.3.2, 3.2.2.2.6, 3.3.2.2.3, 3.3.2.2.5, 3.4.2.2.2, 3.4.2.2.3		LRA further evaluation (FE) sections 3.2.2.2.3.2, 3.2.2.2.6, 3.3.2.2.3, 3.3.2.2.5, 3.4.2.2.2, 3.4.2.2.3 state that loss of material (LOM) due to pitting corrosion, crevice corrosion, and stress corrosion cracking (SCC) do not require aging management because the outdoor air does not contain sufficient halide concentration and events that would increase the halide content are unlikely. However, LRA Sections 3.2.2.2.3.2, 3.3.2.2.5, and 3.4.2.2.3 identify select instances where LOM due to pitting and crevice corrosion (but not SCC) of stainless steel is considered possible and managed by the External Surfaces Monitoring of Mechanical Components Aging Management Program. These include the top of the Condensate Storage Tank, Reactor Makeup Water Storage Tank, and Refueling Water Storage Tank. The discussions for these instances note that contaminants in outdoor air could collect on these components surfaces. LRA guidance identifies criteria for determining if the presence of sufficient halides is likely (e.g., NUREG-1800, Rev. 2, Sections 3.2.3.2.3.2 and 3.2.3.2.6). However, SLR guidance states more broadly that LOM and SCC of stainless steel could occur in environments containing sufficient halides in	For FE Sections 3.2.2.2.3.2, 3.2.2.2.6, 3.3.2.2.3, 3.3.2.2.5, 3.4.2.2.2, 3.4.2.2.3, please explain the criteria used in the LRA for determining whether or not stainless steel components could be exposed to conditions potentially causing pitting, crevice corrosion, or SCC that require management.

			the presence of moisture (e.g., NUREG-2192, Sections 3.2.3.2.2 and 3.2.3.2.4).	
2	3.2.2.2.3.2, 3.2.2.2.6, 3.3.2.2.3, 3.3.2.2.5, 3.4.2.2.2, 3.4.2.2.3	3.2-8 and 3.2-9, for example	In LRA sections 3.2.2.2.3.2, 3.2.2.2.6, 3.3.2.2.3, 3.3.2.2.5, 3.4.2.2.2, 3.4.2.2.3 for stainless steel, in cases where LOM due to pitting and crevice corrosion are considered aging effects requiring management (AERM), SCC is not considered an AERM. For example, Section 3.2.2.2.3.2, "Loss of Material due to Pitting and Crevice Corrosion," includes some cases where LOM for stainless steel in air is considered possible and is managed by the External Surfaces Monitoring of Mechanical Components Aging Management Program. However, LRA Section 3.2.2.2.6, "Cracking due to Stress Corrosion Cracking," states that no stainless steel components exposed to air in ESF Systems require aging management for SCC.	For FE Sections 3.2.2.2.3.2, 3.2.2.2.6, 3.3.2.2.3, 3.3.2.2.5, 3.4.2.2.2, 3.4.2.2.3, please discuss the basis for not considering stainless steel components susceptible to SCC when they are considered susceptible to pitting and crevice corrosion.
3	3.2.2.2.6	3.2-9	LRA Section 3.2.2.2.6 contains a reference to Item 3.2-1, 006, which applies only to BWRs. This appears to be intended to reference 3.2-1, 007, which applies to FE Section 3.2.2.2.6 for SCC of stainless steel.	Please confirm the intended item where Item 3.2-1, 006 is used in Section 3.2.2.2.6.

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LRA Section AMP B.2.3.12: Closed Treated Water Systems

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	B.2.3.12	B-94	LRA Section B.2.3.12, "Closed Treated Water Systems" (CTW) AMP states that an evaluation performed in 2013 of through-wall leaks in carbon steel Turbine Plant Cooling Water (TPCW) welds identified that the associated portion of the TPCW system was constructed with undersized welds and the degradation qualified as Recurring Internal Corrosion (RIC) based on the guidance in LR-ISG-2012-02. The Closed Treated Water Systems AMP is being enhanced to include a 20% inspection of the in-scope TPCW welds (up to a max of 25 welds) every 10 years during the period of extended operation (PEO). LR-ISG-2012-02 states the criteria for RIC is (a) a 10-year search of plant specific OE reveals the aging effect has occurred in three or more refueling outage cycles; or (b) a 5-year search of plant specific OE reveals the aging effect has occurred in two or more refueling outage cycles as a result of which the component either did not meet plant-specific acceptance criteria or experienced a reduction in wall thickness greater than 50 percent (regardless of the minimum wall thickness).	The 20% sample (max of 25) proposed inspection scope in the LRA matches the minimum inspection scope contained in the GALL-SLR for the CTW system, which is proposed for verifying that the water chemistry program is adequately managing aging in a CTW system. As noted in Section 3.2.2.2.7 of the SRPSLR, recurring internal corrosion can result in the need to augment AMPs beyond the recommendations in the GALLSLR Report. Please discuss your technical basis for implementing the minimum inspection scope and not expanding the sample (e.g., larger sample size, more frequent inspections) since recurring internal corrosion has already been identified in the TPCW system.
2	B.2.3.12	B-94	Additional operating experience in Section B.2.3.12 indicates that 25 percent of the TPCW heat exchanger tubes were replaced in 2020, which reduced the number of plugged tubes from 289 to 240. An analysis determined a tube plugging limit of 1,066 tubes, and the worstcase projection of future tube	Since the worst-case plugging projection predicted exceeding the plugging limit in just 4 years (2020-2024), please discuss the best-case and nominal projections. Also, please discuss the cause of the tube plugging and your technical basis for

			plugging predicts that the plugging limit of 1,066 tubes could be exceeded in 2024.	implementing the minimum inspection scope and not expanding the sample, since recurring internal corrosion has already been identified in the TPCW system.

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LRA Section AMP B.2.3.8: Flow-Accelerated Corrosion

Question Number	Background / Issue (As applicable/needed)	Discussion Question / Request
1	STA-170-1 Software QA Form for FAC Manager Software is level 3 “so SQA controls are minimal” A8.6 CPNPP will provide verification that the software has been procured in accordance with STA-151. Ensure error reporting will be provided by supplier to appropriate Vistra personnel.	Discuss what are software QA requirements for both CHECWORKS and FAC Manager software. Both software suppliers appear to provide error reporting and validation and verification by the vendors. Are these required by Level 3 software? If these are not required by CPNPP’s Software QA requirements, is there a contractual requirement for the vendors to provide error reporting or validation and verification?
2	TR-2017-012206 “From data obtained, the component could operate for two full cycles before it should be replaced.	Provide documentation that showed acceptability of operation for 2 more cycles and discuss. Provide scope expansion information and discuss.
3	TR-2019-004795 identified thinning and component would be replaced next outage; however, leak developed prior to outage and had to be temporarily repaired	Provide documentation that showed acceptability of operation until the next outage. Since leak occurred prior to plan, what went wrong?
4	TR-2019-005126 Extent of condition for TR-2019-004731 added warm-up lines to CHECWORKS as susceptible non-modeled. States “lines will be inspected based on inspection priority established per Altran Susceptibility Evaluation Tech Report 11-2235-TR-001, Rev 0.	What is inspection priority of these lines per the Altran Report? Initial leak was on a priority 2 line, that was not frequently inspected. Is priority changed based on leak in TR-2019-004731? Did orifice location play a part in the rupture? If so, was susceptibility of failure appropriately considered and do other lines with orifices (beyond the HD warm-up lines) need to be considered?

5	TR-2021-001032 discusses additional deficiencies and extent of condition and notes that a cursory review of PMs deactivated by AI-TR-2019-008413-11 found that PM 344709 to measure pipe wall thickness in segments CO-2-071 was deactivated. The initial PM was created by CR-2002-002672 because of operational changes creating a region susceptible to erosion when SG Blowdown is aligned to the condenser hotwell.	Discuss whether this activity managing erosion was appropriately deactivated.
6	2RF17 EOC Report for Cycle (May-2017 to Dec-2018) Report was issued in Sep-2020, 21(?) months after the end of the cycle. 1RF20 EOC Report (Nov-6-2017 to Apr-20-2019) [Prepared Sep-2020, Appvd Dec-2020, 17(?) months after end of cycle.	EPG- 9.04 addresses issuance of EOC Reports but does not specify timeliness. Is 21 months acceptable? If not, was a TR written for this? Does additional guidance on EOC issuance timeliness need to be added to EPG-9.04?
7	TERPT ER-ME-093, Flow-Accelerated Corrosion System Susceptibility Analysis for Comanche Peak Unit 1, Rev 0, May 1995, Table 6.2 Non-susceptible systems The following were excluded based on “NW” non-water: Chemical Volume and Control Containment Spray Residual Heat Removal Safety Injection	Discuss how these systems are “non-water” systems.
8	LUM00020-REPT-045, Rev 1, Flow-Accelerated Corrosion (FAC) Program Basis Document, Section 3.3, License Renewal RAI Responses discusses Peach Bottom RAI B.2.1.9-2 relating to legacy errors	The issue in the RAI and the Info Notice deals with legacy errors in the CHECWORKS models. Neither response in the basis document appear to address the potential issue of legacy errors. Other than going back and checking orifice sizes, have there been any other activities to validate the

	in CHECWORKS models. Response says "procedures will be enhanced to <u>require</u> that updates to predictive models are controlled and independently reviewed." The RAI is based on Info Notice 2019-08, which is addressed in Program Basis Document Section 4.10. Discussion states "Both sites indicated that if their models within the FAC program had been accurately validated this [it] is possible these events could have been prevented."	existing CHECWORKS models that would be responsive to the issues in the RAI or Info Notice? Although response associated with the Peach Bottom RAI says the procedures **will be enhanced**, there are no enhancements associated with predictive model updates. The current program (EPG-9.04, Section 6.12) only <u>recommends</u> that changes to the model be review but does not appear to require this. Does there need to be an additional enhancement to the program procedures as stated?
9	LRA Table 3.3.2-12 Station Service Water System includes PVC piping being managed for wall thinning through the Flow-Accelerated Corrosion Program.	Discuss how wall thinning of PVC piping will be monitored by the FAC program.

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LRA Section: B.2.3.4 GALL AMP
XI.M10 Boric Acid Corrosion

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	B.2.3.4	B-42	The LRA says no Enhancements to program. Program basis document, LUM00020-REPT-042, Section 6 “Regulatory Commitments Including Any Enhancements and Inspections,” says the AMP does not require any enhancements to be consistent with the GALL AMP. However, Section 7, “Summary of Implementing Documents,” notes that procedures STA-737 and STI-737.01 have actions to revise them to include several aspects associated with license renewal with implementing schedules associated with the PEO.	If the program basis document identifies changes to be made to the implementing procedures (beyond being consistent with the GALL) (with the implementation schedule tied to PEO), then it appears that you have identified enhancements to the program. Discuss why the identified changes to the implementing procedures are not considered enhancements to the program.

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LRA Section B.2.3.2
Water Chemistry

Question Number	LRA/SLRA Section	LRA/SLRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	N/A	N/A	The 2020 primary water self-assessment, TR-2020-004330, identifies improvement opportunities. Section 4.10.2 of the Water Chemistry AMP basis document, LUM00020-REP-040, states that results of the TR-2020-004330 self-assessment were incorporated into procedures. However, Revision 15 of CHM-120, "Primary Chemistry," the revision referenced in the AMP basis document, is dated 2012.	Please clarify how the TR-2020-004330 self-assessment results were incorporated into the implementing documents.
2	B.2.3.2	B-34	The "Industry OE" section refers to NRC Information Notice (IN) 2007-37 regarding accumulation of deposits in steam generators and the potential effect on tube integrity. The LRA states that CPNPP performs sludge lancing to remove deposits on the secondary side of SGs. Although sludge lancing is a commonly used method of deposit removal, it is used in the tubesheet region. IN 2007-37 addressed deposits in the tubesheet region as well as openings in broached tube support plate (TSP) openings. This example of operating experience in the LRA does not address deposits at TSPs. The operating experience described in IN 2007-37 was related to deposits in broached quatrefoil-shaped TSP openings. Steam generators at CPNPP have	Please describe the strategy at CPNPP for assessing and limiting the accumulation of deposits at TSPs that could affect tube integrity.

			broached quatrefoil (Unit 2) and trifoil (Unit 1) TSP openings.	
3	N/A	N/A	Section 8 of "Primary Chemistry Strategic Plan," identifies differences between practices at CPNPP and practices that the EPRI primary water chemistry guidelines (GL) "suggest." The document references Revision 7 of the GL in Section 9 ("References"), but Section 8 has descriptions of information that matches Revision 6 of the GL rather than Revision 7, such as Table 3-4. As a result, the differences between CPNPP and the EPRI GL referenced in the LRA are not clear to the staff. In addition, the meaning of EPRI GL "suggestions" is unclear. The staff's understanding is that in this context the word "suggest" in "Primary Chemistry Strategic Plan" refers to "recommended" elements in the EPRI GL.	a) Please clarify whether the term, "EPRI Primary Water Chemistry Guidelines suggest …," in Section 8 of "Primary Chemistry Strategic Plan," refers to "recommended" elements in the EPRI GL. b) Please discuss whether the differences between CPNPP practices and Revision 7 of the EPRI GL are limited to "recommended" elements and do not include "mandatory" or "shall" elements. "Primary Chemistry Strategic Plan" refers to both Revisions 6 and 7 of the EPRI GL, but the LRA proposes an exception to use Rev. 7.
4	N/A	N/A	TR-2019-004928 describes a possible departure from EPRI PWR Secondary Water Chemistry GL with respect to the time to establish wet layup chemistry in the Unit 1 SGs in 1RF20. The TR description indicates the seven-day requirement in the GL was met, but the TR also includes a corrosion estimate for 13.73 days. The TR also refers to SG moisture separator manufacturing deviations, feedring repairs, and foreign objects in the Unit 1 feedwater rings, feedwater nozzles, and lower deck plates.	a) Please clarify the circumstances regarding the possible departure from EPRI guidelines and the conclusions from the evaluation of corrosion and other effects. b) Please describe the moisture separator manufacturing deviations and feedring repairs, and discuss any effects on the present management of loose parts and foreign objects in the Unit 1 SGs.
5	N/A	N/A	Attachment 8.G of CHM-130, "……," has a figure depicting the steam generator high head chemical feed tank with dimensions and formulae for volume calculations. The figure shows a portion of the cylindrical section of the tank identified as non-usable. However, this same section of the tank	Please clarify the calculation of the useable portion of the tank since it appears to include a cylindrical portion of the tank defined as non-usable.

			appears to be included in the calculation of usable volume, V_u, since V_u is based on the full height of the cylindrical portion (V_c).	

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LRA Sections B.2.3.10 and 4.7.7
Steam Generators and TLAA SG Tubes Metal Corrosion Allowance

Question Number	LRA/SLRA Section	LRA/SLRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	B.2.3.10	B-77 to B-80	The LRA states that the "Steam Generators" AMP is consistent, with three exceptions, with the NUREG-1801 Section XI.M19, as modified by LR-ISG-2016-01. However, as currently proposed, with exceptions, the "Steam Generators" program is not fully consistent with the XI.M19 (as modified by the ISG). The staff notes the following: • The EPRI IAGL is mentioned on LRA p. B-79, but there is no statement referencing a report number or revision number. NUREG-1801 Section XI.M19, as modified by LR-ISG-2016-01, references Revision 4 of the IAGL (EPRI 3002007571) in Element 3 (Parameters Monitored/Inspected) and Element 6 (Acceptance Criteria). The latest revision of the IAGL (Revision 5 in 2021) could be referenced using an exception to Element 6. • The "Steam Generators" AMP basis document, LUM00020-REPT-047, refers to Revision 4 of the EPRI IAGL, which makes it consistent with NUREG-1801 as modified by LR-ISG-2016-01. An exception could be used to reference Revision 5. • Revision 4 of the EPRI IAGL is also referenced in implementing document STA-733, "Steam Generator Reliability Program." The staff did not	Please discuss the use of the EPRI IAGL in the AMP and other steam generator program documents, and how consistency is being established and maintained for the referenced EPRI guidelines among all of the Comanche Peak implementing documents.

			determine if this is the only implementing document that may need to have references to guideline documents updated.	
2	B.2.3.10	B-78	In LR Section B.2.3.10, the stated inspection intervals for the steam generator divider plates, channel heads, tubesheets, and tube-to-tubesheet welds are consistent with NUREG-1801 Section XI.M19, as modified by LR-ISG-2016-01. However, those intervals are not consistent with the current CPNPP technical specifications, as approved in License Amendment 182 (ML21321A349). Specifically, o The LRA states that visual inspections are performed at least 72 effective full power months (EFPM) or every third refueling outage for Unit 1, and 48 EFPM or every other refueling outage for Unit 2. o In the technical specifications approved in Amendment 182, visual inspections are performed at least every 96 EFPM for Unit 1, and at least every 54 EFPM for Unit 2 (with additional provisions for enhanced probe inspections). In addition, the inspection intervals stated in Section B.2.3.10 of the LRA are not consistent with the intervals stated in Section 4.3.2 of the AMP basis document, LUM00020-REPT-047.	Please discuss the intended inspection intervals for the SG divider plates, channel heads, tubesheets, and tube-to-tubesheet welds, and consistency among the Steam Generators program documents.
3	B.2.3.10	B-79 to B-85	LRA Section B.2.3.10, "Steam Generators," refers to tube repairs and tube sleeves. References to repairs and sleeves are also found in implementing documents for the Steam Generator Program. However, Comanche Peak Units 1 and 2 do not have sleeving or other tube repair methods approved in the technical specifications.	Please discuss the reason for referencing tube repairs and sleeves given they currently are not approved methods for addressing tube degradation.

4	Table 3.1.2-4	3.1-134 3.1-147 3.1-148	LRA Table 3.1.2-4, "Steam Generators – Summary of Aging Management Evaluation," identifies components such as divider plates, tubesheets, tube-to-tubesheet welds, and tube support plates as having a "Reactor coolant (internal)" environment. Because these are solid components with only exterior surfaces, it is unclear to the staff why an internal environment is identified.	Please discuss how an internal environment is defined for steam generator components such as divider plates, tubesheets, tube-to-tubesheet welds, and tube support plates.
5	3.1.2.2.11.2	3.1-16	LR-ISG-2016-01, Appendix D, contains revised guidance for NUREG-1800 (SRP-LR), Revision 2, for managing primary water stress corrosion cracking (PWSCC) of steam generator divider plate assemblies and tube-to-tubesheet welds. With respect to tube-to-tubesheet welds, the acceptance criteria for Section 3.1.2.2.11.2 for units with thermally treated Alloy 690 steam generator tubes depends on the tubesheet material. The guidance states that for units with thermally treated Alloy 690 tubes and Alloy 600 type tubesheet cladding, a plant-specific AMP is necessary unless certain conditions are met, such as a 22 percent minimum chromium content of the tube-to-tubesheet welds). For units with Alloy 690 tubes and tubesheet cladding material, a plant-specific AMP is not necessary. Section 3.1.2.2.11.2 of the SLRA states that the Unit 1 steam generator divider plate assemblies, tubes, and tube-to-tubesheet welds are Alloy 690 material and credits the Water Chemistry and Steam Generators AMPs for managing PWSCC. However, because the tubesheet cladding material is not identified, and the conditions requiring a plant-specific AMP are not addressed, the staff is unable to determine that the acceptance criteria are met for the Unit 1 tube-to-tubesheet welds.	Please clarify the statement that the Unit 1 <u>tube-to-tubesheet welds</u> are Alloy 690 material. The staff is seeking clarification of how the welds conform to the acceptance criteria in LRA Section 3.1.2.2.11.2, which are stated in terms of the <u>tubesheet cladding</u> material.
6	N/A	N/A	CPNPP Procedure NDE-7.10 (Rev. 16, 8/21/2019), "Steam Generator Tube Selection and Examination," Section 4.4.3, "Tube Examination Scope and Sampling Plans," has sampling plan classifications C-1, C-2, and C-3, which is	Please discuss the reference to SG tube examination sampling plans that were replaced by performance-based technical

			different than the scope of inspections required in the current technical specifications.	specifications, including the current CPNPP technical specifications.
7	N/A	N/A	CR-2018-002217 states that a qualification matrix was developed to identify qualification gaps due to the integration of Component Engineering (CE) into the Systems and Strategic Engineering department. The matrix identified CE positions that need a backup. Resolution for this item describes identifying a previous SG program owner who was an expert but no longer qualified. CR-2012-000075 includes a recommendation from INPO for improving aspects of program implementation. One item they identified was the high level of program owner turnover.	Please discuss the status and trends in addressing qualification gaps and high steam generator program turnover.
8			CR-2012-000075 includes a recommendation from INPO for improving aspects of program implementation. Item 3f stated that the Vendor NDE Level III used as the utility NDE oversight representative is not involved early enough during inspection planning, and that the NDE Level III utility representative does not sign off on the degradation assessment (DA), condition monitoring (CM) report, or operational assessments (OAs). Resolution for this item describes how the CPNPP Level III would be more involved in inspection planning and review. The resolution also states, “The representative will be reviewing and signing off on the Degradation Assessment, the Condition Monitoring report, and the Operational Assessments.”	Please discuss the relative roles and timing of the NDE Level III and program owner in the inspection planning and review, including the DA, CM, and OA reports.
9	N/A	N/A	As summarized in ML22020A178, the licensee and NRC staff discussed the fall 2021 steam generator tube inspection activities during the inspection. With respect to the detection of outside diameter stress corrosion cracking at a free span ding, the licensee described how a change	Please discuss the results of any corrective actions or other follow-up actions in response to that experience.

			in the eddy current signal had been detected initially in 2008. Subsequent inspections with bobbin probes in 2011, 2014, and 2017 resulted in dispositioning the indication without identifying it as a crack or performing special interest inspections to improve the ability to detect a crack-like signal in the presence of masking signals. During the call, the licensee indicated changes to the inspection program would be considered.	
10	B.2.3.10	B-83	The first paragraph for Unit 2 plant-specific Operating Experience states that six tubes were plugged during the fall 2021 inspection, and that these six tubes were plugged as a result of previously identified degradation mechanisms. However, according to the report from the fall 2021 inspection: • Two tubes were plugged on the basis that they are high-stress tubes, and no degradation was identified for these tubes. • One tube was plugged due to axial ODSCC at a freespan ding. This degradation mechanism was new to CPNPP, and the report states that is now an existing mechanism based on the fall 2021 inspection.	Please discuss the difference between the fall 2021 inspection report and the LRA with respect to the causes of tube plugging.
11	B.2.3.10	B-79	The last full paragraph on page B-79 has the following typographical error: "Installed plugged are routinely inspected and ……."	Please clarify the intended wording for this sentence, for example, "Installed plugs are routinely inspected and ….."
			Other Plant Specific TLAA – SG Tubes Metal Corrosion Allowance	
12	4.7.7	4.7.14	This section states: The FSAR Section 5.4.2B.5.4 Allowable Tube Wall Thinning Under Accident Conditions, which covers Unit 2, contains the following discussion of the corrosion of steam generator tubing:	Please clarify this calculation. Doesn't the assumed corrosion rate, equivalent to 0.080 mils thinning leave a conservative

			"The corrosion rate is based on a conservative weight loss rate for mill annealed Inconel tubing in flowing 650°F primary side reactor coolant fluid. The weight loss, when equated to a thinning rate and projected over a 40-yr plant life with appropriate reduction after initial hours, is equivalent to 0.080 mils thinning. The assumed corrosion rate of 3 mils leaves a conservative 2.2 mils for general corrosion thinning on the secondary side."	2.9 mils for general corrosion? (i.e., 3 mils – 0.080 mils = 2.92 mils)
13	3.1.2.2.11.1	3.1-13-3.1-14	LRA Section 3.1.2.2.11.1 states that the evaluation determined the 2014 EPRI report TR-3002002850 is applicable and bounding for CPNPP Unit 2 SGs. Based on LR-ISG-2016-01, this determination makes it unnecessary to have a plant-specific AMP for verifying the effectiveness of the Water Chemistry and Steam Generators AMPs. However, the determination that the 2014 EPRI report is bounding, as documented in Westinghouse LTR-CECO-21-081 P/NP Rev. 1, Question 10, states that the loads used in the 2014 EPRI report do not strictly bound the design and transient loads for CPNPP Unit 2. The limiting differential was then used to calculate tubesheet vertical displacements and compare them to the vertical displacements calculated for the limiting SG model in a 2007 EPRI report. The conclusion that the EPRI report is bounding for CPNPP Unit 2 despite the increased pressure differentials, appears to be based largely on the comparison of these vertical displacement calculations. It is unclear to the staff how LTR-CECO-21-081 demonstrates CPNPP Unit 2 is bounded by the 2014 EPRI report. For example: • It is not clear why vertical displacement is considered sufficient to determine whether the CP2 SGs are bounded by the 2014 EPRI report.	For Question 10 in LTR-CECO-21-081, please step through the analysis that demonstrates how EPRI report TR-3002002850 is applicable and bounding for the CPNPP Unit 2 SGs.

			• Section 4 of the 2014 EPRI report evaluated two cracking scenarios representing limiting cases: cracks propagating from the divider plate assembly into the channel head low alloy steel due to fatigue, and cracks propagating through the tube-to-tubesheet welds. The evaluation for crack propagation into the channel head low alloy steel was based on internal pressure loading, multiple thermal transients for the hot leg and cold leg, finite element stress analysis, and fatigue crack growth analysis. The conclusion that structural integrity of the channel head is not compromised based on a crack in the divider plate was based on the fatigue crack growth analysis.	
14 (Also submitted as Water Chemistry Question 4)	N/A	N/A	TR-2019-004928 describes a possible departure from EPRI PWR Secondary Water Chemistry GL with respect to the time to establish wet layup chemistry in the Unit 1 SGs in 1RF20. The TR description indicates the seven-day requirement in the GL was met, but the TR also includes a corrosion estimate for 13.73 days. The TR also refers to SG moisture separator manufacturing deviations, feedring repairs, and foreign objects in the Unit 1 feedwater rings, feedwater nozzles, and lower deck plates.	a) Please clarify the circumstances regarding the possible departure from EPRI guidelines and the conclusions from the evaluation of corrosion and other effects. b) Please describe the moisture separator manufacturing deviations and feedring repairs and discuss any effects on the present management of loose parts and foreign objects in the Unit 1 SGs.

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LRA Structural Scoping and Screening

Question Number	LRA/SLRA Section	LRA/SLRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	LRA Table 2.2-3	2.2-7	LR DWG LR-STRUCT-01 shows the SW Discharge Canal as in-scope for LR; however, LRA Table 2.2-3 indicates the canal is not in-scope for LR.	Please provide a justification for the canal not being in scope of LR or identify how the canal is age-managed.
2	LRA 2.4.8	2.4-23	Per LRA Section 2.4.8 (pdf pg. 279) "Traveling screens perform their function with moving parts, are "active" and are not subject to AMR." The staff agrees the overall traveling screen component is active; however, it is unclear why the actual screen portion of the traveling screen is considered active and how it will be age-managed.	Explain how the screen portion of the traveling screens will be age-managed or why age-management is unnecessary.

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LRA Section B.2.3.9, Bolting integrity

Question Number	LRA/SLRA Section	LRA/SLRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	Bolting Integrity Basis Document, Sections 4.3 and 4.4	Basis doc pg. 16 of 42	NUREG-1801 XI.M18, element 3, notes that this AMP monitors bolting for the effects of aging, including leakage, loss of material, cracking, and loss of preload/loss of prestress. Element 4 notes that periodic inspections are conducted for signs of leakage to ensure age-related degradation is detected. NUREG-2191 also notes that site procedures should include inspection parameters, such as lighting, distance, etc., to ensure an adequate inspection is performed. The AMP basis document notes that these aging effects are monitored by the AMP and references several documents (9.12, 9.62 – 9.65). The staff reviewed these documents; however, it was unclear how these documents captured clear guidance on how to actually implement the program and the appropriate inspections. Reference 9.65 provides general guidance on what should be reviewed during a walkdown, including a sample walkdown checklist; however, there is no clear guidance on how to conduct inspections.	Please identify specific sections in implementing procedures that explain how the appropriate inspections are conducted for this AMP. If this level of detail does not exist in current plant documents, please explain why an enhancement is not necessary for this element.
2	Bolting Integrity Basis	Basis doc pg. 20 of 42	NUREG-1801 XI.M18, element 6, notes that for non-ASME pressure retaining bolting, indications of aging should be dispositioned in accordance with	Please identify site documents that clearly define 'indications of aging' for bolted connections. If this level of detail does not exist in current plant

	Document, Section 4.6		the corrective action process (CAP). NUREG-2191 states that leaking joints do not meet acceptance criteria. The AMP basis document notes that indications of aging are dispositioned in accordance with the CAP and references Ref. 9.65. As noted above, Ref 9.65 provides general guidance on what should be reviewed during a walkdown but does not provide clear guidance on what would be considered 'indications of aging.'	documents, please explain why an enhancement is not necessary for this element.
3	LRA Tables 3.3.2-2, -6, -12 LRA Table 3.3-1	LRA pg. 3.3-139, 187, 279, 280 3.3-32	Several items in the LRA identify loss of preload for carbon or stainless-steel materials in a condensation or wastewater environment and reference generic Note H, which indicates the aging effect is not in GALL for this component, material, and environment. LRA Table 3.3-1, item 15 notes that the bolting integrity AMP manages loss of preload for steel and stainless-steel bolting. It is unclear to the staff why the LRA credits a note H for these items, instead of referencing Table 3.3-1, item 15.	Please explain why a note H is referenced for these items and why no GALL item is referenced.
4	Bolting Integrity Basis Document, Section 4.2	Basis doc pg. 15 of 42	NUREG-1801 XI.M18, element 2, notes that preventive measures include using bolting material that has an actual measured yield strength limited to less than 1,034 MPa (150 ksi). The AMP basis document Section 4.2 states that high strength bolts with actual yield strength of 150 ksi or greater are **not used** as closure bolting for pressure retaining components.	Please clarify whether any high strength bolts with actual yield strength of 150 ksi or greater are currently used as closure bolting. If high strength bolts have been used, discuss procedures to identify these high strength bolts and explain how they will be age managed.

			LRA Section B.2.3.9 notes that procedures will be enhanced to **<u>minimize any future use</u>** of bolting material with an actual yield strength greater than or equal to 150 ksi in portions of systems within the scope of the Bolting Integrity program. It is unclear to the staff whether high strength bolts with actual yield strength of 150 ksi or greater exist in the plant currently.	
5	Bolting Integrity Basis Document, Section 4.3	Basis doc pg. 16 of 42	NUREG-1801 XI.M18, Element 3, notes that this AMP should monitor cracking for high strength closure bolting if used. The AMP basis document Section 4.3 states that high strength bolts with actual yield strength of 150 ksi or greater are **<u>not used</u>** as closure bolting for pressure retaining components. CPNPP Element 3 does not include procedures to monitor cracking for high strength closure bolting. The information in CPNPP Element 3 appears to be inconsistent with the information in CPNPP Element 2 CPNPP regarding minimizing future use of high strength closure bolts.	If future use is to be minimized, as opposed to prohibited, please discuss the need to enhance CPNPP Element 3 to include monitoring for cracking of high-strength closure bolts.

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Sections:
AMP B.2.3.13 Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems
TLAA 4.7.4 Crane Load Cycle Limits
AMP: 24 and TLAA: 116.4

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	AMP B.2.3.13, TLAA 4.7.4, Basis Document Reports, FSAR, and Report DBD-ME-006, Rev. 40	Page B-96, Page 4.7-6, FSAR Table 17A-1 and Table 1-2 of Report	Basis Document Reports, LUM00020-REPT-050, Rev. 1, (AMP for Inspection of Load Handling Systems) and LUM00020-REPT-083, Rev. 3, (TLAA for Cranes) provided the lists of "load handling systems," in Section 4.1 and in Table 4.7.2-2 that are in scope for the LR (AMP/TLAA), respectively. As stated in Section B.2.3.13 of the LRA, *"Table 17A-1 in FSAR was reviewed to identify the cranes that are designed in accordance with CMAA-70, 1975, or considered equivalent to CMAA-70."* Report DBD-ME-006, Rev. 40, also identified "Drum Storage Area Crane (CPX-MESCDS-01)," in Table 1-2 as it does not require additional review and evaluation.	Discuss why "**Drumming Storage Area Crane**," is listed in Table 4.7.2-2 (TLAA) and in Table 17A-1 of the FSAR but was excluded from the scope of AMP in Section 4.1 of the Basis Document Reports, LUM00020-REPT-050, Rev. 1, and consequently, in Section B.2.3.13 of the LRA.
2	AMP B.2.3.13, TLAA 4.7.4, and Basis Document Reports,	Page B-96, Page 4.7-6,	Basis Document Reports LUM00020-REPT-050, Rev. 1, (AMP for Inspection of Load Handling Systems) and LUM00020-REPT-083, Rev. 3, (TLAA for Cranes) provided twenty-five (25) in Section 4.1 and twelve (12) in Table 4.7.2-2 "load handling systems," that are in scope for the LR (AMP/TLAA), respectively. 2. Furthermore, Report DBD-ME-006, Rev. 40, in Table 1-1 identified over twenty-eight (28)	Discuss why some of the "load handling systems," listed in Section 4.1 of the Basis Document Report, LUM00020-REPT-050, Rev. 1, were excluded from the scope in of the Basis Document Report, LUM00020-REPT-083, Rev. 3, (TLAA for Cranes). 2. Discuss why all the "overhead load handling systems," identified in Table

	and Report DBD-ME-006, Rev. 40		“overhead load handling systems,” with potential for heavy loads drop on or near spent fuel or systems required for plant shutdown or decay heat removal. **Note**: The following “overhead load handling systems,” in Table 1-1 of the report DBD-ME-006, Rev. 40, were not considered in the scope of AMP/TLAA: 1.b - Overhead Crane 130Ton Hook Critical Lifting Devises 26 - Mobile Crane 27.b – VCT Special Lifting Devises	1-1 in Report DBD-ME-006, Rev. 40, were not considered in the scope of AMP/TLAA.
3	OE List in portal	Excel File: XI.M23.xlsx	In the electronic portal, the applicant provided a table (file XI.M23.xlsx) listing information related to OEs on the “load handling systems (XI.M23),”. Based on the cursory check of the OEs against the Basis Reports (AMP/TLAA) - as listed below, the staff identified the “load handling systems,” that were not considered in the scope of AMP/TLAA. <u>Examples</u>: 1. CR-2014-001643: The Galion Mobile Crane – XI.M23, related to refueling. 2. CR-2012-012795: CPX-MEMHMB-02, Main Shop Electric Hoist – XI.M23, related to refueling. 3. CR2015-003675/CR2015-003194/CR2016-006515/TR-2016-001599/CR-2015-010930: CPX-MESCCW-01, Circulating Water Intake Structure Gantry Crane X-01 - XI.M23, related to refueling.	Describe why some of the “load handling systems (XI.M23),” listed in OEs (file XI.M23.xlsx) were not in the scope of AMP/TLAA as identified under “Examples,” in the adjacent column.

			4. TR-2016-005487/TR2018-001820: TCX-FHSCMC-01, Fuel Handling Refueling Manipulator Crane 2-01 - XI.M23, related to refueling. 5. TR-2018005800: Galion Crane, 150 Crane 15T - XI.M23 - related to refueling. 6. TR-2018-005799: E0249-20 Ton Glove Crane - related to refueling. 7. CR-2018-005660: TBX-FHSCFB-01, Fuel Handling Refueling Manipulator Crane 1-01 - XI.M23, related to refueling.	
4	TLAA 4.7.4, and Basis Document Report	Page 4.7-6	Basis Document Report, LUM00020-REPT-083, Rev. 3, (TLAA for Cranes), provides a "Project Report Preparation Checklist," that the Item 3 asking *"Have the appropriate review forms/checklists been completed?"* In response, both, "Yes," and "N/A" were checked. **Note**: This issue may be editorial!	It is not clear why two (2) out of three (3) options were chosen!
5	TLAA 4.7.4, and Basis Document Report	Page 4.7-6	In Section 7.1, "LRA Section 4.7.2," of Basis Document Report, LUM00020-REPT-083, Rev. 3, (TLAA for Cranes), refers to Section 4.7.2 for TLAA in the LRA. However, it is Section 4.7.4 in the LRA. **Note**: Editorial.	Refer to the appropriate section of "4.7.4," from LRA in Basis Document Report, LUM00020-REPT-083, Rev. 3, (TLAA for Cranes),
6	OE List in portal	Excel File: XI.M23.xlsx	In the electronic portal, the applicant provided a table (file XI.M23.xlsx) listing information related to OEs on the "load handling systems (XI.M23)," The trending report, TR-2021-006651, describes: *"This IR is to track an STA-748 equipment reliability task team effort on CPNNP cranes. The health of the cranes so far in 2rf19 has <u>not</u> been optimal."* XI.M23, related to refueling.	Elaborate identified specific issues related to the cranes that are not considered in optimal conditions.

7	AMP B.2.3.13,	Page B-97, Under Operating Experience, and "<u>Plant-Specific OE</u>," Forth bullet	Forth bullet, the paragraph starts with *"In November of 2018, during the inspection of the Refueling Machine crane hoist..."* It is not clear whether the "Refueling Machine crane hoist," is in the scope for AMP/TLAA. [CR-2018-005660] **Note**: CR-2018-005660 was also identified above in Question 3, Item 7.	Confirm whether the "Refueling Machine crane hoist," is in the scope for AMP/TLAA. is the crane ID #: TBX-FHSCFB-01?
8	Procedure TPMDA-MDA-402-13	Page 3 Effected page 12 Page 12, Item 6.1.2.3	On page 3 of Procedure MDA-402-13, states *"Update Step 6.1.2.3 for load testing overhead hoists based on latest edition of ASME B30.16."* The Step 6.1.2.3 on page 12 in Procedure MDA-402-13 appears to be not updated as stated on page 3 above. **Note**: In Section 4.7, "Corrective Actions," of the Basis Document Reports, LUM00020-REPT-050, Rev. 1, refers to reference 9.24 "MDA-402, Rev. 13, Control of Load Handling Equipment," for the implementation of ASME B30.16. Furthermore, in AMP B.2.3.13 of the LRA, Element 3, "Parameters Monitored or Inspected," and Element 6, "Acceptance Criteria," will be enhanced to include ASME B30.16.	Discuss why Step 6.1.2.3 on page 12 in Procedure MDA-402-13 was not updated.
9	N/A	N/A	It is not clear whether CPNPP, Units 1 and 2, has any "monorails and underhung cranes," at the site.	Clarify whether CPNPP, Units 1 and 2, has any "monorails and underhung cranes," at the site that can be considered in the APM/TLAA of LRA.

<u>Comanche Peak Nuclear Power Plant, Units 1 and 2</u>
<u>License Renewal Application (LRA) Breakout Audit Questions</u>

LRA Section B.2.3.29 AMP: ASME Section XI, Subsection IWE

Question Number	SLRA Section	SLRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	3.5.2.2.1.3.1, B.2.3.29, A.2.2.29	3.5-21, B-173, A-24	The further evaluation in the referenced LRA section states in part: *"An evaluation of the acceptability of inaccessible areas is required **when conditions exist in augmented areas** that could indicate degradation could also exist or could have extended into the inaccessible areas."* Inaccessible areas are not necessarily always located adjacent to locations that are or may be identified for augmented examination. LRA Section B.2.3.29 claims the LRA IWE program with enhancements will be consistent with the GALL-LR program XI.S1. The LRA statement cited in the 1st paragraph above appears to be inconsistent with corresponding statements in LRA B.2.3.29, A.2.2.29 and the "scope of program" element of GALL-LR AMP XI.S1, which states the following with regard to managing aging effects in inaccessible areas: "…the licensee is to evaluate the acceptability of inaccessible areas ***when conditions exist in accessible areas*** that could indicate the presence of or result in degradation in inaccessible areas."	a) Explain and reconcile the noted potential inconsistency in the referenced LRA description and the "scope of program' element of GALL-SLR AMP XI.S1 with regard to addressing acceptability of inaccessible areas.

2	B.2.3.29 & Table A-3, Item 31(a)	B-174 & A-79	To establish consistency with the GALL-LR Report AMP XI.S1, the LRA AMP includes the following enhancement to the “preventive actions” program element: *“Reconcile the preventive actions in NUREG-1339, EPRI NP-5769, and EPRI TR 10423 with the existing procedures and practices for structural bolting.”* The “preventive actions” program element of the GALL-LR AMP XI.S1 states, in part: *“The program is also augmented to require that the selection of bolting material installation torque or tension and the use of lubricants and sealants are in accordance with the guidelines of EPRI NP-5769, EPRI TR-104213, and the additional recommendations of NUREG-1339 to prevent or mitigate degradation and failure of structural bolting.”* • The language of the enhancement and related LR commitment appear to be very general and non-specific, lacks clarity with regard to the specific objective of the enhancement, and it is unclear how it would make the program element consistent with the GALL-SLR.	a) Discuss potential revised language to the enhancement and related LR commitment that provides specific clarity of actions and objective in a manner that demonstrates consistency with the “preventive actions’ program element of the GALL-LR AMP XI.S1.
3	B.2.3.29 & Table A-3, Item 31(b)	B-174 & A-79	In consideration of operating experience and laboratory examinations that show that the use of molybdenum disulfide (MoS2) and other products containing sulfur as a lubricant is a potential contributor to stress corrosion cracking (SCC) especially in high-strength bolting, the LRA includes the following enhancement to the preventive actions program element*: “Prohibit the use of molybdenum disulfide or other sulfur containing lubricants for structural bolts.”* • It is not clear if MoS2 or other lubricants containing sulfur have been or will used in	a. Confirm or clarify whether or not MoS2 or other lubricants containing sulfur have been or will be used prior to the PEO for high strength bolting (actual tensile strength greater than 150 ksi) within the scope of the LRA B.2.3.29 (IWE) AMP. b. If used in response to a), discuss how the potential for SCC in such bolting will be adequately managed during the PEO as required by 10 CFR

			structural bolting in the scope of the LRA AMP prior to entering the period of extended operation (PEO). If so, it is not clear how the potential for SCC in such bolting be adequately managed during the PEO.	54.21(a)(3). Also, discuss related changes that will be made to the LRA AMP.
4	B.2.3.29	B-173 & B-174	Section 4.3.2 on page 16 of PBD (LUM00020-REPT-066, Rev 1) for AMP Element 3 (Parameters Monitored or Inspected) states under "Category E-C": *"The Containment Areas Visible Surfaces (Item E4.11) will be inspected using the general visual examination method to examine the parts defined during the third interval based on IWE-1241 requirement and Category E-A examinations."* • Contrary to the claim that program element 3 of the LRA AMP is consistent without exception to GALL-SLR AMP XI.S1, the examination method specified for Examination Category E-C (Item 4.11) in Table IWE-2500-1 is VT-1 method, and not "general visual examination' method cited above, and therefore, inconsistent or taking exception with the ASME Code Section XI, Subsection IWE and GALL-LR AMP XI.S1.	1. Discuss and justify how the LRA AMP program element 3 is consistent with the GALL-SLR AMP regarding Examination Category E-C. 2. If determined to be inconsistent, discuss what revision will be made to the program element 3 in the PBD to be consistent with GALL-LR as claimed, or justify the exception to the GALL-LR AMP XI.S1.
5	B.2.3.29 & PBD Section 4.4.2	B-173 & PBD page 18	PBD (LUM00020-REPT-066, Rev 1), Section 4.4.2, states on page 18: *"The frequency of the examinations on the structural components is specified in the 2007 edition through 2008 addenda of ASME Section XI, Subsections IWE [Ref. 9.20, Table IWE-2500-1]."* • Contrary to the above statement in the PBD, the GALL-LR AMP XI.S1 requires the examination methods, frequency and scope to be consistent with the code edition(s) during the PEO in accordance with 10 CFR 50.55a, and is therefore inconsistent with the GALL-LRA AMP	a. Discuss how the "Detection of Aging Effects" program element of the LRA AMP described in the PBD is consistent with GALL-LR AMP XI.S1, as claimed, when the code edition used during the PEO will not be the 2007 Edition with 2008 addenda. Also, discuss the changes that may need to be made to the PBD to achieve consistency.

			XI.S1. There is no mention of 10 CFR 50.55a in Section 4.4.2 of the PBD.	
6	B.2.3.29 & Table A-3, Item 31(c), A.2.2.29	B-174, A-79, A-24	To establish consistency with the GALL-LR Report AMP XI.S1, the LRA AMP includes the following enhancement to the “detection of aging effects program element: *“Monitor cracking due to cyclic loading of non-piping penetrations and DMWs between the stainless steel piping and the steel sleeve/forging by periodic supplemental surface examinations* ***consistent with the frequency of this AMP and the 10 CFR Part 50, Appendix J AMP.****”* • Since the surface examinations are supplemental to the routine examinations required by Subsection IWE and intended to be plant-specific, the GALL-LR AMP does not specify the interval for these examinations. • The LRA does not explicitly state what specific non-piping penetrations are included in this enhancement, and the specific supplemental examination methods that will be used and its capability to detect cracking. • It is not clear how the supplemental surface examinations are related to the 10 CFR 50, Appendix J AMP which is a performance monitoring program, whereas the IWE program is a condition monitoring program.	a. Clearly state in the enhancement and the related LR commitment the frequency at which the proposed supplemental surface examinations to detect cracking will be performed. Also, justify the sufficiency of the examination frequency to provide adequate management of the aging effect. b. State the specific non-piping penetrations that are subject to periodic supplemental surface examinations by this enhancement. Also, state the specific supplemental examinations methods (e.g., MT, PT, E-VT1 etc.) capable of detecting cracking that will be used. c. Clarify why the frequency of the proposed supplemental examination is related to 10 CFR 50 Appendix J AMP cited in the enhancement. If Appendix J tests will be credited to detect cracking due to cyclic loading for certain components, identify these components and the type of Appendix J test capable of detecting crack that will be used, including the frequency of the test. d. Discuss conforming changes, if any, that may need to be made to the LRA

				AMP and/or its FSAR supplement, consistent with the above requests.
7	B.2.3.29 & Table A-3, Item 31(d)	B-174 & A-79	The "detection of aging effects" program element includes an enhancement of a pre-PEO supplemental one-time inspection, using methods capable of detecting cracking, of a representative sample (4 penetrations and 1 transfer tube) to confirm the absence of SCC. • The enhancement does not state the method(s) capable of detecting cracking that will used for the inspection. • There is no information provided of the adequacy of the representative sample, especially considering the population of SS penetrations or DMWs associated with high temperature piping it represents. • The enhancement does not address what additional actions will be taken if the absence of SCC is not confirmed or if SCC is detected from the one-time examination.	a. Clearly state in the enhancement the method(s) capable of detecting cracking that will be for the one-time inspection. b. Discuss the adequacy of the proposed representative sample discussing the total number of the population of penetrations/DMWs it represents. c. Discuss and include in the enhancement or as a separate enhancement what additional actions (e.g., sample expansion, periodic examination with method and frequency, etc) will be taken if absence of SCC is not confirmed or if SCC is detected. d. Clarify and reconcile if there are common components that are subjected to the actions in both LR Commitments 31(c) and 31(d) discussed in Questions 6 and 7 above.
8	B.2.3.29	B-175	The first bullet under "Plant-Specific OE" states, in part: *"…. There is no documented evidence of degraded conditions like bulges in the liner plate, etc.; …"* Contrary to the above statement, the second full bullet on LRA page B-175 beginning with NRC IN 97-10, states, in part: *"… A bulge to the Unit 1 liner was documented and evaluated. …"*	a. Clarify and correct the cited inconsistent statements in the plant-specific operating experience description. b. Clarify if there is physical evidence of age-related degraded conditions (e.g., corrosion, moisture barrier degradations, etc) of significance in

				the containment liner or other containment pressure-retaining boundary components in the scope of the LRA AMP for Comanche Peak Units 1 and 2.
9	B.2.3.29	B-175	The conclusion paragraph under the discussion of "Industry OE" states on LRA page B-175: *"…. These examples provide objective evidence to confirm that* ***station testing procedures*** *are effective to maintain containment integrity."* It is not clear why the statement refers to "station testing procedures" when the industry OE discussed or evaluated therein relate to findings of degradation identified as part of the containment ISI program, which is essentially a condition monitoring program and not a testing or performance monitoring program.	c. Clarify and correct, if needed, how the referenced conclusion statement under "Industry OE" in LRA Section B.2.3.29 is supported by the industry OE discussed above the statement specifically with regard to the reference to station testing procedures.
10	B.2.3.29	B-175	The LRA plant-specific OE description for the LRA AMP includes only a single factual plant-specific OE example, which is of a loose bolt found in 2011 on the handwheel gear box of the Unit 2 Personnel Emergency Air Lock. Based on this, the LRA concludes that: *"The above OE provides objective evidence that the ASME Section XI, Subsection IWE AMP has been and will continue to be effective in ensuring that component intended functions are maintained consistent with the CLB through the PEO."* • The factual description of plant-specific OE provided in the LRA appears to not adequately support the stated conclusion that it provides objective evidence of the effectiveness of the AMP in adequately managing aging effects such that intended functions are maintained through the PEO.	a. Confirm or clarify if the plant-specific OE of a loose bolt is the only and most significant plant-specific OE of aging degradation of the containment pressure-retaining boundary components within the scope of the LRA IWE AMP. If not describe other significant OE that would provide objective evidence in support the LRA conclusion. b. Are there components or surfaces identified for augmented examination under the CISI (IWE) program in the past or currently. If so, describe the areas and conditions based on which they were identified for augmented examination.

			- In support of its OE conclusion, the LRA does not provide explicit supporting statement(s) of the physical material condition of the containment pressure-retaining boundary components within the scope of the AMP observed based on condition monitoring through the containment inservice inspection program and the maintenance rule program. - Section 3.7 "Augmented Examinations" of the "Interval 4 CISI Program Plan – Section XI" on the ePortal, states, in part: ".... *Areas previously identified by the CPNPP Coatings Program as "areas/items of specific interest" are also considered to require special attention for CISI examination. These areas will be further assessed during the fourth interval of examinations and may be designated as augmented areas by the CISI Program Plan."*	c. Confirm whether the containment pressure-retaining boundary components under the scope of the LRA AMP have had no plant-specific aging degradations (e.g., corrosion, moisture barrier degradations etc.) of significance at Comanche Peak Units 1 and 2. d. Discuss the physical material condition of the containment pressure-retaining boundary components under the scope of the LRA AMP based on condition monitoring inspections performed under the CISI (IWE) and maintenance rule programs. If supported, provide supplemental statement(s) based on the observed material condition that would support the stated LRA OE conclusion.
11	B.2.3.29	B-173	LRA Section B.2.3.39 states on page B-173: *"Final Reports are generated for engineering evaluations in accordance with Code Case N-532-4."* This is also reflected in the PBD on the ePortal. - However, Appendix B of the CPNPP Containment ISI Program Plan provided on the ePortal for both the 3rd as well as 4th CISI Intervals state: "No [Section XI] Code Cases Have Been Adopted by CPNPP." - The most recent revision of this code case incorporated by reference in 50.55a via RG 1.147 is N-532-5 dated 1/4/2011. If adopted, 10 CFR 50.55a(b)(5) requires the latest revision of	a. Clarify the referenced discrepancy between LRA Section B.2.3.29 and the CPNPP CISI Program Plans with regard to use of stated code case N-532-4. b. Clarify if the referenced code case or any other Section XI code cases will be adopted for the LRA AMP for the PEO, and if so clarify how the appropriate revision to be used will be determined.

			code cases incorporated by reference therein to be used in subsequent inspection intervals.	
12	Interval 4 CISI Program Plan on ePortal: Section 3.6, 3.12, 3.15, and Tables 3.12-1 & 3.15-1	CISI Plan pages 3-10 thru 3-16; 3-24 thru 3-30	• 4th Interval CISI Plan document on the ePortal, Section 3.6 *“Accessible/Inaccessible Areas” states, in part: “Inaccessible areas are exempt from examination and are shown on CISI Drawings titled “Metal Containment Inaccessible Areas Sh. 1 of 1” in Appendix E (CPNPP1) and Appendix F (CPNPP2).”* (For drawings, refer pdf pages 76 & 101). It is not clear to the staff what the areas indicated as inaccessible on these drawings are and why they are inaccessible. • 4th Interval CISI Plan document on the ePortal, Tables 3.12-1 (U1) and 3.15-1 (U2) list Containment Penetrations for Examination Category E-A, Item 1.11 general visual examination. The staff notes in the remarks column that some of these penetrations are identified as having Insulation or Radiant Energy Shielding. It is not clear to the staff the process used in the field to adequately conduct the general visual examinations of the penetrations with Insulation or Radiant Energy Shielding.	a. Briefly discuss the areas indicated as inaccessible on the referenced CISI Plan drawings on pdf pages 76 & 101, and briefly explain why they are inaccessible. b. Briefly discuss the process used in the field to adequately conduct the general visual examinations of the penetrations, listed in CISI Plan Tables 3.12-1 & 3.15-1, with Insulation or Radiant Energy Shielding.
13 *(TRP 77)*	Table 3.5-1, Item 3.5-1, 052 and FE Section 3.5.2.2.2.4	3.5-54, 3.5-34	The referenced AMR item and FE relates to cracking due to SCC, and loss of material due to pitting and crevice corrosion of <u>*stainless steel*</u> material components of Group 7 (concrete with liner) & Group 8 (steel) tanks in a <u>*water-standing*</u> environment. The Table 3.5-1, 052 line item states in the Discussion column: *“Not Applicable.”* It also provides some additional discussion for Group 7 tanks and for Group 8 tanks references item 3.3-1, 067and the Fire Water System AMP. Further, LRA	a. Clarify and justify the “non-applicable” claim for item 3.5-1, 052 in a manner that objectively demonstrates that the component material and environment required for the aging mechanism/effect(s) does not exist. b. Clarify what aging effect(s) in the line item 3.5-1, 052(cracking due to SCC, loss of material due to corrosion) is applicable and identify the

			3.5.2.2.2.4 states, in part: "Loss of material due to pitting and crevice corrosion in the treated borated water…..in Group 7 tanks will be managed by the Water Chemistry (B.2.3.2) AMP and One-Time Inspection (B.2.3.19). Cracking due to SCC ……does not require management as temperatures of the water is less than 140°F." • From the descriptions in the LRA, there is lack of clarity regarding the non-applicability claim for item 3.5-1, 052, which aging effect(s) covered by the line item is not applicable, and which aging effect(s) is managed by the referenced AMPs, and which specific corresponding Table 2 AMR items manage the applicable aging effects. Also, referring to the harsh environment threshold description for SCC in stainless steel on page IX-14 of GALL-LR, Rev. 2, the LRA does not appear to provide discussion on the presence of a harsh environment or operating experience with regard to SCC.	corresponding Table 2 AMR line items that manages the aging effect

<u>Comanche Peak Nuclear Power Plant, Units 1 and 2</u>
<u>License Renewal Application (LRA) Breakout Audit Questions</u>

LRA Section B.2.3.30, ASME Section XI, Subsection IWL AMP

Question Number	LRA/SLRA Section	LRA/SLRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	A.2.2.30	A-25	The FSAR summary lists types of concrete degradations but does not identify the appropriate reference documents (ACI 349.3R).	Explain why these documents are not included in the FSAR summary.
2	LRA Table 3.5-1, 016	3.5-41	Items 3.5-1, 016 and 024 note that increase in porosity and permeability is not an aging effect requiring management because the concrete is not exposed to acidic solutions. Item 3.5-1, 067 notes that the SMP will manage this aging effect for similar concrete. It is not clear why the aging effects in items 016 and 024 are not applicable. At a minimum, the staff would expect to see an AMP identified to look for these aging effects in accessible areas, even if they are generally not expected to occur.	Explain why this aging effect is not inspected for in accessible areas of concrete structures or update the LRA to credit these items and identify an AMP to manage aging.
3	IWE/IWL Final Report (1RF22)	Pg. 38	The last IWL ISI report (spring 2022) identifies several locations where an area was accepted by evaluation of the Responsible Engineer. The report notes that these evaluations are captured in EV-TR-2022-002161-1 and EV-TR-2022-002161-2.	Please post these documents on the portal for staff review. Be prepared to discuss how these evaluations meet the acceptance criteria guidance of IWL.
4	TX-ISI-IWL Rev. 8		The IWL inspections are implemented by Westinghouse PROP procedure TX-ISI-IWL.	Please verify this document is on the PROP portal or have it added to the portal if it is not already there.

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section B.2.3.31. ASME Section XI, Subsection IWF

Question Number	LRA Section	LRA Page	Background/Issue (As applicable/needed)	Discussion Question/Request
1	B.2.3.31 Table A-3	B-180 A-80	LRA Section B.2.3.31 provides several enhancements and a commitment (Commitment 33) that those enhancements will be implemented to the existing ASME Section XI IWF AMP NLT six months prior to entering the PEO. The staff could not locate current procedures for review. Please make these available (preferably prior to breakout) to aid discussion in understanding the proposed enhancements rendering the AMP consistent with that of GALL-LR, Revision 2, AMP, XI.S3.	Please make available current procedures for review and further discussion as noted in Background/Issue.
2	B.2.3.31	B-182 B-183	The OE Section of LRA AMP B.2.3.31 states that during 2RF19 Fall 2021 and 1RF22 Spring 2022 Outages of Units 2 and 1, respectively, the RV supports were found acceptable but with minor issues of peeling and flaking paint and boric acid deposits. It is not clear what is the current status of the supports and how the aforementioned issues were resolved (particularly of boric acid deposits) to make them acceptable.	1. Please provide pictures taken during the inspections of the RV supports indicating the described in LRA AMP OE “minor” issues. 2. Please discuss resolution of issues noted in this question.
3	B.2.3.31	B-182	The LRA AMP in its OE Section states that the The Unit 2 supports were reinspected after a failure of the Material Stress Improvement Process (MSIP) hardware, with no apparent change from previous inspections. It is not clear how the MSIP is relevant to the structural integrity for the noted RV supports. Was there an expectation that the MSIP failure could impact the structural integrity of the supports? If so, shouldn’t the MSIP hardware be included in the scope of LRA AMP B.2.2.31?	1. Please clarify the relevance of the MSIP hardware to the structural supports (provide documentation as necessary). 2. Indicate the importance of the hardware regarding structural integrity of the noted RV supports.

4	B.2.3.31	B-182	Boric acid solution could lead to corrosion of carbon or low alloy steels with rates potentially reaching 1 in or greater annually. These corrosion rates are exacerbated particularly when there is a substantial loss water by evaporation and/or wet/dry oxygenation cycles. The increased concentration of the acid then leads to loss of material that is an applicable aging effect that requires management. CCNP CR-2021-003041 states that for 28 years the RV supports were not examined as they "were determined to be inaccessible." CPNP 2RF19 Fall 2021 ISI Final Report dated October 29, 2021 and CPNP 1RF22 ISI Final Report dated May 5, 2022 state that remote VT-3 examination performed with a fiberscope and flashlight, found the supports acceptable despite the existence of corrosion and boric acid deposits. It is not clear how the boric acid was deposited on the columns and whether the deposition was/is reoccurring or was it a onetime event. Given that the VT-3 exams were remote and the challenges of the process in estimating corrosion levels, it is not also clear how the applicant determined that the observed corrosion at the supports was limited with no further action of cleaning/corrective measures needed to satisfy requirements of the ASME Code Section XI and 10CFR Part 50 and Appendix B, Criterion XVI. In addition, it is not clear whether the minimum design code-required RV column support section thickness and integrity of bolting/welding connections had been maintained or will be maintained for the period of extended operation.	1. Discuss how boric acid was deposited on the RV columns and state its frequency of deposition, if any. 2. Clarify how CPNPP determined that the boric acid deposits at the supports did not constitute a non-conforming condition or did not necessitate cleaning/corrective action(s) otherwise required by ASME Section XI, Subsection IWF-3410(a) requirements, particularly when the performed VT-3 inspections were remote. 3. If boric acid corrosion was/is a reoccurring event, there could be a measurable loss of material on the RV steel supports. Delineate measures taken and to be taken to maintain the RV supports free of corrosion boric acid accumulation so that they remain acceptable for service for the period of extended operation, consistent with acceptance criteria of ASME Code Section XI, Subsection IWF 3410(a) included in GALL-LR Report, Revision 2, AMP XI.S3, "Acceptance Criteria" program element.

<u>Comanche Peak Nuclear Power Plant, Units 1 and 2</u>
<u>License Renewal Application (LRA) Breakout Audit Questions</u>

LRA Section B.2.3.32 AMP: 10 CFR Part 50, Appendix J

Question Number	SLRA Section	SLRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	B.2.3.32; 3.5.2.1.1	B-185; 3.5-3	The SLRA and PBD (LUM00020-REPT-069, Rev 1, Section 4.1.2) state that some components are excluded from local leak rate testing (LLRT) under the CLB (as shown in FSAR Table 6.2.4-2). In those cases, aging effects associated with those components are managed by the following AMPs (list provided therein) • ASME XI ISI – IWB, IWC, IWD • ASME XI ISI – IWE • Water Chemistry • Closed Treated Water Systems • One-Time Inspection • External Surfaces Monitoring • Boric Acid Corrosion • Flow-Accelerated Corrosion • Fatigue Monitoring It is not clear to the staff where and how these excluded components are included in applicable SLRA Table 2s to demonstrate adequate aging management of such excluded components. Additionally, SLRA Section 3.5.2.1.1 "Containment	a. Walk through on Teams and discuss specific examples of where and how the containment pressure-retaining boundary components excluded from LLRTs are captured for aging management in the applicable SLRA Table 2s by the listed AMPs. That is, identify the specific line items in the applicable Table 2s that include the exempted component and each of the credited AMP(s). b. Explain if and how AMPs such as ISI – IWB, IWC, IWD; Closed Treated Water Systems, External Surfaces Monitoring, Flow-Accelerated Corrosion, Fatigue Monitoring credited for containment pressure-retaining boundary components excluded from Appendix J LLRTs include these components within its scope.

			Buildings" on page 3.5-3 does not include several of the AMPs listed above among the AMPs that manage aging effects for the containment building components. The staff needs this information to verify consistency of the "scope of program" element of the 10 CFR 50, Appendix J AMP will the GALL-LR Report.	
2	TS 5.5.16, Amendment 173; LRA B.2.3.32	5.5-14, B-184	TS 5.5.16.a states: "*A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in* ***NEI 94-01****, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,"* ***Revision 3-A****, dated July 2012, and the conditions and limitations specified in* ***NEI 94-01, Revision 2A****, dated October 2008, as modified by the following exceptions:"* However, inconsistent with the above TS requirements, the program implementing procedures STA-743 (Rev 2-5), TSP-743 (0-3), and PPT-S2-7014 (Rev. 1) continue to reference RG 1.163 (Rev. 0 Sept 95), NEI 94-01 (1995, Rev 0) and ANSI N58.2-1994 as the implementing documents.	Explain the inconsistency between TS 5.5.16 and the referenced program implementing procedures with respect to the implementing documents for the program and whether changes are required to the implementing procedures.
3	B.2.3.32	B-184	LRA states that the 10 CFR 50, Appendix J program is an existing performance monitoring program. The SLRA does not provide a summary of the results of the most recent U1 ILRT that was scheduled for April 2022.	To verify adequate performance regarding the containment leakage rate, provide a summary of the results and date of the most recent ILRT for Unit 1.

4	B.2.3.32	B-186 thru B-188	LRA B.2.3.32, under Plant-Specific OE, states, in part: *"… In addition AMPs, such as the 10 CFR Part 50, Appendix J AMP, will receive effectiveness reviews every 5 years or as appropriate, in accordance with the guidance in NEI 14-12."* However, no documentation of recent program health/effectiveness self-assessment(s) for the AMP is discussed or provided on the ePortal.	Provide 3 most recent program health self-assessments for the LRA B.2.3.32 AMP.

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section B.2.3.33, Masonry Wall

Question Number	LRA/SLRA Section	LRA/SLRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	Basis Document LUM00020-REPT-070 Section 4.3	7 of 20	GALL AMP XI.S5, Masonry Walls, element 3, notes that "gaps between supports and masonry walls" should be monitored. GALL-SLR also mentions cracking or loss of material at mortar joints. Section 4.3 in the basis document references a degradation mechanism checklist for masonry walls which makes no mention of gaps between supports or mortar joints.	Please discuss why mortar joints and gaps between supports are not discussed in the implementing procedure.
2	Basis Document LUM00020-REPT-070 Section 4.5	9 of 20	Element 5 of the basis document identifies several recommended actions from GALL-SLR and notes that these actions are not necessary to meet the GALL guidance but are warranted and addressed in Section 7 of the basis document. It was unclear to the staff how these items were addressed in Section 7.	Please explain how these additional actions are being addressed or why it is unnecessary to implement them for the PEO.
3	LRA Table 3.5-1, 071	3.5-62	Item 3.5-1, 071 addresses freeze-thaw for masonry walls and is noted as not applicable. Comanche Peak is in a moderate weathering environment. The GALL Report recommends aging management for this aging effect for accessible areas in this weathering environment.	Please explain why this aging effect is not applicable to masonry walls exposed to an outdoor environment.

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section B.2.3.34: Structures Monitoring

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	Table 3.5.2-15	3.5-188	***Note: This breakout question also applies to TRP 27, Fire Protection.*** LRA Table 3.5.2-15 includes silicate radiant energy shield, subliming compound, ceramic fiber/blanket, and stainless steel insulation and wrap exposed to outdoor air. This table also includes elastomer penetration seals and carbon steel and stainless steel penetration sleeves exposed to outdoor air. Loss of material of the stainless steel insulation and wrap and penetration sleeve exposed to outdoor air in LRA Table 3.5.2-15 are managed by the Structures Monitoring program (AMR Item 3.5-1, 093) and cites plant-specific Note 5, which states, “Relative to stainless-steel components located outdoors, the Structures Monitoring (B.2.3.34) AMP is focused on areas with potential for frequent or prolonged water pooling and communicates with the Fire Protection (B.2.3.15) AMP as warranted.” This plant-specific note does not provide details on how the Structures Monitoring program is adequate to manage the aging effects for the stainless steel insulation and wrap and penetration sleeve to ensure the Fire Barrier intended function is maintained. The staff notes that the discussion of AMR Item 3.5-1, 093 states, “For stainless steel and aluminum, the focus	1. Please discuss where these insulations and wraps, penetration seals, and penetration sleeves are located to be exposed to outdoor air. If these fire barriers are not protected from weather, please discuss any impacts on the aging effects to be managed. 2. Please discuss what the specific ceramic fiber/blanket materials are. 3. Please discuss how the Structures Monitoring program is adequate to manage the aging effects for the stainless steel insulation and wrap and penetration sleeve to ensure the Fire Barrier intended function is maintained. 4. Please discuss how the program descriptions and procedures for the Structures Monitoring and Fire Protection programs reflect this. *Please note: Note E is an error.*

			is on areas where water could pool or get within insulation jacketing." The staff did not find information related to these stainless steel insulation and wrap and penetration sleeve in Structures Monitoring or Fire Protection program descriptions or documents on the portal *Also Note: Table 2 items in Table 3.5.2-15, associated with NUREG-1801 Item III.B2.TP-6 and AMR item 3.5-1, 093, lists stainless steel penetration sleeve and insulation and wrap, which their aging effects are managed by the Structures Monitoring program. It cites Notes E, and C, respectively.* *The staff verified that NUREG-1801 Item III.B2.TP-6 uses the Structures Monitoring program,* ***Note E is an error****.*	
2	B.2.3.34	B-192	The GALLLR Report AMP XI.S5 states that the steel edge supports and steel bracings for masonry walls are considered component supports and aging effects are managed by the Structures Monitoring program. The staff could not locate the inspection of steel edge supports and steel bracings for masonry walls in the Structures Monitoring AMP basis document. However, Section 8.5.2.3 in Procedure No. STI-744.09 states that steel members laterally support the masonry walls and these connections between the masonry and the support members should be examined.	**Scope of the Program:** Clarify whether existing Structures Monitoring program manages aging effects for the steel edge supports and steel bracing for masonry walls.
3	B.2.3.34 Table 2.2-3 2.4.11	B-192 2.2-7 2.4-29	1. Section 6.2 in Procedure No. STI-744.09 states that all structures identified in Section 7.0 will be included within the scope of the Structural Monitoring program. Section 7.0 in Procedure No. STI-744.09 lists the following inspection areas:	**Scope of the Program:** 1. Clarify whether inspection areas in question (from a to f) are within the scope of LR.

			a. Circulating water intake structure b. Service water discharge structure c. Plant effluent tanks d. Demineralized water storage tank e. foundation & appurtenances f. Block wall enclosures for the fire water deluge valves for the transformers The staff notes that the AMP basis document does not include these inspection areas within the scope of LR. LRA Table 2.2-3 indicates that the circulating water intake structure and service water discharge structure are not in the scope of LR. 2. LRA Section 2.4.11 includes the Firewater Valve Houses in the scope of LR. However, the staff could find the Firewater Valve Houses in the Section 7.0 of Procedure No. STI-744.09 and AMP basis document.	2. Clarify whether existing Structures Monitoring program include the Firewater Valve Houses. If not, provide the enhancement.
4	B2.3.34 AMP Basis document	B-193 15 of 42	The GALL-LR Report AMP XI.S6 states that preventative actions emphasize proper selection of bolting material, lubricants, and installation torque or tension to prevent or minimize loss of bolting preload and cracking of high strength bolting. AMP basis document states that design requirements and maintenance practices provide for proper selection of bolting material and installation torque or tension of structural bolts and it also states that the storage requirements of MDA-404 partially meet the storage requirement in Section 2.2 of the RCSC, and AMP provides an enhancement “ Reconcile the preventive actions in NUREG-1339, EPRI NP-5769, and EPRI	**Preventative Actions:** 1. Evaluate and revise to ensure that the enhancement to preventive actions in AMP will be consistent with the GALL-LR report recommendations. 2. Check enhancements to preventative actions for structural bolting integrity in other AMPs to ensure the consistency.

			TR104213 with the existing procedures and practices for structural bolting. It appears that AMP does not include preventative action of proper selection of lubricants, and AMP's enhancement is vague and needs to be specific.	
5	B.2.3.34	B-192	The GALL-LR Report AMP XI.S6 states that structural sealants are monitored for cracking, loss of material, and hardening, and structural sealants are acceptable if the observed loss of material, cracking, and hardening will not result in loss of sealing. The staff could not locate structural sealants in the Structures Monitoring AMP basis document and Procedure No. STI-744.09. It is unclear whether there are structural sealants at CPNPP and how their aging effects are adequately managed.	**Parameters monitored or inspected, and Acceptance criteria** Clarify whether there are structural sealants at CPNPP. If yes, provide enhancements to Parameters monitored or inspected, and Acceptance criteria for structural sealants, also provide Table 1 and 2 items if necessary.
6	B.2.3.34 STI-744.09	B-192 11 of 34	1. The GALL-LR Report AMP XI.S6 states that "In general, all structures and ground water quality are monitored on a frequency not to exceed 5 years. Some structures of lower safety significance, and subjected to benign environmental conditions, may be monitored at an interval exceeding five years; however, they should be identified and listed, together with their operating experience". LR Section B.2.3.34 only states that the frequency of monitoring groundwater chemistry (pH, chlorides, and sulfates) is once every 5 years. It appears that the Structures Monitoring program in Section B.2.3.34 lacks information of inspection frequency.	**Detection of Aging Effect:** 1. Clarify the inspection frequencies of all structures in the LR Section B.2.34 2. Provide the justification why the inspection frequencies every 10 years for the settlement of Category I structures and component supports are acceptable. and clarify what component supports are inspected once every ten years.

			2. Section 7.0 in procedure No. STI-744.09, Revisions 0, "Structural Monitoring Inspection Guide," provides the inspection frequency of structures and structural components, including the inspection frequency of every 10 years for the settlement of Category I structures. Section 12.2 in procedure No. STI-744.09 also states that the component supports (Sampled) will be inspected 1 every ten years. It is unclear to the staff why the inspection frequency of every 10 years is acceptable, and what's component supports are inspected once every ten years?	
7	B2.3.34 AMP Basis document	B-192 10 of 42	The GALL-LR Report AMP XI.S6 states that the program includes provisions for more frequent inspections of structures and components categorized as (a)(1) in accordance with 10 CFR 50.65. Section 3.3 in AMP basis document states, "As described in Section 4.4.2 below the CPNPP Structures Monitoring AMP includes provisions for increased inspection frequencies if a structure or component cannot meet its applicable acceptance criteria." The staff could not locate provisions for more frequent inspections in Section 4.4.2 of the AMP basis document.	**Detection of Aging Effect:** Clarify whether the existing Structures Monitoring program includes provisions for more frequent inspections if a structure or component cannot meet its applicable acceptance criteria. If yes, provide procedures and examples. Otherwise, provide the enhancement.
8	B.2.3.34	B-192	Section 3.3 in AMP basis document indicates Section 4.4.2 to include an enhancement to the CPNPP Structures Monitoring AMP that outlines subsequent steps if groundwater leakage is identified. The staff could not locate the above-mentioned enhancement in the AMP basis document and LRA.	**Detection of Aging Effect:** Provide the enhancement to the CPNPP Structures Monitoring AMP that outlines subsequent steps if groundwater leakage is identified.
9	B.2.3.34	B-194	The Structures Monitoring program has an enhancement: provide guidance for documentation and archival requirements in accordance with ACI 349.3R	**Acceptance Criteria:** Clarify the ACI Section for this enhancement.

			Section 3.5. The staff reviewed ACI 349.3R and found that it is related to ACI 349.3R Section 3.4 instead of Section 3.5.	
10	Table 3.3-1	3.3-76 3.3-80	The applicant claims that AMR item 3.3-1, 111 is not used, but states that the CPNPP new fuel storage racks are stainless steel and addressed with item 3.3-1, 120. Loss of material of structural steel exposed to indoor air is addressed with item 3.5-1, 077 below. AMR item 3.3-1, 111 is to address aging effect of loss of material due to general, pitting, and crevice corrosion. AMR 3.3-1, 120 has no aging effect without requiring AMP. It is unclear to the staff how AMR item 3.3-1, 120 is related to AMR item 3.3-1, 111 due to different aging effects. In addition, the staff could not find new fuel storage racks Table 2 items associated with AMR item 3.3-1, 120.	1. Evaluate the applicability of AMR item 3.3-1, 120 for AMR item 3.3-1, 111, and explain why aging management is not required for the new fuel storage racks. 2. Clarify where the Table 2 items for new fuel storage racks are located.
11	Table 3.3-1 Table 3.5.2-5 Table 3.5.2-8	3.3-73 3.5-119 3.5-138	AMR item 3.3-1, 106 is to manage aging effect of loss of material due to general, pitting, crevice, and microbiologically influenced corrosion, which is managed by the Buried and Underground Piping and Tanks (B.2.3.27) AMP. AMR item 3.3-1, 106 in Table 3.3-1 states that "Additionally, as listed in Tables 3.5.2-5 and 3.5.2-8, below grade piping penetrations for the FB and SWIS have coated steel plate collars exposed to soil that are managed by the Structures Monitoring AMP, which includes inspection of inaccessible components when excavated for other reasons. A generic note E and plant-specific note are used." The Buried and Underground Piping and Tanks AMP is an existing preventive,	Explain why the Structures Monitoring program can be used to manage this aging effect instead of using the Buried and Underground Piping and Tanks AMP.

			mitigative, and condition monitoring AMP that manages the aging effects associated with the external surfaces of buried and underground piping and tanks for loss of material and loss of coating integrity. Components addressed by this program, fabricated of steel, use preventive and mitigative techniques including external coatings, cathodic protection, and quality backfill. However, the Structures Monitoring program is only a condition monitoring AMP, it does not provide preventive and mitigative measures.	
12	Table 3.5-1 and 3.5.2.2.2	3.5-55, 3.5-59 3.5-61 and 3.5-27	AMR item 3.5-1, 054 in Table 3.5-1 states "Consistent with NUREG-1801 for Group 1, 3, 4, 5, and 7 structures, as well as accessible areas of the SWIS (Group 6 structure) that are above-grade/water-line." AMR item 3.5-1, 063 in Table 3.5-1 states that the Structures Monitoring AMP will be used to manage increase in porosity and permeability and loss of strength for accessible exterior concrete in Group 1, 3, 5 and 7 structures exposed to flowing water in the form of heavy drainage of rainwater. AMR item 3.5-1, 067 in Table 3.5-1 states the Structures Monitoring AMP will be used to manage increase in porosity and permeability, cracking, and loss of material of inaccessible concrete in Groups 1, 3 through 7 structures. LRA Section 3.5.2.2.2 states that Group 8: Missile doors (The steel FWSTs are addressed in Section 3.5.2.1.11 whereas the foundations are considered with Group 3), and Group 3 includes AB, DGBs, SGBs, Switchgear Buildings, Switchyard Structures, TBs, Yard Structures.	1. Revise Table 1 items 3.5-1, 054, 063 and 067 to include the Group 8: foundations. 2. Clarify where Table 2 items for the foundations in Group 8 are located for the AMR items 3.5-1, 054, 063, and 067.

			It appears that AMR items 3.5-1, 054, 063, and 067 shall include Group 8, foundations. The staff also could not locate Table 2 items for the foundations in Group 8.	
13	Table 3.5-1	3.5-60 3.5-62	The applicant claims AMR item 3.5-1, 070 to be consistent with NUREG-1801. The Masonry Walls AMP will be used to manage cracking of masonry walls exposed to indoor air and outdoor air. However, the staff noticed in AMR item 3.5-1, 066 that the applicant uses the Structural Monitoring program to manage concrete aging effect for masonry walls conservatively. The staff also noticed that not all of masonry walls are managed by the Structures Monitoring program. For example, masonry walls in Table 3.5.2-3 have AMR item 3.5-1, 070 without AMR item 3.5-1, 066. In addition, two AMR items for masonry block commodity "Wall, floor, and ceiling" in Table 3.5.2-15 associated with GALL item III.A3.T-12 have AMR item 3.5-1, 070 without AMR item 3.5-1, 066. It is unclear why some masonry walls are managed by both AMPs but other masonry walls are only managed by the Masonry Walls AMP.	Explain how the applicant determined what masonry walls are managed by both Masonry Walls and Structures Monitoring programs.
14	A.2.2.34	A-27	1. LRA Section A.2.2.34 states that the structures are monitored on an interval not to exceed 5 years. 2. SRP-LR FSAR Supplement in the Table 3.0-1 states that this program is implemented in accordance with NUMARC 93-01, Rev. 2 and RG 1.160, Rev. 2. The staff could not find this statement in LRA Section A.2.2.34.	1. Update the inspection frequency in UFSAR Supplement Section A.2.2.34 based on the applicant's response to question number 6. 2. Update the LRA Section A.2.2.34 to be consistent with SRP-LR FSAR Supplement.
15	Table 3.5-1	3.5-37	SRP-LR shows the title of Table 3.5-1 as "Summary of Aging Management Programs for Containments, Structures and Component Supports Evaluated in Chapters II and III of the GALL Report."	Table 3.5-1 title is confusing and not accurate. This is an editorial comment.

			The title of LRA Table 3.5-1 is "Summary of Aging Management Programs for Containment Building and Internal Structural Components."	

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section B.2.3.35: Inspection of Water-Control Structures

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	A.2.2.35 B.2.3.35 2.4.7 2.4.8 SRP-LR Table 3.0-1	A-27 B-197 2.4-21 2.4-22 3.0-24	LRA Section A.2.2.35 states that the structures within the scope of the RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP include the Safe Shutdown Impoundment (SSI) and Dam, Service Water Intake Channel, debris and fish barrier system, the discharge canal, and the SWIS interior concrete exposed to water-flow. LRA Section B.2.3.35 states that the structures within the scope of the RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP include the SSI and Dam, portions of the Service Water Intake structure exposed to water-flowing, the Service Water Intake Channel, debris and fish barrier system and the discharge canal. Section 2.4.7 is about Safe Shutdown Impoundment and Dam. Section 2.4.8 states that The SWIS provides housing to the nuclear safety related service water pumps and a Non-Nuclear Safety Related (NNS) fire pump and is equipped with trash racks, traveling screens, stop gates, and screen wash pumps. But it does not discuss the Service Water Intake Channel and the discharge canal. SRP-LR Table 3.0-1 FSAR Supplement states that the program includes structural steel and structural bolting	**Scope of the Program and UFSAR Supplement:** 1. Clarify the scope of the Water-Control Structures 2. Confirm that Tables 2.4-7 and 2.4-8 include all components subject to aging management review for the water-control structures based on the response to the question above, otherwise revise Table 1 and 2 items accordingly. 3. Update the LRA and UFSAR supplement to ensure the consistency.

			associated with water-control structures, steel or wood piles and sheeting required for the stability of embankments and channel slopes, and miscellaneous steel, such as sluice gates and trash racks. It appears that LRA UFSAR supplement is not consistent with the SRP-LR report recommendations.	
2	B.2.3.35	B-197	The Inspection of Water-Control Structures program is an existing AMP and is implemented as part of the Structures Monitoring program. Both AMPs use the same procedure. The staff notes that the Structures Monitoring program has some enhancements that the Inspection of Water-Control Structures does not have. For example, the AMP basis document in Section 4.4.2 states that the qualifications requirement for the inspection of structures and components as well the requirements for the reviewer will be updated to match the ACI 349.3R current code requirements.	**Preventive Actions, Detection of Aging Effects, and Acceptance Criteria:** Evaluate what enhancements of the Structures Monitoring program will be applied to the Inspection of Water-Control Structures program and provide the enhancements to the Inspection of Water-Control Structures program if necessary.
3	Table 3.5-1 Table 3.5.2-8	3.5-56 3.5-136	AMR item 3.5-1, 056 in GALL-LR report includes the concrete elements for exterior above grade and below grade concrete, foundation, and interior slab. AMR item 3.5-1, 056 in Table 3.5-1 states that the RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP will be used to manage loss of material for exterior above grade and below-grade concrete exposed to flowing water. It appears that AMR item 3.5-1, 056 misses the concrete elements for foundation and interior slab. Table 2 item associated with AMR item 3.5-1, 056 in Table 3.5.2-8 only shows component as “concrete: interior.” This Table 2 item does not include other concrete components requiring aging management.	Evaluate and identify the concrete components subject to AMR for this aging effect, and revise Table 1 and 2 items accordingly.

4	Table 3.5-1 Table 3.5.2-8	3.5-57 3.5-135	AMR item 3.5-1, 059 in Table 3.5-1 states that the RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP will be used to manage cracking, loss of bond, and loss of material of the accessible above-grade, below-grade, and interior concrete in the SWIS. Table 2 item associated with AMR item 3.5-1, 059 in Table 3.5.2-8 only shows component as “concrete: interior.” This Table 2 item does not include other concrete components requiring aging management. Please note that GALL-LR report refers to all accessible concrete areas for this AMR item.	Evaluate and identify the concrete components subject to AMR for this aging effect, and revise Table 2 items accordingly.
5	Table 3.5-1	3.5-58	The applicant claims AMR item 3.5-1, 060 to be not applicable. CPNPP is located in a region where weathering conditions are considered moderate, as shown in ASTM C33-90, Figure 1. Therefore, loss of material (spalling, scaling) and cracking due to freeze-thaw is an applicable aging effect and subject to AMR.	Evaluate the claim of no-applicability of AMR item 3.5-1, 060, and provide table 2 items if necessary.
6	Table 3.5.2-13	3.5-176	Table 3.5.2-13 lists the carbon steel support for ASME Class 3 component exposed to raw water environment that cites Note E, its aging effect will be managed by ASME Section XI, Subsection IWF program. This AMR item is associated with GALL item III.A6.TP-221 and AMR item 3.5-1, 083. The staff noted that GALL item III.A6.TP-221 and AMR item 3.5-1, 083 is used for the structural bolting in the water-control structures. It is unclear the staff whether the applicant uses GALL item properly.	Clarify where this commodity (ASME Class 3 component) is used.

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Sections 4.6.1 & 4.6.2 TLAAs: Containment Liner Plate and Penetrations Fatigue Analyses

Question Number	SLRA Section	SLRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	4.6.1, A.3.5.1	4.6-1, 4.6-2, A-39, A-40	LRA Sections 4.6.1 & A.3.5.1 related to containment liner plate fatigue do not make any mention of the material of the liner plate based on which the fatigue waiver analysis described was performed.	a. State the liner plate material based on which the described fatigue waiver analysis was performed. b. Explain or clarify if the material used in the analyses is bounding or enveloping, with regard to fatigue, of all materials used for the containment liner plate and its anchorage or integral attachments, as applicable, for which the LRA 4.6.1 TLAA is credited.
2	4.6.1, A.3.5.1	4.6-1, 4.6-2, A-39, A-40	• LRA Section 4.6.1 references LRA Section 4.3.1 and LRA Table 4.3.1-2 with regard to transient cycles considered for the containment liner plate. Althougj it is consistent with the cycles used in the audited Calculation 16345/6-CS(B)-028, Rev 1 on the ePortal, the staff notes an inconsistency regarding the number of OBE and SSE cycles between LRA Section 4.6.1 and referenced LRA Table 4.3.1-2. • The LRA Section A.3.5.1 FSAR supplement description for the containment liner plate TLAA makes no mention of the OBE/SSE cycles evaluated in the containment liner plate fatigue waiver analysis; therefore, the FSAR summary description appears incomplete.	a. Explain or clarify the noted inconsistency between the number of OBE and SSE cycles used in the TLAA in LRA 4.6.1 and the referenced LRA Table 4.3.1-2, specifically in terms of number of OBE/SSE cycles and events. b. Provide a revised LRA A.3.5.1 FSAR supplement liner plate summary description that includes a summary of all the transient cycles considered in the fatigue waiver analysis of the containment liner plate consistent with LRA Section 4.6.1.

3	4.6.2, A.3.5.2	4.6-2 thru 4.6-4, A-40	LRA Sections 4.6.2 & A.3.5.2 related to containment penetrations fatigue do not make any mention of the material of the penetrations based on which the fatigue waiver analysis described was performed.	a. State the containment penetrations material based on which the described fatigue waiver analysis was performed. b. Explain or clarify if the material used in the analyses is bounding or enveloping, with regard to fatigue, of all materials used for the containment process piping penetrations, as applicable, for which the LRA 4.6.2 TLAA is credited.

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section 3.5.2.2, “AMR Results for Which Further Evaluation is Recommended by the GALL Report”

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	3.5.2.2.1.2 Table 3.5.2-1	3.5-20 3.5-85	The GALL Report recommends that a plant-specific evaluation be performed if any portion of the concrete containment components exceeds specified temperature limits, i.e., general temperature greater than 66°C (150°F) and local area temperature greater than 93°C (200°F). 1. Section 3.5.2.2.1.2 states that local area temperatures may be elevated above general area temperatures due to process piping carrying high temperature fluids (e.g., MS, and feedwater piping RCB penetrations and RCL cold and hot leg piping through the biological shield tunnel in the primary shield wall), for example, feedwater temperatures are above 250ºF when main feedwater is preheated prior to this switch. To comply with the requirements of ASME Section III, Division 2, paragraph CC-3430 on concrete temperatures, the unit 2 “cold” penetrations (MV-17, MV-18, MV-19, and MV-20) along the preheater bypass flow path, are subject to an administrative time limit of 24 hours for operating temperatures above 250ºF. Section 3.5.2.2.1.2 also states that the RCB penetration and reactor coolant piping insulation contributes to keeping the local concrete temperatures of the RCB and PSW below 200ºF during normal plant operation.	1. Explain how the elevated temperatures (above 250ºF) at local areas will be adequately managed not to exceed the specified temperature limit, and what’s the AMP? 2. Explain why the thermal insulations with material of calcium silicate are not subject to aging management. 3. Explain why thermal insulation can use SRP-LR item III.B1.1.TP-8, and why the thermal insulations with material of stainless steel are not subject to aging management. 4. Evaluate whether a plant-specific program is required?

			2. Table 2 item in Table 3.5.2-1 lists a component of thermal insulation (high temperature penetrations) with material of calcium silicate and cites generic note H without aging management. It is unclear why aging management is not required. 3. Table 2 item associated with AMR item 3.5-1, 095 in Table 3.5.2-1 lists a component of thermal insulation (high temperature penetration) with material of stainless steel and cites generic note C without aging management. SRP-LR item III.B1.1.TP-8 (AMR item 3.5-1, 095) lists a component (Aluminum, galvanized steel and stainless steel Support members; welds; bolted connections; support anchorage to building structure exposed to air-indoor uncontrolled environment. This component does not include thermal insulation. It is unclear how thermal insulation can use SRP-LR item III.B1.1.TP-8, and it is not clear why thermal insulations are not subject to AMR.	
2	3.5.2.2.1.7 Table 3.5-1 Table 3.5-1	3.5-24 3.5-39 3.5-41	SRP-LR Section 3.5.3.2.1.7 states that a plant-specific program is not required if documented evidence confirms that where the existing concrete had air content of 3% to 8% and subsequent inspection did not exhibit degradation related to freeze-thaw. 1. The applicant claims AMR item 3.5-1, 011 and 018 to be not applicable. Section 3.5.2.2.1.7 states that CPNPP is located in a region where weathering conditions are considered moderate, as shown in ASTM C33-90, Figure 1. Therefore, loss of material (spalling, scaling) and cracking due to freeze-thaw is an applicable aging effect and subject to AMR. 2. Section 3.5.2.2.1.7 states that air entrainment content conformed to the design requirements of ACI 211.1 and was determined by ASTM C231. It is unclear to the staff	1. Evaluate the claim of no-applicability of AMR item 3.5-1, 011 and 018 and provide table 2 items if necessary. 2. Clarify air content of the concrete mix used for the containment structure at CPNPP. 3. Evaluate whether a plant-specific program is required. 4. Explain what AMPs will be used and how these AMPs will manage aging effect of freeze-thaw in inaccessible areas.

			what's air content of concrete mix used for the containment structure at CPNPP. 3. Section 3.5.2.2.1.7 lacks the evaluation whether plant-specific program is required for managing this aging effect. 4. Section 3.5.2.2.1.7 lacks information of how aging effect in inaccessible areas will be managed.	
3	Table 3.5-1	3.5-42	***<u>Note: This breakout question also applies to TRP 42, ASME XI Subsection IWL program</u>*** AMR 3.5-1, 019 in Table 3.5-1 states that the ASME Section XI, Subsection IWL AMP will continue to inspect and monitor for cracking and indications of ASR-induced degradation. The staff notes that the Structures Monitoring program has an enhancement to the Parameters Monitored or Inspected, " Visually inspect concrete structures for unique cracking such as "craze", "mapping" or "patterned" cracking to determine the presence of alkali-silica gel." The staff does not find similar enhancement in the ASME XI Subsection IWL program.	Clarify whether the ASME XI Subsection IWL program needs an enhancement for the inspection of ASR.

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section 3.5.2.2, "AMR Results for Which Further Evaluation is Recommended by the GALL Report"
Inaccessible areas

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	3.5.2.2.2.1.1 Table 3.5-1	3.5-27 3.5-48	SRP-LR Section 3.5.3.2.2.1.1 states that a plant-specific program is not required if documented evidence confirms that where the existing concrete had air content of 3% to 8% and subsequent inspection did not exhibit degradation related to freeze-thaw. 1. The applicant claims AMR item 3.5-1, 042 to be not applicable. Section 3.5.2.2.2.1.1 states that CPNPP is located in a region where weathering conditions are considered moderate, as shown in ASTM C33-90, Figure 1. Therefore, loss of material (spalling, scaling) and cracking due to freeze-thaw is an applicable aging effect and is subject to AMR. 2. Section 3.5.2.2.2.1.1 states that non-containment structures at CPNPP consist of Groups 1, and 3 through 8. However, AMR item 3.5-1, 042 in Table 3.5-1 states that CPNPP is located in a moderate weathering region with concrete, for group 1, 3-5 and 7 structures, that meet the air-entrainment and water-cement-ratio of ACI 318-71, and the Structures Monitoring AMP is credited with management of other aging effects for the concrete foundations of group 1, 3-5, and 7 structures.	1. Evaluate the claim of no-applicability of AMR item 3.5-1, 042, and provide table 2 items if necessary. 2. Revise Table 1 item 3.5-1, 042 to include Group 8: foundations, and clarify whether the foundations in Group 8 meet the air-entrainment and water-cement-ratio of ACI 318-71. 3. Clarify air content of the concrete mix used for the non-containment structures at CPNPP. 4. Evaluate whether a plant-specific program is required.

			LRA Section 3.5.2.2.2 states that Group 8: Missile doors (The steel FWSTs are addressed in Section 3.5.2.1.11 whereas the foundations are considered with Group 3), and Group 3 includes AB, DGBs, SGBs, Switchgear Buildings, Switchyard Structures, TBs, Yard Structures. It appears that AMR items 3.5-1, 042, shall include Group 8, foundations. In addition, it is not clear to the staff whether the foundations in Group 8 meet the air-entrainment and water-cement-ratio of ACI 318-71. 3. Section 3.5.2.2.2.1.1 states that air entrainment content conformed to the design requirements of ACI 211.1 and was determined by ASTM C231. It is unclear to the staff what's air content of concrete mix used for the non-containment structures at CPNPP. 4. Section 3.5.2.2.2.1.1 lacks the evaluation whether a plant-specific program is required for managing this aging effect.	
2	3.5.2.2.2 3.5.2.2.2.1.2 Table 3.5-1	3.5-27 3.5-28 3.5-49	1. Section 3.5.2.2.2.1.2 mentioned petrographic examination and its results. 2. The AMR item 3.5-1, 043 in Table 3.5-1 states that the Structures Monitoring AMP is credited with managing cracking due to reaction with aggregates (such as ASR), for CPNPP group 1, 3-5, and 7 structures, including inaccessible areas. Section 3.5.2.2.2.1.2 states that non-containment structures at CPNPP consist of Groups 1, and 3 through 8. Section 3.5.2.2.2 lists "Group 8: Missile doors (The steel FWSTs are addressed in Section 3.5.2.1.11	1. Provide the results of petrographic examination for the staff to review. 2. Revise Table 1 item 3.5-1, 043 to include Group 8: foundations, and clarify where Table 2 items for Group 8 foundations are located. 3. Explain why different notes A and E are cited for the Structures Monitoring program for the further evaluation sections.

			whereas the foundations are considered with Group 3). It appears that AMR items 3.5-1, 042, shall include Group 8, foundations. In addition, the staff could not locate Table 2 items for the foundation in Group 8. 3. Table 2 items associated with AMR item 3.5-1, 043 in Table 3.5.2-1 cite Note E, while the rest of Table 2 items associated with AMR item 3.5.1-043 in Tables 3.5.2-2 thru 3.5.2-6, and Tables 3.5.2-9 thru 3.5.2-12 cite Note A for Structures Monitoring program. The staff noticed that other AMR items cite Note E for the AMP when further evaluation is required to determine if a plant-specific aging management program is needed. Per SLR-ISG-2021-03-STRUCTURES, AMP for AMR item 3.5-1, 043 is revised to "Plant-specific aging Management program or AMP XI.S6, Structures Monitoring, enhanced as Necessary" when further evaluation is required.	
3	3.5.2.2.2.1.3 Table 3.5-1	3.5-29 3.5-50	1. Section 3.5.2.2.2.1.3 mentioned that the Seismic Category I concrete structure walls and columns are supported on thick continuous concrete base mats which rest on the rock subgrade with the exception of the Category I tanks and pipe tunnels. It is not clear to the staff whether porous concrete sub-foundations are present at CPNPP site. 2. AMR 3.5-1, 046 is related to aging effect of reduction of foundation strength and cracking due to differential settlement and erosion of porous concrete sub-foundation. The applicant claims AMR item 3.5-1, 046 to be not applicable. This aging effect is related to a). differential settlement, b) porous sub-	1. Clarify whether porous sub-foundations are used at CPNPP site. 2. Evaluate the claim of non-applicability of AMR item 3.5-1, 046, and provide table 2 items if necessary. 3. Clarify the note citation for Table 2 items associated with AMR item 3.5-1, 044 for the Structures Monitoring program.

			foundation. This aging effect exists if either or both conditions present. 3. Table 2 items associated with AMR item 3.5-1, 044 cite Note A for the Structures Monitoring program, while Table 2 items associated with 3.5-1, 047 cite Note E for the Structures Monitoring program. It is not consistent for the AMP note citation in the further evaluation. Per SLR-ISG-2021-03-STRUCTURES, AMP is revised to “Plant-specific aging Management program or AMP XI.S6, Structures Monitoring, enhanced as Necessary” when further evaluation is required.	
4	3.5.2.2.2.1.4 Table 3.5-1	3.5-30 3.5-51	1. Section 3.5.2.2.2.1.4 states Leaching of calcium hydroxide or carbonation has not occurred for above-grade concrete at CPNPP. It is unclear to the staff whether there is OE in below-grade interior concrete walls. 2. AMR item 3.5-047 in Table 3.5-1 states “Consistent with the OE reflected in SLR-ISG-2021-03-STRUCTURES, a plant-specific AMP is not required for inaccessible concrete areas of Group 1, and 3 structures with foundations exposed to groundwater. It appears that it does not include Groups 4, 5, 7 and 8. SRP-LR Section 3.5.3.2.2.1.4 states that a plant-specific aging management program is not required for the reinforced concrete exposed to flowing water if (1) there is evidence in the accessible areas that the flowing water has not caused leaching of calcium hydroxide and carbonation or (2) evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no	1. Clarify whether there is OE for the increase in porosity and permeability and loss of strength due to leaching of calcium hydroxide or carbonation in below-grade interior concrete walls or any other locations at CPNPP. If yes, provide evaluation whether the observed leaching of calcium hydroxide and carbonation in accessible areas has impact on the intended function at CPNPP site. 2. Explain why a plant-specific program is not required for Groups 1-5, 7-9, and update the LRA accordingly.

			impact on the intended function of the concrete structure. Note: Per GALL-LR definition in Table IX.D, water-flowing includes rainwater, raw water, ground water, or water flowing under a foundation.	
5	3.5.2.2.2.3.1 Table 3.5-1	3.5-32 3.5-52	SRP-LR Section 3.5.3.2.2.3 states that a plant-specific program is not required if documented evidence confirms that where the existing concrete had air content of 3% to 8% and subsequent inspection of accessible areas did not exhibit degradation related to freeze-thaw. 1. AMR item 3.5-1, 049 claims to be not applicable, and the loss of material (spalling, scaling) and cracking due to freeze-thaw does not apply to the CPNPP SWIS or SSI and Dam. Section 3.5.2.2.2.3.1 states that CPNPP structures, including SWIS, are located in a region where weathering conditions are considered moderate, as shown in ASTM C33-90, Figure 1. Therefore, loss of material (spalling, scaling) and cracking due to freeze-thaw is an applicable aging effect and subject to AMR. 2. Section 3.5.2.2.2.3.1 states that air entrainment content conformed to the design requirements of ACI 211.1 and was determined by ASTM C231. AMR item 3.5-1, 049 in Table 3.5-1 states that SWIS concrete meet the air-entrainment and water-cement ratio of ACI 318-71. It is unclear to the staff what's air content of concrete mix used for the water-control structures at CPNPP.	1. Evaluate the claim of no-applicability of AMR item 3.5-1, 049, and provide table 2 items if necessary. 2. Clarify air content of the concrete mix used for the water-control structures at CPNPP. 3. Evaluate whether a plant-specific program is required. 4. Explain how and what AMP will manage this aging effect in below grade inaccessible concrete areas.

			3. Section 3.5.2.2.2.3.1 lacks the evaluation whether a plant-specific program is required for managing this aging effect. 4. Section 3.2.2.2.2.3.1 mentions the Inspection of Water-Control Structures AMP and the Structures Monitoring AMP for certain inspections. It is not clear to the staff how and what AMP will manage this aging effect in below grade inaccessible concrete areas of Group 6 structures.	
6	3.5.2.2.2.3.2 Table 3.5-1	3.5-33 3.5-53	1. Section 3.5.2.2.2.3.2 states that cracking due to expansion and reaction with aggregates is an applicable aging effect in below-grade inaccessible concrete areas for CPNPP Group 6 structures and will be managed by the Inspection of Water-Control Structures AMP and the Structures Monitoring AMP. However, AMR item 3.5-1, 050 in Table 3.5-1 only credits the Structures Monitoring program for this aging effect. 2. Section 3.5.2.2.2.3.2 lacks information how to manage this aging effect in below grade inaccessible concrete areas.	1. Clarify which AMP is used for the aging management. 2. Explain how AMP will manage aging effect in below-grade inaccessible concrete areas.
7	3.5.2.2.2.3.3 Table 3.5-1	3.5-33 3.5-54	1. Section 3.5.2.2.2.3.3 states that the Inspection of Water-Control Structures program and the Structures Monitoring program will manage the aging effect. However, AMR item 3.5-1, 051 in Table 3.5-1 and associated Table 2 items only credit the Structures Monitoring program for this aging effect. 2. Section 3.5.2.2.2.3.3 does not present OE related to the leaching in accessible areas of the water-control structures, which is the basis for evaluating whether a plant-specific AMP is required.	1. Clarify which AMP is used for the aging management. 2. Clarify whether OE related to leaching exists in accessible areas of the water-control structures. If yes, provide evaluation whether the observed leaching of calcium hydroxide and carbonation in accessible areas has impact on the intended function. 3. Explain why a plant-specific AMP is not required.

			3. AMR item 3.5-1, 051 in Table 3.5-1 states that a plant-specific AMP is not required for the inaccessible areas of the CPNPP SWIS (Group 6) structure, but it does not explain the reason. 4. Section 3.5.2.2.2.3.1 lacks information how to manage aging effects in below grade inaccessible concrete areas.	4. Explain how AMP will manage aging effect in below-grade inaccessible concrete areas.
8	Table 3.5-1	3.5-41, 3.5-43	***<u>This breakout question also applies to TRP 42.</u>*** AMR items 3.5-1, 016 and 024 note that increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack is not an aging effect requiring management because the concrete is not exposed to acidic solutions with a pH < 5.5, chloride solutions > 500ppm, or sulfate solutions > 1500ppm. AMR item 3.5-1, 067 notes that the SMP will manage this aging effect for similar concrete. It is not clear why the aging effect in items 016 and 024 are not applicable. This aging effect exists in the environment as long as acidic solutions present, concrete deterioration may differ.	Evaluate the claim of no-applicability of AMR item 3.5-1, 016 and 024, and explain what and how AMP will manage this aging effect in below grade inaccessible and accessible concrete areas as applicable and provide table 2 items if necessary.
9	Table 3.5-1	3.5-59	The applicant claims AMR item 3.5-1, 064 to be not applicable. CPNPP is located in a region where weathering conditions are considered moderate, as shown in ASTM C33-90, Figure 1. Therefore, loss of material (spalling, scaling) and cracking due to freeze-thaw is an applicable aging effect and subject to AMR.	Evaluate the claim of no-applicability of AMR item 3.5-1, 064, and provide table 2 items if necessary.
10	Table 3.5-1	3.5-60	SRP-LR Report lists the following components for AMR item 3.5-1, 065: a. Groups 1-3, 5, 7-9: concrete (inaccessible areas): below-grade exterior; foundation;	1. Revise AMR 3.5-1, 065 notes in Table 3.5-1 to ensure the consistency with SRP-LR report recommended Groups and concrete areas. 2. Clarify whether aging effects for Groups 5, 7, and 8 structures, as well as certain

			b. Groups 1-3, 5, 7-9: concrete (accessible areas): below-grade exterior; foundation, c. Groups 6: concrete (inaccessible areas): all AMR item 3.5-1, 065 in Table 3.5-1 states that the Structures Monitoring AMP will be used to manage cracking, loss of bond, and loss of material of inaccessible concrete in Groups 1, 3 and 6 structures exposed to a groundwater/soil environment. It also states that Group 5, and 7 structures, as well as certain Group 3 structures are founded above the water-table. Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel is an applicable aging effect, where the water table is one of factors causing this aging effect. Section 3.5.2.2.2 lists “Group 8: Missile doors (The steel FWSTs are addressed in Section 3.5.2.1.11 whereas the foundations are considered with Group 3). It is unclear whether aging effects for Groups 5, 7, and 8 structures, as well as certain Group 3 structures are managed by the Structures Monitoring program.	Group 3 structures are managed by the Structures Monitoring program, and provide Table 2 items if necessary

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

LRA Section 3.5.2.2, "AMR Results for Which Further Evaluation is Recommended by the GALL Report"
Settlement

Question Number	LRA Section	LRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1	3.5.2.2.1.2 Table 3.5-1	3.5-19 3.5-37	1. Section 3.5.2.2.1.2 states that reduction of foundation strength and cracking due to differential settlement and erosion of porous concrete sub-foundation is not an applicable aging effect for the CPNPP RCB, and AMR item 3.5-1, 002 claims to be not applicable. The staff understands that RCB is not founded on a porous concrete sub-foundation. Differential settlement might be small but requires Structures Monitoring to monitor. Therefore, reduction of foundation strength and cracking due to differential settlement and erosion of porous concrete sub-foundation is an applicable aging effect. 2. Table 2 items associated with AMR item 3.5-1, 001 cite Note A for the Structures Monitoring program, while Table 2 items associated with other AMR items cite Note E for the Structures Monitoring program when the further evaluation is required. Per SLR-ISG-2021-03-STRUCTURES, AMP is revised to "Plant-specific aging Management program or AMP XI.S6, Structures Monitoring, enhanced as Necessary" when further evaluation is required.	1. Evaluate the claim of non-applicability of AMR item 3.5-1, 002, and provide table 2 items if necessary. 2. Clarify the note citation for Table 2 items associated with AMR item 3.5-1, 001 for the Structures Monitoring program.

Comanche Peak Nuclear Power Plant, Units 1 and 2
License Renewal Application (LRA) Breakout Audit Questions

Supports (LRA Section B.2.3.34. Structures Monitoring)

Question Number	**LRA Section**	**LRA Page**	**Background / Issue** **(As applicable/needed)**	**Discussion Question/Request**
1	OE TR-2021-003756 CR-2017-004316 CR-2015-009667 CR-2013-004651	Applicant provided OE	A number of routine walkdowns identified pipe supports that were degraded, loose, or hangers having missing pins, and/or needing pins replaced/reworked. CPNPP Units 1 and 2 "Specifications for Structural Steel/Miscellaneous Steel (Cat I, II & Non-Seismic) 2323-SS-16B," Rev 14, paragraph 3.13.7.5 emphasizes shall be sufficiently deformed to prevent withdrawal. It is not clear whether pins were missing because they were not sufficiently deformed and withdrew under applied loads. It is also not clear how supports lacking cotter pins can maintain their structural integrity.	Discuss steps taken to ensure that pins are adequately deformed so that they can remain in place following application of design basis loads. For those hangers identified as having missing pins discuss steps taken to rectify/ensure their operability.