Oconee Nuclear Station, Units 1, 2, and 3

SLRA: Breakout Questions

SLRA Section 3.5.2.2.2.6 “Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation”
AMR TRP: 76

Question Number	SLRA Section	SLRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request
1 JM	FE 3.5.2.2.2.6 (Fluence/Gamma Dose - Cavity/Pedestal Concrete Evaluation)	3-1321	Topic: **Fluence, Cavity Fluence Uncertainty** Background: SLRA Section 3.5.2.2.2.6 reports reactor cavity neutron fluence and gamma dose projections. Purpose: To ensure the fluence and gamma dose calculations include sufficient margin to account for associated uncertainties.	Please provide an estimate of the uncertainty in the reactor cavity neutron fluence and gamma dose calculations.
2 AP	FE 3.5.2.2.2.6 (Cavity/Pedestal Concrete Evaluation)	3-1321	Topic: **Configuration, Reactor Cavity Concrete and RV Pedestal** SLRA Section 3.5.2.2.2.6, Subsections “Reactor Cavity Wall Concrete Evaluation” and “Reactor Vessel Embedment Pedestal Concrete Evaluation” discuss the reactor cavity and pedestal concrete but do not provide information for the compressive strength of concrete (including that of RV support grout) and grade of reinforcing bars used. In addition, there is no discussion for reactor cavity/pedestal liners (if any) and how these are anchored to concrete.	Provide the compressive strength of concrete (and that of grout) and grade of reinforcing bars used in the construction of the reactor cavity concrete and pedestal. By sharing screen show in relevant drawings the reactor cavity/pedestal configuration and liners (if any) and how the liners are anchored to concrete.

3 AP	FE 3.5.2.2.2.6 (Cavity/Pedestal Concrete Evaluation)	3-1321	Topic: **RV Support Components/Embedment, RV Skirt Load Transfer, Loading Conditions** The SLRA states: The reactor vessel support assembly is defined as the reactor vessel support skirt and the reactor vessel support flange, which were attached to the reactor vessel during fabrication of the reactor vessel. The reactor vessel embedment includes the anchor bolts and associated washers and hex nuts, sole plate, vertical bearing plate and associated nelson studs, grout, and reinforced concrete pedestal that contain the embedded anchor bolts. Based on configuration of the reactor vessel support assembly and embedment detail, the following items were determined to directly support the reactor vessel support intended function (i.e., provide structural support for the reactor vessel): • RPV support skirt (SA-516 Grade 70) • RPV support flange (SA-515 Grade 70) • Anchor bolts (A490) • Anchor bolt jamb nuts, hex nuts, and washers The Framatome ANP-3898P states that the [[XX]].	1. Discuss SLRA's omission of base/vertical plates from the listed structural components determined to directly support the reactor vessel, as these components are considered to be essential to transfer loads to underlying concrete. 2. Explain the underlined Framatome ANP-3898P statement above with the aforementioned statements made in Topical Report BAW-1621/B&W 177-FA Owners Group Report regarding the role of the vertical bearing plate in resisting lateral loads. 3. Discuss the approach taken in ANP-3898P for transferring loads to concrete and its conservatism when compared to the aforementioned methodology in Topical Report BAW-1621/B&W 177-FA as supported with ePortal posted documents/drawings (e.g., Note 10 and Section C-C in

			The Babcock & Wilcox Topical Report BAW-1621/B&W 177-FA Owners Group Report, "Effects of Asymmetric LOCA Loadings - Phase II Analysis," identifies the primary lateral loads to be LOCA and seismic. For plants supported with skirts, generically it discusses the distribution of lateral forces at the RV pedestal (embedment). It states that "[t]he lateral load applied through the skirt can be transmitted across the skirt/pedestal interface in three ways: by friction between the sole plate and the underlying concrete, by direct thrust of the vertical bearing plates against the concrete, and through shear anchors beneath the sole plate." In the analysis of lateral force distribution the Topical Report BAW-1621/B&W 177-FA also states that a coefficient of friction of 10 percent was assumed to act between the sole plate and the concrete with the remaining lateral force applied as a radial thrust to the vertical bearing plate." It concludes that the "shear failure plane was assumed to extend from the vertical bearing plate to the inner face of the pedestal. It then states that "[a]ny slippage along such a plane results in wedge-like interaction between the moving block and the intact pedestal concrete."	DWG O-68 A, shear pin distribution on the outer perimeter of the saddle shown in DWG 128702 Rev D18, and the partial IWF inspections included in Framatome 86-9314756-001 Appendix A).
4 AP	FE 3.5.2.2.2.6 (Cavity/Pedestal Concrete Evaluation)	3-1319 – 3-1322	Topic: **RV Support Embedment, RV Concrete Anchorage, Inspections**	Identify and post on ePortal for review and discussions the procedure(s) for performing

			AR 01868962 dated 4/17/2019 discusses RV annulus inspections and the need to include skirt anchor bolts in ISI examinations. ePortal document NDE-NE-ALL-7302 Rev 001 in its Attachment 1, “Component Support Boundaries,” states that boundaries for VT-3 examinations include mechanically connected anchor bolts and that examinations shall include the base plate and any grout between the base plate and the building structure.	inspections for each of the aforementioned components.
5 AP	FE 3.5.2.2.2.6 (Cavity/Pedestal Concrete Evaluation)	3-1319 – 3-1322	Topic: **Thermal loads, insulation** ONS UFSAR Chapters 5 and 7 state that the hot leg design temperature(s) to be as high as 650ºF with normal coolant operating temperatures at inlet/(hot leg) outlet of 554/604°F. ePortal ONS DWG 128702, Revision D18, shows the outlet nozzle to be 36 inch in diameter. ePortal ONS DWG O69D /1069D indicate penetrations in the PSW for the hot leg pipes to pass through. ePortal ONS DWG O-70X shows such penetration openings initially to be 36 inches and subsequently enlarged to more than 5 feet. ONS DWG O-66G shows the hot leg pipe to expand within a sleeve at the PSW. The SLRA does not state what the PSW annulus temperature is, particularly when the normal coolant operating temperatures at inlet/outlet (hot leg) to be 554/604 °F (with Measurement Uncertainty Recapture Power Uprate – MUR PU* subtracting and adding	1. If insulation at that location exists, provide PSW drawings (sketches if drawings are not available) detailing its arrangement in the penetrations and include size and locations of PSW reinforcing steel, type of insulation (e.g., metal, fiberglass, reflective type) used and how it is age managed during the subsequent period of extended operation for the aforementioned potential aging effects. In addition, state if the insulation is of the reflective type discussed in Information Notice (IN) 2007-21, Supplement 1: “Pipe Wear

			0.4 ^{o}F), respectively. However, SLRA Section 3.5.2.2.2.2 states that "analysis performed to determine the maximum concrete temperature of the primary shield wall illustrates that the concrete will not exceed 200 °F for local loads." Given that the hot leg temperature is at about 605 °F, it is not clear whether the pipes/sleeves at the PSW are insulated to limit effects of elevated temperature and coincidental radiation effects on concrete at that location. (*not yet initiated)	due to Interaction of Flow-Induced Vibration Metal Insulation," and how did ONS responded to IN 2007-21. 2. If the penetration/pipes/sleeves are not insulated, provide on the ePortal for review and discussion the thermal analysis performed indicating that concrete at the penetration does not exceed sustained general temperatures of 150 ^{o}F (or 200 ^{o}F locally); and hence demonstrate that the effects of elevated temperature and radiation exposure of concrete at hot leg penetrations are aging effects that need not be addressed during the subsequent period of extended operation.
6 AP	FE 3.5.2.2.2.6 (Cavity/Pedestal Concrete Evaluation)	3-1319 – 3-1322	Topic: **Thermal Loads, Insulation** ARs 01821889 and 01786270 discuss mirror insulation in the annulus, below the RV. It is not clear whether the ex-vessel insulation is positioned to protect the concrete pedestal and RV attachments and pedestal embedments (e.g., skirt, bolts, pins) to the pedestal from radiation, temperature effects,	Clarify the function of the insulation below the RV. If the insulation is that of type described in IN 2007-21, how did ONS address the IN 2007-21, and how does ONS manage the effects of aging for such insulation at that location

			or both. It is also not clear whether it is of the reflective type discussed in Information Notice (IN) 2007-21, Supplement 1: “Pipe Wear due to Interaction of Flow-Induced Vibration Metal Insulation.”	consistent with the guidance promulgated in GALL-SLR.
7 AP	FE 3.5.2.2.2.6 (Cavity/Pedestal Concrete Evaluation)	3-1321	Topic: **Thermal Loads and Analysis of Reactor Cavity** Framatome 0402-01-F01, Rev. 021 and 32-9319316-000, “Oconee Reactor Vessel Cavity Concrete Temperature,” includes a sketch of RV skirt attachment to the concrete pedestal. The detail shows that the grout and pedestal concrete are “not in scope.”	In light of statements made in the SLRA that “[t]he reactor vessel support assembly and embedment consist of … grout, and reinforced concrete pedestal…,” clarify the note in the aforementioned Framatome document sketch.
8 AP	FE 3.5.2.2.2.6 (Cavity/Pedestal Concrete Evaluation)	3-1321	Topic: **Thermal Calculations and Analysis of Reactor Cavity** Framatome 0402-01-F01, Rev. 021 and 32-9319316-000 states that the 2-D model used in the thermal analysis was developed in Framatome 32-9311203-000, “Oconee Reactor Vessel Skirt Lowest Service Temperature.” It is clear (referenced Framatome document 32-9292931-002, “Oconee SLR Fluence and Gamma Heating Analyses”) that the RV cavity concrete analysis includes gamma heating. The Framatome 0402-01-F01, Rev. 021 and 32-9319316-000 report, however, concludes that “[a]dditional work may be required to	1. By sharing screen show in the model (or in other relevant drawings) location of mirror insulation and discuss how it was considered in the heat loss analysis (e.g., direct modeling, boundary condition). 2. Discuss the statement made for potential additional work. Was it made in anticipation of revised calculations for ex-vessel radiation effects? Summarize the additional

			provide a more refined estimate of the foundation concrete temperatures, and/or determine the acceptability of the temperatures calculated therein.”	work performed, if any, and state conclusions reached regarding concrete pedestal/grout temperatures.

Question Number	SLRA Section	SLRA Page	Background / Issue (As applicable/needed)	Discussion Question / Request	Outcome of Discussion
1 JM	FE 3.5.2.2.2.6 (Fluence/dose - RV Support Steel Evaluation)	3-1322	Topic: **Fluence, Ex-vessel fluence methodology** Background: Section 5.3.2 of ANP-3899NP discussing reactor vessel internals (RVI) fluence calculations describes known deficiencies in the BAW-2241P DORT methodology at locations “far away from the core” and “regions above and below the core.” As a result, a hybrid MCNP (or more specifically Framatome’s SVAM methodology) and DORT methodology was used, reporting the larger of the two results for the RVI. Purpose: To confirm that the chosen fluence methodology is applicable.	Please explain why using only the DORT methodology for the reactor vessel support steel, pedestal concrete, and cavity is acceptable, given that these regions may be considered “far away from the core” and includes areas “below the core.”	
2 JM	FE 3.5.2.2.2.6 (dpa - RV Support Steel Evaluation)	3-1322	Topic: **Fluence, Its Effects, Support Structure dpa** Background: In Section 9.4.4 of ANP-3898NP it is stated that “The dpa of 5.53E-04 is conservatively assumed to be applicable to all the RPV support assembly and embedment	Please explain the specific location that was calculated to receive the 5.53E-04 dpa and why it is conservative for all of the RPV support assembly and embedment materials.	

			materials (i.e., skirt, flange, anchor bolts, nuts, washers, and shear pins)." Purpose: To ensure that the dpa estimate is sufficiently conservative to bound the actual dpa anticipated for all RPV support assembly and embedment materials.		
3 AP	FE 3.5.2.2.2.6 (RV Support Steel Evaluation)	3-1322	Topic: **Calculations, RV Steel Support Assembly Configuration, Loads and Loading Conditions, Acceptance Limits** SLRA Section 3.5.2.2.2.6 Subsection "Reactor Vessel Support Steel Evaluation" states that "[t]he reactor vessel support assembly and embedment consist of a support skirt, a support flange, anchor bolts and associated washers and hex nuts, sole plate, vertical bearing plate and associated nelson studs, grout, and reinforced concrete pedestal that contains the embedded anchor bolts." According to Framatome ANP-3898P, the Babcock & Wilcox Topical Report BAW-1621/B&W 177-FA Owners Group, "Effects of Asymmetric LOCA Loadings - Phase II Analysis," [[XX]]. Topical Report BAW-1621/B&W 177-FA states that "[t]he Oconee 1, 2, and 3 reactor vessel support exceeded acceptance limits by 10%..."	1. Discuss whether the applied loads for Normal/Upset/Faulted loading conditions considered in Framatome ANP-3898NP/P for the subsequent period of extended operation are the same as those discussed as applicable for pipe ruptures in Topical Report BAW-1621/B&W 177-FA. 2. Confirm that the ONS RV support system evaluated in the Topical Report BAW-1621/B&W 177-FA addressed the same safety related components as those identified in SLRA Section 3.5.2.2.2.6 Subsection "Reactor Vessel Support Steel Evaluation." 3. Discuss the cited "acceptance limits" in Topical Report BAW-1621/B&W 177-FA and identify to which of the SLRA	

				Section 3.5.2.2.2.6 Subsection “Reactor Vessel Support Steel Evaluation” RV steel or concrete support components the 10 percent exceedance applied (or still applies) and state the reason for the exceedance. 4. Discuss whether the cited “acceptance limits” of Topical Report BAW-1621/B&W 177-FA remain in effect and are the same as those in ANP-3898P Section 9.4.2, “Design Stress Summary,” or have been conservatively revised including their potential modifications for radiation, and thermal effects for all applied loadings for the subsequent period of extended operation.	
4 DD	FE 3.5.2.2.2.6 (RV Support Steel Evaluation)	3-1322	Topic: **Calculations, RV Steel Support Assembly Design Stresses** a) Section 9.4.2 of ANP-3898P shows calculated values of stress intensities. b) Figure 9-2 of ANP-3898NP presents a diagram of the RPV support assembly that shows the vertical bearing plate and nelson studs; Section 9.3 of ANP-3898NP states that the vertical bearing plate and nelson studs do not support the RPV support assembly.	a) Clarify how these stress intensities were calculated (e.g., through finite element analysis or hand calculations). b) Confirm that the vertical embedment bearing plate and nelson studs were not included when calculating the stress intensities in Section 9.4.2 of ANP-3898P (since these components claimed to not	

			c) In Section G5 of Drawing No. 128668E, Rev. D7, a cross-hatched component just above the RPV inlet/outlet nozzles appears to provide lateral support for the RPV.	perform a support function for the RPV) and the stresses in the evaluation of the support flange and associated welds discussed in Section 9.4.5 of ANP-3898NP. c) Clarify the function of the cross-hatched component in terms of support for the RPV. If it does not support the RPV, confirm that this component was not included when calculating the stress intensities in Section 9.4.2 of ANP-3898P and the stresses in the evaluation of the support flange and associated welds in Section 9.4.5 of ANP-3898P.	
5 DD	FE 3.5.2.2.2.6 (RV Support Steel Evaluation)	3-1322	Topic: **Calculations, Initial NDT of RV Steel Support Materials** Section 9.4.4.3 of ANP-3898NP shows that the equation for adjusted reference temperature includes the initial NDT and associated margin.	Discuss the sources of the initial NDT and margin of the materials of the RPV steel support assembly listed in Tables 9-4 and 9-5 of ANP-3898NP.	
6 DD	FE 3.5.2.2.2.6 (RV Support Steel Evaluation)	3-1322	Topic: **Calculations, Determination of Lowest Service Temperature (LST)** Section 9.4.3 of ANP-3898NP discusses the determination of LST for the evaluation of the RPV steel supports. A 2D finite element thermal model was used.	Go over modeling details such as material properties, thermal boundary conditions, 2D symmetry assumptions, potential impact of ventilation holes in the RPV support skirt (especially the larger holes) in the determination of LST.	

7 DD	FE 3.5.2.2.2.6 (RV Support Steel Evaluation)	3-1322	Topic: **Inspections, Required ASME Code Section XI for RV steel supports** There is no discussion of the results of the required ASME Code Section XI examinations for the RPV support steel assembly (including weldments) in the main part of the SLRA. Section 9.4.1 of ANP-3898NP (Enclosure 4, Attachment 1 to the SLRA) discussed results of visual examinations performed. The ePortal document "OISI-0169.10-0050 BASIS DOC" indicates that the dutchman-to-skirt welds are inspected per IWB of ASME Code Section XI (examination category B-K, Item No. B10.10). However, page 72/89 in DUKE-002-002 "Assessment of the Oconee Nuclear Station ONS3 ISI Assessment Report," Revision 0, states that the weld between the lower head and support skirt (weld WR36) is not included in the B-K category.	Discuss the applicable ASME Code Section XI examination requirements (IWB or IWF or both) used for all the components of the RV support steel assembly (including any approved relief requests) and the results (recordable indications or conditions detected, if any) of the most recent examinations. The staff noted Relief Request 15-ON-004 discussed in Section 9.4.1 of ANP-3898NP, but what about any reliefs from IWB of ASME Code Section XI for the dutchman-to-skirt welds?	
8 AP	FE 3.5.2.2.2.6 (RV Support Steel Evaluation)	3-1322	Topic: **Inspections, Embrittlement evaluations consistent with NUREG-1509** Framatome 86-9314756-001 (page 9/44) states that [[XX]] Section 4.3.1.1 of NUREG-1509 states that physical examination of the RPV supports is essential to the evaluation. The purpose of the examination is to detect visible signs of degradation of the supports, including, but not limited to, rust, corrosion, cracks, or permanent deformation of the members. Figure 4-2 of NUREG-1509 identifies "evaluate existing physical condition" as one of the key inputs to the "preliminary evaluation". Examinations performed to date on the RV supports as part of	All things being equal, it is not clear why the Unit 1 VT-3 examination results are rejected, while those for Unit 2 and 3, were found acceptable. It is also not clear whether historical records exist dating back to initial RV structural steel support IWF examinations to indicate area covered, their effectiveness to identify aging effects or clearly indicating that no aging effects are occurring or if they do their rate at which it occurs are so slow so that they are of no	

			the ASME Section XI, Subsection IWF ISI Program, which consists at the minimum of VT-3 exams for Units 1, 2, and 3 are summarized in WOs 02057934, 02026125, and 01985270, respectively. The OE in WO 02057934 states that the outside [anchor] bolts not incased in concrete are painted and the paint is cracked. However, with no obvious rust or material loss the Level II inspector rejects the results citing 66.5% of VT-3 coverage and "limitations in NDE-91." The Level II inspector in WO 02026125 for same coverage and with no rust corrosion evident on the bolts accepts the results and further identifies flaking of paint skirt. Same holds for Unit 3 WO 01985270.	concern and hence there is no need to manage any of the slow observed aging effects during the subsequent period of extended operation. Please explain.	
9 DD	FE 3.5.2.2.2.6 (RV Support Steel Evaluation)	3-1322	Topic: **Inspections, Corrosion of Anchor Bolts stated in ISI Examination Report** This question is related to the ISI examination in #7. Appendix A (page A-7) of Document 86-9314756-001 states that [[XX]]. Note that these two questions are specifically for the anchor bolts. See other questions on the topic of boric acid corrosion on the RPV steel support assembly below.	a) Discuss how potential corrosion issues on the anchor bolts impact the evaluation in ANP-3898NP. b) What is the identified [[XX]] on page 24 of Document 86-9314756-001.	
10 AP	FE 3.5.2.2.2.6 (RV Support Steel Evaluation)	3-1322	Topic: **Inspections, Boric Acid Corrosion, RV Anchorage** AR 01809387 refers to ONS Unit 2 RV annulus cavity states that boric acid affected equipment including the reactor vessel skirt and annulus area (e.g., annulus shielding). It states that although dry boron residue exists on the skirt, it	Discuss how the boron residue on the RV skirt was discovered. Was it a part of maintenance rule inspections, part of an ISI scheduled inspection, or a periodic inspection promulgated by the Boric Acid AMP, another AMP, or multiple AMPs working	

			did not exhibit signs of material degradation. AR 02300737 discusses borated water penetration beneath base plates resulting in potential corrosion of under support base plates and anchor bolting. For metal components, such as base plates, with residue of dried boric acid, corrosion rates vary. The reactions are environment dependent with rates varying accordingly. Loss of material in base plates for dry acid reaction could be as low as 0.0002 in/yr. (see AR 01809387) with higher rates to 0.007 (see AR 01910016) in/yr. experienced in wetter environments. Given the potential reduction in base plate thickness (loss of material) due to boric acid corrosion, it is possible the vessel anchorage to experience loss of preload. Detection of aging effects program element of every AMP in GALL-SLR suggests methods and techniques (i.e., visual, volumetric, surface inspection), frequency, sample size, data collection, and timing of new/one-time inspections to ensure timely detection of aging before there is a loss of a component's or structure's intended function.	in concert to ensure that corrosion is not occurring or if it is, its rate is sufficiently slow so that it is not of a concern, because of steps taken during anchor bolt installation (e.g., sole plate shimmed). If not, discuss measures taken for the base plate to remain intact and the anchor bolts to maintain their preload during the subsequent period of extended operation.	
11 AP	FE 3.5.2.2.2.6 (RV Support Steel Evaluation)	3-22, 3-41	Topic: **RV Skirt TLAA** SLRA Table 1, AMR item 3.1.1-004, states that the "[s]teel pressure vessel support skirt and attachment welds" are subject to "[c]umulative fatigue damage: cracking due to fatigue, cyclic loading. The referenced SLRA Further Evaluation Section 3.1.2.2.1, "Cumulative Fatigue Damage," states that Table 1, AMR item, [3.1.1-004] – addresses the evaluation of fatigue of the steel reactor vessel support skirt	Discuss how the TLAA for the skirt is dispositioned consistent with 10 CFR 54.21(c)(1). If the TLAA is dispositioned as a: (i) please provide calculations indicative that the design basis for fatigue analysis of RV support skirt assembly remains valid for the subsequent period of extended operation; if it is dispositioned as a (ii) please	

			components in “reactor vessel, internals, and reactor coolant system, and is addressed in Section 4.3, “Metal Fatigue”. A review of SLRA Table 3.1.2-1, “Reactor Vessel, Reactor Internals, and Reactor Coolant System - Reactor Vessel - Aging Management Evaluation,” indicates an AMR item for fatigue analysis for the support skirt and its weldments exists (Figure 5-10 of UFSAR confirm that) and that the aging effect and aging mechanisms are as noted in SLRA item 3.1.1-004. However, a review of SLRA Section 4.3, does not indicate whether the TLAA for the skirt and its weldments are dispositioned consistent with 10 CFR 54.21(c)(1) as a (i), (ii), or (iii).	provide calculations indicative of the fatigue analysis projected to the end of the subsequent period of extended operation; if it is dispositioned as a (iii) identify the relevant AMP or provide a plant specific AMP to manage the effects of aging on the intended function(s) of the support skirt assembly to the end of the subsequent period of extended operation. Supplement the SLRA as needed. If this is an error in the SLRA provide a supplement with updates as needed.	
12 DD	FE 3.5.2.2.2.6 (RV Support Steel Evaluation)	3-1322	Topic: **Confirm Applicability of Evaluation to All ONS Units** Whenever the SLRA or Section 9 of ANP-3898P/NP doesn’t refer to a specific ONS unit, confirm that the subject topic is applicable to all three ONS units (e.g., summary of configuration on page 3-1322 of the SLRA and Section 9.3 of ANP-3898P/NP).	Confirm that the subject topic is applicable to all three ONS units (e.g., summary of configuration on page 3-1322 of the SLRA and Section 9.3 of ANP-3898P/NP).	
13 AP	FE 3.5.2.2.2.6	4-24, 4-26, 4-28	Topic: **Confirm Applicability of Evaluation to All ONS Units (Follow-up to BOQ 12)** SLRA Tables 4.2.1-1, 4.2.1-2, 4.2.1-3, for 72 EFPY ONS Units 1, 2, and 3 for “Reactor Vessel Shell Locations With Fluence > 1.0E+17 n/cm^2 (E > 1.0 MeV)” and SLRA Figures 4.2-2, 4.2-3, 4.2-4 provide information on ONS Unit 1, 2, and	Clarify if there are differences in construction/methods of construction for the three ONS RVs and if so how would these affect each of the RV steel support (skirt) assembly (including welds) embrittlement,	

			3 Reactor Vessel construction. It appears that the shell for Unit1 is constructed from (bent?) plates, while shells for Units 2 and 3 are manufactured through forging. It is not clear whether forging compressive stresses, if any, have been accounted for embrittlement in subsequent welding of the skirt.	underlying RV concrete anchorage and pedestals.	
14 AP	FE 3.5.2.2.2.6 (RV Support Steel Evaluation)	3-1322	Topic: **Confirm Applicability of Evaluation to All ONS Units (Follow-up to BOQ 12).** The SLRA states that "[t]he evaluation of reactor vessel support steel determined that the reactor vessel support flange (all three ONS units) and the ONS Unit 1 and Unit 2 reactor vessel support flange welds connecting the 90 degree segments together to form a circular support plate are potentially susceptible to reduction of fracture toughness by irradiation embrittlement at 72 EFPY of operation." There is no discussion, however, for the Unit 3 welds at that location. It appears that SLRA 3.5.2.2.6 have omitted discussion of RV support flange welds for Unit 3. It appears from Table 9-7 of ANP-3898P, that the [[XX]].	Confirm that ONS Unit 3 reactor vessel support flange welds connecting the 90 degree segments together to form a circular support plate although are similar to those of Unit 1 and Unit 2, and unlike those for Units 1 and 2, [[XX]].	