From:	Sayoc, Emmanuel
To:	Paul Aitken; Eric A Blocher
Cc:	Wu, Angela
Subject:	Surry SLRA Draft RAIs - Irradiation Effects on CBS and RV Steel Supports, ASME Section XI, Subsection IWF
Date:	Tuesday, June 04, 2019 12:36:00 PM
Attachments:	076 SPS RAI AMR Irradiation Effects CBS Wall RV Steel Support (Consolidated - 11 RAIs) AP SK MY AC AB GT 5-31-19 ERO.docx
Importance:	High

Paul, Eric,

Surry SLRA Draft RAIs outlined below. We will assume you want a clarification call until you indicate otherwise.

TRP	RAI Number	Issue	Branch	Reviewer
76	3.5.2.2.2.6-1	Irradiation Effects on CBS and RV Steel Support	ESEB	Thomas, Prinaris, Buford
76	3.5.2.2.2.6-2	Irradiation Effects on CBS and RV Steel Support	ESEB	Thomas, Prinaris, Buford
76	3.5.2.2.2.6-3	Irradiation Effects on CBS and RV Steel Support	ESEB	Thomas, Prinaris, Buford
76	3.5.2.2.2.6-4	Irradiation Effects on CBS and RV Steel Support	ESEB	Thomas, Prinaris, Buford
76	3.5.2.2.2.6-5	Irradiation Effects on CBS and RV Steel Support	ESEB	Thomas, Prinaris, Buford
76	3.5.2.2.2.6-6	Irradiation Effects on CBS and RV Steel Support	ESEB	Thomas, Prinaris, Buford
76	3.5.2.2.2.6-7	Irradiation Effects on CBS and RV Steel Support	ESEB	Thomas, Prinaris, Buford
76	3.5.2.2.2.6-8	Irradiation Effects on CBS and RV Steel Support	ESEB	Thomas, Prinaris, Buford
76	3.5.2.2.2.6-9	Irradiation Effects on CBS and RV Steel Support	ESEB	Thomas, Prinaris, Buford
76	3.5.2.2.2.6-10	Irradiation Effects on CBS and RV Steel Support	ESEB	Thomas, Prinaris, Buford
43	B.2.1.31-1	ASME Section XI, Subsection IWF	ESEB	Prinaris, Buford

Thanks

Emmanuel “Manny” Sayoc
Safety Project Manager
NRR/DLR
301-415-4084

RAIs: TRP 76 – Irradiation Effects on CBS and RV Steel Support (10 RAIs), and TRP 43 – IWF (1 RAI)

Regulatory Basis:

10 CFR 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. One of the findings that the staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. In order to complete its review and enable making a finding under 10 CFR 54.29(a), the staff requires additional information regarding the matters described below.

RAI 3.5.2.2.2.6-1 (Concrete and NST Fluence/Dose Estimates)

Background:

Dominion's Subsequent License Renewal Application (SLRA) Section 3.5.2.2.2.6, as supplemented by Change Notice 1 dated January 29, 2019 (ADAMS Accession No. ML19042A137), discusses its "Further Evaluation" of the aging effects of irradiation on the Concrete Biological Shield (CBS) Wall (or Primary Shield Wall) and the Reactor Vessel (RV) Support Steel Assembly (consisting of the Neutron Shield Tank (NST) and sliding foot assembly. The SLRA concludes that no plant-specific aging management program to manage the effects of irradiation is required. The SLRA, as supplemented, discusses evaluations in support of Dominion's estimation of projected fluence and dose to the end of the subsequent period of extended operation (SPEO) at the CBS wall and at the NST, respectively, for comparison against the applicable threshold criteria for concrete in the SRP-SLR Section 3.5.2.2.2.6, and as input to the fracture mechanics evaluation for embrittlement of the RV support steel assembly.

Issue:

The conclusions made in the SLRA with respect to the need for aging management of the concrete CBS wall, NST, and related RV support structures depends, in part, on the projected fluence/dose at the end of the SPEO. The information presented in the SLRA is not sufficient to allow the NRC staff to determine whether reasonable assurance exists that the limiting fluence/gamma dose values have been identified, with sufficient margin and conservatism to accommodate uncertainties due to the relative lack of validation for fluence analysis methodologies directly applicable to the regions of interest. Therefore, with respect to the fluence/dose values presented in the SLRA and the context stated below, the NRC staff needs additional information:

1) During the audit, the NRC staff reviewed information from calculations performed in 2018 (LTR-REA-18-88 referenced in ETE-SLR-2018-1271) to determine the fluence/gamma dose at selected locations at Surry to the end of SPEO. These values provide additional validation of the fluence/dose values cited in the SLRA and SLRA

supplement for the CBS wall and NST. However, the SLRA does not provide details of this model and its results.

2) The SLRA provides information for fluence/gamma dose at the vessel side surface of the CBS wall at the limiting location for the RV traditional beltline region. This location includes attenuation of the fluence through the NST. Based on a review of relevant figures and drawings (e.g., 11448/11548-FV-7A, 11448-FM-1G), there are regions of the CBS wall above and below the NST. The fluence incident on these regions do not to the staff to be attenuated by the steel or water present in the NST, so even though these regions are further from the traditional RV beltline, they may experience greater fluence than the part of the CBS wall closest to the RV traditional beltline region. This is especially true for neutron fluence, since a large number of neutrons would not be moderated by the NST water to energies below the lower threshold for inclusion in the fluence estimates.

<u>Request:</u>

1) Provide a brief summary of the origin, details, and validation of the model used in the calculations in LTR-REA-18-88 referenced in ETE-SLR-2018-1271, including the methodology used and relevant model characteristics, to allow the NRC staff to evaluate the adequacy of the model to compute fluence in areas beyond the traditional beltline region of the RV (i.e., the area of applicability envisioned by the NRC approved methodology in the available regulatory guidance in Regulatory Guide 1.190). In addition, provide a summary of the key limiting results for the CBS wall and the NST.

2) Provide an estimate for the maximum neutron fluences (E > 0.1 MeV) and gamma doses associated with the regions on the vessel side surface of the CBS wall above and below the NST, <u>or</u> a justification for why the fluence/dose in these regions is bounded by other available fluence estimates.

3) If the limiting values of fluence/gamma dose in any portion of the CBS exceed the threshold criteria in SRP-SLR, describe how the aging effects of irradiation on concrete will be adequately managed, pursuant to 54.21(a)(3) in those areas; or, provide a summary of a structural evaluation and its results that demonstrate that the CBS wall will remain capable of performing its intended function through the end of the SPEO.

RAI 3.5.2.2.2.6-2 (Operating Experience Bases)

<u>Background:</u>

One criterion, among others, in SRP-SLR Section 3.5.2.2.2.6 for not requiring a plant-specific program for managing aging effects of irradiation is for the applicant to demonstrate that there is no plant-specific operating experience (OE) of irradiation degradation that may impact intended function(s) of applicable materials and components.

SLRA Section 3.5.2.2.2.6, as supplemented by Change Notice 1 dated January 29, 2019 (ADAMS Accession No. ML19042A137), states "no plant-specific OE [operating experience] of concrete irradiation degradation has been identified." The SLRA supplement Section 3.5.2.2.2.6, also states that "[t]here is no plant-specific or industry OE of reactor vessel support assembly irradiation degradation that would impact a license renewal intended function."

Issue:

It is not clear what actions may have formed the bases for SPS to make the above plant-specific OE statements related to irradiation degradation of CBS wall and RV steel support assemblies.

Request:

State what actions (e.g., surveillances, inspections, observations, tests), if any, were taken by SPS to provide justification for the plant-specific OE statements made above for irradiation degradation of CBSW and RV steel support assemblies.

RAI 3.5.2.2.2.6-3 (Whether Structural Consequence Analyses Exists in CLB)

Background:

SLRA Section 3.5.2.2.2.6, as supplemented by Change Notice 1 dated January 29, 2019 (ADAMS Accession No. ML19042A137), states:

> The PTR fracture mechanics evaluation on the reactor vessel support steel assembly predated resolution of Generic Safety Issue 15 (GSI-15), "Radiation Effects on Reactor Pressure Vessel Supports," in 1996, as reported in NUREG-0933 which states in part:
>
> > The preliminary conclusion indicated that the potential problem did not pose an immediate threat to public safety. The tentative results indicated that plant safety could be maintained despite reactor vessel support structures (RVSS) radiation damage. In order to encompass the uncertainties in the various analyses and provide an overall conservative assessment, several structural analyses conducted demonstrated the following:
> >
> > (1) Postulating that one of the four RPV supports was broken in a typical PWR, the remaining supports would carry the reactor vessel and the load even under safe-shutdown earthquake (SSE) seismic loads;
> > (2) If all supports were assumed to be totally removed (i.e., broken), the short span of piping between the vessel and the shield wall would support the load of the vessel.

Issue:

It is not clear if supporting plant-specific structural consequence analyses, that postulate failure of one or more RV support assemblies, like those cited above from NUREG-0933, exists in the current licensing basis (CLB) for SPS Units 1 and 2.

Request:

State if plant-specific structural consequence analyses, postulating failure of one or more RPV support assemblies, exists in the CLB of SPS Units 1 and 2. If they do exist, describe in sufficient technical detail the consequence analyses performed and its results.

RAI 3.5.2.2.2.6-4 (Apparent Discrepancy of Certain Fluence Values cited in SLRA)

Background:

The criteria in SRP-SLR Section 3.5.2.2.2.6 requires a plant-specific program for managing aging effects of irradiation in concrete if the estimated (calculated) fluence levels or irradiation dose received by any portion of the concrete from neutron **(fluence cutoff energy E > 0.1 MeV)** or gamma radiation exceeds the respective threshold level stated therein during the subsequent period of extended operation, or if there is plant-specific operating experience (OE) of irradiation degradation that may impact intended functions. SLRA Section 3.5.2.2.2.6, as supplemented by Change Notice 1 dated January 29, 2019 (ADAMS Accession No. ML19042A137), states on page 4 of 6 of Enclosure 2: "The maximum neutron fluence at the CBS wall surface of 1.18 x 10^{13} n/cm^2 (**E > 1.0 MeV**)" (emphasis added).

Further, the SLRA supplement Section 3.5.2.2.2.6, under sub-title "Irradiation of the RV Support Steel Assembly," of Enclosure 2 states that "[t]he PTR [Project Topical Report] was conservatively estimated for 100 years of plant operation (76.8 EFPY [effective full power years]) that yields a fast neutron fluence (E > 1 MeV) of 9.5 x 10^{19} n/cm^2 at the inside surface of the RV and a fast neutron fluence (E > 1 MeV) of 5.0 x 10^{19} n/cm^2 at the outside surface of the RV."

Additionally, Enclosure 2 of the SLRA supplement states: "The projected EFPY for SPS SLR is 68 EFPY which yields a fast neutron fluence (E > 1.0 MeV) of 3.42 x 10^{18} n/cm^2 at the inside surface of the NST."

Issue:

1) The estimated neutron fluence level on the CBS wall is cited in the SLRA in terms of cutoff energy E > 1.0 MeV, whereas the neutron fluence acceptance threshold in the SRP-SLR Section 3.5.2.2.2.6 is in terms of cutoff energy E > 0.1 MeV; for appropriate comparison, they need to be stated based on the same cutoff energy as the threshold criteria in the SRP-SLR.
2) The staff audited the Project Topical Report (PTR) 2178-1525314-B4 "Unit No. 1 Surry Power Station – Life Extension Evaluation of the Reactor Vessel Support," dated October 10, 1986, and noted that the fast neutron fluence (E > 1.0 MeV) at the outside surface of the RV, used for the evaluation for 100 calendar years of operation (76.8 EFPY) is 5.0 x 10^{18} n/cm^2. This fluence value is inconsistent with that of 5.0 x 10^{19} n/cm^2 cited in the SLRA.
3) The staff audited ETE-SLR-2018-1271, "Assessment of Radiation Effects on Reactor Vessel Supports for SPS Units 1 & 2," Revision 0, and noted that in its Table 3 the reported fast neutron fluence (E > 1.0 MeV) at the inside vessel side surface of the NST is 3.82 x 10^{19} n/cm^2 for Unit 2 at 68 EFPY. This fluence value is inconsistent with that of 3.42 x 10^{18} n/cm^2 cited in the SLRA.

Request:

1) Provide the maximum calculated neutron fluence values for the CBS wall for SPS Units 1 and 2 based on the cutoff energy for concrete damage as defined in SRP-SLR Section 3.5.2.2.2.6 (i.e., E > 0.1 MeV).

2) Clarify the inconsistency between the fast neutron fluence (E > 1 MeV) at the outside surface of the RV, cited in the SLRA with that used in the PTR for 100 calendar years of operation (76.8 EFPY), and provide the correct value to the end of the SPEO.

3) Clarify the inconsistency between the fast neutron fluence (E > 1 MeV) at the inside (vessel side) surface of the NST cited in the SLRA and that reported in ETE-SLR-2018-1271 for 68 EFPY. State which reactor Unit experiences the bounding fluence and provide the bounding fluence value.

RAI 3.5.2.2.2.6-5 (Applied Stresses and Fracture Mechanics Evaluation - Methodology and Results)

Background:

SLRA Change Notice 1, dated January 29, 2019 (ADAMS Accession No. ML19042A137), supplemented SLRA Section 3.5.2.2.2.6 with a new subsection entitled, "Irradiation of the Reactor Vessel Support Steel Assembly," to address the aging effect of loss of fracture toughness due to neutron irradiation embrittlement of the reactor vessel (RV) support steel materials in the neutron shield tank (NST). The applicant's evaluation is based up on the audited Project Topical Report (PTR): "Reactor Vessel Support for Unit No 1 Surry Power Station, Life Extension Evaluation of the Reactor Vessel Support, including Appendix 3, Resistance to Brittle Fracture of the Neutron Shield Tank Materials," October 10, 1986.

This supplemental discussion in the SLRA states that, in this PTR evaluation, the applied stresses for the area of the NST subject to high neutron fluence were developed and compared to the critical (allowable) stresses derived from the fracture toughness evaluation. These evaluations were performed to determine the structural integrity of the Surry Unit 1 NST through the end of projected plant life or to the end of the SPEO. The applied stresses were updated in the audited report ETE-SLR-2018-1270, "Review of Loads on Neutron Shield Tank for SPS Units 1 & 2 Reactor Vessel Supports," Revision 0. The assessment of the PTR to support the supplemented SLRA is discussed in audited report ETE-SLR-2018-1271, "Assessment of Radiation Effect on Reactor Vessel Supports for SPS Units 1 & 2," Revision 0. The evaluations concluded that: a) the applied stresses calculated from the peak stress values for the associated loads of the NST were demonstrated to be below the critical (allowable) stress for a through wall flaw and a surface flaw, and b) loss of fracture toughness due to irradiation embrittlement is not an aging effect requiring management for the NST. The supplemental discussion further states that the PTR evaluation was updated for SLR in ETE-SLR-2018-1271, which validated that the original PTR evaluation is bounding for: a) the Surry Unit 2 NST, b) the applied stresses for both units through the subsequent license renewal period, and c) the 80-year projected fluence values at the inner surface of the NSTs.

NUREG-1509, “Radiation Effects on Reactor Pressure Vessel Supports,” provides an engineering approach, including screening criteria and technical evaluation procedures, to reassess the structural integrity of the reactor pressure vessel supports.

The staff noted that the audited Attachment 2 of CM-AA-ETE-101, Technical Report CE-0087, “Condition Monitoring of Structures,” Revision 7, includes up to 10 percent (minor) loss of material in the design of all SPS steel structures. The staff also noted that this report is being revised to include the NST steel structure.

Issue:

In the supplemented SLRA, the applicant provided the conclusions from the PTR and the updated evaluation that addresses the SLR period. However, the SLRA did not provide sufficient docketed details regarding the methodology used in the updated evaluation of the PTR, including derivation of the critical (allowable) and controlling applied stresses to assess the NST structural integrity during the SPEO. It is also not clear if this evaluation was performed consistent with the NRC staff guidelines in NUREG-1509. Therefore, the NRC staff needs additional information to determine the adequacy of the fracture mechanics and applied stress evaluations (subject to a 10 percent reduction in cross sectional areas as noted in Technical Report CE-0087) of the NSTs and the evaluations remain valid through the end of the SPEO.

Request:

1) Identify and justify the specific loads (e.g., seismic, LOCA, anticipated thrust forces exerted by friction if any), loading conditions/loading combinations used or omitted as not applicable in the above postulated fracture mechanics evaluation(s) of the NST for all calculated applied stresses. State the controlling load combination, the limiting applied stresses and its location for the NSTs.

2) State whether all applied stresses considered for the fracture mechanics calculations of the NST were augmented to include the 10 percent reduction in steel section for loss of material due to corrosion as promulgated in Technical Report CE-0087.

3) In regard to the update to the PTR evaluation in report ETE-SLR-2018-1271 to support subsequent license renewal,
    a. Describe in detail the methodology used to perform the fracture mechanics evaluation and to calculate the corresponding critical (allowable) stresses with flaws for the NST. Include in this summary the key assumptions and inputs used, and how the evaluation accounted for the complete neutron fluence spectrum (i.e., slow and fast neutrons), added factors of safety to satisfy margins if any, alloy metals in NST steel, and other additional applicable variables.
    b. Provide the calculated critical (allowable) stresses for both a through-wall flaw and surface flaw and state whether these are bounding for any anticipated crack growths to the end of the SPEO.

4) Demonstrate that the fracture mechanics evaluation accounts for the effects of irradiation embrittlement of the weld metals used and developed heat affected zones of the parent metal in NST.

RAI 3.5.2.2.2.6-6 (AMPs for aging management of NST for structural support function)

Background:

Scoping and Screening results of mechanical systems in the SLRA describes the functions of the Neutron Shield Tank (NST). SLRA Section 2.3.1.1, "Reactor Vessel [RV]" states that "[t]he reactor coolant system includes a neutron shield tank [(NST)] located inside the primary shield wall around the reactor vessel." It also states that "[t]he tank provides support for the reactor vessel as described in the structural section of the application, but aging management of the neutron shield tank is addressed in this section [mechanical section] of the application."

Section IX.E, "Aging Effects," and Section IX.F, "Significant Aging Mechanisms," respectively, of the GALL-SLR Report discuss irradiation by neutrons that could result in embrittlement (aging mechanism) of carbon and low-alloy steels that may decrease their fracture toughness (aging effect).

Title 10 of Code of Federal Regulations (10 CFR) Part 54.4 requires that systems, structures, and components including those that assure the integrity of the reactor coolant pressure boundary remain functional during and following design-basis events and that their intended functions form the basis for including them within the scope of license renewal and subject to aging management review (AMR) such that they continue to fulfill their intended function consistent with the CLB in accordance with 10 CFR 54.21(a).

Issue:

UFSAR Section 15.6.2.2.1 states that the "reactor vessel support […] consists of six sliding foot assemblies mounted on the neutron shield tank [and that t]he neutron shield tank is a double-walled cylindrical structure that transfers the loadings to the heavy reinforced-concrete mat of the Containment structure." The UFSAR section also discusses the criteria and codes used for the design of the NST for structural support of the RV. Sections 3.3 and 4.3 of audited SPS "Technical Report CE-0087," Revision 7, states that NST steel "embrittlement due to neutron exposure" is a possible aging (degradation) mechanism in addition to others such as corrosion and chemical attack. Attachment 2 of the audited SPS procedure 0-NSP-BS-005 identifies the NST as a safety related structure being monitored for aging effects consistent with the license renewal rule (10 CFR 54) and required to remain functional during and after design basis events consistent with the definitions of 10 CFR 50.49(b)(1).

However, the SLRA does not identify whether an aging management program (AMP) and aging management results (AMRs) are proposed to manage aging effects as discussed in the aforementioned internal SPS procedure(s), including embrittlement due to neutron exposure aging mechanism, consistent with the requirements of 10 CFR 54.21(a)(3) and the so that the intended structural integrity and structural support function of the NST is maintained during the SPEO.

Request:

1) Discuss the SLRA AMP(s) or provide a plant-specific AMP, and AMRs that manage aging effects associated with the structural integrity and structural support function of the NST during the SPEO consistent with 10 CFR 54.21(a), or provide a justification that one is not required for the NST during the SPEO.

DRAI 3.5.2.2.2.6-7 (Impact of NST Leakage)

Background:

Scoping and Screening results of mechanical systems of SLRA describes the cooling functions of the Neutron Shield Tank (NST) system. SLRA Section 2.3.1.3, "Reactor Coolant," states that "[t]he reactor coolant system includes a neutron shield tank located inside the primary shield wall around the reactor vessel," and that aging management of the neutron shield tank is addressed in the mechanical section of the application. SLRA Section 2.3.3.9, "Neutron Shield Tank Cooling," states that "[t]he neutron shield tank cooling system provides cooling for the neutron shield tank fluid which is heated by attenuation of neutron and gamma radiation in the vicinity of the reactor vessel. Heat removal is provided by the component cooling system. The neutron shield tank cooling system also removes heat from the primary shield wall." SLRA Section 3.5.2.2.2.6, "Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation," identifies the heated water to be contained within the 1-1/2-inch-thick steel shell walls of the tank.

Table 3.3.2-9, "Auxiliary Systems - Neutron Shield Tank Cooling - Aging Management Evaluation," of the SLRA lists several NST cooling components and aging effects, but none directly related to loss of material or cracking of the NST.

Title 10 of Code of Federal Regulations (10 CFR) Part 54.4 requires that systems, structures, and components including those that assure the integrity of the reactor coolant pressure boundary remain functional during and following design-basis events and that their intended functions form the basis for including them within the scope of license renewal and subject to aging management review (AMR) such that they continue to fulfill their intended function consistent with 10 CFR 54.21(a).

Issue:

UFSAR Section 11.3, states that "[p]rimary shielding is provided to limit radiation emanating from the reactor vessel." It also states that "[t]he primary shield consists of a water-filled neutron shield tank […which…] designed to prevent overheating and dehydration of the concrete primary shield wall and to prevent activation of the plant components within the reactor containment. In its OE audit, the staff reviewed CA238320 included in CR479576 for SPS Unit 2 and noted that the NST has been experiencing chromated water leakage of up to two and one-half gallons per day since 1989. It is not clear how the NSTs can perform their radiation and thermal shielding functions to protect the reactor primary shield wall effectively when they experience unmitigated leakage. It is also not clear what corrective actions the applicant has taken to remedy leakage such as that experienced in the Unit 2 NST or plans to take for any potential NST leakages during the SPEO. It is further not clear what AMPs and AMRs address management of relevant aging associated with NST leakage.

Request:

1) Discuss proposed plans to maintain structural integrity of the primary shield wall (PSW) (i.e., reduce/eliminate overheating, dehydration, and radiation induced degradation of the reactor primary shield wall) when NSTs experience fluid leakage of fluid conducive to shielding of PSW.

2) Clarify what mechanical AMPs and AMRs address management of relevant aging associated with NST leakage issues.

DRAI 3.5.2.2.2.6-8 (NST Water Chemistry Sampling for Corrosion)

Background:

Scoping and Screening results of mechanical systems of SLRA Section 2.3.3.9, “Neutron Shield Tank [NST] Cooling” describes cooling of the NST fluid heated by attenuation of neutron and gamma radiation near the reactor vessel. SLRA Section B2.1.12., “Closed Treated Water Systems [CTWS]” describes activities including chemistry of the fluid used to prevent loss of material to the NST. The audited SPS procedure CH-93.400 “Closed Cooling Water Chemistry Program,” further delineates the fluid chemistry of the steel NST and indicates that it is monitored every refueling outage. SPS CH-93.400 procedure also states that the NST mitigating fluid chemistry is examined for its alkalinity and contents of chlorides, chromates, and iron.

The enhancement to the “detection of aging effects” program element of SLRA Section B2.1.12 AMP states that a new SPEO procedure will be developed to inspect a 20% sample of various populations (each material, water treatment program, and aging effect combination) every 10 years. The enhancement also states that if opportunistic inspections will not fulfill the minimum number of inspections by the end of each 10-year period, the program owner will initiate work orders as necessary to request additional inspections.

Issue:

The SLRA identifies NST to be subject to corrosion mechanism and its mitigating fluid to heat and radiation. Given that the CTWS program is a sampling program, it is not clear from the SLRA how the chemistry of the NST fluid is sampled (i.e., at the NST or at other components having the same material, environment, and aging effect characteristics). It is also not clear how the adverse localized environment of heat and radiation affect the chemistry of the contained fluid and if such chemistry has affected the NST internal (e.g., steel, seals, and other materials) construction.

Request:

1) Discuss how, where (including location if sampled within NST), and at what frequency the NST fluid is sampled. If chemistry data are not directly obtained at the NST but at other sampled components discuss the relevance of such components in providing accurate data that can be used to interpret loss of material at the NST.

2) Provide several years trending of relevant NST chemistry data to asses for loss of material OE evaluation. If chromate data has changed since the beginning of plant operation explain why and justify how so.

3) State the actions SPS will take if the NST fluid chemistry concentration(s) falls outside the acceptance limits noted in procedure CH-93.400, such that the intended function of the NST fluid is maintained during the SPEO.

4) Discuss to what extent heat and radiation affects the NST fluid chemistry. State whether radiolysis of the NST fluid alters the characteristics of the fluid chemistry (e.g., that of water and chromated products used and added oxidation potential from the chromated water breakup) such that it could impact the intended functions of the NST.

RAI 3.5.2.2.2.6-9 (Potential Degradation of Lubrite® Lubricant)

Background:

In its SLRA, Section 3.5.2.1.36, "Component Supports," the applicant stated that Lubrite is a material of construction used in structural support subcomponents within the containment. The applicant also stated that aging effects such as loss of mechanical function require aging management for component support subcomponents. The applicant proposed to manage the effects of aging of Lubrite® exposed to air used to lubricate the sliding foot assemblies for the reactor vessel (RV) supports with the ASME Section XI, Subsection IWF AMP. However, the applicant did not identify whether the Lubrite® at the sliding foot assemblies is susceptible to degradation when exposed to radiation.

During the On-Site audit, the staff reviewed excerpts from EPRI Technical Report 3002013084, "'Structural Tools' – Long Term Operations: Subsequent License Renewal Aging Effects for Structures and Structural Components," (EPRI Report) and the Project Topical Report (PTR) for Unit No. 1 Surry Power Station, "Life Extension Evaluation of the Reactor Vessel Support dated October 10, 1986, (Life Extension Report) and had discussions with the applicant's staff. As stated in the Life Extension Report, the applicant uses Lubrite® Type II lubricant to lubricate the bottom of the sliding block for the reactor vessel supports. The Lubrite® is described as a solid lubricant comprised of graphite and an organic binder.

However, the staff has not previously accepted the full EPRI Report or the Life Extension Report for use in subsequent license renewal and has not determined the applicability of the statements in these documents to potential aging effects of Lubrite® for this application. Additionally, both documents discuss the potential for organic materials to degrade when exposed to radiation, and the need to consider this as a potential aging effect.

The EPRI Report contained an excerpt that stated "…humidity, high temperature, and radiation are not significant in the aging of Lubrite." However, the EPRI Report also states that change in materials properties due to radiation is an applicable aging effect if the gamma radiation exceeds a previously defined limit. Additionally, the EPRI Report recommends that "[e]ach plant should review specific material types of manufacturer's data for detailed information" regarding gamma radiation effects.

The PTR Life Extension Report states that if visual inspections under the ASME Code were implemented, they would not provide an indication of an impending lubrication failure. The PTR goes on to state that due to consequences of binding in the sliding foot assemblies and the potential for lubricant degradation, further study or monitoring for binding is recommended. The PTR Life Extension Report also states that at the time the report was written, it was unknown if radiation tests were performed on the lubricant and that "[t]he radiation stability of bonded solid lubricants, like Lubrite II, depends on the properties of the binder." Further, the Life Extension Report states that "…on-line monitoring to detect stick-slip behavior may be implemented, especially if the long-term properties of the lubricant cannot be reliably ascertained."

Additionally, the PTR Life Extension Report states that it is possible to tolerate radiolytic degradation of the binder if it does not produce an adverse effect on the binder's cohesion or adhesion properties. However, the Life Extension Report does not discuss the binder's cohesion or adhesion properties for the staff to assess whether the specific binder used in the Lubrite® at Surry would be able to withstand radiolytic degradation.

Issue:

As noted in both the EPRI and the PTR Life Extension Reports, Lubrite® has the potential to degrade when exposed to radiation. Additionally, audited literature from Lubrite® Technologies as provided by the applicant states that radiation can degrade lubricants and therefore each lubricant must be designed to meet the specific conditions encountered. Because the applicant has not provided information that demonstrates the lubricant used at Surry was designed to withstand the expected radiation fluence/dose over 80 years, it is not clear to the NRC staff that the Lubrite® used in the construction of RV sliding shoe assemblies will continue to perform its intended function throughout the SPEO, and whether its degradation will not impose additional applied stresses on the NSTs and RVs. Potential loss of lubricating ability of the Lubrite® may need to be considered in conjunction with the RAIs dealing with applied stresses for the RPV sliding shoe assemblies.

Request:

1) Clarify which Lubrite® lubricant is used in the sliding foot assemblies for the RV supports.

2) Clarify whether the organic binder is designed to sustain degradation and still ensure the lubricant can perform its intended function for the subsequent period of extended operation:
    a) If so, provide the technical justification as to why the binder degradation can be tolerated at Surry. The justification should account for aging effects due to radiation and fluence exposure that would be encountered by the lubricant during the SPEO (60 – 80 year span) at Surry. Discuss whether such degradation would impose additional adverse stress effects and the impact the stresses would have on the ability for the supports to perform their intended function.
    b) If not, provide qualification data, and compare to site-specific conditions, for the specific lubricant used at Surry that demonstrates the lubricant will not experience significant degradation due to environmental factors such as temperature, accumulated gamma radiation dose and flux, and neutron fluence and flux that this material is projected to receive (or be exposed to) through the SPEO. Note that for any qualification data provided it should include aging effects due to both slow and fast neutrons, if applicable.
    c) Considering the answers to a. and/or b. above, is the depletion rate of the lubricant sufficiently low to ensure the lubricant can perform its intended function through the end of the SPEO?

3) If the organic binder for the lubricant contains halogens, provide a discussion on how production of acids may impact corrosion of components in contact with the lubricant and justify why it will not contribute significantly to corrosion of these components.

4) State whether the accumulated gamma radiation dose, and neutron fluence the lubricant is projected to receive through the SPEO will degrade the graphite component of the lubricant. Include qualification data, and compare to site-specific conditions, for the lubricant that demonstrates the graphite component of the lubricant will not experience significant degradation that would impact the intended function of the lubricant or provide a justification for not needing to do so.

5) Based on operating experience data, provide confirmation that no degradation of the Lubrite lubricant (i.e. loss of mechanical function) has been observed at Units 1 and 2 of Surry.

DRAI 3.5.2.2.2.6-10 (Stress Corrosion Cracking of RV Support Sliding Foot Components)

<u>Background:</u>

The NRC staff audited CE-1653, “Review of Structural Adequacy of the Reactor Vessel Support Sliding Foot Assemblies – Surry Units 1 and 2” dated May 27, 2003. The report states that major components of each RV sliding foot assemblies (i.e., ball, socket plates, sliding block, stationary saddle block, and hold down plates) are fabricated from high strength maraging Vascomax ® 300 or 350 steels. The report also states that Vascomax ® 300 or 350 steels are susceptible to stress corrosion cracking (SCC) subject to environmental conditions.

The GALL-SLR Report Section IX.F, “Aging Mechanisms,” and its references state that for certain steels (in particular those containing Nickel) SCC is an aging effect that needs to be managed. SPS UFSAR indicates that Vascomax ® is a maraging iron-based steel alloy that includes a large percentage of nickel as an alloy strengthening agent.

In addition, the SLRA supplement by letter dated January 29, 2019 (ADAMS Accession No. ML19042A137), references Project Topical Report (PTR) 2178-1525314-B4 “Unit No. 1 Surry Power Station – Life Extension Evaluation of the Reactor Vessel Support,” dated October 10, 1986, states that the components of the sliding foot assembly were coated with Heresite™ VR 514 (a phenolic coating). The NRC staff audited the PTR and noted that it states that the Heresite™ coating may not be needed to prevent stress corrosion cracking of the maraging steel components of the sliding foot assembly unless normal operating loads are exacerbated by lubrication failure. The Surry SLRA as revised, does not appear to discuss the Heresite™ coating, or whether it has applicable aging effects requiring management.

<u>Issue:</u>

The Life Extension Report discusses the use of Heresite™ VR 514 as a preventive coating to manage Vascomax ® steels susceptibility for SCC. It is not clear how the applicant would manage SCC of Vascomax ® steels used in the RV shoe assembly components, if the coating cannot provide the required adequate protection for SCC of Vascomax ® steels subject to environmental conditions, including radiation exposure, during the SPEO. The staff noted that there was no AMP or AMRs that address the susceptibility of Vascomax ® steels to SCC. It is also unclear whether the Heresite™ VR 514 coating is subject to any aging effects requiring management, and if so, whether degradation of the coating is being managed by any AMPs.

Request:

1) Identify what AMP and AMRs will SPS use to manage the effects of aging due to SCC for the Vascomax ® steels used in the fabrication of the RV shoe assembly components.

2) State whether the Heresite™ coating(s) used, is (are) subject to any aging effects requiring management or credited for corrosion control of components that are in-scope for the SLRA.

3) Clarify and justify if no management of aging effects for Vascomax ® steels and/or of the Heresite™ coating(s) used in the RV shoe assembly components is required.

RAI B2.1.31-1 (TRP 43 – IWF: Expansion/Post-installed Anchors)

Background:

GALL-SLR Report AMP XI.S3, ASME Code, Section XI, Subsection IWF, in "scope of program," "parameters monitored," "acceptance criteria," and "operating experience" program elements discusses anchor bolts and/or support anchorage to building structure of ASME Class 1, 2, 3, and MC component supports.

Regulatory Guide (RG) 1.199, "Anchoring Components and Structural Supports in Concrete" provides guidance to licensees and applicants on methods acceptable to the NRC staff for complying with the NRC's regulations in the design, evaluation, and quality assurance of anchors (steel embedments) used for component and structural supports on concrete structures. In addition, subsequent to the OE review of improperly installed pipe support expansion anchors, the NRC communicated to the industry its concerns regarding inadequate installation of post-installed anchors (expansion anchors) in pipe supports in Information Notice (IN) 2010-01, "Pipe Support Anchors Installed Improperly." Specifically, IN 2010-01 emphasizes the importance of anchor installation procedures and states that "[i]nsufficient oversight and training, improper tools, lack of detailed checklists, insufficient written instructions, inadequate supervision during construction, lack of inspection after work was completed, and inadequate quality assurance controls and quality control measures" are contributing factors to failure of the bolts.

Issue:

During the audit, the staff reviewed SPS OE relevant to pipe support post-installed anchors and noted that CR1096216 discusses a VT-3 exam leading to identification of a loose anchor bolt defined as a rejectable condition. The staff also noted that the applicant has in place procedures and guidelines to address installation of wedge-type concrete expansion anchor bolts manufactured by HILTI, Drillco, Maxi Bolts and others. In its review of audited procedure/specification NAS-1023/NUS-3007, Revision 2, dated March 19, 2015, the staff noted that the procedure/specification does not reference RG 1.199 nor IN 2010-01. The staff is not clear to what extent the applicant's procedure/specification NAS-1023/NUS-3007 aligns with the guidance provided in RG 1.199. The staff is also not clear how IN 2010-01 was dispositioned by the applicant.

Request:

1) Discuss how the applicant resolved the rejectable condition identified in CR1096216.

2) Clarify whether applicant's procedures/specifications regarding installation of expansion/post-installed anchors are consistent (in its entirety or in parts) with those described in RG 1.199.

3) Discuss how IN 2010-01 has been dispositioned and provide a summary of actions taken for improperly installed post-installed pipe support anchors, if any, at Surry.