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10 CFR 50.55a

RS-22-084

June 17, 2022

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Braidwood Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN-50-455

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Renewed Facility Operating License Nos. DPR-53 and DPR-69
NRC Docket Nos. 50-317 and 50-318

R.E. Ginna Nuclear Power Plant
Renewed Facility Operating License Nos. DPR-18
NRC Docket Nos. 50-244

Subject: Response to Request for Additional Information - Proposed Alternative for Examinations of Examination Categories B-B, B-D, and C-A Steam Generator Pressure Retaining Welds and Full Penetration Welded Nozzles

References:

1) Letter from D. Gudger (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Proposed Alternative for Examinations of Examination Categories B-B, B-D, and C-A Steam Generator Pressure Retaining Welds and Full Penetration Welded Nozzles," dated December 14, 2021, ADAMS Accession No. ML21348A078.

2) Email Letter from J. Wiebe (U.S. Nuclear Regulatory Commission) to T. Loomis (Constellation Energy Generation, LLC), "Draft RAIs for Requests for Alternatives I4R-17, I4R-23, ISI-05-018, I6R-10 (EPID Nos.: L-2021-LLR-091, L-2021-LLR-092, L-2021-LLR-093, L-2021-LLR-094)," dated May 6, 2022.

In the Reference 1 letter, Exelon Generation Company, LLC (Exelon) (now known as Constellation Energy Generation, LLC (CEG)) requested approval of a proposed alternative for Braidwood Generating Station (Braidwood), Units 1 and 2, Byron Generating Station (Byron), Units 1 and 2, Calvert Cliffs Nuclear Power Plant (Calvert Cliffs), Units 1 and 2, and R.E. Ginna Nuclear Power Plant (Ginna). This proposed alternative requested approval to defer the ASME Code, Section XI, Table IWB- 2500-1, Examination Categories B-B and B-D, and Table IWC-2500-1, Examination Category C-A steam generator examinations for the remainder of the currently licensed operating periods.

In the Refence 2 email, the U.S. Nuclear Regulatory Commission requested additional information. Attached is our response.

There are no commitments contained in this letter.

If you have any questions or require additional information, please contact Tom Loomis at (610) 765-5510.

Respectfully,

David T. Gudger
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David T. Gudger
Senior Manager - Licensing
Constellation Energy Generation, LLC

Attachment: Response to Request for Additional Information

cc: Regional Administrator - NRC Region I
Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Braidwood Station
NRC Senior Resident Inspector - Byron Station
NRC Senior Resident Inspector - Calvert Cliffs Nuclear Power Plant
NRC Senior Resident Inspector – Ginna Nuclear Power Plant
NRC Project Manager - Braidwood Station
NRC Project Manager - Byron Station
NRC Project Manager - Calvert Cliffs Nuclear Power Plant
NRC Project Manager - Ginna Nuclear Power Plant
Illinois Emergency Management Agency – Division of Nuclear Safety

Attachment

Response to Request for Additional Information

Background

By letter dated December 14, 2021 (Agencywide Document Access and Management System Accession Number ML21348A078), Exelon Generation Company, LLC (the licensee) submitted to the United States Nuclear Regulatory Commission (NRC), a proposed alternative to the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) of the steam generator (SG) welds of Braidwood Station (Braidwood), Units 1 and 2, Byron Station (Byron), Units 1 and 2, Calvert Cliffs Nuclear Power Plant (Calvert Cliffs), Units 1 and 2, and R.E. Ginna Nuclear Power Plant (Ginna). On February 1, 2022 (ADAMS Accession No. ML22032A333), Exelon Generation Company, LLC was renamed Constellation Energy Generation, LLC.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations*, Part 50 (10 CFR Part 50), Paragraph 50.55a(z)(1), the licensee is proposing to defer the ISI examinations for the SG welds of the subject units from the current ASME Code, Section XI 10-year requirement to the end of the currently approved periods of extended operation for the units. The licensee referred to the results of the probabilistic fracture mechanics (PFM) analyses in the following Electric Power Research Institute (EPRI) non-proprietary report as the primary basis for the deferral of the ISI examinations: 3002015906 “Technical Bases for Inspection Requirements for PWR [Pressurized Water Reactor] Steam Generator Class 1 Nozzle-to-Vessel Welds and Class 1 and Class 2 Vessel Head, Shell, Tubesheet-to-Head and Tubesheet-to-Shell Welds,” 2019 (hereinafter referred to as “EPRI report 15906” (ADAMS Accession No. ML20225A141). The NRC staff needs additional information to complete its review and approval of the licensee’s submittal.

Regulatory Basis

The NRC has established requirements in 10 CFR Part 50 to protect the structural integrity of structures and components in nuclear power plants. Among these requirements are the ISI requirements of Section XI of the ASME Code incorporated by reference in 10 CFR Part 50.55a to ensure that adequate structural integrity of the SG vessel (including welds) is maintained through the service life of the vessel. Therefore, the regulatory basis for the following requests for additional information (RAIs) has to do with demonstrating that the proposed alternative ISI requirements would ensure adequate structural integrity of the SG welds of the subject units for which EPRI report 15906 is referenced, and thereby would provide an acceptable level of quality and safety per 10 CFR 50.55a(z)(1) for these welds.

RAI-1

Issue

The licensee referenced probabilistic and deterministic analyses in EPRI report 15906 to estimate potential fatigue growth in the subject SG welds. The licensee presented plant-specific information to demonstrate that the referenced EPRI analyses would bound the subject SG welds, including high-level results from previous ISI of the welds. The licensee also provided limited discussion of performance monitoring, primarily focused on justifying application of the EPRI analyses to the proposed ISI interval extension for the subject SG welds (i.e., that leakage would be detected).

Leveraging PFM analyses to define the basis for risk-informing inspection requirements requires knowledge of both the current and future behavior of the material degradation and the associated uncertainties applicable to the subject SG welds. Confidence in the results of these analyses hinges on the assurance that the PFM model adequately represents, and will continue to represent, the degradation behavior in the subject SG welds. The NRC staff has determined that, when considering extended examination intervals, adequate performance monitoring through inspections is needed to ensure that the PFM model continues to predict the material behavior and that emergent degradation is discovered and dispositioned in a timely fashion.

The licensee discusses the system leakage test as "providing further assurance" for the proposed alternative. However, the NRC staff notes that the visual examinations performed during system leakage tests may not provide sufficient information to ensure that the PFM model continues to predict the material behavior and that emergent degradation is discovered and dispositioned in a timely fashion. Specifically, visual examinations may not directly detect pertinent integrity conditions (e.g., presence or extent of degradation); may not provide direct detection of aging effects prior to potential loss of structure or intended function; and do not provide sufficient validating data necessary to confirm the modeling of degradation behavior in the subject SG welds.

Request

a. Describe the performance monitoring that will be implemented with this proposed alternative to ensure that the PFM model adequately represents, and will continue to represent, the degradation behavior in the subject components commensurate with the duration of the requested alternative (i.e., plant-specific end date).

b. Justify that this performance monitoring will meet this objective and address the concerns discussed above.

c. Explain how this performance monitoring will provide, over the extended examination interval, (1) direct evidence of the presence and extent of degradation, (2) validation and confirmation of the continued adequacy of the PFM model; and (3) timely detection of novel or unexpected degradation.

d. Describe any actions that will be taken if issues are identified through this performance monitoring to ensure that the integrity of the component is adequately maintained.

Response

a. In the Reference [1] letter, Exelon Generation Company, LLC (now known as Constellation Energy Generation, LLC (CEG)) requested relief from the examination of Category B-B, B-D, and C-A SG pressure retaining welds and full penetration welded nozzles for Braidwood Generating Station (Braidwood), Units 1 and 2, Byron Generating Station (Byron), Units 1 and 2, Calvert Cliffs Nuclear Power Plant (Calvert Cliffs), Units 1 and 2, and R.E. Ginna Nuclear Power Plant (Ginna) for the remainder of the currently licensed operating periods. In the Reference [2] email, the U.S. Nuclear Regulatory Commission (NRC) requested additional information (RAI). In the Reference [3] letter, CEG provided a performance monitoring plan for the CEG PWR fleet for the examination Category C-B SG main steam and feedwater nozzle-to-shell welds and inside radius sections per the request from the NRC. When comparing EPRI Reports 3002015906 (Reference [7]) and 3002014590 (Reference [11]) the probability of rupture and leakage are comparable between the secondary side steam generator shell and nozzle welds for the base case and among the various sensitivity studies. In fact, the FW nozzles exhibited the highest probability of rupture when the combined effects of fracture toughness, stress, and nozzle flaw density were evaluated in Table 8-28 of Reference [11] compared to the SG shell welds in Table 8-32 of Reference [7]. Also comparing the DFM results in Table 8-31 of Reference [7] with those in Table 8-3 of Reference [1] shows that the maximum stress intensity factor (K) is higher for the SG FW nozzles than the SG shell welds and the time to leakage is considerably higher for the SG shell welds than the SG FW nozzles; therefore, CEG believes that the performance monitoring plan of the FW nozzles detailed in Reference [3] is adequate in addressing the NRC's concerns for all other secondary side SG welds for the CEG PWR fleet covered by the Reference [1] proposed alternative since the materials, environment, and operating conditions are similar. In the Reference [4, 5] letters, CEG provided a performance monitoring plan for the Class 1 pressurizer welds which are on the primary side. The Reference [6] letter documents the NRC verbal authorization for Byron and Braidwood to defer inspection of pressurizer nozzle-to-vessel and pressure retaining welds for the current inspection interval using the same DFM and PFM methodologies. CEG believes that the performance monitoring plan detailed in References [4, 5] adequately addresses the NRC's concerns for the SG primary side welds for all the CEG plants in the Reference [1] proposed alternative since the materials, environment, and operating conditions are similar to the pressurizer.

b. The Reference [1] proposed alternative is supported by the evaluations and conclusions presented in EPRI report 15906 [Reference 7] which are summarized as follows:

- A comprehensive industry survey involving 47 PWR units (US and international) was conducted by EPRI to determine the degradation history of these components. The survey reviewed examination results from the start of plant operation. Most of these plants have operated for over 30 years and in some cases over 40 years. A summary of the survey results is provided below:

Category	Item No.	Exams Performed	Results
B-B	B2.40	183	Two units (at a single site) reported flaws determined to be fabrication related. Flaw evaluation performed and flaws found acceptable as-is.
B-D	B3.130	135	No flaws exceeding ASME XI criteria.
C-A	C1.10	305	No flaws exceeding ASME XI criteria.
C-A	C1.20	319	Two units reported flaws determined to be fabrication related. Flaws evaluated per IWC-3600 and determined acceptable for continued service.
C-A	C1.30	163	No flaws exceeding ASME XI criteria.

The survey results showed that no examinations identified any unknown degradation mechanisms (i.e., mechanisms other than those listed in Section 6.0 of the EPRI Report). As shown in Table B6 of Reference [1], Byron Unit 2 identified embedded indications in weld 2RC-01-BB/SGC-03, 2RC-01-BB/SGC-05, 2RC-01-BC/SGC-05 and 2RC-01-BD/SGC-06 during PSI. Relief was requested to accept these flaws by evaluation and approved through an NRC safety evaluation on October 29, 1986 (Reference [8]). Successive examinations were performed, and the embedded indications were either found to have no change in sizing or were found to be non-recordable due to improved inspection methods during later ISI intervals. No other CEG PWR station covered by the proposed alternative has identified any flaws that exceed ASME Code, Section XI acceptance criteria for the SG welds. Based on this exhaustive industry survey, it is concluded that although the emergence of an unknown degradation mechanism cannot be completely ruled out, the possibility of the occurrence of such an unknown degradation mechanism is highly unlikely.

- The deterministic fracture mechanics (DFM) evaluation presented in Section 8.2 of the EPRI Report (Reference [7]) indicates that it would take a minimum of 641 years for a postulated initial flaw (with a depth equal to the ASME Code, Section XI acceptance standards) in the SG welds to reach 80% through-wall (assumed as leakage). The maximum stress intensity factor (K) obtained from the analysis remained below the ASME Code, Section XI allowable fracture toughness. This demonstrates that the SG nozzle-to-vessel and pressure retaining welds are very flaw tolerant.

- Demonstrating the plant-specific applicability of the EPRI Report along with the probabilistic fracture mechanics (PFM) evaluations presented in Section 8.3 of the EPRI Report, as supplemented by this CEG RAI response, indicate that the SG nozzle-to-vessel and pressure retaining welds at Braidwood, Byron, Calvert Cliffs, and Ginna can operate safely for over 80 years.

- The maximum proposed inspection deferral for Braidwood, Byron, Calvert Cliffs and Ginna is significantly lower (36 years vs over 600 years) than those justified by the results of the DFM and the PFM evaluations in the EPRI Report. These conservative inspection deferrals provide defense-in-depth for the analytically-determined safe operating period.

- Operating conditions at all CEG stations covered by the proposed alternative have been satisfactory over the life of the SGs and are bounded by the analysis in the EPRI Report. As shown in the tables of Appendix A of the proposed alternative (Reference [1]), as supplemented by this RAI response, the number of actual cycles experienced is significantly less that what was evaluated in the EPRI Report. In almost all cases the number of actual cycles experienced is less than or equal to half of what was used in the EPRI Report and in most cases the number is significantly less than half. The same is true of the projected number of cycles expected over a 60-year operating life. This adds an additional layer of confidence in the extension of the SG nozzle-to-vessel and pressure retaining weld inspections.

As shown in Table 1 of the proposed alternative (Reference [1]), Braidwood Unit 1, Category B-B, Item B2.40 and Category C-A, Item C1.30 request a maximum deferral of 36 years from the last ASME Code, Section XI inspection. As stated above, the DFM analysis indicates that it would take a minimum of 641 years for a postulated initial flaw, with a depth equal to the ASME Code, Section XI acceptance standards, to reach 80% through wall, which is assumed as leakage. This overall limiting scenario is applicable to Category B-B, Item B2.40. The limiting scenario for other welds covered by the proposed alternative is considerably longer. Category B-B, Item B3.130 (CE) has a limiting “years to leak” of 848 years, Category C-A, Item C1.10 has a limiting “years to leak” of 3533 years, Category C-A, Item C1.20 has a limiting “years to leak” of 1.05E+4 years, and Category C-A, Item C1.30 has a limiting “years to leak” of 1940 years. Significant margin exists between the maximum requested deferral (36 years) and the analytically-determined safe operating period (641 years minimum). The strong technical basis provided by the results of the DFM and PFM evaluations of the EPRI Report, as supplemented by this CEG RAI response, along with the satisfactory inspection history and relatively short duration of the proposed examination deferrals compared to the analytically-determined safe operating period provide sufficient assurance that the SG pressure retaining welds at Braidwood, Byron, Calvert Cliffs, and Ginna can operate safely for the remainder of plant life and will continue to provide an acceptable level of quality and safety with no additional performance monitoring examinations.

The satisfactory completion of the ASME Code, Section XI examinations to-date have provided sufficient direct evidence of the absence of any active degradation mechanisms (known or unknown) experienced by these components. In addition to the direct evidence provided by examinations completed to-date, examination of Category B-B, B-D, and C-A SG components will continue to be performed by other units across the domestic and international PWR fleet. Continued examination of Category B-B, B-D, and C-A SG components across the industry will provide additional opportunities to detect known degradation mechanisms, as described in Section 6.0 the EPRI Report, and will also provide the opportunity to detect any new or unexpected degradation mechanisms that may occur in the future for the subject components. If a new degradation mechanism is identified during continued industry examinations, CEG will follow the industry guidance to address the new degradation mechanism.

The combination of the significant margin between the maximum requested deferral (36 years) and the analytically-determined safe operating period (641 years minimum), extensive satisfactory inspection history to-date, and the very low likelihood of any future unexpected degradation mechanism affecting these components across the operating fleet, provides validation that the assumptions and methods of the PFM model used in the EPRI Report are adequate to predict the future behavior of the subject components. The strong technical basis provided by the results of the PFM model and EPRI Report provide sufficient assurance that the SG nozzle-to-vessel and pressure retaining welds at Braidwood, Byron, Calvert Cliffs, and Ginna can operate safely for the remainder of plant life and will continue to provide an acceptable level of quality and safety with no additional performance monitoring examinations.

Furthermore, the materials of construction for the components in the proposed alternative are low alloy steel which has been used in the fabrication of nuclear vessels for nearly 60 years without any material degradation issues solely attributed to the chemical or mechanical properties of these materials. Throughout the years, industry and NRC research has shown that these materials are very flaw tolerant and resistant to many of the degradation mechanisms listed in Section 6 of the EPRI report. These materials are among the most heavily studied materials in the industry and their behaviors are very well understood.

c. (1) The performance monitoring plans provided in References [3, 4, 5] include inspection sampling that will provide, over the extended examination interval, direct evidence of the presence and extent of any degradation.

(2) The components in the proposed alternative have operated for a minimum of 19 years up to a maximum of 47 years without the occurrence of any service-induced degradation. The embedded indications identified at Byron Unit 2 in welds 2RC-01-BB/SGC-03, 2RC-01-BB/SGC-05, 2RC-01-BC/SGC-05 and 2RC-01-BD/SGC-06 during PSI have operated for 34 years without further growth. All other welds in the proposed alternative have operated for a minimum of 19 years up to a maximum of 47 years without any identified inservice flaws. This excellent operating history is validation and confirmation of the conservative nature of

the PFM and DFM models used in the Reference [7] EPRI Report which concluded that a minimum of 641 years would be required for a postulated flaw to reach 80% of wall thickness. This also shows that the models will predict future behavior conservatively.

(3) The performance monitoring plans in References [3, 4, 5] will provide timely detection of any novel or unexpected degradation in these components.

d. The performance monitoring plans provided in References [3, 4, 5] describe actions that will be taken if issues are identified to ensure that the integrity of the pressurizer and SG components are adequately maintained.

Based on the above discussion, it is CEG's position that no additional performance monitoring examinations for the components covered by the proposed alternative of Reference [1] are required to ensure the safe operation of the components for the remainder of the current operating licenses.

RAI-2

Issue

The PFM analyses in the EPRI report 15906 assume certain examination histories, e.g., preservice inspection (PSI) followed by 10-year inspections and assume that 100% coverage was assumed during the PSI examination. The licensee provided actual examination histories for the subject units in Appendix B to the submittal. However, Appendix B does not provide information on PSI history for the units. Also, there seems to be gaps in the SG weld examinations at the subject units.

Request

a. Confirm that PSI examinations were performed on all SG welds at the subject units and that 100% coverage was achieved for the examinations.

b. Discuss whether any of the subject SG welds at the units contain fabrication defects.

c. Discuss any SG welds of the units that required repairs during the fabrication of the SGs prior to commercial operation at the units.

d. Discuss why for some SG welds of the subject units, the examinations of earlier ISI intervals are not shown in Appendix B to the submittal (e.g., the first ISI interval for Braidwood Unit 1).

Response

a. As used in the Reference [7] report, preservice examination (PSI) refers to the collective examinations required by ASME Code, Section III during fabrication and any ASME Code, Section XI examinations performed prior to service. The ASME Code, Section III fabrication examinations required for these components were robust and any ASME Code, Section XI preservice examinations further contributed to thorough initial examinations. The SG components covered by the proposed alternative were designed, fabricated, and certified in accordance with various Editions and Addenda of ASME Code, Section III. This certification includes an N-1 or N-2, Certificate Holders Data Report, as applicable, which certifies that the SGs were fabricated (including completing all necessary shop inspections) in accordance with ASME Code, Section III. Additionally, ASME Code, Section XI preservice examinations were performed prior to initial service. Except for Byron Unit 2, as shown in Table B6 of the proposed alternative, no recordable indications exceeding the acceptance standards were identified for the subject welds during ASME Code, Section XI preservice examinations.

b. Pre-service examination results for all units other than Byron Unit 2 did not reveal any indications exceeding ASME Section XI acceptance criteria. Byron Unit 2 identified multiple surface and subsurface indications exceeding the ASME Section XI acceptance criteria that were either removed by grinding, with no subsequent welding, or accepted by analytical evaluation. Relief was requested to accept the remaining flaws that exceeded ASME XI acceptance standards by evaluation and approval was granted by an NRC safety evaluation dated October 29, 1986 (Reference [8]). Table B6 of the Reference [1] proposed alternative identifies Byron Unit 2 embedded indications in weld 2RC-01-BB/SGC-03, 2RC-01-BB/SGC-05, 2RC-01-BC/SGC-05 and 2RC-01-BD/SGC-06 found during PSI. Successive examinations were performed, and the embedded indications were either found to have no change in sizing or were found to be non-recordable due to improved inspection methods during later ISI intervals.

c. Review of plant records indicate that weld repairs were performed during fabrication on some of the welds covered by the proposed alternative even though the details are not readily available. However, the records also show that in process fabrication repairs had post weld heat treatment (PWHT) performed which serves to relieve the residual stresses from in process weld repairs. In the DFM and PFM evaluations in the EPRI Report, the residual stresses after PWHT of the vessel were considered (Section 8.8.8.3.2 and 8.3.2.4); therefore, the effect of any in process fabrication weld repairs have been considered in the Reference [7] evaluations.

d. As stated in Section 1 of the proposed alternative, all of the SGs, with the exception of Braidwood Unit 2 and Byron Unit 2, have been replaced, or partially replaced, at some point during the plant's service life. Three ten-year inspections have been performed at Braidwood Unit 2 and Byron Unit 2. The PSI/ISI scenario for these units is therefore:

   - Braidwood Unit 2 and Byron Unit 2 PSI/ISI Scenario: (PSI+10+20+30+Deferral Period)

   The replacement of the Braidwood Unit 1 SGs occurred during the 1998 outage. The first inspection interval for Braidwood Unit 1 concluded in July 1998; therefore, any inspection history from the first inspection interval at Braidwood Unit 1 is not applicable to the proposed alternative. The replacement of the Byron Unit 1 SGs occurred in the 1997 outage. The first inspection interval for Byron Unit 1 concluded in August of 1998; therefore, any inspection history from the first inspection interval at Byron Unit 1 is not applicable to the proposed alternative. Two ten-year interval inspections have been performed at Braidwood Unit 1 and Byron Unit 1 after the SG replacement. The PSI/ISI scenario for these units is therefore:

   - Braidwood Unit 1 and Byron Unit 1 PSI/ISI Scenario: (PSI+10+20+Deferral Period)

   The Calvert Cliffs Unit 1 SGs were partially replaced during the 2002 outage and the Unit 2 SGs were partially replaced during the 2003 outage. The steam drum, which includes welds SG-11(12)-3 and SG-11(12)-4 for Unit 1 and welds SG-21(22)-3 and SG-21(22)-4 for Unit 2, was retained and refurbished leaving only Item Numbers C1.10 and C1.20 for the original (non-replaced) SGs. No examination records prior to the SG replacements were able to be located; however, two inspection intervals of examinations have been performed with no unacceptable indications identified. For the remaining welds, only the examinations performed since the partial SG replacements in 2002 (Unit 1) and 2003 (Unit 2) are applicable to the proposed alternative. Since Calvert Cliffs SGs were replaced mid-3rd interval and not all examinations for all components were required during the 3rd interval, as described in the response to RAI 3c, only the PSI inspection is credited and for conservatism, the ISI performed during the replacement interval is not credited for evaluation purposes. Without taking credit for the 1st and 2nd interval inspection at Calvert Cliffs, due to lack of inspection records, and not taking credit for the ISI during the replacement interval, the PSI/ISI scenarios are:

- Calvert Cliffs PSI/ISI Scenario – Non-replaced Portion: (PSI+30+40+Deferral Period)
- Calvert Cliffs PSI/ISI Scenario – Replaced Portion: (PSI+20+Deferral Period)

The Ginna SGs were replaced during the 1996 outage. The second inspection interval for Ginna concluded in December of 1989; therefore, any inspection history from the first interval, second interval, and most of the third interval (prior to 1996) are not applicable to the proposed alternative. Two ten-year interval inspections have been performed at Ginna after the SG replacement; therefore, the PSI/ISI scenario is:

- Ginna PSI/ISI Scenario: (PSI+10+20+Deferral Period)

The limiting PSI/ISI scenarios from the above are those associated with Calvert Cliffs (both the replaced and non-replaced portions) since there was a long time period between the PSI and first ISI inspection (due to records not being found for the first two interval inspections and not taking credit for the replacement interval inspections). These two scenarios were not considered in the Reference [7] EPRI Report, so evaluations were performed for the critical Case ID from the EPRI Report (SGPTH-4A) associated with Item No. B2.40 with these limiting PSI/ISI scenarios. The maximum deferral period of 36 years in Table 1 of the proposed alternative [1] associated with Braidwood Unit 1 was conservatively used in the evaluation and therefore the PSI/ISI scenarios are (PSI+30+40+76) for the non-replaced portion and (PSI+20+56) for the replaced portion. Fracture toughness of 200 ksi√in and standard deviation of 5 ksi√in as recommended by the NRC in the Safety Evaluation for Millstone Unit 2 (Reference [9]) and flaw density of 1.0 (since this Case ID is associated with a shell weld) are used in the evaluation. The PFM evaluations were performed using the **PROMISE**, Version 2.0 software code (the same version used in the Reference [7] EPRI Report) to determine the maximum stress multipliers that will meet the acceptance criteria of 1.0×10^{-6}. The results of the evaluations are presented in Tables RAI 2-1 and RAI 2-2 for the non-replaced and replaced portions, respectively. From these tables, stress multipliers of 1.6 and 1.8 can be applied to the non-replaced and replaced portion, respectively, before the acceptance criteria are reached.

Table RAI 2-1. Sensitivity to Combined Effects of Fracture Toughness, Stress, and Flaw Density for 80 Years for the CEG Plants Tubesheet-to-Shell Weld (Case ID SGPTH-P4A from Reference [7], PSI+30+40+76)

Time (year)	Probability per Year for Combined Case K_{IC} = 200 ksi√in. SD = 5 ksi√in. Stress Multiplier = 1.60 Flaw Density = 1 PSI+30+40+76	
	Rupture	Leak
10	1.00E-08	1.00E-08
20	5.00E-09	5.00E-09
30	2.77E-07	3.33E-09
40	2.10E-07	2.50E-09
50	1.68E-07	2.00E-09
60	1.40E-07	1.67E-09
70	1.20E-07	1.43E-09
80	1.05E-07	1.25E-09

Table RAI 2-2. Sensitivity to Combined Effects of Fracture Toughness, Stress, and Flaw Density for 80 Years for the CEG Plants Tubesheet-to-Shell Weld (Case ID SGPTH-P4A from Reference [7], PSI+20+56)

Time (year)	Probability per Year for Combined Case K_{IC} = 200 ksi√in. SD = 5 ksi√in. Stress Multiplier = 1.80 Flaw Density = 1 PSI+20+56	
	Rupture	Leak
10	1.00E-08	1.00E-08
20	3.65E-07	5.00E-09
30	2.53E-07	3.33E-09
40	2.85E-07	2.50E-09
50	4.02E-07	2.00E-09
60	5.32E-07	1.67E-09
70	4.56E-07	1.43E-09
80	4.01E-07	1.25E-09

As discussed in Sections 4.3.3 and 4.6 of Reference [7] EPRI Report and noted by the NRC in Section 5.1, page 8, first paragraph of the Safety Evaluation for Millstone Unit 2 (Reference [9]), the dominant stress is the pressure stress; therefore, the variation in the R_i/t ratio relative to that of the model used in Reference [7] EPRI Report (stress multiplier) can be used to scale up the stresses of the Reference [7] EPRI Report to obtain the plant-specific stresses for each unit and component. From the Reference [7] EPRI Report, the critical Case ID is

SGPTH-P4A which is associated with the primary side tubesheet-to-head weld. Table RAI 2-3 provides the plant-specific R_i/t comparison to that in the Reference [7] EPRI Report for the primary side tubesheet-to-head welds. As shown in Table RAI 2-3, the stress multipliers for all the units in the proposed alternative are less than unity and considerably less than the limiting stress multiplier of 1.6 obtained from the PFM evaluation indicating that the stress analysis in the Reference [7] EPRI Report is conservative in application to all the units in the Reference [1] proposed alternative.

Table RAI 2-3. CEG SG Primary Side Channel Head Dimensions in Comparison with Reference [7] EPRI Report

Plant	Primary Side Channel Head ID (in)	Primary Side Channel Head Thk. (in)	Primary Side Channel Head R_i/t	Stress Multiplier = $(R_i/t)_{plant}$ / $(R_i/t)_{EPRI}$
EPRI Report (Figure 4-4 of [7])	157.24	6.94	11.33	n/a
Braidwood Unit 1	123.625	6.25	9.89	0.87
Braidwood Unit 2	124.375	5.6	11.10	0.98
Byron Unit 1	123.625	6.25	9.89	0.87
Byron Unit 2	124.375	5.6	11.10	0.98
Calvert Cliffs Unit 1	151.37	7.25	10.44	0.92
Calvert Cliffs Unit 2	151.37	7.25	10.44	0.92
Ginna	117.5	6.125	9.59	0.85

RAI-3

Issue

The following issues have to do with the plant-specific inspection histories included in Appendix B to the submittal.

Section 1 of Attachment 1 to the submittal shows that each SG at Braidwood Unit 1 contains five welds that are required to be examined. Tables B1, B2, B3, and B4 of Appendix B to the submittal show the welds that have been examined. The NRC staff noted that in the Braidwood Unit 1 second and third ISI interval, the licensee examined only four, not five, subject SG welds. The licensee did not examine the Upper Secondary Shell – Shell Cone weld (component ID numbers 1SG-05-SGC-05, 1SG-06-SGC-05, 1SG-07-SGC-05, and 1SG-08-SGC-05, item number C1.10) during the second and third ISI intervals. The NRC staff recognizes that footnote 4 to the ASME Code, Section XI, Table IWC-2500-1, Examination Category C-A, Item No. C1.10 permits welds from only one SG to be examined. However, the ASME Code does not state that a weld in the required SG weld sample can be unexamined.

Also, Section 1 of Attachment 1 to the submittal shows the SG welds at the Byron Units 1 and 2. However, Tables B5, B6, B7, and B8 of Appendix B to the submittal, which show the SG welds of the Byron units that were examined, do not show all the welds in Section 1 of Attachment 1 to the submittal for the Byron Units 1 and 2.

The note in Table B10 of Appendix B to the submittal for the Calvert Cliffs units refers to Paragraph IWB-2412(b) of Section XI of the ASME Code (2013 Edition for the Calvert Cliffs code of record stated in the submittal), but there is no IWB-2412(b) in of Section XI of the ASME Code. The same note in Table B10 states that the referenced Paragraph of the ASME Code allowed the licensee to forego the inspection of the W7 SG welds of the Calvert Cliffs units. The NRC staff noted that, however, it appears from Table B10 that other welds were not inspected. For example, welds SG-12-W6, SG-21-W5, and SG-21-W6 were inspected only during the fourth ISI interval, and not during the third ISI interval per Table B10.

Request

a. Discuss why the Upper Secondary Shell – Shell Cone welds (1SG-05-SGC-05, 1SG-06-SGC-05, 1SG-07-SGC-05, and 1SG-08-SGC-05) in the Braidwood Unit 1 SG weld sample were not examined in the second and third ISI interval.

b. Discuss why Tables B5, B6, B7, and B8 of Appendix B to the submittal do not show all the welds in Section 1 of Attachment 1 to the submittal for the Byron Units 1 and 2.

c. Clarify which Paragraph of Section XI of the ASME Code the note in Table B10 of Appendix B to the submittal is supposed to refer to.

d. Discuss why welds SG-12-W6, SG-21-W5, and SG-21-W6 of the Calvert Cliffs units were inspected only during the fourth ISI interval.

Response

It should be noted that compliance with ASME Code, Section XI scheduling requirements for the service life of a given unit, or weld, is not a prerequisite to apply the results and conclusions of the Reference [7] EPRI Report.

a. As stated in Note 3 of Table B2 of Reference [1], the upper secondary shell – shell cone welds (1SG-05-SGC-05, 1SG-06-SGC-05, 1SG-07-SGC-05, and 1SG-08-SCG-05) were not examined during the Second or Third interval on the basis that these welds are not located at gross structural discontinuities as defined in ASME Code, Section III, NB-3213.2. The extent of examination for Category C-A, Item No. C1.10 is "welds[1] at gross structural discontinuity[2] only". Note 1 states "Includes essentially 100% of the weld length" and Note 2 states in part "Gross structural discontinuity is defined in NB-3213.2." Since it was determined that these welds do not meet the definition of a gross structural discontinuity, no examination is required in accordance with ASME Code, Section XI, Category C-A, Item No. C1.10. The location of weld SGC-05 is shown in Figure A4 of Reference [1] and the weld configuration is shown in Figure A2, secondary shell to shell cone W80.

   It should be noted that the extent of examination for Category C-A, Item No. C1.10 was revised in later Edition and Addenda of ASME Code, Section XI to be "Cylindrical-shell-to-conical-junction welds" and shell (or head)-to-flange welds. Based upon the configuration of the upper secondary shell cone welds at Braidwood, examination of these components is no longer required by ASME Code, Section XI.

b. Section 1 of Attachment 1 of Reference [1] list all Category B-B, Item No. B2.40 and Category C-A, Item No. C1.10, 1.20, and C1.30 components at Byron Units 1 and 2. In accordance with ASME Code, Section XI, not all welds of all SGs require inspection each inspection interval. Table B5, B6, B7, and B8 of Reference [1] represents all examination history that was available for the subject components. The examination requirements of ASME Code, Section XI, Category B-B and Category C-A are clarified with the explanation below.

The extent of examination for successive inspection intervals for the tube-sheet-to-head weld are defined in Table IWB-2500-1, Examination Category B-B, Item No. B2.40. The extent of examination is defined as "Weld" and is clarified by Note (1) and Note (4). Note (1) of the 2013 Edition states, "The examination may be limited to one vessel among the group of vessels performing a similar function" and Note (4) states "Includes essentially 100% of the weld length"; therefore, for Byron Unit 1 and 2 the tube-sheet-to-head weld of only one of the four SGs needs to be inspected at each unit. Similar notes to Note (1) and Note (4) of the 2013 Edition have appeared in previous Edition and Addenda of ASME Code, Section XI.

Similarly, the extent of examination for successive inspection intervals for the secondary side shell circumferential welds, head circumferential welds, and tubesheet-to-shell welds are defined in Table IWC-2500-1, Examination Category C-A, Item No. C1.10, C1.20, and C1.30, respectively. The extent of examination for each Item No. is clarified by Note (3) and Note (4). Note (3) of the 2013 Edition states, "Includes essentially 100% of the weld length" and Note (4) states "In the case of multiple vessels of similar design, size, and service (such as steam generators, heat exchanges), the required examinations may be limited to one vessel or distributed among the vessels"; therefore, for the Byron Units 1 and 2 secondary side pressure retaining welds only the number of welds corresponding to one SG need to be examined. These welds may be performed on one SG or may be distributed among the four SGs. Similar notes to Note (3) and Note (4) have appeared in previous Edition and Addenda of ASME Code, Section XI.

c. The reference to IWB-2412(b) in Table B10, Note 1 is to the 1998 Edition of ASME Code, Section XI which was the Code of Record for the 3rd Inspection Interval at the time the SGs were replaced. IWB-2412(b)(1) of the 1998 Edition states, "When items or welds are added during the first period of an interval, at least 25% of the examinations required by the applicable Examination Category and Item Number for the added items or welds shall be performed during each of the second and third periods of that interval. Alternatively, if deferral of the examinations is permitted for the Examination Category and Item Number, the second period examinations may be deferred to the third period and at least 50% of the examinations required by the applicable Examination Category and Item Number for the added items or welds shall be performed during the third period." The Calvert Cliffs Unit 1 SGs were partially replaced during the 2002 outage, which was the second outage of the first inspection period; therefore, Calvert Cliffs Unit 1 was required to meet the requirements of IWB-2412(b)(1) for the remainder of the third inspection interval for the new welds.

   IWB-2412(b)(2) of the 1998 Edition states, "When items or welds are added during the second period of an interval, at least 25% of the examinations required by the applicable Examination Category and Item Number for the added items or welds shall be performed during the third period of the interval." The Calvert Cliffs Unit 2 SGs were partially replaced during the 2003 outage, which was the first outage of the second inspection period; therefore, Calvert Cliffs Unit 2 was required to meet the requirements of IWB-2412(b)(2) for the remainder of the third inspection interval for the new welds.

   Note that in later editions of ASME Code, Section XI, IWB-2412 does not exist due to the elimination of the requirements for "Inspection Program A." The equivalent requirements to those stated above are now contained in ASME Code, Section XI, IWB-2411(b).

d. As stated in the response to RAI-3c, the requirements of IWB-2412 applied to the new SG welds added to the program as part of the partial SG replacements. Each replacement SG

(2 per unit) had three Category B-D, Item No. B3.130 components for a total of six new Category B-D, Item No. B3.130 components per unit added to the program as part of the SG replacements.

As stated in the response to RAI-3c, the requirements of IWB-2412(b)(1) applied to Unit 1 which required 50% of the new Category B-D, Item No. B3.130 components to be inspected by the end of the third inspection interval. As shown in Table B10 of Reference [1], three (of six) Category B-D, Item No. B3.130 components (SG-11-W5, SG-11-W6, and SG-12-W5) were inspected during the third interval which meets the 50% inspection requirement; therefore, inspection of the remaining Category B-D, Item No. B3.130 components (SG-11-W7, SG-12-W6, and SG-12-W7) was not required during the third inspection interval.

As stated in the response to RAI-3c, the requirements of IWB-2412(b)(2) applied to Unit 2 which required 25% of the new Category B-D, Item B3.130 components to be inspected by the end of the third inspection interval. As shown in Table B10 of Reference [1], two (of six) Category B-D, Item B3.130 components (SG-22-W5 and SG-22-W6) were inspected during the third interval which meets the 25% inspection requirement; therefore, inspection of the remaining Category B-D, Item No. B3.130 components (SG-21-W5, SG-21-W6, SG-21-W7, and SG-22-W7) was not required during the third inspection interval.

RAI-4

<u>Issue</u>

Section 1 of Attachment 1 to the submittal states that the affected components are SG pressure retaining welds and full penetration welded nozzles. However, the welded nozzles are not identified in the affected components tables for the Braidwood, Byron, and Ginna units.

<u>Request</u>

a. Discuss whether Braidwood, Byron, and Ginna have any full-penetration welded nozzles attaching to the SG shell.

b. If the answer is affirmative, discuss whether they are classified under Examination Category B-D and Item Number B 3.130 in accordance with the ASME Code, Section XI, Table IWB-2500-1.

c. If the answer is affirmative, confirm that these full-penetration welded nozzles are not required to be examined.

Response

a. The primary inlet and outlet nozzles for the Braidwood, Byron, and Ginna SGs are integrally forged with the SG channel head; therefore, no full-penetration welded nozzles exist for these stations and there are no applicable examination requirements contained in ASME Code, Section XI, Table IWB-2500-1, Category B-D, Item No. B1.130 for these components. Since there are no applicable ASME Code, Section XI examination requirements, the SG primary nozzle-to-shell welds are not included in the ISI program for Braidwood, Byron, and Ginna. The Reference [1] proposed alternative is not applicable to the SG secondary side nozzle-to-vessel welds.

b. Based upon the response to RAI-4a this question is not applicable.

c. Based upon the response to RAI-4a this question is not applicable.

RAI-5

Issue

Tables A4 and A8 of Appendix A to the submittal shows transient cycles for SG secondary side vessel welds at the Braidwood and Byron units. Footnotes 4, 5, and 6 state that pressure values for various transients could not be found in the design documentation; therefore, the licensee selected pressure values from other sources.

Request

Discuss whether the pressure values selected for the transients in Footnotes 4, 5, and 6 in Tables A4 and A8 are bounding, i.e., if these pressure values were used in the analysis, the results would still be within the probability of failure values calculated in EPRI report 15906.

Response

As discussed in the Response to RAI-2d and shown in Table RAI 2-3, the stress multipliers for all the units included in the proposed alternative are considerably less than the stress multiplier of 1.6 required to reach the acceptance criteria; hence, the pressure can be increased by at least a factor of 1.6. The maximum increase in the pressure values in Table A4 and A8 of the proposed alternative relative to the values in the Reference [7] EPRI report is 1077 vs. 1000 psi resulting in a factor of 1.077. This is considerably less than the stress multiplier of 1.6 and is therefore within the probability of failure values calculated in the Reference [7] EPRI Report.

RAI-6

Issue

For the B2.40 and B3.130 welds of the subject units requested in the submittal, the tables in Appendix A to the submittal that show the transient cycles for each of the units also show the lowest temperatures for the Heatup/Cooldown transient for each unit: 120°F for Braidwood and Byron (Tables A3, A4, A7, and A8), 70°F for Calvert Cliffs (Tables A12 and A13), and 100°F for Ginna (Tables A16 and A17). Section 8.2.2.5 of EPRI report 15906 states that the minimum temperature (200°F) during this transient corresponds to Figure 7-22 of EPRI report 15906; therefore, a fracture toughness (K_{IC}) set at the upper shelf value of the ASME Code K_{IC} curve, 200 ksi$\sqrt{\text{in}}$, may be used. The NRC staff noted that Figure 7-22 of EPRI report 15906 is at the end of heatup (as noted in the figure). During the ramp periods at the beginning and end of the Heatup/Cooldown transient, the temperatures at the subject B2.40 and B3.130 welds of the subject units could be lower than 200°F as indicated in the above tables, and thus, K_{IC} could be lower than 200 ksi√in. A similar was issue was addressed for Millstone 2 in letter dated March 19, 2021 (ADAMS Accession No. ML21081A136) through plots that compare the K_{IC} history for the Heatup/Cooldown transient, which were determined based on a plant-specific RT_{NDT} value of 0°F for the Millstone 2 SG, with the corresponding applied stress intensity factor history (Figures RAI-4-1 and RAI-4-2 in the March 19, 2020 letter). The comparison showed that the applied stress intensity factor during anytime of the Heatup/Cooldown transient did not exceed the lowest K_{IC} plant-specific value for Millstone 2 and was accepted by the NRC staff in the safety evaluation dated July 16, 2021 (ADAMS Accession No. ML21167A355).

<u>Request</u>

Show that the applied stress intensity factor at the B2.40 and B3.130 SG welds of the subject units during anytime of the Heatup/Cooldown transient does not exceed the lowest plant-specific K_{IC} value for each subject unit that occur at the beginning and end of the Heatup/Cooldown transient.

<u>Response</u>

Information on the RT_{NDT} values for the SG plates, forgings and welds for the various plants were obtained from the plants' UFSARs as follows:

<u>Braidwood Unit 1</u>: The limiting RT_{NDT} for the replacement SGs is 0°F.

<u>Braidwood Unit 2</u>: The maximum RT_{NDT} value for the SGs is guaranteed to be no greater than 60°F. The actual values are expected to be lower but records are not readily available to determine the actual RT_{NDT} values.

<u>Byron Unit 1</u>: The limiting RT_{NDT} for the replacement SGs is 0°F.

<u>Byron Unit 2</u>: The maximum RT_{NDT} value for the SGs is guaranteed to be no greater than 60°F. The actual values are expected to be lower but records are not readily available to determine the actual RT_{NDT} values.

<u>Calvert Cliffs Unit 1</u>: For the replacement SGs, the RT_{NDT} for the plate, forging and weld is equal to or less than 0°F.

For the original SGs, the design specification states, in part, that impact properties of all ferritic steel materials which form a part of the pressure boundary shall meet the requirements of ASME Code Section III, at a temperature of +10°F; alternate higher temperature levels up to 40°F may be used only if the material fails a +10°F. From this requirement, the maximum RT_{NDT} value for the original SG pressure boundary material is 40°F.

<u>Calvert Cliffs Unit 2</u>: For the replacement SGs, the RT_{NDT} for the plate, forging and weld is equal to or less than 0°F.

For the original SGs, the design specification states, in part, that impact properties of all ferritic steel materials which form a part of the pressure boundary shall meet the requirements of ASME Code Section III, at a temperature of +10°F; alternate higher temperature levels up to

40°F may be used only if the material fails a +10°F. From this requirement, the maximum RT_{NDT} value for the original SG pressure boundary material is 40°F.

Ginna: For the replacement SG pressure boundary plate, forging and weld, the RT_{NDT} is equal to or less than 0°F; typically, these range from -70°F to -20°F.

From the above, the replacement SG materials at Braidwood Unit 1, Byron Unit 1, Calvert Cliffs Units 1 and 2 and Ginna have RT_{NDT} values of 0°F or less, similar to Millstone Unit 2; therefore, the evaluation performed in response to RAI 4 for Millstone Unit 2 in ADAMS Accession No. ML21081A136 (Reference [10]) referenced in the "Issue" section of this RAI is applicable to these plants.

For the original SG materials at Braidwood Unit 2 and Byron Unit 2, the maximum RT_{NDT} value is guaranteed to be no greater than 60°F. Although the actual values are expected to be lower, records were not readily available to determine the exact value and therefore the maximum value of 60°F is assumed. For the original SG material at Calvert Cliffs Units 1 and 2, the maximum RT_{NDT} is 40°F. It should be noted that the original SGs at Calvert Cliffs Units 1 and 2 do not contain the critical Case ID SGPTH-4A in the Reference [7] EPRI Report which is associated with the tubesheet-to-head weld (Item No. B2.40) on the primary side. The Item Numbers associated with the original SGs at Calvert Cliffs Units 1 and 2 are C1.10 and C1.20. Tables 8-3 and 8-13 through 8-16 of the Reference [7] EPRI Report indicate that there are significant margins in the analysis for these Item Nos with respect to fracture toughness. As such, the SGs at Braidwood Unit 2 and Byron Unit 2 are bounding since they contain the critical Case ID and have the highest RT_{NDT} value. An evaluation similar to that performed in response to RAI 4 for Millstone Unit 2 in ADAMS Accession No. ML21081A136 [10] was repeated using the limiting RT_{NDT} value of 60°F to address the original SG materials at Braidwood Unit 2 and Byron Unit 2. The two Case IDs considered for Millstone Unit 2 (SGPNV-P1A(N) and SGPTH-4A) were also evaluated in this case. The results are shown in Figures RAI 6-1 and RAI 6-2 for Case IDs SGPNV-P1A(N) and SGPTH-P4A, respectively.

As can be seen in Figure RAI 6-1, for Case ID SGPNV-P1A(N), the applied stress intensity factor (K) is below the fracture toughness throughout the transient; however, for Case ID SGPTH-4A, the applied K is higher than the allowable fracture toughness in the beginning part of the heatup transient and at the end of the cooldown transient, with the allowable fracture toughness between 72 ksi√in to 86 ksi√in. Figure 8-6 of the Reference [7] EPRI Report shows that the 80% and 50% through-wall flaws evaluated in Figure RAI 6-2 would have been detected by the POD curve in the PFM evaluation, resulting in the allowable fracture toughness of 86 ksi√in being the most realistic case. Therefore, conservative lower bound fracture toughness of 80 ksi√in at the beginning of heatup and end of cooldown for the Case ID SGPTH-P4A is used to perform a sensitivity study on fracture toughness. This is similar to the sensitivity study performed in Table 8-14 of the Reference [7] EPRI Report, where it was demonstrated that fracture toughness as low as 80 ksi√in would meet the acceptance criteria.

The results of the DFM evaluation presented in Table 8-3 of the Reference [7] EPRI Report showed that for Case ID SGPTH-4A, the maximum K obtained in the transient analysis is 77.8 ksi√in, which further justifies the use of an allowable fracture toughness 80 ksi√in. However, the sensitivity study presented in Table 8-14 of the Reference [7] EPRI Report was performed for a PSI/ISI scenario of (PSI +20+40+60). It was therefore repeated using the plant-specific PSI/ISI scenario for Braidwood Unit 2 and Byron Unit 2 (PSI+10+20+30+66) representing the deferral period of 35.2 years for Braidwood Unit 2 from Table 1 of the proposed alternative [1]. The results of this sensitivity study are presented in Table RAI 6-1. The results show that with a fracture toughness as low as 80 ksi√in representing a reasonable lower bound from Figure RAI 6-2, the acceptance criteria of $1.0x10^{-6}$ is met.

Table RAI 6-1. Sensitivity to Combined Effects of Fracture Toughness, Stress, and Flaw Density for 80 Years of Operation – Braidwood Unit 2 and Byron Unit 2 (Case ID SGPTH-4A from Reference [7], PSI+10+20+30+66)

Time (year)	Probability per Year for Combined Case K_{IC} = 80 ksi√in. SD = 5 ksi√in. Stress Multiplier = 1.0 Flaw Density = 1 PSI+10+20+30+66	
	Rupture	Leak
10	5.40E-07	1.00E-08
20	2.75E-07	5.00E-09
30	1.83E-07	3.33E-09
40	1.38E-07	2.50E-09
50	1.14E-07	2.00E-09
60	1.27E-07	1.67E-09
70	1.56E-07	1.43E-09
80	1.38E-07	1.25E-09

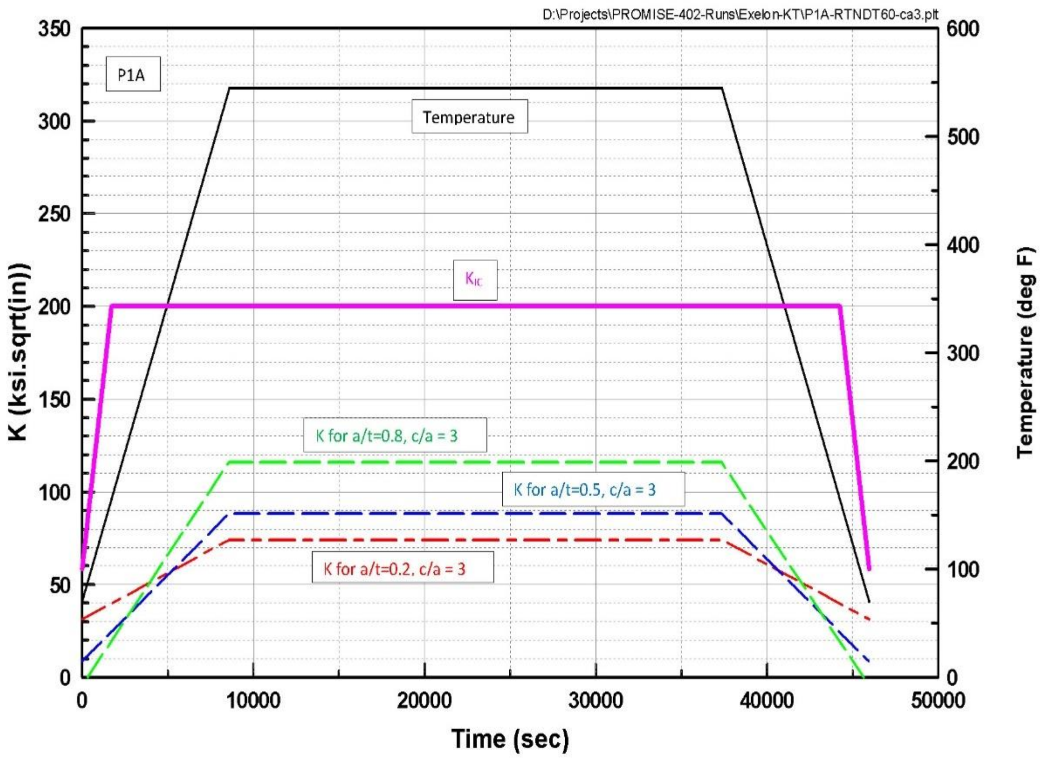

Figure RAI 6-1: Applied K vs. Fracture Toughness as a Function of Temperature for Case ID SGPNV-P1A-(N)
(Using Braidwood Unit 2 and Byron Unit 2 Plant-Specific Upper Bound SG Material RT_{NDT} Value of 60°F)

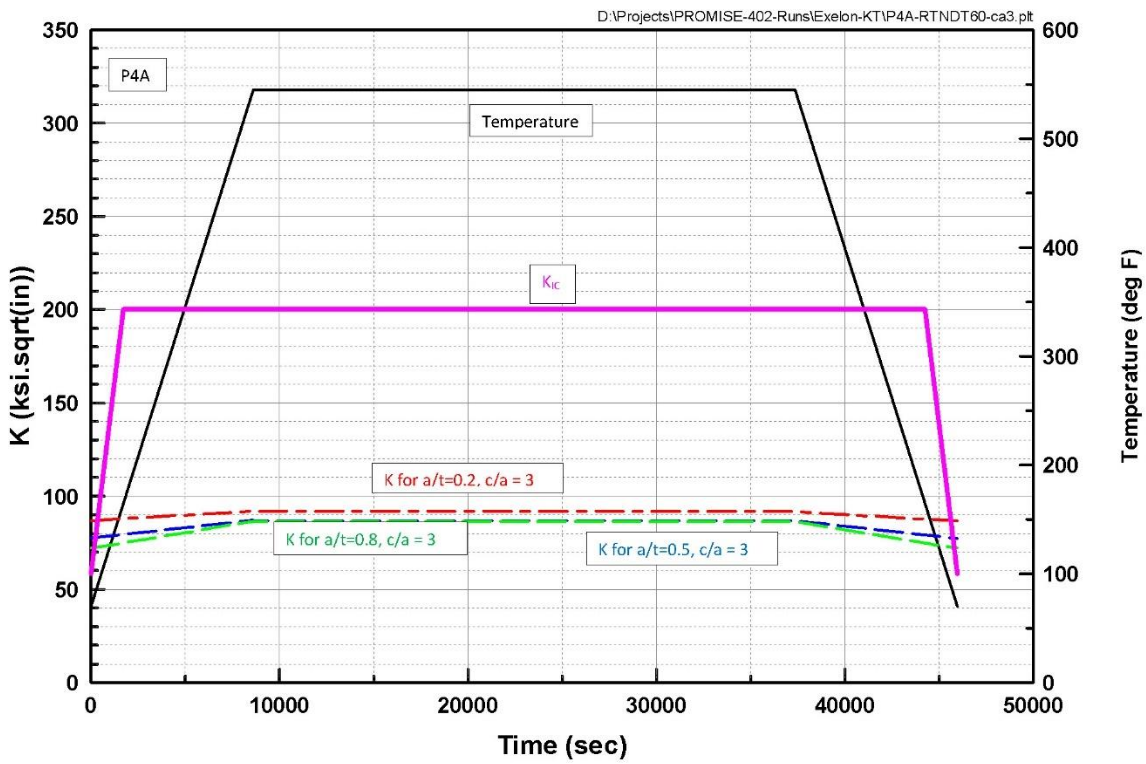

Figure RAI 6-2: Applied K vs. Fracture Toughness as a Function of Temperature for Case ID SGPTH-4A)
(Using Braidwood Unit 2 and Byron Unit 2 Plant-Specific Upper Bound SG Material RTNDT Value of 60°F)

RAI-7

Issue

Tables A1, A2, A5, A6, A9, A10, A11, A14, and A15 in Appendix A to the submittal, the licensee states that the subject units have not resulted in an event (unheated auxiliary feedwater being introduced into a hot SG) that can result in a thermal shock of the SG vessel. The NRC noted that the auxiliary feedwater event referred to in these tables affects the secondary side welds (Item Nos. C1.10, C1.20, and C1.30) requested for the units. The NRC staff noted that a potential thermal shock event from the primary side can affect the primary side SG welds requested for the subject units in the submittal (Item Nos. B2.40 and B3.130).

Request

Similar to the thermal shock applicability discussion in Tables A1, A2, A5, A6, A9, A10, A11, A14, and A15 in Appendix A to the submittal, explain whether each of the subject units in the submittal has experienced a thermal shock event in the *primary* side of the reactor coolant system that can affect the K_{IC} value assumed in the analysis in EPRI report 15906 referenced for the primary side SG welds requested for the subject units in the submittal.

Response

For the secondary side of the SG a condition for the application of the Reference [7] EPRI Report is to confirm that a thermal shock caused by the introduction of unheated auxiliary feed water into a hot SG boiled dry following station blackout has not occurred. Confirming that a similar event has not occurred on the primary side of the SG is not a requirement to apply the results and conclusions of the Reference [7] EPRI Report. The occurrence of such an event occurring on the primary side of the SG is extremely unlikely due to the fact that there are no safety injection sources that inject directly into the primary side of the SG. The safety injection sources for Braidwood, Byron, Calvert Cliffs, and Ginna inject into some combination of the RCS hot leg, RCS cold leg, or directly into the reactor vessel. Additionally, the amount of water injected into the RCS from these safety injection sources is small compared to the overall volume of the RCS. Given the significantly higher temperature of the RCS and the small volume, by comparison, of safety injection flow, a rapid decrease of RCS temperature such that a thermal shock would occur in the primary loop is unlikely. Given that the sources of safety injection are upstream of the primary side of the SG, there is an opportunity for the RCS to increase the temperature of the incoming water before it comes into contact with the components covered by the proposed alternative, thereby reducing the chances of these components experiencing a thermal shock event.

RAI-8

Issue

In Section 5.2 of EPRI report 15906, EPRI stated that the analyses in the report did not consider test conditions beyond a system leak test and that no separate test conditions were included in the evaluation. EPRI noted that pressure testing (i.e., hydrostatic tests or system leakage tests) requirements are in Subsubarticle IWA-4540 of ASME Code, Section XI. Subsubarticle IWA-4540 states in part that pressure-retaining boundary shall include a hydrostatic or system leakage test, prior to, or as part of, returning to service. The NRC staff noted that in many PWRs, the system leakage test is integrated to the heatup process after a refueling outage to meet the pressure testing requirements, and that thus, pressure testing need not be analyzed separately from the Heatup/Cooldown transient. However, it is not clear from the submittal whether the practice of integrating the system leakage test to the heatup process is being performed at the subject units.

Request

a. For the subject units in the submittal, clarify whether the system leakage test is integrated to the heatup process.

b. If the system leakage test is not integrated to the heatup process for the subject units, and since pressure testing is not analyzed separately in EPRI report 15906, explain how the transients in EPRI report 15906 selected for analyses bound pressure testing and confirm that the temperature during the test is high enough such that the assumption of an upper shelf K_{IC} value of 200 ksi$\sqrt{\text{in}}$ is appropriate.

Response

a. The Class 1 system leakage test required by IWB-5000 each refueling outage is performed with the plant in Mode 3 which is defined as “Hot Standby” (Braidwood, Byron, and Calvert Cliffs) or “Hot Shutdown” (Ginna). Any pressure tests required by repair/replacement activities (IWA-4540) performed during an outage on the Class 1 boundary are typically performed in conjunction with the Class 1 system leakage test. The conditions for Mode 3 are a reactivity less than 0.99, no thermal power, and an average reactor coolant temperature of greater than or equal to 300°F (Calvert Cliffs and Ginna) or 350°F (Braidwood and Byron). Since the conditions of the plant during the Class 1 system leakage test are between the conditions associated with Cold Shutdown and rated temperature and pressure, they are bounded by the definition of the heatup and cooldown transient as defined in Section 5 of the Reference [7] EPRI Report.

b. Based upon the response to RAI-8a this question is not applicable. Furthermore, the variation of applied K compared to the fracture toughness during the heatup and cooldown transient was evaluated in response to RAI-6 and was found to be acceptable.

RAI-9

Issue

Section 5 of Attachment 1 to the submittal states that for the Calvert Cliffs units, “PSI has been followed by the performance of four 10-year interval inspections for welds W3 and W4, but records were only found for the Third and Fourth Interval examinations.”

Request

Confirm that the coverages achieved for the 1st and 2nd 10-year ISI intervals for the SG welds W3 and W4 of the Calvert Cliffs units listed in Tables B11 and B12 of Appendix B to the submittal were not less than the smallest coverage shown in the tables.

Response

Welds W3 and W4 at Calvert Cliffs are associated with Item Nos. C1.20 and C1.10, respectively. As stated in the proposed alternative (Reference [1]), inspection records could not be found for the first and second interval inspections for these welds; therefore, it cannot be confirmed that the coverages achieved for the 1st and 2nd 10-year ISI intervals for these welds were not less than the smallest coverage shown in Tables B11 and B12. As shown in Tables B11 and B12 of the proposed alternative (Reference [1]), the minimum coverage achieved for these welds during the 3rd and 4th ISI inspections is 93.1% (which is essentially 100% coverage as defined in ASME Code, Section XI, IWA-2200(c)). To resolve this issue, an evaluation was performed using the conservative assumption that the 1st and 2nd interval inspections were not performed, resulting in a PSI/ISI scenario of (PSI+30+40+Deferral Period). From Table 1 in the proposed alternative (Reference [1]), the maximum deferral period for Item Nos. C1.10 and C1.20 at Calvert Cliffs is 25.5 years (corresponding to Item No. C1.10 at Calvert Cliffs Unit 2); therefore, the PSI/ISI scenario is (PSI+30+40+66). Also, from Table 8-3 of the Reference [7] EPRI Report, the critical Case ID for Item Nos. C1.10 and C1.20 is SGSSC-P9C corresponding to Item No. C1.10. A **PROMISE** run was performed with the PSI/ISI scenario of (PSI+30+40+66), coverage of 93.1%, flaw density of 1.0 and stress multiplier of 1.24. The stress multiplier of 1.24 was obtained from Table RAI 2-1 of Reference [3] for the secondary side shell welds for Calvert Cliffs. The results of the evaluation are presented in Table RAI 9-1 and show that after 80 years of operation, the probabilities of rupture and leakage are at least two orders of magnitude below the acceptance criteria of $1.0x10^{-6}$; therefore, even if the 1st and 2nd interval inspections at Calvert Cliffs are not taken into consideration, the acceptance criteria for probabilities of rupture and leakage are met.

Table RAI 9-1. Sensitivity to Combined Effects of Fracture Toughness, Stress, and Flaw Density for 80 Years of Operation - Calvert Cliffs Welds W3 and W4 (Case ID SGSSC-P9C from Reference [7], PSI+30+40+66)

Time (year)	Probability per Year for Combined Case K_{IC} = 200 ksi√in. SD = 5 ksi√in. Stress Multiplier = 1.24 Flaw Density = 1 Coverage 93.1% PSI+30+40+66	
	Rupture	Leak
10	1.00E-08	1.00E-08
20	5.00E-09	5.00E-09
30	3.33E-09	3.33E-09
40	2.50E-09	2.50E-09
50	2.00E-09	2.00E-09
60	1.67E-09	1.67E-09
70	1.43E-09	1.43E-09
80	1.25E-09	1.25E-09

REFERENCES

1. Letter from D. Gudger (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, “Proposed Alternative for Examinations of Examination Categories B-B, B-D, and C-A Steam Generator Pressure Retaining Welds and Full Penetration Welded Nozzles,” dated December 14, 2021, ADAMS Accession No. ML21348A078.

2. Email Letter from J. Wiebe (U.S. Nuclear Regulatory Commission) to T. Loomis (Constellation Energy Generation, LLC), “Draft RAIs for Requests for Alternatives I4R-17, I4R-23, ISI-05-018, I6R-10 (EPID Nos.: L-2021-LLR-091, L-2021-LLR-092, L-2021-LLR-093, L-2021-LLR-094),” dated May 6, 2022.

3. Letter from D. Gudger (Constellation Energy Generation, LLC) to U.S. Nuclear Regulatory Commission, “Response to Request for Additional Information - Proposed Alternative for Examinations of Examination Category C-B Steam Generator Nozzle-to-Shell Welds and Nozzle Inside Radius Sections,” dated May 20, 2022, ADAMS Accession No. ML22140A055.

4. Letter from D. Gudger (Constellation Energy Generation, LLC) to U.S. Nuclear Regulatory Commission, “Supplemental Information - Proposed Alternative for Examination of Pressurizer Circumferential and Longitudinal Shell-to-Head Welds and Nozzle-to-Vessel Welds,” dated March 10, 2022, ADAMS Accession No. ML22069A580.

5. Letter from D. Gudger (Constellation Energy Generation, LLC) to U.S. Nuclear Regulatory Commission, “Supplemental Information - Proposed Alternative for Examination of Pressurizer Circumferential and Longitudinal Shell-to-Head Welds and Nozzle-to-Vessel Welds,” dated April 8, 2022, ADAMS Accession No. ML22098A179.

6. “Verbal Authorization by the Office of Nuclear Reactor Regulation Proposed Alternative I4R-15 for Braidwood Station Units 1 and 2; Proposed Alternative I4R-21 for Byron Station Units 1 and 2 Constellation Energy Generation, Docket Nos. 50-456, 50-457, 50-454, 50-455,” dated April 15, 2022.

7. *Technical Bases for Inspection Requirements for PWR Steam Generator Class 1 Nozzle-to-Vessel Welds and Class 1 and Class 2 Vessel Head, Shell, Tubesheet-to-Head, and Tubesheet-to-Shell Welds*. EPRI, Palo Alto, CA: 2019. 3002015906, ADAMS Accession No. ML20225A141.

8. Letter from L. N. Olshan (U.S. Nuclear Regulatory Commission) to D. L. Farrar (Commonwealth Edison Co.), “Approval of Byron 2 Preservice Inspection Program,” dated October 29, 1986.

9. Letter from J. G. Danna (U.S. Nuclear Regulatory Commission) to D. G. Stoddard (Dominion Energy Nuclear Connecticut, Inc.), “Millstone Power Station Unit 2 – Authorization and Safety Evaluation for Alternative Request No. RR-05-06 (EPID L-2020-LLR-0097),” dated July 16, 2021, ADAMS Accession No. ML21167A355.

10. Letter from G. T. Bischof (Dominion Energy Nuclear Connecticut, Inc.) to U.S. Nuclear Regulatory Commission, “Dominion Energy Nuclear Connecticut, Inc., Millstone Power Station Unit 2 – Response to Request for Additional Information for Alternative Request RR-05-06 - Inspection Interval Extension for Steam Generator Pressure Retaining Welds and Full-Penetration Welded Nozzles,” dated March 19, 2021, ADAMS Accession No. ML21081A136.

11. *Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzles-to-Shell Welds and Nozzle Inside Radius Sections*. EPRI, Palo Alto, CA: 2019. 3002014590, ADAMS Accession No. ML19347B107.