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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 Category C-B Steam Generator Nozzle-to-Shell Welds and Nozzle Inside Radius Sections

References:
1) Letter from D. Gudger (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, " Proposed Alternative for Examinations of Examination Category C-B Steam Generator Nozzle-to-Shell Welds and Nozzle Inside Radius Sections," dated September 1, 2021 (ML21244A328)
2) Email from J. Wiebe (U.S. Nuclear Regulatory Commission) to T. Loomis (Constellation Energy Generation, LLC), "Final RAIs 9.1.2021 Constellation Relief Request," date May 6, 2022

In the Reference 1 letter, Exelon Generation Company, LLC (now known as Constellation Energy Generation, LLC (CEG)) requested Nuclear Regulatory Commission (NRC) 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). Specifically, the proposed alternative concerns ISI Class 2, Examination Category C-B, Pressure Retaining Nozzle Welds in Pressure Vessels, Item Numbers C2.21, Nozzle-to-Shell (Nozzle to Head or Nozzle to Nozzle) Weld, and C2.22, Nozzle Inside Radius Section.

In the Reference 2 email, the NRC requested additional information. Attachment 2 contains our response.

Attachment 1 contains a regulatory commitment.

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

Respectfully,

David T. Gudger

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

Attachments: 1) Summary of Commitments
2) 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
NRC Project Manager - Braidwood Station
NRC Project Manager - Byron Station
NRC Project Manager - Calvert Cliffs Nuclear Power Plant
NRC Project Manager – Ginna
Illinois Emergency Management Agency – Division of Nuclear Safety
W. DeHaas, Commonwealth of Pennsylvania
S. Seaman, State of Maryland
A. L. Peterson, NYSERDA

Attachment 1

Summary of Commitments

Attachment

Summary of Commitments

The following table identifies commitments made in this document. (Any other actions discussed in the submittal represent intended or planned actions. They are described to the NRC for the NRC's information and are not regulatory commitments.)

COMMITMENT	COMMITTED DATE OR "OUTAGE"	COMMITMENT TYPE	
		ONE-TIME ACTION (Yes/No)	Programmatic (Yes/No)
As part of the performance monitoring plan, CEG will examine one (1) Examination Category C-B, Item C2.21 feedwater nozzle-to-shell weld and one (1) Examination Category C-B, Item C2.22 feedwater nozzle inside radius section at Braidwood Unit 1 to the maximum extent possible. Any new unacceptable indications identified as part of the performance monitoring plan at Braidwood Unit 1 will result in the same population of welds being examined at Braidwood Unit 2 and Byron Units 1 and 2 during the next regularly scheduled outage. The components available for examination are provided in the Table below.	The required examinations will be completed by the end of 2030 to ensure that no more than 20 years elapses between the performance of an ASME XI examination for the Examination Category C-B components at Braidwood Station, Unit 1.	Yes	No

Unit	SG	Component ID	Item Number	Description
1	A	1SG-05-SGN-04	C2.21	Feedwater Nozzle
1	A	1SG-05-SGN-04 (NIR)	C2.22	Feedwater Nozzle Inner Radius
1	B	1SG-06-SGN-04	C2.21	Feedwater Nozzle
1	B	1SG-06-SGN-04 (NIR)	C2.22	Feedwater Nozzle Inner Radius
1	C	1SG-07-SGN-04	C2.21	Feedwater Nozzle
1	C	1SG-07-SGN-04 (NIR)	C2.22	Feedwater Nozzle Inner Radius
1	D	1SG-08-SGN-04	C2.21	Feedwater Nozzle
1	D	1SG-08-SGN-04 (NIR)	C2.22	Feedwater Nozzle Inner Radius

Attachment 2

Response to Request for Additional Information

RAI 1

Issue

The applicant referenced probabilistic and deterministic analyses (EPRI Report No. 3002014590 noted above) estimating potential fatigue growth in the subject components. The applicant presented information to demonstrate that the referenced analysis would bound the subject components. This information included high-level results from previous inservice inspections (ISI) of the subject components. The applicant provided limited discussion of performance monitoring, primarily focused on justifying application of analyses to components with low inspection coverages (e.g., that leakage would be detected and plants safely shut-down).

The applicant proposed, based on the above, that the ISI interval for the subject components could be extended to the end-of-license, ranging from 12.4 years to 36 years depending on plant unit. For components for which this period would extend substantially beyond 20 years, there appears to be a lack of necessary performance monitoring to approve the request.

Leveraging probabilistic fracture mechanics to define the basis for risk-informing inspection intervals requires knowledge of both the current and future behavior of the material degradation and the associated uncertainties applicable to the subject components during the requested alternative period. Confidence in the results of these analyses hinges on the assurance that the model used adequately represents, and will continue to represent, the degradation behavior in the subject components. Proper performance monitoring through inspections is needed to ensure that the model continues to predict the behavior and that unknown/unpredicted degradation behavior 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

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

<u>Response</u>

It should be noted that all examinations reported in the Constellation Energy Generation, LLC (CEG) proposed alternative [1] have coverages greater than 90% as shown in Appendix B of the proposed alternative. Therefore, the statement “The applicant provided limited discussion of performance monitoring, primarily focused on justifying application of analyses to components with low inspection coverages (e.g., that leakage would be detected and plants safely shut-down)” in the first paragraph of the “Issue” section of RAI 1 is incorrect as no low inspection coverages exist. If the request for performing monitoring is related to low coverage as alluded to in this paragraph of RAI 1, then there should be no need for performance monitoring for the welds and components covered by the proposed alternative. Nevertheless, a performance monitoring plan is provided in the paragraphs below as requested.

In the Reference [1] letter, CEG requested relief from the examination of steam generator main steam and feedwater nozzle-to-shell welds and nozzle inside radius sections 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). In the Reference [2] email, the U.S. Nuclear Regulatory Commission requested additional information. The Reference [1] relief request is supported by the evaluations and conclusions presented in EPRI Report 3002014590 [3] 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. Of the plants surveyed, 483 examinations have been performed on Item Number C2.21 and 232 examinations have been performed on Item Number C2.22. 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). Only 1 examination for Item Number C2.21 identified flaws exceeding the acceptance criteria of Section XI, and no flaws have been identified for Item Number C2.22. As shown in Table B4 of Reference [1], Byron Unit 2 identified two linear indications by surface examination on the “B” steam generator main steam nozzle-to-vessel weld during the 2013 refueling outage. One indication was acceptable as-is and the other was ground to an acceptable size and accepted in accordance with IWC-3122.2 and IWC-3511. No indications were identified with ultrasonic examination. No other CEG PWR station covered by the proposed alternative has identified any flaws that exceed ASME Code, Section XI acceptance criteria for the steam generator welds or components. 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 8.3 of the EPRI Report [3] indicates that it would take a minimum of 147 years for a postulated initial flaw (with a depth equal to the ASME Code, Section XI acceptance standards) in the steam generator 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 steam generator main steam and feedwater welds and components are very flaw tolerant.

- Demonstrating the plant-specific applicability of the EPRI Report along with the probabilistic fracture mechanics (PFM) evaluations presented in Section 8.2 of the EPRI Report, as supplemented by this CEG RAI response, indicate that the steam generator main steam and feedwater welds and components at Braidwood, Byron, Calvert Cliffs, and Ginna can operate safely for over 80 years.

- The maximum proposed inspection deferrals for Braidwood, Byron, Calvert Cliffs and Ginna are significantly lower (36 years vs 147 years) than those justified by the results of the DFM and the PFM evaluations in the EPRI Report. These conservative inspection deferrals combined with the performance monitoring plan for Braidwood and Byron below, 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 steam generators and are bounded by the analysis in the EPRI Report. As shown in the tables of Appendix A of 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 steam generator main steam and feedwater inspections.

As shown in Table 1 of Reference [1], Ginna, Category C-B, Item C2.21 and C2.22 request a deferral of 12.4 years from the last ASME Code, Section XI inspections while Calvert Cliffs Unit 2 Category C-B, Item C2.21 and C2.22 request a maximum deferral of 21.5 years from the last ASME Code, Section XI inspections. For the purposes of performance monitoring 21.5 years is not considered substantially longer than 20 years and therefore no additional examinations are proposed for Ginna or Calvert Cliffs for the subject components for the remainder of the plant life. The strong technical basis provided by the results of the PFM Model and EPRI Report 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 steam generator main steam and feedwater nozzle welds and components at 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.

For Braidwood and Byron, in addition to the results of the DFM and PFM evaluations of the EPRI Report, which demonstrate that the steam generators are very flaw tolerant, CEG has developed an additional performance monitoring plan. The performance monitoring plan will validate the continued adequacy of the PFM model and verify that no unexpected degradation mechanisms have developed over time.

As shown in Table 1 of Reference [1], Braidwood Unit 1, Category C-B, Item C2.22 requests a maximum deferral of 36 years from the last ASME Code, Section XI inspections while Byron Unit 2, Category C-B, Item C2.21 requests a maximum deferral of 33.6 years. As a performance monitoring plan, CEG will examine one (1) Category C-B, Item C2.21 feedwater nozzle-to-shell weld and one (1) Category C-B, Item C2.22 feedwater nozzle inside radius section at Braidwood Unit 1 to the maximum extent possible. The proposed performance monitoring plan for Braidwood Unit 1 will be performed by the end of 2030. This will ensure that no more than 20 years elapses between the performance of an ASME Code, Section XI examination for the Category C-B components at Braidwood Station, Unit 1.

With the proposed performance monitoring plan by CEG, the required ASME Code, Section XI, Category C-B examination of steam generator components covered by the proposed alternative will be performed during the analytically determined safe operating period for Braidwood Unit 1. This performance monitoring plan represents a sample of two (2) of the ten (10) Category C-B examinations required by ASME Code, Section XI that are covered by the proposed alternative at Braidwood and Byron. The components selected for examination as part of the performance monitoring plan are considered representative of the remaining components covered by the proposed alternative given the similarities in design, materials, construction methods, service conditions, and operating strategies between Braidwood and Byron. Given the number of examinations (two) and the representative nature of the components selected, the performance monitoring plan is considered to adequately represent the material condition of the remaining components covered by the proposed alternative at Braidwood and Byron.

Performing an ASME Code, Section XI examination of the components included in the performance monitoring plan by the specified date will provide direct evidence to the presence of, or extent of, any unexpected degradation experienced by these components. Due to similarities between the components and operating conditions at

Braidwood and Byron, the results of the performance monitoring plan for Braidwood Unit 1 are considered to accurately represent the material condition for Category C-B components in Braidwood Unit 2 as well as Byron Unit 1 and 2.

In the unlikely event of any new unacceptable indications (i.e., indications exceeding the acceptance standards of IWC-3500, that are accepted by Repair/Replacement Activity or Analytical evaluation) are identified during the performance monitoring plan at Braidwood Unit 1, these indications will be evaluated as required by ASME Code, Section XI and the corrective action program. The additional examination and successive inspection requirements of ASME Code, Section XI also apply. Any new unacceptable indications identified as part of the performance monitoring plan at Braidwood Unit 1 will result in the same population of welds being examined at Braidwood Unit 2 and Byron Units 1 and 2 during the next regularly scheduled outage.

In addition to the direct evidence provided by the performance monitoring plan, examination of Category C-B steam generator components is expected to continue to be performed by other units across the domestic and international PWR fleet. These examinations will provide additional monitoring and opportunities to detect any degradation in the components covered by the proposed alternative. Continued examination of Category C-B steam generator 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 this new degradation mechanism.

The absence of any new unacceptable indications in the Braidwood Unit 1 components selected for examination as part of the performance monitoring plan and the absence of any unexpected degradation 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, along with the implementation of the proposed performance monitoring plan, including scope expansion criteria, will provide additional assurance that the steam generator welds and components at Braidwood and Byron can operate safely for the remainder of plant life and will continue to provide an acceptable level of quality and safety.

A regulatory commitment is contained in the Attachment 1.

RAI 2

Issue

The sensitivity studies and analysis provided in the EPRI report indicate significant potential for risks higher than the base case for Westinghouse steam generator feedwater nozzle cases FEW-P1N and FEW-P3A. As the potential for uncertainties to increase risk for these cases is relatively higher than for the other cases presented in the EPRI report, the relationship between these cases as modeled and the subject plant components is of significant interest in reaching regulatory conclusions.

Request

Compare and contrast the plant specific parameters equivalent to the FEW-P1N and FEW-P3A case components. Indicate where the plant specific parameters suggest plant specific evaluation would result in reduced probabilities relative to the EPRI report base case and/or substantiation of results being materially improved relative to the reported sensitivity results in the EPRI report.

Response

The sensitivity studies which are most relevant are those presented in Tables 8-28 and 8-29 of the EPRI Report [3]. In these tables, sensitivity studies were performed considering the combined effects of fracture toughness, stress, and flaw density. A PSI/ISI scenario of (PSI+20+40+60) was considered in the sensitivity study of Table 8-28 and a scenario of (PSI +10+20+40+60) was considered in the sensitivity study in Table 8-29. In both tables, a nozzle flaw density of 0.1, stress multiplier of 1.5 and fracture toughness of 200 ksi√in with standard deviation of 30 ksi√in were used. In the Vogtle SER [4], the NRC indicated that the nozzle flaw density of 0.1 should be used for the inside radius sections and a fracture toughness of 200 ksi√in with standard deviation of 5 ksi√in in the sensitivity studies. These recommended parameters are used to perform a plant specific evaluation for the bounding CEG plant using the limiting plant specific PSI/ISI scenario. As detailed in the proposed alternative, for the CEG fleet steam generators, the limiting PSI/ISI scenario is PSI followed by two ten-year ISI examinations followed by the requested deferral period. For the inside radius sections, the maximum requested deferral period is 36 years for Braidwood U1. For the nozzle-to-shell weld, the maximum requested deferral period is 33.6 years for Byron U2. Plant-specific evaluations were therefore performed with the limiting PSI/ISI scenarios of (PSI +10 +20 +56) and (PSI +10+20+54) for Case IDs FEW-P1N and FEW-P3A, respectively, as detailed below.

In the Reference [3] stress analyses, representative geometries were used to develop finite element models of the steam generator feedwater and main steam nozzles. As discussed in Sections 4.3.3 and 4.6 of Reference [3] and noted by the NRC in Section 3.8.3.1, page 9, third paragraph of the Safety Evaluation for Vogtle [4], 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 [3] (stress multiplier) can be used to scale up the stresses of the Reference [3] report to obtain the plant-specific stresses for each unit and component. Tables RAI 2-1 and RAI 2-2 provides the plant-specific R_i/t comparison to that in the Reference [3] report. As shown in Table RAI 2-1, the maximum stress multiplier for the SG shell (corresponding to Case ID FEW-P3A) is 1.24 and is associated with the configuration at Calvert Cliffs. Table RAI 2-2 shows that for the feedwater nozzle (corresponding to FEW-P1N), the maximum stress multiplier is 1.31 and associated with the configuration at Ginna.

Table RAI 2-1. CEG SG Shell Dimensions in Comparison with Reference [3]

Plant	Secondary Lower Shell ID (in)	Secondary Lower Shell Thk (in)	Secondary Lower Shell R_i/t	Stress Multiplier = $(R_i/t)_{plant}$ / $(R_i/t)_{EPRI}$
EPRI Report (Figure 4-10 of [3])	168.88	3.52	23.98	n/a
Braidwood Unit 1	129.0	3.6875	17.4	0.78
Braidwood Unit 2	129.0	3.1875	20.2	0.90
Byron Unit 1	129.0	3.6875	17.4	0.78
Byron Unit 2	129.175	3.2	20.1	0.90
Calvert Cliffs Unit 1	159.1875	2.875	27.68	1.24
Calvert Cliffs Unit 2	159.1875	2.875	27.68	1.24
Ginna	122	2.875	21.21	0.95

Table RAI 2-2. CEG PWR SG Nozzle Dimensions in Comparison with Reference [3]

Plant	FW Nzl ID (in)	FW Nzl Thk (in)	FW Nzl R_i/t	Stress Multiplier = $(R_i/t)_{plant}$ / $(R_i/t)_{EPRI}$
EPRI Report (Figure 4-10 of [3])	16.5	6	1.38	n/a
Braidwood Unit 1	14.81	4.5	1.65	1.20
Braidwood Unit 2	16.5	4.75	1.74	1.26
Byron Unit 1	14.81	4.5	1.65	1.20
Byron Unit 2	16.5	4.75	1.74	1.26
Calvert Cliffs Unit 1	16.75	6.46	1.26	0.91
Calvert Cliffs Unit 2	16.75	6.46	1.26	0.91
Ginna	16.75	4.625	1.81	1.31

Case ID FEW-P1N

An evaluation similar to that shown in Tables 8-28 and 8-29 of Reference [3] was performed for this Case ID for a nozzle inside radius section assuming a nozzle flaw density of 0.1, a fracture toughness of 200 ksi√in and a standard deviation 5 ksi√in, as recommended by the NRC in Reference [4], with the limiting PSI/ISI scenario of (PSI+10+20+56). A stress multiplier of 1.75 was used in the evaluation. This stress multiplier was chosen to result in probability of rupture or probability of leakage close to the acceptance criteria of 1.0×10^{-6} after 80 years. All other input parameters are the same as detained in Table 8-7 of Reference [3]. The evaluation was performed using **PROMISE** Version 1.0 (the same version of **PROMISE** that was used in the Reference

[3] report). The results are summarized in Table RAI 2-3 and show that after 80 years of plant operation, the limiting probability of rupture is leakage 3.97E-07 and the limiting probability of leakage is 1.58E-07, both just below the acceptance criterion of 1.0x10^{-6}. The applied stress multiplier of 1.75 is higher than the maximum stress multiplier of 1.31 associated with Ginna in Table RAI 2-2, indicating that on a plant-specific basis the probabilities of rupture and leakage are well below the acceptance criteria for all the CEG PWR units in the proposed alternative. It should be noted that the evaluation period of 80 years is more than twice the maximum requested deferral period which provides additional margin.

Table RAI 2-3. Sensitivity to Combined Effects of Fracture Toughness, Stress, and Nozzle Flaw Density for 80 Years for CEG Plants Feedwater Nozzle Inside Radius Section (Case ID FEW-P1N from Reference [3], PSI+10+20+56)

Time (yr)	Probability per Year for Combined Case K_{IC} = 200 ksi√in. SD = 5 ksi√in. Stress Multiplier = 1.75 Nozzle Flaw Density = 0.1 PSI+10+20+56	
	Rupture	Leak
10	3.97E-07	1.58E-07
20	2.41E-07	9.80E-08
30	1.60E-07	6.57E-08
40	1.22E-07	4.98E-08
50	1.06E-07	4.26E-08
60	9.93E-08	4.03E-08
70	8.53E-08	3.46E-08
80	7.46E-08	3.03E-08

Case ID FEW-P3A

For the feedwater nozzle-to-shell weld (FEW-P3A), a flaw density of 1.0 flaw per weld was used, consistent with the evaluations in Reference [3]. A fracture toughness of 200 ksi√in and standard deviation of 5 ksi√in were also used with the limiting PSI/ISI scenario of (PSI+10+20 +54). A stress multiplier of 1.35 was applied in this evaluation (this stress multiplier was chosen to result in probability of rupture or probability of leakage close to the acceptance criteria after 80 years). All other input parameters are the same as those detailed in Table 8-7 of Reference [3]. The results of the evaluation, using **PROMISE** Version 1.0, are summarized in Table RAI 2-4 and show that after 80 years of plant operation the probabilities of rupture and leakage are well below the acceptance criterion of 1.0x10^{-6}. The applied stress multiplier of 1.35 is higher than the maximum stress multiplier of 1.24 associated with Calvert Cliffs in Table RAI 2-1 indicating that on plant-specific basis, the probabilities of rupture and leakage are well below the acceptance criteria for all the CEG PWR units in the proposed alternative. It should be noted that the evaluation period of 80 years is more than twice the maximum requested deferral period which provides additional margin.

Table RAI 2-4. Sensitivity to Combined Effects of Fracture Toughness, Stress, and Nozzle Flaw Density for 80 Years for the CEG Plants Feedwater Nozzle-to-Shell Weld (Case ID FEW-P3A from Reference [1], PSI+10+20+54)

Time (year)	Probability per Year for Combined Case K_{IC} = 200 ksi√in. SD = 5 ksi√in. Stress Multiplier = 1.35 Nozzle Flaw Density = 1 PSI+10+20+54	
	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	1.00E-08
50	2.00E-09	4.44E-07
60	1.67E-09	9.35E-07
70	1.43E-09	8.23E-07
80	1.25E-09	7.46E-07

RAI 3

Issue

The probabilistic fracture mechanics (PFM) analysis in the EPRI report assumes a certain number of fatigue cycles for each analyzed transient. The licensee compared corresponding actual fatigue cycles to those assumed in the analysis in Tables A.2, A.3, A.5, A.6, A.8, and A.10 of the submittal. However, Table A.8 does not provide the cycle count for the Loss of Power transient at Calvert Cliffs 1 and 2. The licensee must demonstrate that the assumptions of the generic PFM in the EPRI report are reasonable for Calvert Cliffs 1 and 2.

Request

Provide justification that the transient types and associated cycle numbers analyzed in the EPRI report are reasonable for Calvert Cliffs 1 and 2, including the Loss of Power transient.

Response

The Loss of Power transient is associated with the loss of offsite power and occurs very infrequently. Review of plant event records indicate that there has been one total Loss of Power event [5] and three partial Loss of Power events [6, 7, 8] at Calvert Cliffs 1 and 2. Conservatively grouping all these events as total Loss of Power events results in four such events. In the evaluation in the Reference [3] report, 60 cycles of the Loss of Power transient were considered which is considerably higher than the Loss of Power events experienced at Calvert Cliffs.

The infrequency of the Loss of Power transient is confirmed by the limited number of occurrences documented in the Request for Alternative for the other CEG PWR units: Braidwood 1 (one in 35.5 years), Braidwood 2 (two in 34 years), Byron 1 (one in 34 years), Byron 2 (three in 34 years), and Ginna (twelve projected for 60 years). Information available at other plants that have submitted similar Relief Requests also confirm the infrequency of this transient:

- Vogtle Unit 1: three for 60-year projection (ADAMS Accession No. ML19347B105)
- Vogtle Unit 2: three for 60-year projection (ADAMS Accession No. ML19347B105)
- Millstone Unit 2: zero occurrences to-date (ADAMS Accession No. ML20198M682)
- Davis-Besse: six projected for 60-year (ADAMS Accession No. ML21256A119)

Based on the information provided above and the limited number of loss of power transients experienced to date, Calvert Cliffs is determined to be bounded by the number of cycles for the Loss of Power transient evaluated in the Reference [3] report for the period of the requested deferral.

RAI 4

Issue

The PFM analysis in the EPRI report assumes certain examination histories, e.g., PSI followed by 10-year inspections. The licensee provided actual examination histories for the subject plants in Appendix B of the submittal. However, there are apparent gaps in the provided examination histories, as described below.

- Appendix B does not provide information on preservice examination history.
- Appendix B does not provide examination history for the Braidwood 2 main steam nozzle inner radius.
- Appendix B does not provide examination history for the Braidwood 1 and 2 inner radius examinations for the 1st ISI interval.
- Appendix B does not provide examination history for the Byron 2 main steam nozzle inner radius.
- Appendix B does not provide examination history for Calvert Cliffs 1 and 2 1st and 2nd ISI interval examinations.

- Appendix B does not provide examination history for Ginna 1st and 2nd ISI interval examinations.

The licensee must demonstrate that the assumptions of the generic PFM in the EPRI report are reasonable for the subject plants.

Request

Describe how the reported examination histories are compliant with Section XI requirements. Describe how the assumed examination histories in the EPRI report's PFM analysis are appropriate for Braidwood 1 and 2, Byron 1 and 2, Calvert Cliffs 1 and 2, and Ginna, given their respective examination histories.

Response

Compliance with ASME Code, Section XI scheduling requirements for the service life of a given unit is not a prerequisite to apply the results and conclusions of the Reference [3] report. As detailed in the Request for Alternative, and the response to RAI 2, the limiting PSI/ISI scenario is PSI followed by two ten-year ISI examinations followed by the requested deferral period. The following provides the response to the bulleted items in the "Issues" section of RAI 4.

- *Preservice Examination History*

As used in the Reference [3] report, preservice examination (PSI) refers to the collective examinations required by ASME Code, Section III during fabrication and any ASME Section XI examinations performed prior to service. The 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 steam generator 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 steam generators were fabricated (including 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 the Byron U2 FW Nozzle (C2.21), shown in Table B5 of the proposed alternative and discussed in RAI 5, no recordable indications exceeding the acceptance standards were identified for the subject welds during ASME Code, Section XI preservice examinations.

- *Examination History for the Braidwood 2 Main Steam Nozzle Inner Radius*

The design for the Braidwood U2 main steam nozzle does not include an inside radius section as defined in ASME Code, Section XI and is not included as part of the proposed alternative. The Tables in Section 1 of the proposed alternative, ASME Code

Components Affected, do not include main steam nozzle inside radius section components for Braidwood 2 and therefore no inspection history was provided.

- *Examination History for the Braidwood 1 and 2 Inner Radius Examinations for the 1st ISI Interval*

As stated in Section 1 of the proposed alternative, replacement of the Braidwood U1 steam generators occurred during the 1998 outage. The first inspection interval for Braidwood U1 concluded in July of 1998, therefore any inspection history from the first inspection interval at Braidwood U1 is not applicable to the proposed alternative. As stated above the Braidwood U2 main steam nozzle does not include an inside radius section component and is not included as part of the proposed alternative. Inspection records for the Braidwood U2 feedwater nozzle inside radius section from the first inspection interval could not be located, however two inspection intervals of examinations have been performed with no unacceptable indications identified. An evaluation similar to that in the response to RAI 2 for Case ID FEW-P1N was performed by conservatively assuming that no inspection was performed for the Braidwood Unit 2 feedwater nozzle inside radius section for the first interval. From Table 1 of the proposed alternative, the deferral period for Braidwood Unit 2, Item C.22 is 30.6 years. The scenario considered is therefore (PSI+20+30+61) with a stress multiplier of 1.26 associated with Braidwood Unit 2 from Table RAI 2-1. The results of the evaluation are presented in Table RAI 4-1 and shows that after 80 years of operation, the probabilities of rupture and leakage are below the acceptance criteria of 1.0×10^{-6}. This indicates that even if no credit is taken for the first interval inspection, the acceptance criteria will still be met for Braidwood Unit 2 feedwater nozzle inside radius section.

Table RAI 4-1. Sensitivity to Combined Effects of Fracture Toughness, Stress, and Nozzle Flaw Density for 80 Years for Braidwood Unit 2 Feedwater Nozzle Inside Radius Section (Case ID FEW-P1N from Reference [3], PSI+20+30+61)

Time (yr)	Probability per Year for Combined Case K_{IC} = 200 ksi√in. SD = 5 ksi√in. Stress Multiplier = 1.26 Nozzle Flaw Density = 0.1 PSI+20+30+61	
	Rupture	Leak
10	1.00E-09	3.00E-09
20	5.00E-10	2.83E-07
30	3.33E-10	1.92E-07
40	2.50E-10	1.44E-07
50	2.00E-10	1.15E-07
60	1.67E-10	9.63E-08
70	1.43E-10	8.26E-08
80	1.25E-10	7.23E-08

- *Examination History for the Byron 2 Main Steam Nozzle Inner Radius*

The design for the Byron U2 main steam nozzle does not include an inside radius section as defined in ASME Code, Section XI and is not included as part of the proposed alternative. The Tables in Section 1 of the proposed alternative, ASME Code Components Affected, do not include main steam nozzle inside radius section components for Byron 2 and therefore no inspection history was provided.

- *Examination History for Calvert Cliffs 1 and 2 1st and 2nd ISI Interval Examinations*

As stated in Section 1 of the proposed alternative, partial replacement of the Calvert Cliffs U1 steam generators was performed during the 2002 outage and partial replacement of the U2 steam generators was performed during the 2003 outage. The steam drum, which includes the MS and FW nozzles, was retained and refurbished. No inspection records prior to the steam generator replacements were able to be located; however, two inspection intervals of examinations have been performed with no unacceptable indications identified. Evaluations similar to those in the response to RAI 2 for Case IDs FEW-P1N and FEW-P3A were performed by conservatively assuming that no inspections were performed at Calvert Cliffs Units 1 and 2 feedwater and main steam nozzle-to-shell welds and inside radius sections for the first and second intervals. From Table 1 of the proposed alternative, the maximum deferral period for Items C2.21 (Case ID FEW-P3A) and C2.22 (Case ID FEW-P1N) is 21.5 years for Calvert Cliffs. The scenario considered for both Case IDs is therefore (PSI+30+40+62). Stress multipliers of 1.24 and 0.91 from Tables RAI 2-1 and RAI 2-2 were used for FEW-P3A and FEW-P1N respectively. The results of the evaluation are presented in Table RAI 4-2 and RAI 4-3 and show that after 80 years of operation, the probabilities of rupture and leakage are below the acceptance criteria of $1.0x10^{-6}$. This indicates that even if no credit is taken for the first and second interval inspections at Calvert Cliffs, the acceptance criteria will still be met for feedwater nozzle-to-shell welds and inside radius sections.

Table RAI 4-2. Sensitivity to Combined Effects of Fracture Toughness, Stress, and Nozzle Flaw Density for 80 Years for Calvert Cliffs Feedwater Nozzle Inside Radius Section (Case ID FEW-P1N from Reference [3], PSI+30+40+62)

Time (yr)	Probability per Year for Combined Case K_{IC} = 200 ksi√in. SD = 5 ksi√in. Stress Multiplier = 0.91 Nozzle Flaw Density = 0.1 PSI+30+40+62	
	Rupture	Leak
10	1.00E-09	1.00E-09
20	5.00E-10	1.00E-09
30	3.33E-10	1.83E-08
40	2.50E-10	1.38E-08
50	2.00E-10	1.10E-08
60	1.67E-10	9.17E-09
70	1.43E-10	7.86E-09
80	1.25E-10	6.88E-09

Table RAI 4-3. Sensitivity to Combined Effects of Fracture Toughness, Stress, and Nozzle Flaw Density for 80 Years for Calvert Cliffs Feedwater Nozzle-to-Shell Weld (Case ID FEW-P3A from Reference [3], PSI+30+40+62)

Time (year)	Probability per Year for Combined Case K_{IC} = 200 ksi√in. SD = 5 ksi√in. Stress Multiplier = 1.24 Nozzle Flaw Density = 1 PSI+30+40+62	
	Rupture	Leak
10	1.00E-08	1.00E-08
20	5.00E-09	6.00E-08
30	3.33E-09	1.78E-06
40	2.50E-09	1.35E-06
50	2.00E-09	1.08E-06
60	1.67E-09	9.00E-07
70	1.43E-09	7.71E-07
80	1.25E-09	6.76E-07

- *Examination History for Ginna 1st and 2nd ISI Interval Examinations*

As stated in Section 1 of the proposed alternative, replacement of the Ginna Unit 1 Steam Generators was performed during the 1996 outage. The second inspection interval for Ginna concluded in December of 1989; therefore, any inspection history from the first or second inspection interval at Ginna is not applicable to the proposed alternative.

RAI 5

Issue

The licensee reported an indication in the Byron Unit 2 feedwater nozzle. The licensee did not provide sizing information for the flaw. The EPRI report's PFM analysis assumes a certain flaw distribution. The licensee must demonstrate that the assumptions of the generic PFM in the EPRI report are reasonable for Byron Unit 2.

Request

Provide sizing information for the flaw reported in Table B5 of the licensee's submittal. Describe how the assumed flaw distribution in the PFM analysis compares with the actual flaw size.

Response

The flaw identified in Component 2RC-01-BC/SGN-02 during PSI at Byron 2 is a near surface flaw with length (l) = 0.5 inches, depth (2a) = 0.29 inches and distance from the inside surface = 0.04 inches [9]. In the flaw evaluation of Reference [9], a surface flaw depth of 0.31 inches and flaw length of 0.5 inches were used resulting in an aspect ratio (a/l) of 0.62 and the flaw depth-to-thickness ratio of 9.7%. The final disposition of ultrasonic indications identified in the Byron U2 steam generators during PSI was provided to the NRC via letter dated May 21, 1986 [10]. Supplemental non-destructive examinations and metallurgical analyses performed to determine the root cause concluded that the indications resulted from small slag inclusions formed during the fabrication process [10]. A flaw evaluation in accordance with IWB-3600 concluded that the flaw in component 2RC-01-BC/SGN-02 would not grow to the allowable flaw size during the life of the plant [9, 10]. The NRC approved this final disposition via letter dated October 29, 1986 [11]. The NRC concluded that the disposition of the 2RC-01-BC/SGN-02 flaw provided an acceptable level of quality and safety provided the area containing the flaw in the steam generator was reexamined during the first two inspection periods. The examination history provided in Table B5 of the proposed alternative indicates that these examinations were completed in January of 1989 and September of 1993 with no change in the flaw size.

The PVRUF flaw distribution model used in the PFM analysis is represented by Equation 8.1 in the Reference [3] EPRI report. The evaluated flaw depth of 0.31 associated with the identified flaw at Byron Units 2 is contained in the PVRUF model and has a probability of occurrence of 2.5%. From Figure 8-2 of the Reference [3] EPRI report, the probability of detection of this flaw is about 90%.

References

1. Letter No. RS-21-093 from D. T. Gudger (Exelon Generation) to USNRC, "Proposed Alternative for Examinations of Examination Category C-B Steam Generator Nozzle-to-Shell Welds and Nozzle Inside Radius Sections," dated September 1, 2021, ADAMS Accession No. ML 21244A328.
2. Email Letter from J. Wiebe (USNRC) to T. R. Loomis (Exelon Generation), "Final RAIs 9.1.2021 Constellation Relief Request," dated May 6, 2022.
3. *Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Nozzle Inside Radius Sections.* EPRI, Palo Alto, CA: 2019. 3002014590, ADAMS Accession No. ML19347B107.
4. Letter from Michael T. Markley (USNRC) to Cheryl A. Gayheart (Southern Nuclear), "Vogtle Electric Generating Plant, Units 1 & 2 - Relief Request for Proposed Inservice Inspection Alternative VEGP-ISI-ALT-04-04 to the Requirements of ASME," dated January 11, 2021, ADAMS Accession No. ML20352A155.
5. INPO Significant Event Notification (SEN) 17, "Calvert Cliffs Units 1 and 2, Loss of Offsite Power," dated June 24, 1987.
6. Licensee Event Report (LER) No. 93-003, Calvert Cliffs Unit 1, "Reactor Trip Caused by Closure of Turbine Stop Valves," dated June 10, 1993.
7. Letter from P. E. Katz (Baltimore Gas and Electric Co.) to USNRC, "Calvert Cliffs Nuclear Power Plant Unit 2; Docket No. 50-318, License No. DPR-69, Licensee Event Report 96-001, Automatic Plant Trip Due to Partial Loss of Offsite Power," dated March 28, 1996.
8. Letter from T. E. Trepanier (Constellation Energy) to USNRC, "Calvert Cliffs Nuclear Power Plant Unit 2; Docket No. 50-318, License No. DPR-69, Licensee Event Report 2010-001, Plant Trip Due to Partial Loss of Offsite Power," dated April 15, 2010.
9. Letter from K. A. Ainger (Commonwealth Edison Co.) to H. R. Denton (USNRC), "Byron Station Unit 2, Preservice Inspection, NRC Docket N-. 50-455," dated October 11, 1985.
10. Letter from K. A. Ainger (Commonwealth Edison Co.) to H. R. Denton (USNRC), "Byron Station Unit 2, Preservice Inspection, Steam Generators and Pressurizer, NRC Docket N-. 50-455," dated May 21, 1986.
11. Letter from L. N. Olshan (USNRC) to D. L. Farrar (Commonwealth Edison Co.), "Approval of Byron Unit 2 Preservice Inspection Program," dated October 29, 1986.