UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 23, 2024

David P. Rhoades
Senior Vice President
Constellation Energy Generation, LLC
President and Chief Nuclear Officer
Constellation Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2; BYRON STATION, UNIT NOS. 1 AND 2; CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2; AND R. E. GINNA NUCLEAR POWER PLANT - ISSUANCE OF RELIEF RE: PROPOSED ALTERNATIVE REQUEST ASSOCIATED WITH STEAM GENERATOR EXAMINATIONS (EPIDS L-2023-LLR-0053, L-2023-LLR-0054, L-2023-LLR-0055, L-2023-LLR-0056)

Dear David Rhoades:

By letters dated October 10, 2023 and October 11, 2023 (Agencywide Documents Access and Management System Accession Nos. ML23283A003 and ML23284A259, respectively), Constellation Energy Generation, LLC (the licensee), submitted requests for the Braidwood Station, Units 1 and 2 (Braidwood); Byron Station, Units 1 and 2 (Byron); Calvert Cliffs Nuclear Power Plant, Units 1 and 2 (Calvert Cliffs); and R.E. Ginna Nuclear Power Plant (Ginna), to the U.S. Nuclear Regulatory Commission (NRC or Commission) for proposed alternatives to certain requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code (ASME Code), Section XI.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), paragraph 50.55a(z)(1), “Acceptable level of quality and safety,” the licensee proposed to defer the ASME Code, Section XI inspection requirements pertaining to steam generators welds and nozzles from the current 10-year requirement to the end of the currently licensed operating periods of the subject units. The regulation in 10 CFR 50.55a(z)(1) requires the licensee to demonstrate that the proposed alternatives provide an acceptable level of quality and safety.

The proposed alternatives are based probabilistic fracture mechanics and a proposed performance monitoring plan. In the October 10, 2023, letter, the licensee requested to use the proposed alternative I4R-16, Revision 1 for Braidwood; I4R-22, Revision 1 for Byron; ISI-05-017, Revision 1 for Calvert Cliffs; and I6R-09, Revision 1 for Ginna. These proposed alternatives concern steam generator nozzle-to-shell and nozzle inside radius sections volumetric and surface examinations. In the October 11, 2023, letter, the licensee requested to use the proposed alternative I4R-17, Revision 1 for Braidwood, I4R-23, Revision 1 for Byron, ISI-05-018, Revision 1 for Calvert Cliffs, and I6R-10, Revision 1 for Ginna. These proposed alternatives concern volumetric examinations of steam generator pressure retaining welds and full penetration welded nozzles.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the use of proposed alternatives I4R-16 and I4R-17, Revision 1 for Braidwood; I4R-22 and I4R-23, Revision 1 for Byron; ISI-05-017 and ISI-05-018, Revision 1 for Calvert Cliffs; and I6R-09 and I6R-10, Revision 1 for Ginna, for the remainder of their currently approved operating license, currently scheduled to end on October 17, 2046 (Braidwood, Unit 1), December 18, 2047 (Braidwood, Unit 2), October 31, 2044 (Byron, Unit 1), November 6, 2046 (Byron, Unit 2), July 31, 2034 (Calvert Cliffs, Unit 1), August 13, 2036 (Calvert Cliffs, Unit 2), and September 18, 2029 (Ginna).

All other ASME BPV Code, section XI, requirements for which relief was not specifically requested and approved remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact the Senior Project Manager, Scott Wall, at 301-415-2855 or e-mail at Scott.Wall@nrc.gov.

Sincerely,

Jeffrey A. Whited

Digitally signed by Jeffrey A. Whited
Date: 2024.07.23 13:21:08 -04'00'

Jeffrey A. Whited, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-456, STN 50-457,
STN 50-454, STN 50-455, 50-317,
50-318, and 50-244

Enclosure:
Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUESTS FOR ALTERNATIVE REGARDING

INSPECTION INTERVAL EXTENSION FOR STEAM GENERATOR WELDS AND NOZZLES

CONSTELLATION ENERGY GENERATION, LLC

BRAIDWOOD STATION, UNITS 1 AND 2

BYRON STATION, UNIT NOS. 1 AND 2

CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NOS. STN 50-456, STN 50-457, STN 50-454, STN 50-455,

50-317, 50-318, AND 50-244

INTRODUCTION

By letters dated October 10, 2023 and October 11, 2023 (the submittals) (Agencywide Documents Access and Management System Accession Nos. ML23283A003 and ML23284A259, respectively), Constellation Energy Generation, LLC (CEG, the licensee), submitted requests for the Braidwood Station, Units 1 and 2 (Braidwood); Byron Station, Units 1 and 2 (Byron); Calvert Cliffs Nuclear Power Plant, Units 1 and 2 (Calvert Cliffs); and R.E. Ginna Nuclear Power Plant (Ginna), to the U.S. Nuclear Regulatory Commission (NRC or Commission) for proposed alternatives to certain requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code (ASME Code), Section XI.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee proposed to defer the ASME Code, Section XI inspection requirements pertaining to steam generators (SG) welds and nozzles from the current 10-year requirement to the end of the currently licensed operating periods of the subject units. The proposed alternatives are based probabilistic fracture mechanics (PFM) and a proposed performance monitoring plan. In the October 10, 2023, letter, the licensee requested to use the proposed alternative I4R-16, Revision 1 for Braidwood; I4R-22, Revision 1 for Byron; ISI-05-017, Revision 1 for Calvert Cliffs; and I6R-09, Revision 1 for Ginna. These proposed alternatives concern SG nozzle-to-shell and nozzle inside radius sections volumetric and surface examinations. In the October 11, 2023, letter, the licensee requested to use the proposed alternative I4R-17, Revision 1 for Braidwood, I4R-23, Revision 1 for Byron, ISI-05-018, Revision 1 for Calvert Cliffs, and I6R-10, Revision 1 for Ginna. These proposed alternatives concern volumetric examinations of SG pressure retaining welds and full penetration welded nozzles.

Enclosure

2.0 REGULATORY EVALUATION

The SG pressure-retaining welds and nozzles at the subject CEG units are ASME Code Class 1 and Class 2 components, whose inservice inspections (ISIs) are performed in accordance with the applicable edition of Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the ASME Code, as required by 10 CFR 50.55a(g).

The regulations in 10 CFR 50.55a(g)(4) state, in part, that components that are classified as ASME Code Class 1, 2, and 3 must meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, to the extent practical within the limitations of design, geometry, and materials of construction of the components.

The regulations in 10 CFR 50.55a(z) state, in part, that alternatives to the requirements in paragraphs (b) through (h) of 10 CFR 50.55a may be used when authorized by the NRC if the licensee demonstrates that: (1) the proposed alternative would provide an acceptable level of quality and safety, or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Proposed Alternative

Applicable Code Edition and Addenda

The current ISI interval and associated codes of record for the subject plants are summarized in Table 1.

Table 1: Section XI Codes of Record for Subject Plants

Plant	Interval	Current Edition and Addenda	Interval Start	Interval End
Braidwood	Fourth	2013 Edition	August 29, 2018 (Unit 1) November 5, 2018 (Unit 2)	July 28, 2028 (Unit 1) October 16, 2028 (Unit 2)
Byron	Fourth	2007 Edition with the 2008 Addenda	July 16, 2016	July 15, 2025
Calvert Cliffs	Fifth	2013 Edition	July 1, 2019	June 30, 2029
Ginna	Sixth	2013 Edition	January 1, 2020	December 31, 2029

American Society of Mechanical Engineers (ASME) Code Components Affected

ASME Code Class:	Class 1 and 2
Examination Category:	B-B, "Pressure Retaining Welds in Vessels Other Than Reactor Vessels" B-D, "Full Penetration Welded Nozzles in Vessels" C-A, "Pressure Retaining Welds in Pressure Vessels" C-B, "Pressure Retaining Nozzle Welds in Pressure Vessels"
Item Numbers:	B2.40 for the SG vessel primary side, tubesheet-to-head welds B3.130 for the SG vessel primary side, nozzle-to-vessel Welds C1.10, C1.20, and C1.30 for SG vessel secondary side welds C2.21 for the SG feedwater (FW) and main steam (MS) nozzle-to-shell welds C2.22 for the SG FW and MS nozzle inside radius (NIR) sections
Component IDs:	The tables in Section 1 of Attachment 1 to the submittals list the component identifications (IDs) affected for each subject plant.

ASME Code Requirements for Which Alternative Is Requested

For ASME Code Class 1 welds listed below, the ISI requirements are those specified in Subarticle IWB-2500 of the ASME Code, Section XI, which requires the licensee to perform volumetric examinations as specified in ASME Code, Section XI, Table IWB-2500-1, for each Examination Category and Item No. listed below once every 10-year ISI interval. As noted in Table IWB-2500-1 for Examination Category B-B, cases of multiple vessels of similar design, size, and service (such as SGs), the required examinations may be limited to one vessel or distributed among the vessels.

- Examination Category B-B, Item No. B2.40, SG Primary Side Tubesheet-to-Head Welds

For ASME Code Class 1 welds listed below, the ISI requirements are those specified in Subarticle IWB-2500 of the ASME Code, Section XI, which requires the licensee to perform volumetric examinations as specified in ASME Code, Section XI, Table IWB-2500-1, for each Examination Category and Item No. listed below once every 10-year ISI interval.

- Examination Category B-D, Item No. B3.130, SG Primary Side Nozzle-to-Vessel Welds

For ASME Code Class 2 welds and NIR sections listed below, the ISI requirements are those specified in Subarticle IWC-2500 of the ASME Code, Section XI, which requires the licensee to perform volumetric and surface examinations as specified in ASME Code, Section XI, Table IWC-2500-1, for each Examination Category and Item No. listed below once every 10-year ISI interval. As noted in Table IWC-2500-1 for Examination Categories C-A and C-B, cases of multiple vessels of similar design, size, and service (such as SGs), the required examinations may be limited to one vessel or distributed among the vessels.

- Examination Category C-A, Item No. C1.10, Shell Circumferential Welds
- Examination Category C-A, Item No. C1.20, Head Circumferential Welds
- Examination Category C-A, Item No. C1.30, Tubesheet-to-Shell Welds

- Examination Category C-B, Item No. C2.21, Nozzle-to-Shell Welds
- Examination Category C-B, Item No. C2.22, Nozzle Inside Radius Sections

The NRC staff confirmed that the ASME Code ISI requirements listed above did not change in the latest edition of ASME Code, Section XI incorporated by reference in 10 CFR 50.55a.

Reason for Proposed Alternative

In Section 4 of Attachment 1 to the submittals, the licensee stated that the Electric Power Research Institute (EPRI) performed assessments in the following non-proprietary reports of the basis for the ASME Code, Section XI examination requirements for the subject SG welds and nozzles.

- EPRI Technical Report 3002014590, “Technical Bases for Inspection Requirements for PWR [Pressurized Water Reactor] Steam Generator Class 1 Feedwater and Main Steam Nozzle-to-Shell Welds and Nozzle Inside Radius Sections,” 2019 (hereafter “EPRI report 14590,” ML19347B107).

- EPRI Technical Report 3002015906, “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,” 2019 (hereafter “EPRI report 15906,” ML20225A141).

The assessments include a survey of inspection results from 74 domestic and international nuclear units and flaw tolerance evaluations using PFM and deterministic fracture mechanics (DFM). The licensee stated that these reports were developed consistent with EPRI’s White Paper on PFM (ML19241A545). The licensee stated that the EPRI reports concluded that the current ASME Code, Section XI ISI frequency of 10 years can be increased significantly with no impact to plant safety, and that this is the basis for which an alternate inspection frequency is being requested.

The NRC staff noted that EPRI reports 14590 and 15906 were not submitted or reviewed as topical reports. The NRC staff reviewed the proposed alternative requests as a plant-specific alternative for each of the subject plants. The NRC did not review the EPRI reports for generic use, and this review does not extend beyond the plant-specific authorization.

Proposed Alternative

In Section 5 of Attachment 1 to the submittals, the licensee described the proposed alternative, summarized as follows:

The licensee is requesting an inspection alternative to the examination requirements of the ASME Code, Section XI, described above. The proposed alternative is to defer the inspection of the subject examination items from the current ASME Code, Section XI requirements to the end of the currently approved period of extended operation Braidwood, Byron, Calvert Cliffs, and Ginna, as summarized in Table 1 of Attachment 1 to the submittals. The licensee stated that the responses to NRC requests for additional information (RAIs) in letters dated May 20, 2022 (ML22140A055), June 17, 2022 (ML22168A005), and September 20, 2022 (ML22263A440), (response to RAI 2a), that were developed during the review of the first versions (Revision 0) of the subject requests (ML21244A328 and ML21348A078, respectively) remain valid, except those that pertain to the performance monitoring plan. The licensee also stated that the supplemental information in the letter dated February 21, 2023 (ML23052A068), is no longer valid. The requests in the current submittals include a description of a fleetwide performance monitoring plan, graphically shown in Attachment 1 to the submittals and reproduced in Figure 1 below.

Basis for Proposed Alternative

In Section 5 of Attachment 1 to the submittals, the licensee referred to the results of the PFM analyses in EPRI reports 14590 and 15906 and performance monitoring plan as the bases for the proposed alternative.

Duration of Proposed Alternative

In Section 6 of Attachment 1 to the submittals, the licensee stated that the proposed alternatives are requested for Braidwood, Byron, Calvert Cliffs, and Ginna for the remainder of their currently approved operating license, currently scheduled to end on October 17, 2046 (Braidwood, Unit 1), December 18, 2047 (Braidwood, Unit 2), October 31, 2044 (Byron, Unit 1), November 6, 2046 (Byron, Unit 2), July 31, 2034 (Calvert Cliffs, Unit 1), August 13, 2036 (Calvert Cliffs, Unit 2), and September 18, 2029 (Ginna).

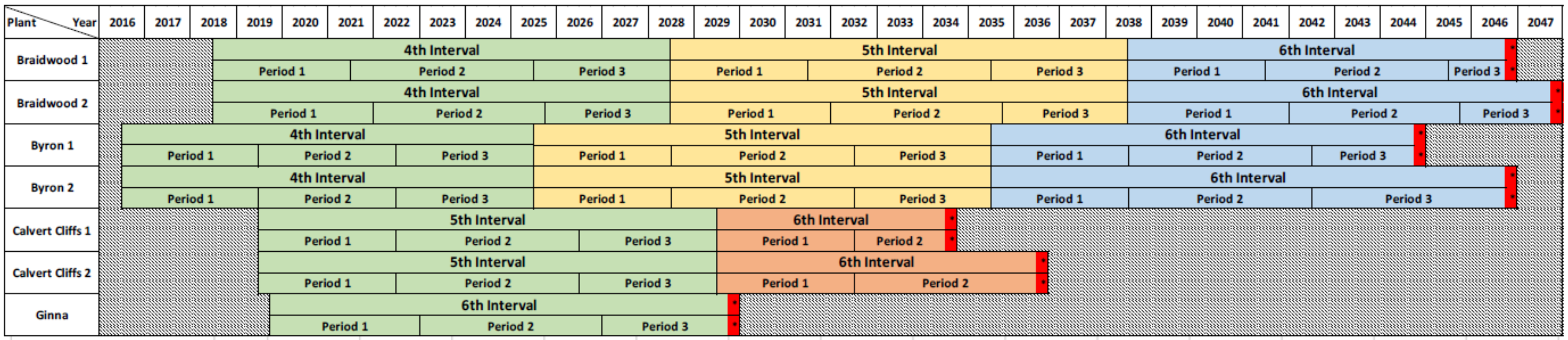
3.2 NRC Staff Evaluation

The NRC staff's review focused on evaluating the applicability of the PFM analyses in Section 8.2 of EPRI report 14590 and Section 8.3 of EPRI report 15906, and verifying whether the DFM and PFM analyses in the reports support the proposed alternatives. The NRC staff previously reviewed similar requests based on EPRI reports 14590 and 15906. These requests were in support of a Vogtle Electric Generating Plant, Units 1 and 2, submittal (ML20253A311, hereafter "Vogtle submittal") and a Millstone Power Station, Unit 2 submittal (ML20198M682, hereafter "Millstone submittal"). As part of the previous reviews of these submittals, the NRC staff conducted a thorough review of the applicable aspects of the EPRI reports and documented its review in the associated, plant-specific SEs (Vogtle (ML20352A155) and Millstone (ML21167A355)). For the CEG review, the NRC staff considered the referenced information and focused on the plant-specific application of the EPRI reports for the subject CEG units. Using a risk-informed approach, the NRC staff also confirmed that the proposed alternative provides sufficient performance monitoring.

In Section 5 of both submittals, the licensee stated that the responses to the NRC RAIs in letters dated May 20, 2022, June 17, 2022, and September 20, 2022 (response to RAI 2a), that were developed during the review of the first versions (Revision 0) of the subject requests (ML21244A328 and ML21348A078, respectively) remain valid, except those that pertain to the performance monitoring plan. To confirm that these responses remain valid for the current submittals, the NRC staff compared the relevant content of the current submittals with that of the corresponding first version of the submittal, and determined that, excepting the information on the performance monitoring plan, the relevant content of the current submittals are identical to that of the corresponding first version of the submittal with respect to geometric configurations, materials, transients, and inspection histories of the subject CEG SG welds and nozzles. Any additional inspections that were performed between the first and current versions of the submittals were included in the inspection history section of the current submittals. The NRC staff reviewed these additional inspections and determined that they are acceptable

Figure 1: Licensee’s Proposed Alternative and Performance Monitoring Plan for SG Welds and Nozzles

<u>Graphical Representation of Constellation Proposed Performance Monitoring Plan</u>



Legend	
Current Intervals	Sample of examinations performed across CEG PWR Fleet during current intervals to date, along with examinations completed during previous ISI Intervals, provide sufficient performance monitoring to support the proposed alternative for remainder of current intervals.
Successive Interval Performance Monitoring	Successive 5th Interval Performance Monitoring at Byron/Braidwood, 1 SG at 1 Braidwood Unit will be examined in accordance with ASME XI requirements. Examinations will be distributed across the periods as requried by Table IWB-2411-1 or Table IWC-2411-1, as applicable.
Successive Interval Performance Monitoring	Successive 6th Interval Performance Monitoring at Byron/Braidwood, 1 SG at 1 Byron Unit will be examined in accordance with ASME XI requirements. Examinations will be distributed across the periods as requried by Table IWB-2411-1 or Table IWC-2411-1, as applicable.
Successive Interval Performance Monitoring	Successive 6th Interval Performance Monitoring at Calvert Cliffs, a 25% sample will be performed at Unit 2, prorated to account for EOL dates. At least one component from each Examination Category will be selected, examinations will be distributed between the first and second period.
*	Identifies approximate expiration of the current operating license
Note: successive interval start and end dates and period start and end dates are approximate based on current interval scheduling	

Note: The end dates for the current interval and the start and end dates for successive intervals are estimates based on the current interval. Interval start and end dates may be adjusted as allowed by ASME Section XI. Successive interval period start and end dates are approximate based on current interval scheduling.

because greater than 90 percent examination coverage was achieved with no recordable indications. Therefore, the NRC staff finds that the licensee's responses to the NRC RAIs in letters dated May 20, 2022, June 17, 2022, and September 20, 2022 (response to RAI 2a), remain valid and pertinent and evaluated the responses within the applicable topics described in the sections that follow.

3.2.1 Degradation Mechanisms

The NRC staff reviewed the submittal for plant-specific circumstances that may indicate presence of a degradation mechanism sufficiently unique to the subject CEG units to merit additional consideration. The NRC staff found no evidence of conditions at the subject CEG units that would require consideration of a unique degradation mechanism beyond application of the information the licensee referenced from EPRI reports 14590 and 15906. Specifically, the NRC staff reviewed the materials, stress states, and consistency of chemical environment (i.e., reactor coolant) of the subject SG welds and nozzles and found them to be consistent with the assumptions made in the EPRI reports. Therefore, the NRC staff finds that consideration of additional degradation mechanisms beyond those from the EPRI reports is not necessary.

3.2.2 PFM Analysis

The NRC staff confirmed that the PFM analysis referenced by the licensee for the CEG submittal is consistent with the approach taken in the technical arguments presented in the Vogtle and Millstone submittals and cited as precedents in the CEG requests. The original review of this approach is documented in the Vogtle and Millstone SEs. The NRC staff reviewed the application of this approach, as proposed in the CEG requests, and determined that the PFM analysis is consistent with the previously approved precedent in the Vogtle and Millstone submittals. Therefore, the NRC staff finds the proposed PFM analysis to be appropriate for the CEG application.

The NRC staff noted that the acceptance criterion of $1{\times}10^{-6}$ failures per year (also termed Probability of Failure, (PoF)) is tied to that used by the NRC staff in the development of 10 CFR 50.61a, "Alternate fracture toughness requirements for protection against pressurized thermal shock events" and other similar reviews. In that rule, the reactor vessel through-wall crack frequency (TWCF) of $1{\times}10^{-6}$ per year for a pressurized thermal shock event is an acceptable criterion because reactor vessel TWCF is conservatively assumed to be equivalent to an increase in core damage frequency, and as such meets the guidance in Regulatory Guide (RG) 1.174, "An Approach to for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." This assumption is conservative because a through-wall crack in the reactor vessel does not necessarily increase the likelihood of core damage. The discussion of TWCF is explained in detail in the technical basis document for 10 CFR 50.61a, NUREG-1806 "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)," August 2007 (ML072830074).

The NRC staff also noted that the TWCF criterion of $1{\times}10^{-6}$ per year was generated using a very conservative model for reactor vessel cracking. In addition, this criterion exists within a context of reactor pressure vessel surveillance programs and inspection programs. The NRC staff finds that the licensee's use of $1{\times}10^{-6}$ failures per year based on the reactor vessel TWCF criterion is acceptable for the requested SG welds and NIR because (a) the impact of a SG vessel failure would be less than the impact of a reactor vessel failure on overall risk; (b) the subject SG welds and NIR have substantive, relevant, and continuing inspection histories and programs; and (c) the estimated risks associated with the individual welds or NIR are mostly much lower than the system risk criterion (i.e., the system risk is dominated by a small sub-population which can be considered the principal system risk for integrity). The NRC staff further noted that comparing the probability of leakage to the same criterion is conservative because leakage is less severe than rupture. The NRC staff noted that the use of a PoF criteria such as 1×10^{-6} per year for individual welds may not be appropriate generically, but based on the discussion above, the NRC staff finds the application of this criterion acceptable for this plant-specific review for the subject SG welds and NIR for the CEG units subject to this review.

Lastly, the NRC staff noted that the acceptance criterion of 1×10^{-6} failures per year is lower than the criterion the NRC staff accepted in proprietary report BWRVIP-05, "BWR [Boiling Water Reactor] Vessel and Internals Project: BWR Reactor Pressure Vessel Weld Inspection Recommendation, September 1995"; non-proprietary report BWRVIP-108NP-A, "BWR Vessel and Internals Project: Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii, October 2018" (ML19297F806); and non-proprietary report BWRVIP-241NP-A, "BWR Vessel and Internals Project: Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii, October 2018" (ML19297G738). These EPRI reports were developed prior to or around the time the rules for PTS were reevaluated, and as such the acceptance criterion for failure frequency in the reports is based on the guidelines for PTS analysis in RG 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors" that were available at the time. The NRC staff also noted that the BWR Vessel and Internals Project topical reports included substantive inspection aspects that were critical to the NRC's findings.

Based on the discussion above, the NRC staff finds the use of the acceptance criterion of 1×10^{-6} failures per year for PoF acceptable for the CEG plant-specific alternative requests.

3.2.3 Parameters Most Significant to PFM Results

The NRC staff reviewed the submittal for plant-specific aspects of the CEG alternative request that may diverge from the Vogtle and Millstone submittals, concerning parameters most significant to PFM results in EPRI reports 14590 and 15906. The NRC staff confirmed that the review conclusions in the Vogtle and Millstone SEs applied to the CEG submittals and found that the parameters most significant to PFM results would be the same and consistent with the NRC staff's reviews documented in the Vogtle and Millstone SEs, and consequently the approach taken in those reviews appropriately applies to the current review for CEG.

As discussed in the Vogtle and Millstone SEs, the sensitivity analysis (SA), sensitivity study (SS), and the NRC staff's observations on the PROMISE software identified the following significant parameters or aspects of the PFM analyses that warrant a close evaluation: stress analysis, fracture toughness, flaw crack growth (FCG) rate coefficient (or simply FCG rate), and effect of ISI schedule and examination coverage. The NRC staff also evaluated the flaw density of 1.0 per weld used by the licensee for the subject SG welds. Similarly, the NRC staff evaluated a flaw density of 0.1 flaw per NIR used by the licensee for the subject SG NIR. The NRC staff discussed and closely evaluated each in the next five sections of this SE. The NRC staff also evaluated other parameters or aspects of the analyses in Section 3.2.9 of this SE.

3.2.4 Stress Analysis

3.2.4.1 Selection of Components and Materials

In Appendix A of both submittals, the licensee evaluated the plant-specific applicability of the components and materials selected and analyzed in EPRI reports 14590 and 15906 to the subject SG welds and NIR. The acceptability of meeting the criteria, however, depends on the acceptability of the component and material selection described in the EPRI reports, which the NRC staff evaluated below.

In Section 4 of EPRI reports 14590 and 15906, EPRI discussed the variation among SG shell and SG nozzle designs. EPRI used this information for finite element analyses (FEA, see Section 3.2.4.4 of this SE) to determine stresses in the analyzed components, which the licensee referenced for the corresponding plant-specific SG components. In selecting the components, EPRI considered geometry, operating characteristics, materials, field experience with respect to service-induced cracking, and the availability and quality of component-specific information. EPRI concluded that variations in the configuration of SG lower heads, shells, and nozzles among the various designs are not significant, and that the most important parameter, ratio of radius-to-thickness (R/t) of the primary side nozzles, primary side lower head, secondary side shell, and secondary side nozzles, can be addressed through SS on stress in the PFM evaluations in both reports.

The NRC staff reviewed Section 4 of EPRI reports 14590 and 15906 and finds that the SG configurations selected in the reports for stress analysis are acceptable representatives for the corresponding SG components in the plant-specific alternative requests because differences in R/t ratios in the SG configurations discussed in both reports are small and, therefore, differences in stresses would be reasonably addressed through the SS on stress. To verify the dominance of the R/t ratio, the NRC staff reviewed the through-wall stress distributions in Section 7 of the EPRI reports to confirm that the pressure stress is dominant, which would confirm the dominance of the R/t ratio. For some of the SG shell welds modeled in EPRI report 15906, the NRC noted that the thermal stress is also potentially high, as discussed in the Millstone SE. However, the NRC staff determined that EPRI report 15906 is adequate for the subject plant-specific SG welds because they are adequately bounded by the SS on stress in the report. Accordingly, the NRC staff finds that EPRI's conclusion about the R/t ratio being the dominant parameter in evaluating the various configurations to be acceptable for the CEG plant-specific alternative request.

Tables A1, A4, A7, and A9 of the October 10, 2023, submittal and Tables A1, A2, A5, A6, A9, A10, A11, A14, and A15 of the October 11, 2023, submittal show that the plant-specific SG configurations meet the SG configuration criteria in Section 9.4 of the EPRI reports. The NRC staff confirmed that the SG configuration criteria in the EPRI reports are met. In the June 17, 2022, response to RAI 4, the licensee clarified that the primary inlet and outlet nozzles for the Braidwood, Bryon, and Ginna SGs are integrally forged with the SG channel head, and that therefore, there are no primary-side full-penetration welded nozzles in the SGs of these three plants.

Section 9.4 of EPRI reports 14590 and 15906 also address criteria for plant-specific applicability with respect to SG component materials. The plant-specific SG component materials are acceptable if they conform to ASME Code, Section XI, Nonmandatory Appendix G, paragraph G-2110. The licensee addressed these criteria in Tables A1, A4, A7, and A9 of the October 10, 2023, submittal and Tables A1, A2, A5, A6, A9, A10, A11, A14, and A15 of the

October 11, 2023, submittal. The licensee stated that the materials of construction for the SG components are as reported in Table 2.

Table 2: Materials of Construction

Unit	SG Component	Material
Braidwood, Unit 1	vessel heads	SA-508, Class 3
	vessel shell	SA-508, Class 3; SA-533, Grade B, Class 1
	tubesheet	SA-508, Class 3
	feedwater nozzles	SA-508, Class 3
	main steam nozzles	SA-508, Class 3
Braidwood, Unit 2	vessel heads	SA-216, Grade WCC; SA-533, Grade A, Class 2
	vessel shell	SA-533, Grade A, Class 2
	tubesheet	SA-508, Class 2A
	feedwater nozzles	SA-508, Class 2A
	main steam nozzles	SA-508, Class 2A
Byron, Unit 1	vessel heads	SA-508, Class 3
	vessel shell	SA-533, Grade B, Class 1
	tubesheet	SA-508, Class 3
	feedwater nozzles	SA-508, Class 3
	main steam nozzles	SA-508, Class 3
Byron, Unit 2	vessel heads	SA-216, Grade WCC; SA-533, Grade A, Class 2
	vessel shell	SA-533, Grade A, Class 2
	tubesheet	SA-508, Class 2A
	feedwater nozzles	SA-508, Class 2
	main steam nozzles	SA-508, Class 2
Calvert Cliffs, Units 1 and 2	vessel heads	SA-508, Class 3A; SA-533, Grade B, Class 1
	vessel shell	SA-508, Class 3A
	tubesheet	SA-508, Class 3A
	feedwater nozzles	SA-508, Class 1
	main steam nozzles	SA-508, Class 2
	primary inlet/outlet nozzles	SA-508, Class 3A
Ginna	vessel heads	SA-508, Class 3
	vessel shell	SA-533, Type B, Class 1
	tubesheet	SA-508, Class 3
	feedwater nozzles	SA-508, Class 3
	main steam nozzles	N/A

The NRC staff verified that these materials conform with ASME Code Section XI, Nonmandatory Appendix G. Therefore, the NRC staff finds that the materials for CEG meet the material applicability criterion.

Based on the discussion above, the NRC finds that CEG SGs are acceptable with regard to SG component configuration and materials.

3.2.4.2 Selection of Transients

In Section 5.2 of EPRI reports 14590 and 15906, EPRI discussed the thermal and pressure transients under normal and upset conditions considered relevant to the SG shell and SG nozzles. EPRI developed a list of transients for analysis applicable to the SG shell and SG nozzles analyzed in the reports, based on transients that have the largest temperature and pressure variations.

The NRC staff evaluated the transient selection in the EPRI reports in detail, as discussed in the Vogtle and Millstone SEs. The NRC staff confirmed that the applicable aspects of the transients discussed in these SEs apply equally to this review for CEG. The NRC staff reviewed the discussion of transients in Section 5.2 of EPRI reports 14590 and 15906 and determined that the transient selection defined in the reports is reasonable for the CEG plant-specific alternative request because the selection was based on large temperature and pressure variations that are conducive to FCG and are expected to occur in PWRs. The NRC staff then compared the analysis in the EPRI reports to plant-specific information provided in the licensee's submittal.

In Tables A2, A3, A5, A6, A8, and A10 of the October 10, 2023, submittal and Tables A3, A4, A7, A8, A12, A13, A16, and A17 of the October 11, 2023, submittal, the licensee evaluated the plant-specific applicability of the transients selected in EPRI reports 14590 and 15906 to the CEG SGs. In the May 20, 2022, response to RAI 3, the licensee clarified that the Loss of Power transient at the Calvert Cliffs units are bounded by the number of cycles of the Loss of Power transient analyzed in EPRI report 14590. In the June 17, 2022, response to RAI 5, the licensee clarified that pressure values applicable to the SG secondary side vessel welds at Braidwood and Byron are bounded by a sensitivity study that showed that the PoF criterion of 1×10^{-6} per year is met up to a stress multiplier of up to 1.6. In the June 17, 2022, response to RAI 7, the licensee clarified that a thermal shock on the primary side of the SG is extremely unlikely because there are no safety injection sources that inject directly into the primary side of the SG. The NRC staff noted that the absence of safety injection sources that inject directly into the primary side of the SG means that the possibility of water that is relatively colder than the reactor coolant at normal operating temperature causing a rapid temperature decrease (i.e., thermal shock) of the primary side SG shell is highly unlikely. The NRC staff reviewed the tables in both submittals and the RAI responses and confirmed that the CEG SGs are bounded by the criteria in the EPRI reports.

In the analyses in the EPRI reports there were no separate test conditions included in the transient selection. The licensee clarified in the June 17, 2022, response to RAI 8 that the Class 1 system leakage test is performed during plant Mode 3, defined as "Hot Standby" for Braidwood, Byron, and Calvert Cliffs, or "Hot Shutdown" for Ginna, at conditions between Cold Shutdown and rated temperature/pressure. The licensee stated that any required pressure tests are performed in conjunction with the Class 1 system leakage test. The licensee stated that the average reactor coolant temperature is at least 300°F (Calvert Cliffs and Ginna) and 350°F (Braidwood and Byron) at the Mode 3 conditions. The NRC staff noted that since pressure tests are performed during the Class 1 system leakage tests, at conditions between Cold Shutdown and rated temperature/pressure normal operating conditions, they would be bounded by Heatup/Cooldown, and therefore, test conditions need not be analyzed as a separate transient.

Based on the discussion above, the NRC staff finds that CEG meets the transient applicability criteria in the EPRI reports. Therefore, the analyzed transient loads for the requested CEG SG components are acceptable.

3.2.4.3 Other Operating Loads

Weld residual stress and cladding stresses are addressed in EPRI reports 14590 and 15906. The NRC staff documented the review of these aspects of the EPRI reports in the Vogtle and Millstone SEs. The NRC staff confirmed that no CEG plant-specific aspects warranted additional consideration, noting in particular (1) the relatively low sensitivity of the EPRI results on residual stress (Table 8-12 of the EPRI reports); and (2) the small impact of clad residual stress on the PFM results. Based on this, the NRC staff finds that there is a very low probability that plant-specific aspects of other operating loads would have a significant effect on the probability of leakage or rupture beyond the studies documented in the EPRI reports.

Based on the discussion above, the NRC staff finds the treatment of other loads described in this section of the SE acceptable for the requested SG welds and NIR of CEG.

3.2.4.4 Finite Element Analysis

The NRC staff reviewed the FEA conducted in EPRI reports 14590 and 15906 and documented its review in detail in the Vogtle and Millstone SEs. The NRC staff confirmed that no CEG plant-specific aspects warranted further review. Based on this, the NRC staff determined that the pressure and thermal stresses calculated through FEA in the EPRI reports are acceptable for referencing for the requested SG welds and NIR of CEG.

3.2.5 Fracture Toughness

In EPRI reports 14590 and 15906, EPRI assumed for fracture toughness of ferritic materials an upper-shelf K_{IC} value of 200 ksi√in based on the upper-shelf fracture toughness value in the ASME Code, Section XI, A-4200. EPRI treated K_{IC} as a random parameter normal distribution with a mean value of 200 ksi√in and a standard deviation of 5 ksi√in, stating that these assumptions are consistent with the BWRVIP-108 project. Further discussion of this topic as it relates to the EPRI reports, and to plant-specific applications, is contained in the Vogtle and Millstone SEs. The NRC staff confirmed that the evaluations documented in the Vogtle and Millstone SEs apply to the CEG submittal without further plant-specific considerations. In the June 17, 2022, response to RAI 6, the licensee performed an additional plant-specific sensitivity study with a fracture toughness of 80 ksi√in and plant-specific ISI scenario to show that the 80-year probability of rupture is less than the acceptance criterion of 1×10^{-6} per year. As discussed in Section 3.2.4 of this SE, CEG meets the material criteria in the EPRI reports, and thus the NRC staff determined that the assumed fracture toughness parameters above are applicable to CEG.

Based on the discussion referenced above and the discussion in Section 3.2.4 of this SE, which confirmed that the materials are acceptable for the requested CEG SG welds and NIR, the NRC staff finds the fracture toughness models in the referenced EPRI reports acceptable for the CEG SG welds and NIR.

3.2.6 Flaw Density

In Section 8.2.2.2 of EPRI report 14590, EPRI stated that 0.001 flaw per nozzle is assumed at the NIR. The NRC staff noted in the Vogtle SE that the acceptable number of flaws in the NIR is 0.1 flaw per nozzle. Further discussion of this topic as it relates to EPRI report 14590 is contained in the Vogtle SE. The NRC staff confirmed that the evaluation documented in the

Vogtle SE applies to the CEG October 10, 2023, submittal regarding the SG NIR included in the request without further plant-specific considerations.

The flaw density in the SG welds (1.0 flaw per weld) is based on the flaw density the NRC staff determined acceptable as documented in the December 19, 2007, SE for BWRVIP-108 (ML073600374). Using this flaw density and estimated volumes of the subject SG welds, the NRC staff finds that the assumed flaw density for the SG welds is reasonable.

Based on the discussion above and the discussion in Section 3.2.4 of this SE, which confirmed that the materials and geometric criteria are acceptable for the requested SG welds and NIR of CEG, the NRC staff finds the appropriate flaw density has been considered, and therefore acceptable, for the requested SG welds and NIR of CEG.

3.2.7 Fatigue Crack Growth Rate

The NRC staff reviewed the FCG rate used in EPRI reports 14590 and 15906 and documented its review in detail in the Vogtle and Millstone SEs. The NRC staff confirmed that no plant-specific aspects of the CEG submittals warranted further review with regards to FCG rate. Therefore, the NRC staff finds that the ASME Code, Section XI, A-4300 FCG rate used in EPRI reports 14590 and 15906 is acceptable for the requested SG welds and NIR of CEG.

3.2.8 ISI Schedule and Examination Coverage

In Section 5 of Attachment 1 to both submittals, the licensee provided the ISI schedule scenarios of the subject CEG SGs. The licensee stated that the limiting scenario in both requests is preservice inspection (PSI) followed by two 10-year ISIs (PSI+10+20+50) and that this case has similar PoF results as the ISI scenario of PSI+20+40+60 analyzed in EPRI reports 14590 and 15906. The licensee indicated that some SGs of the subject CEG units have been replaced. Given the implementation of ISI in the PFM analyses in the EPRI reports, the NRC staff explained in the Vogtle and Millstone SEs that in terms of PFM modeling, ISIs with replacement would be at least as beneficial as only ISIs because replacement is essentially “repair” of a postulated flaw, while the outcomes of ISI are either repair of a postulated flaw or non-detection and growth of a postulated flaw.

The licensee also provided plant-specific ISI examination results of the requested SG welds and NIR of CEG in the following tables: Tables B1 through B12 of the October 10, 2023, submittal and Tables B1 through B17 of the October 11, 2023, submittal. The results of the examinations are either no recordable indication or the indication was acceptable per the ASME Code, Section XI flaw acceptance standards, except for the following Byron, Unit 2 welds: 2RC-01-BC/SGN-02, 2RC-01-BB/SGC-03, 2RC-01-BB/SGC-05, 2RC-01-BC/SGC-05 and 2RC-01-BD/SGC-06. In the May 20, 2022, response to RAI 5, the licensee provided additional information on the indication detected in the 2RC-01-BC/SGN-02 weld at Byron, Unit 2 that clarified its acceptability and confirmed that the indication is bounded by the postulated flaw distribution in the PFM analysis in the EPRI reports.

In the June 17, 2022, response to RAI 2, the licensee provided information on the indications in the 2RC-01-BB/SGC-03, 2RC-01-BB/SGC-05, 2RC-01-BC/SGC-05 and 2RC-01-BD/SGC-06 welds at Byron, Unit 2 that clarified acceptability of the indications. In the response the licensee also provided information on fabrication repairs and additional PFM sensitivity studies (discussed in Section 3.2.10 of this SE) of replaced/non-replaced portion of the SGs. In the September 20, 2022, response to RAI 2a, the licensee provided explanation to justify the assumption of 100 percent examination coverage during PSI examinations. In the June 17, 2022, response to RAI 3, the licensee clarified information pertaining to the plant-specific inspection history in the October 11, 2023, submittal. The NRC staff determined that there is nothing in the responses to these RAIs that would invalidate the modeling assumptions in EPRI reports 14590 and 15906 with regard to the plant-specific ISI schedule and examination coverage.

Finally, the inspection history shows that for the Calvert Cliffs B3.130 welds, the examination coverage was as low as 74 percent. The NRC staff noted that the sensitivity study in Table 8-33 of EPRI report 15906 showed that an examination coverage of 50 percent resulted in a probability of rupture that is less than the acceptance criterion of 1×10^{-6} per year for Item No. “B3.130 (CE)” that is applicable to the Calvert Cliffs B3.130 welds. Therefore, the NRC staff determined that, provided that Calvert Cliffs met the configuration, materials, and transient selection criteria discussed in Section 3.2.4.1 and 3.2.4.2 of this SE. The Calvert Cliffs examination coverage of 74 percent for some of the B3.130 welds is acceptable because the probability of rupture is less than the acceptance criterion of 1×10^{-6} per year.

Based on this discussion, the NRC staff finds the ISI scenarios of the subject CEG SG welds and NIR to be acceptable and adequately represented by the ISI scenarios analyzed in EPRI reports 14590 and 15906.

3.2.9 Other Considerations

The NRC staff reviewed the application and associated references concerning initial flaw depth and length distribution, probability of detection, models, uncertainty, convergence, flaw density, and DFM analysis. The NRC staff previously reviewed the applicable aspects of these topics as used in EPRI reports 14590 and 15906 and documented their review in detail in the Vogtle and Millstone SEs. The NRC staff confirmed that no plant-specific aspects of the submittals warranted further review. Therefore, the NRC staff finds that the submittals are acceptable with regards to these aspects as used in the EPRI reports, and thus, these aspects are acceptable for the requested SG welds and NIR of the subject CEG units.

3.2.10 PFM Results Relevant to Proposed Alternative

In Section 5 of Attachment 1 to both submittals, the licensee stated that the PFM results in EPRI reports 14590 and 15906 indicated that after a PSI followed by subsequent ISIs, no other inspections are required for up to 60 years of plant operation to meet the PoF acceptance criterion of 1×10^{-6} per year. The NRC staff did not find this conclusion acceptable since it does not account for the effect of the combination of the most significant parameters or the added uncertainty of low probability events. More significantly, the NRC staff considers this conclusion to be a solely risk-based approach inconsistent with NRC policy that calls for risk insights to be considered together with other factors rather than sole reliance on risk-based approaches. Post fabrication examinations are critical in supporting necessary performance monitoring goals including monitoring and trending; bounding uncertainties; validating/confirming analytical results; and providing timely means to identify novel and/or unexpected degradation. The NRC staff finds that the licensee must include adequate performance monitoring as part of the basis for its proposed alternative. The NRC staff evaluated the licensee’s proposed performance monitoring plan in Section 3.2.11 of this SE.

The PFM analyses in both EPRI reports investigated several ISI examination schedule scenarios, which include PSI followed by various ISI examinations. The PFM results relevant to the proposed alternatives for the CEG SGs are those resulting from the ISI scenario (i.e., PSI+20+40+60) that closely matches that of the limiting ISI scenario (i.e., PSI+10+20+50) for the CEG SGs discussed in Section 3.2.8 of this SE. The relevant PFM results show that the probability of rupture is below the acceptance criterion of 1×10^{-6} per year. In the May 20, 2022, response to RAIs 2 and 4, and the June 17, 2022, response to RAIs 2 and 9, the licensee presented additional PFM sensitivity studies that address the impact of plant-specific parameters in the limiting EPRI PFM results and lack of plant-specific examination data for some ISI intervals. The 80-year PoF results of these additional PFM sensitivity studies are all below the acceptance criterion of 1×10^{-6} per year. Based on the above and the discussions in Sections 3.2.1 through 3.2.9 of this SE, the NRC staff finds that the proposed alternatives for the CEG SG welds and NIR would result in a PoF per year that is below the acceptance criterion of 1×10^{-6} per year.

3.2.11 Performance Monitoring

Performance monitoring, such as ISI programs, is a necessary component such as described by the NRC five principles of risk-informed decision making as discussed in RG 1.174. Analyses, such as PFM, work along with performance monitoring to provide a mutually supporting and diverse basis for facility condition and maintenance that is within its licensing basis. An adequate performance monitoring program must provide direct evidence of the presence and extent of degradation, validation of continued appropriateness of associated analyses, and a timely method to detect novel/unexpected degradation. The NRC staff described these characteristics at several public meetings (ML22060A277, ML23033A667, and ML23114A034, respectively).

In Section 5 of Attachment 1 to the submittals in the section titled “Performance Monitoring Requirements,” the licensee described the performance monitoring plan for the subject alternative requests. The licensee addressed performance monitoring for the current ISI intervals and successive ISI intervals separately, the combination of which is graphically shown in Attachment 1 to the submittals and reproduced in Figure 1 of this SE.

For the current ISI intervals for Braidwood, Byron, Calvert Cliffs, and Ginna, the licensee provided a table in Attachment 1, Section 5, describing the completed examinations by unit and ASME Code Item No., reproduced here as Figure 2,

Figure 2: Licensee’s Current Interval Examinations Completion Status

Constellation Steam Generator Examination Requirements and Current Interval Completion Status																
ASME Cat.	ASME Item	BRD U1		BRD U2		BYR U1		BYR U2		CAL 1		CAL 2		GIN		
		Req.	Comp.	Req.	Comp.	Req.	Comp.	Req.	Comp.	Req.	Comp.	Req.	Comp.	Req.	Comp.	
B-B	B2.40	1	0	1	0	1	1	1	1	1	0	1	0	1	0	
B-D	B3.130	n/a	n/a	n/a	n/a	n/a	n/a	n/a	n/a	6	3	6	0	n/a	n/a	
C-A	C1.10	2	0	3	1	2	2	3	2	1	0	1	0	2	0	
C-A	C1.20	1	0	1	0	1	1	1	0	1	0	1	0	1	0	
C-A	C1.30	1	1	1	0	1	1	1	1	1	1	1	0	1	0	
C-B	C2.21	1	0	2	0	1	1	2	1	2	1	2	0	1	0	
C-B	C2.22	1	1	1	0	1	1	1	1	2	1	2	0	1	0	CEG PWR Fleet
Exams Required Per Int.		7		9		7		9		14		14		7		67
Exams Completed Current Int.		2		1		7		6		6		0		0		22
																33%

The NRC staff confirmed that for the current interval, a fleetwide sampling percentage of 33 percent has been achieved by inspecting across all pertinent ASME Code Item Nos. This sampling includes welds and NIRs at all sites except Ginna. The staff noted that performance monitoring at Ginna would depend on inspections during the previous inspection interval and inspections that have been conducted at the other subject units. This is consistent with the premise that fleet-based sampling would provide a timely method to identify novel degradation should such begin to occur. Further, the proposed applicability of the alternative to Ginna covers only the current inspection interval in contrast to the proposal for the other units. This relatively shorter period of alternative treatment reduces the uncertainty associated with long periods without plant-specific sampling.

The NRC staff reviewed the proposed performance monitoring plan in terms of SG examination set equivalents, rather than in terms of number of required Item No. examinations. This is to ensure that some sampling occurs of all Item Nos. required by ASME Code, Section XI. The ASME Code, Section XI requires that one SG (i.e., all welds, NIRs, etc. specified in the ASME Code, Section XI, for the SG) be examined per unit per ISI interval. Accordingly, Table 3 shows the NRC staff's calculation of the number of total SG examination sets required to be examined by the ASME Code, Section XI for the licensee's proposed alternative for current intervals.

Table 3: Calculation of Total ASME Code Required SG Inspections for Current Intervals

Site	# of units	# SGs required per unit	# of ISI intervals	ASME Code Required SG = units × SGs required per unit × intervals
Braidwood	2	1	1	2
Byron	2	1	1	2
Calvert Cliffs	2	1.43 [1]	1	2.86
Ginna	1	1	1	1
Total				7.86

Note:
1. This accounts for the Item No. B3.130 components in Calvert Cliffs, all of which are required to be inspected by ASME Code, Section XI.

The NRC staff determined, through binomial statistics and Monte Carlo methods, that a 25 percent sample of the total ASME Code required number of SGs would be an adequate performance monitoring sample over the subject alternative period. This leads to a required sample of 0.25 × 7.86 = 2 SGs (rounding up). The SG equivalents proposed in the licensee's performance monitoring plan for the current intervals is 2.3 (shown in Table 4), which is greater than the required sample of 2 SGs for the current intervals. Note that because ASME Code inspection requirements for B3.130 components require inspection of all B3.130 (that is, not a single SG worth) the calculation of "# SGs required per unit" in Table 3 shows that more than one SG "worth" of components is needed to be examined per interval at Calvert Cliffs (the other units do not have B3.130 components).

Table 4: Number of SG Equivalent Exams for Current Intervals

Unit	# of Section XI required exams	# of performance monitoring exams	SG Equivalents = PM exams / required exams
Braidwood 1	7	2	0.2
Braidwood 2	9	1	0.1
Byron 1	7	7	1.0
Byron 2	9	6	0.6
Calvert Cliffs 1	14	6	0.4
Total			2.3*

* While these are not true SG equivalents due to the irregular distribution of examinations by Item No. they are shown as such for demonstration purposes (see Note 1 on Table 3).

For the subsequent (i.e., successive) ISI intervals for Braidwood, Byron, and Calvert Cliffs, the licensee proposed to conduct inspections of all subject components of a single SG at Braidwood during the 5th interval, and a single SG at Byron during the 6th interval; and a prorated 25 percent sample of a SG at Calvert Cliffs, Unit 2 for the 6th interval there (the Braidwood and Byron 5th intervals overlap the Calvert Cliffs 6th interval). The Calvert Cliffs Unit 2 inspections are prorated due to the expiry of the current operating license part-way through the 6th interval.

Table 5 shows the NRC staff's calculation of the number of total SG examination sets required to be examined by the ASME Code, Section XI for the licensee's proposed alternative for subsequent intervals. Table 6 shows the licensee's proposed performance monitoring plan for the subsequent intervals.

Table 5: Calculation of Total ASME Code Required SG inspections for Subsequent Intervals

Site	# of units	# SGs required per unit	# of ISI intervals	ASME Code Required SG = units × SGs required per unit × intervals
Braidwood	2	1	2	4
Byron	2	1	2	4
Calvert Cliffs 1 & 2	2	1.43 [1]	1/2 and 2/3 respectively	1.43*(1/2 + 2/3) = 1.67
Total				9.67

Note:
1. This accounts for the Item No. B3.130 components in Calvert Cliffs, all of which are required to be inspected by ASME Code, Section XI.

Table 6: Number of SG Equivalent Exams for Subsequent Intervals

Unit	# of Section XI required exams	# of performance monitoring exams	SG Equivalents [1] = PM exams / required exams
Braidwood 1	Note 2	Note 2	1
Braidwood 2			
Byron 1	Note 3	Note 3	1
Byron 2			
Calvert Cliffs 1	Note 4	Note 4	5/14 = 0.3
Calvert Cliffs 2			
Total			2.3

Notes:
1. For Braidwood and Byron these are true SG equivalents, for Calvert Cliffs it is a prorated SG equivalent.
2. For the 5th ISI interval, the licensee will inspect either one SG from Braidwood Unit 1 or one SG from Braidwood Unit 2. See Figure 1 of this SE.
3. For the 6th ISI interval, the licensee will inspect either one SG from Byron Unit 1 or one SG from Byron Unit 2. See Figure 1 of this SE.
4. For Calvert Cliffs, the licensee stated that the performance monitoring sample size will be prorated as follows: (14 exams per unit per interval) * (2 units) * (0.25 PM sample) * (12 yr./20 yr proration) = 5 examinations. The licensee stated that the proration of (12 yr./20 yr) is to account for the current operating license of each unit expiring midway through the 6th ISI interval. The five examinations resulting from the prorated sample size accounts for both Calvert Cliffs units even though the licensee stated that the examinations will be performed at Calvert Cliffs Unit 2.

Based on the NRC staff's calculations in Tables 5 and 6, the licensee proposes to conduct 2.3 SG equivalents worth of inspections for subsequent intervals. Uniquely, Calvert Cliff units have B3.130 components. The sampling of B3.130 components will be less than the 25 percent sample discussed otherwise. This is because the ASME Code B3.130 requirements apply to all B3.130 components, and not just those in a single SG. In effect, the NRC staff noted that inspecting other component types at Calvert Cliffs will achieve more than the 25 percent sampling based on the proposed alternative. This is due to the alternative proposal including that at least one of each component type at Calvert Cliffs will be selected for examination in the 6th interval (the portion of which subject to this approval).

The NRC staff noted the relatively higher emphasis on inspection of B3.130 in the ASME Code as relating to the relatively higher safety significance of these components. However, based on the PSI and ISI inspections conducted to date, the individual uncertainties regarding these components have already been substantially reduced. The purpose of the 25 percent sampling is to appropriately manage the population level uncertainties of all the components in SGs. Consequently, it is appropriate to emphasize a better overall coverage of the component. That is, instead of focusing more of the 25 percent sampling on just B3.130 components, the licensee proposed to evenly sample the broad set of SG component types, locations, conditions, etc. The NRC staff finds that this will provide a stronger basis of assurance that no incipient novel degradation is occurring (and its timely detection if it does begin to occur).

Based on the above, a 25 percent sample for the units subject to subsequent ISI intervals under the alternative would require 0.25 × 9.67 = 2.4 SG equivalents (not rounded due to partial 6th interval of applicability at Calvert Cliffs). The proposed inspections of 2.3 SG equivalents in Table 6 constitute a nearly equivalent sampling meeting the 25 percent sample.

Given the unique situation with Calvert Cliffs discussed above, the NRC staff finds that the proposed sampling meets the necessary quantity of acceptable performance monitoring for the current and subsequent inspection intervals subject to this alternative because a statistically sufficient quantity of examinations will be performed.

The NRC staff reviewed the timing of examinations to ensure that the proposed examinations in the performance monitoring plan would provide a reasonably continuous source of data supporting the characteristics of acceptable performance monitoring. Specifically, data would continue to become available on a cadence reasonably commensurate with ASME Code requirements, but on a fleet basis rather than an individual unit basis. This is based on the alternative including inspections for all ASME Code Item Nos. and distributing the proposed examinations across all subject periods per ASME Code, Section XI, Tables IWB-2411-1 or Table IWC-2411-1 for the subsequent intervals (e.g., other than the current intervals). Based on the proposed examinations during the alternative periods the NRC staff finds that the examinations proposed in the performance monitoring plan will provide an acceptably continuous stream of data because the inspections will be spread out by interval and period.

As part of the proposed performance monitoring plan in both submittals, the licensee described actions they would take in the event that new unacceptable indications are identified as part of performance monitoring activities. The licensee stated that detected indications would be evaluated according to the rules of ASME Code, Section XI (which include additional examination and successive inspection requirements), and the CEG corrective action program. These additional activities are described in detail in pages 13 to 15 of Attachment 1 to the October 10, 2023, submittal and in pages 17 to 19 of Attachment 1 to the October 11, 2023, submittal.

In addition, the licensee described system leakage tests as “providing further assurance of safety” for the proposed alternatives. The NRC staff noted that the visual examinations performed during system leakage tests may not directly detect the 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 and nozzles. However, the NRC staff noted that leakage tests provide complementary additional performance monitoring to the ISI examinations. This additional assurance increases confidence that the proposed quantity of examinations, in concert with other on-going activities, will provide an acceptable level of performance monitoring for the subject SG components.

Based on the above discussion, the NRC staff determined that inspections for the subject SG components could be deferred during the proposed period because an adequate level of performance monitoring is maintained for the components.

4.0 CONCLUSION

As set forth above, the NRC staff determined that the licensee’s proposed alternative for the requested SG components provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the use of proposed alternatives I4R-16 and I4R-17, Revision 1 for Braidwood; I4R-22 and I4R-23, Revision 1 for Byron; ISI-05-017 and ISI-05-018, Revision 1 for Calvert Cliffs; and I6R-09 and I6R-10, Revision 1 for Ginna, for the remainder of their currently approved operating license, currently scheduled to end on October 17, 2046 (Braidwood, Unit 1), December 18, 2047 (Braidwood, Unit 2), October 31, 2044 (Byron, Unit 1), November 6, 2046 (Byron, Unit 2), July 31, 2034 (Calvert Cliffs, Unit 1), August 13, 2036 (Calvert Cliffs, Unit 2), and September 18, 2029 (Ginna).

All other ASME Code, Section XI requirements for which relief has not been specifically requested and approved in this relief request remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

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Date: July 23, 2024

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2; BYRON STATION, UNIT NOS. 1 AND 2; CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2; AND R. E. GINNA NUCLEAR POWER PLANT - ISSUANCE OF RELIEF RE: PROPOSED ALTERNATIVE REQUEST ASSOCIATED WITH STEAM GENERATOR EXAMINATIONS (EPIDS L-2023-LLR-0053, L-2023-LLR-0054, L-2023-LLR-0055, L-2023-LLR-0056) DATED JULY 23, 2024

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DATE	06/27/2024	06/28/2024	06/20/2024
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NAME	JWhitted	SWall	
DATE	07/23/2024	07/23/2024	

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