Exelon Generation

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10 CFR 50.55a

RS-21-093

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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: Proposed Alternative for Examinations of Examination Category C-B Steam Generator Nozzle-to-Shell Welds and Nozzle Inside Radius Sections

In accordance with 10 CFR 50.55a(z)(1), Exelon Generation Company, LLC (Exelon) hereby requests 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 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.

The proposed alternative is to extend the frequency of steam generator nozzle-to-shell and nozzle inside radius sections volumetric and surface examinations for the remainder of the currently licensed operating periods for Braidwood, Units 1 and 2; Byron, Units 1 and 2; Calvert Cliffs, Units 1 and 2; and Ginna. Exelon requests authorization to use the proposed alternative pursuant to 10 CFR 50.55a(z)(1) on the basis that the alternative provides an acceptable level of quality and safety.

NRC review and approval of the proposed alternative is requested by August 31, 2022, to support the Fall 2022 outage season when the subject examinations are currently scheduled.

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
Exelon Generation Company, LLC

Attachment 1: 10 CFR 50.55a Proposed Alternative I4R-16 for Braidwood Station, Units 1 and 2; Proposed Alternative I4R-22 for Byron Station, Units 1 and 2; Proposed Alternative ISI-05-017 for Calvert Cliffs Nuclear Power Plant, Units 1 and 2; and Proposed Alternative I6R-09 for Ginna, Revision 0

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

Braidwood Station
Byron Station
Calvert Cliffs
Ginna

10 CFR 50.55a Proposed Alternative I4R-16 for Braidwood Station, Units 1 and 2,
Proposed Alternative I4R-22 for Byron Station, Units 1 and 2,
Proposed Alternative ISI-05-017 for Calvert Cliffs Nuclear Power Plant, Units 1 and 2,
and Proposed Alternative I6R-09 for Ginna, Revision 0

Proposed Alternative for Examinations of Nozzle-to-Shell Welds and Nozzle Inside Radius Sections Examinations (Examination Category C-B) in Accordance with 10 CFR 50.55a(z)(1)

1 ASME CODE COMPONENTS AFFECTED:

Code Class: Class 2
Description: Steam Generator (SG) Nozzle-to-Shell Welds and Nozzle Inside Radius Sections
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
C2.22 - Nozzle Inside Radius Section

Braidwood Component IDs:

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
2	A	2SG-01-SGN-02	C2.21	Feedwater Nozzle
2	A	2SG-01-SGN-02 (NIR)	C2.22	Feedwater Nozzle Inner Radius
2	A	2SG-01-SGN-03	C2.21	Main Steam Nozzle
2	B	2SG-02-SGN-02	C2.21	Feedwater Nozzle
2	B	2SG-02-SGN-02 (NIR)	C2.22	Feedwater Nozzle Inner Radius
2	B	2SG-02-SGN-03	C2.21	Main Steam Nozzle
2	C	2SG-03-SGN-02	C2.21	Feedwater Nozzle
2	C	2SG-03-SGN-02 (NIR)	C2.22	Feedwater Nozzle Inner Radius
2	C	2SG-03-SGN-03	C2.21	Main Steam Nozzle
2	D	2SG-04-SGN-02	C2.21	Feedwater Nozzle
2	D	2SG-04-SGN-02 (NIR)	C2.22	Feedwater Nozzle Inner Radius
2	D	2SG-04-SGN-03	C2.21	Main Steam Nozzle

*Differences in number of components between Unit 1 and Unit 2 is due to the replacement of the Unit 1 Steam Generators during the 1998 outage.

Byron Component IDs:

Unit	SG	Component ID	Item Number	Description
1	A	1RC-01-BA/N-3	C2.21	Feedwater Nozzle
1	A	1RC-01-BA/N-3-NIR	C2.22	Feedwater Nozzle Inner Radius
1	B	1RC-01-BB/N-3	C2.21	Feedwater Nozzle
1	B	1RC-01-BB/N-3-NIR	C2.22	Feedwater Nozzle Inner Radius
1	C	1RC-01-BC/N-3	C2.21	Feedwater Nozzle
1	C	1RC-01-BC/N-3-NIR	C2.22	Feedwater Nozzle Inner Radius
1	D	1RC-01-BD/N-3	C2.21	Feedwater Nozzle
1	D	1RC-01-BD/N-3-NIR	C2.22	Feedwater Nozzle Inner Radius
2	A	2RC-01-BA/SGN-02	C2.21	Feedwater Nozzle
2	A	2RC-01-BA/SGN-02-NIR	C2.22	Feedwater Nozzle Inner Radius
2	A	2RC-01-BA/SGN-03	C2.21	Main Steam Nozzle
2	B	2RC-01-BB/SGN-02	C2.21	Feedwater Nozzle
2	B	2RC-01-BB/SGN-02-NIR	C2.22	Feedwater Nozzle Inner Radius
2	B	2RC-01-BB/SGN-03	C2.21	Main Steam Nozzle
2	C	2RC-01-BC/SGN-02	C2.21	Feedwater Nozzle
2	C	2RC-01-BC/SGN-02-NIR	C2.22	Feedwater Nozzle Inner Radius
2	C	2RC-01-BC/SGN-03	C2.21	Main Steam Nozzle
2	D	2RC-01-BD/SGN-02	C2.21	Feedwater Nozzle
2	D	2RC-01-BD/SGN-02-NIR	C2.22	Feedwater Nozzle Inner Radius
2	D	2RC-01-BD/SGN-03	C2.21	Main Steam Nozzle

*Differences in number of components between Unit 1 and Unit 2 is due to the replacement of the Unit 1 Steam Generators during the 1997 outage.

Calvert Cliffs Component IDs:

Unit	SG	Component ID	Item Number	Description
1	11	SG-11-FW	C2.21	Feedwater Nozzle
1	11	SG-11-FW-IRS	C2.22	Feedwater Nozzle Inner Radius
1	11	SG-11-MS	C2.21	Main Steam Nozzle
1	11	SG-11-MS-IRS	C2.22	Main Steam Nozzle Inner Radius
1	12	SG-12-FW	C2.21	Feedwater Nozzle
1	12	SG-12-FW-IRS	C2.22	Feedwater Nozzle Inner Radius
1	12	SG-12-MS	C2.21	Main Steam Nozzle
1	12	SG-12-MS-IRS	C2.22	Main Steam Nozzle Inner Radius
2	21	SG-21-FW	C2.21	Feedwater Nozzle
2	21	SG-21-FW-IRS	C2.22	Feedwater Nozzle Inner Radius
2	21	SG-21-MS	C2.21	Main Steam Nozzle
2	21	SG-21-MS-IRS	C2.22	Main Steam Nozzle Inner Radius
2	22	SG-22-FW	C2.21	Feedwater Nozzle
2	22	SG-22-FW-IRS	C2.22	Feedwater Nozzle Inner Radius
2	22	SG-22-MS	C2.21	Main Steam Nozzle
2	22	SG-22-MS-IRS	C2.22	Main Steam Nozzle Inner Radius

* Partial replacements of the Unit 1 Steam Generators were performed during the 2002 outage. Partial replacement of the Unit 2 Steam Generators was performed during the 2003 outage. The steam drum, which includes the main steam and feedwater nozzles, was retained and refurbished as part of the steam generator replacements. Therefore, the nozzle to vessel and inside radius welds were retained, as well. Examination records prior the Steam Generator replacements could not be located; however, two inspection intervals of examinations have been performed since the Steam Generators' replacements with no unacceptable indications identified.

Ginna Component IDs:

Unit	SG	Component ID	Item Number	Description
1	A	FWIN-AR	C2.21	Feedwater Nozzle
1	A	FWIN-AR	C2.22	Feedwater Nozzle Inner Radius
1	B	FWIN-BR	C2.21	Feedwater Nozzle
1	B	FWIN-BR	C2.22	Feedwater Nozzle Inner Radius

*Replacement of the Unit 1 Steam Generators were performed during the 1996 outage.

2 APPLICABLE CODE EDITION AND ADDENDA:

The following table identifies the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Section XI Code of Record for performing Inservice Inspection (ISI) activities at Braidwood, Byron, Calvert Cliffs, and Ginna:

Plant	Interval	Current Edition and Addenda	Interval Start	Interval End
Braidwood Station, Units 1 and 2	Fourth	2013 Edition	August 29, 2018 (Unit 1) November 5, 2018 (Unit 2)	July 28, 2028 (Unit 1) October 16, 2028 (Unit 2)

Plant	Interval	Current Edition and Addenda	Interval Start	Interval End
Byron Station, Units 1 and 2	Fourth	2007 Edition with the 2008 Addenda	July 16, 2016	July 15, 2025
Calvert Cliffs Nuclear Power Plant, Units 1 and 2	Fifth	2013 Edition	July 1, 2019	June 30, 2029
R.E. Ginna Nuclear Power Plant	Sixth	2013 Edition	January 1, 2020	December 31, 2029

The 2019 Edition of ASME Section XI, Table G-2110-1 will be utilized to extend the use of Figure G-2110-1, Reference Critical Stress Intensity Factor for Material, to material SA-533, Grade A, Class 2. (Note: The 2019 Edition of ASME Section XI is approved in the proposed rules of the Federal Registry, Vol 86, No. 57 [14]. Currently there are conditions in the proposed NRC Rulemaking (86 FR 16087) regarding Table G-2110-1, but these conditions do not pertain to the use of Table G-2110-1 for material SA-533, Grade A, Class 2 as used in this proposed alternative.)

3 APPLICABLE CODE REQUIREMENTS:

Table IWC-2500-1 of the 2007 Edition with the 2008 Addenda and 2013 Edition of ASME Code, Section XI (ASME Section XI), provides requirements and acceptance standards for examining Class 2 nozzle-to-shell welds and nozzle inside radius sections. Table IWC-2500-1, Examination Category C-B, Item Number C2.21 requires surface and volumetric examination of all nozzles of a representative steam generator at terminal ends of piping runs once during each Section XI inspection interval. Table IWC-2500-1, Examination Category C-B, Item Number C2.22 requires volumetric examination of all nozzle inside radius sections of a representative steam generator at terminal ends of piping runs once during each Section XI inspection interval. The examination volumes and surfaces for Item Numbers C2.21 and C2.22 are shown in Figures IWC-2500-4(a), (b), and (d).

4 REASON FOR REQUEST:

The Electric Power Research Institute (EPRI) performed a technical evaluation in Reference [1] of the basis for the ASME Section XI examination requirements specified for Examination Category C-B of ASME Section XI for Steam Generator (SG) Main Steam (MS) and Feedwater (FW) Nozzle-to-Shell Welds and Nozzle Inside Radius Sections. The evaluation includes a survey of inspection results from 74 units as well as flaw tolerance evaluations using probabilistic fracture mechanics (PFM) and deterministic fracture mechanics (DFM). The Reference [1] report concluded that the current ASME Section XI inspection interval of ten years can be increased significantly with no impact to plant safety. It is upon the basis of this conclusion that an alternate inspection interval is being requested. The Reference [1] report was developed consistent with the recommendations provided in EPRI's White Paper on PFM [2].

5 PROPOSED ALTERNATIVE AND BASIS FOR USE:

Exelon Generation Company, LLC (Exelon) is requesting an inspection alternative to the examination requirements of ASME Section XI, Table IWC-2500-1, Examination Category C-B, Item Numbers C2.21 and C2.22. The proposed alternative is to defer the inspection of these examination items from the current ASME Section XI 10-year requirement to the end of the currently approved Period of Extended Operation (PEO) for Braidwood Station (Braidwood), Units 1 and 2, Byron Station (Byron), Units 1 and 2, Calvert Cliffs Nuclear Power Plant (Calvert Cliffs), Units 1 and 2, and Ginna, as summarized in the following table.

Table 1. Summary of Inspection Deferrals in this Proposed Alternative

Station	Unit	ASME Category	Item No.	Description	Date of Last Inspection	End of Current Licensed Operating Period (60 Years)	Length of Time Until Next Inspection for This Request (Years)
Braidwood	1	C-B	C2.21	FW Nozzle-to-Shell Weld	4/8/2015	10/17/2046	31.5
		C-B	C2.22	FW Nozzle Inside Radius Section	10/10/2010		36.0
Braidwood	2	C-B	C2.21	MS Nozzle-to-Head Weld	5/10/2014	12/18/2047	33.6
		C-B	C2.21	FW Nozzle-to-Shell Weld	5/30/2017		30.6
		C-B	C2.22	FW Nozzle Inside Radius Section	5/30/2017		30.6
Byron	1	C-B	C2.21	FW Nozzle-to-Shell Weld	3/19/2014	10/31/2044	30.6
		C-B	C2.22	FW Nozzle Inside Radius Section	3/5/2017		27.7
Byron	2	C-B	C2.21	MS Nozzle-to-Head Weld	4/20/2013	11/6/2046	33.6
		C-B	C2.21	FW Nozzle-to-Shell Weld	4/12/2019		27.6
		C-B	C2.22	FW Nozzle Inside Radius Section	4/12/2019		27.6
Calvert Cliffs	1	C-B	C2.21	MS Nozzle-to-Head Weld	3/3/2018	7/31/2034	16.3
		C-B	C2.22	MS Nozzle Inside Radius Section	3/3/2018		16.3
		C-B	C2.21	FW Nozzle-to-Shell Weld	2/22/2016		18.4
		C-B	C2.22	FW Nozzle Inside Radius Section	2/22/2016		18.4
Calvert Cliffs	2	C-B	C2.21	MS Nozzle-to-Head Weld	2/25/2019	8/13/2036	17.5
		C-B	C2.22	MS Nozzle Inside Radius Section	2/25/2019		17.5
		C-B	C2.21	FW Nozzle-to-Shell Weld	2/23/2015		21.5
		C-B	C2.22	FW Nozzle Inside Radius Section	2/24/2015		21.5
Ginna	1	C-B	C2.21	FW Nozzle-to-Shell Weld	4/30/2017	9/18/2029	12.4
		C-B	C2.22	FW Nozzle Inside Radius Section	4/30/2017		12.4

As indicated in Table 1, the proposed alternative results in a maximum effective operating period of 36.0 years from the last inspection for Item Number C2.22 for Braidwood Unit 1 included in this Proposed Alternative. As summarized in Attachment 1, the EPRI report demonstrates that after PSI and two 10-year interval inspections, the failure probabilities (for both rupture and leakage) are significantly below the acceptance criteria of $1x10^{-6}$ failures per year for time periods exceeding 36 years of plant operation.

The key aspects in the technical basis for this request are summarized below. The applicability of the technical basis for Braidwood, Units 1 and 2, Byron, Units 1 and 2, Calvert Cliffs, Units 1 and 2, and Ginna, is shown in Appendix A.

Degradation Mechanism Evaluation

An evaluation of degradation mechanisms that could potentially impact the reliability of the SG MS and FW Nozzle-to-Shell Welds and Nozzle Inside Radius Sections was performed in Reference [1]. Evaluated mechanisms include stress corrosion cracking (SCC), environmental assisted fatigue (EAF), microbiologically influenced corrosion (MIC), pitting, crevice corrosion, erosion-cavitation, erosion, flow accelerated corrosion (FAC), general corrosion, galvanic corrosion, and mechanical/thermal fatigue. Other than the potential for EAF and mechanical/thermal fatigue, there were no active degradation mechanisms identified that significantly affect the long-term structural integrity of the SG MS and FW nozzles.

Stress Analysis

Finite element analysis (FEA) was performed in Reference [1] to determine the stresses in the SG MS and FW Nozzle-to-Shell Welds and Nozzle Inside Radius Sections. The analysis was performed using representative pressurized water reactor (PWR) geometries, bounding transients, and typical material properties. The results of the stress analyses were used in a flaw tolerance evaluation. The applicability of the FEA analysis to Braidwood, Units 1 and 2, Byron, Units 1 and 2, Calvert Cliffs, Units 1 and 2, and Ginna, are shown in Appendix A and confirms that all plant-specific requirements are met. Therefore, the evaluation results and conclusions of Reference [1] are applicable to Braidwood, Units 1 and 2, Byron, Units 1 and 2, Calvert Cliffs, Units 1 and 2, and Ginna.

Flaw Tolerance Evaluation

Flaw tolerance evaluations were performed in Reference [1] consisting of PFM evaluations and confirmatory DFM evaluations. The results of the PFM analyses indicate that, after the preservice inspection (PSI) and two 10-year interval inspections, no other inspections are required for up to 60 years of plant operation to meet the U.S. Nuclear Regulatory Commission's (NRC's) safety goal of 10^{-6} failures per year. For the specific case of Braidwood and Byron, Unit 1, PSI has been followed by the performance of two 10-year interval inspections. For Braidwood and Byron, Unit 2, PSI has been followed by the performance of three 10-year interval inspections. For Calvert Cliffs, Units 1 and 2, PSI has been followed by the performance of four 10-year inspection intervals, but records were only found for the Third and Fourth Interval examinations. For Ginna, PSI has been followed by the performance of two 10-year interval inspections.

The most limiting case requested within this proposed alternative, analyzed by the EPRI report, is the scenario of PSI followed by two years of inservice inspections, followed by a 30-year inspection (PSI + 10 + 20 + 50). The EPRI report does not contain Probability of Failure results for the limiting case “PSI + 10 + 20 + 50”, but the base case scenario “PSI + 20 + 40 + 60” has similar Probability of Failure results, as shown in Figure 8-9 and Table 8-10 of the EPRI report. The Probability of Failure results are larger for the limiting “PSI + 10 + 20 + 50” case than the base case scenario “PSI + 20 + 40 + 60”, but they are of the same magnitude. The Probability of Failure results in Table 8-8 of the EPRI report for “PSI + 20 + 40 + 60” show a large margin from the criterion of $1x10^{-6}$ per year. The Probability of Failures listed in Table 8-10 only address the base case scenario and it does not address the uncertainties that are addressed in the sensitivity studies in Section 8.2.4.3 of the EPRI report. Therefore, results of the sensitivity studies should be applied to the limiting case “PSI + 10 + 20 + 50” requested in this proposed alternative.

A sensitivity study was performed on the effects of three key parameters used in the EPRI report. For the sensitivity study, the mean fracture toughness was changed to 200 ksi√in. with a conservative standard deviation of 30 ksi√in., the stress multiplier was changed to 1.5, and the nozzle inner radius flaw density was changed to 0.1. The limiting case for this sensitivity study was found to be the nozzle inner radius location, with a probability of rupture of $5.3x10^{-6}$, for the base case scenario “PSI + 20 + 40 + 60” (which is similar to the limiting case “PSI + 10 + 20 + 50”). All other locations reviewed by the sensitivity study were below the $1x10^{-6}$ per year acceptance criteria.

Another sensitivity study was performed on stress within the EPRI report. Table 8-15 shows that the probabilities of rupture are below the criterion of $1x10^{-6}$ per year, for all stress multipliers, even when considering the 0.1 flaw per nozzle at the FW nozzle NIR. Table 8-16 shows the limiting probability of leakage of $1.04x10^{-6}$ per year at the FW nozzle FEW-P3A. Even though the probability is greater than $1x10^{-6}$ per year, this is a probability of leakage and not a probability of rupture. Additionally, the results for the FW nozzle NIR are slightly conservative because the applied loads include the effect of welding residual stress.

The sensitivity study results in the EPRI report are assuming 80 years of operation, but this proposed alternative only requests extension of the ISI examinations for no more than 60 years of operation. If the PFM analysis was performed for only 60 years of operation, the probability of failure values would be lower than the determined values.

The DFM evaluations confirm the PFM results by demonstrating that it takes approximately 80 years for a postulated flaw with an initial depth equal to the ASME Section XI acceptance standards to grow to a depth where the maximum stress intensity factor (K) exceeds the ASME Section XI allowable fracture toughness.

Based on the information provided, the proposed alternative of PSI + 10 + 20 + 50 for the requested SG nozzles for the plants request, would result in a Probability of Failure per year that is reasonably below the criterion of $1x10^{-6}$ per year.

Inspection History

Braidwood, Units 1 and 2, Byron, Units 1 and 2, Calvert Cliffs, Units 1 and 2, and Ginna operating experience (including examinations performed to date, examination findings, inspection coverage, and Request for Alternatives) are presented in Appendix B. As shown in Appendix B, Byron recorded one examination with limited coverage for the applicable components. The lowest recorded coverage for Byron was 90% for weld 2RC-01-BC/SGN-02, Item Number C2.21, Nozzle-to-Shell Weld. Based on the information provided in the previous section, no other inspections are required for the applicable components, for the remainder of the current life of the plants requested, due to the relatively low failure probabilities for rupture and leakage. Also as shown in Appendix B, Byron was the only plant to identify recordable indications that exceeded the ASME Section XI acceptance standards. Weld 2RC-01-BC/SGN-02 was found to have one indication during PSI that exceeded the ASME Section XI acceptance standards. Relief to accept these flaws was approved through an NRC Safety Evaluation on 10/29/1986 [17]. Successive examinations were performed for the weld and the indication was found to have little or no change in sizing.

Industry inspection history for the applicable components (as obtained from the industry survey summarized in Reference [1]) is presented in Appendix C. The results of the survey in Reference [1] indicate that these components are very flaw tolerant.

Conclusion

It is concluded that the SG MS and FW Nozzle-to-Shell Welds and Nozzle Inside Radius Sections are very flaw tolerant. PFM and DFM evaluations performed as part of the technical basis in Reference [1] demonstrate that, after PSI followed by two 10-year intervals, no other inspection is required until 60 years to meet the NRC safety goal of $1x10^{-6}$ failures per reactor year. Plant-specific applicability of the technical basis to Braidwood, Units 1 and 2, Byron, Units 1 and 2, Calvert Cliffs, Units 1 and 2, and Ginna are demonstrated in Appendix A. While the technical bases demonstrate longer inspection intervals are possible, Exelon considers that deferral of these inspections until the end of the currently approved PEO, as shown in Table 1, provides an acceptable level of quality and safety in lieu of the current ASME Examination Category C-B, Item Numbers C2.21 and C2.22 surface and volumetric examination 10-year inspection frequency.

Operating and examination experience demonstrates that these components have performed with very high reliability, mainly due to their robust design. As shown in Appendix B, to date, Exelon has performed over 50 inspections of SG MS and FW Nozzle-to-Shell Welds and Nozzle Inside Radius Sections at Braidwood, Units 1 and 2, Byron, Units 1 and 2, Calvert Cliffs, Units 1 and 2, and Ginna. Byron weld 2RC-01-BC/SGN-02 was found to have one flaw exceeding the ASME Section XI acceptance standards. It was a fabrication flaw detected during PSI and is documented in Appendix

B. Subsequent examinations have been performed for the recordable indication and there has been no increase in flaw sizing. The lowest inspection coverage listed in Appendix B was 90%. Section 8.2.5 of Reference [1] discusses limited coverage and determines that the conclusions of the report are applicable to components with limited coverage. In addition, it is important to note all other inspection activities, including the system leakage test (Examination Category C-H) conducted each ISI period will continue to be performed providing further assurance of safety.

Finally, as discussed in Reference [3], for situations where no active degradation mechanism is present, it was concluded that subsequent inservice inspections do not provide additional value after PSI has been performed and the inspection volumes have been confirmed to have no flaws that exceeded the ASME Section XI acceptance standards. The Exelon Steam Generator MS and FW nozzles have received the required PSI examinations and 50 subsequent inservice inspections with one flaw that exceeded the ASME Section XI acceptance standards. The one flaw that exceeded the ASME Section XI acceptance standard is shown in Appendix B and successive examination has been performed for the recordable indication with no increase in size.

Therefore, Exelon requests that the NRC authorize this proposed alternative in accordance with 10 CFR 50.55a(z)(1).

6 <u>DURATION OF PROPOSED ALTERNATIVE:</u>

The proposed alternative is 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), as summarized in Table 1.

7 <u>PRECEDENT:</u>

To-date, there has been two previous approved alternatives that requested relief from the ASME Examination Category C-B, Item Numbers C2.21 and C2.22 surface and volumetric examinations on the basis of the Reference [1] technical basis:

- Southern Nuclear Letter No. NL-20-1011, “Vogtle Electric Generating Plant, Units 1 & 2, Proposed Inservice Inspection Alternative VEGP-ISI-ALT-04-04 Version 2.0,” September 9, 2020, ADAMS Accession No. ML20253A311.

- U.S. NRC, “Vogtle Electric Generating Plant, Units 1 and 2 - Relief Request for Proposed Inservice Inspection Alternative VEGP-ISI-ALT-04-04 to the Requirements of the ASME Code (EPID L-2020-LLR-0109),” January 11, 2021, ADAMS Accession No. ML20352A155.

- Dominion Energy Letter No. 20-167, “Dominion Energy Nuclear Connecticut, Inc. Millstone Power Station Unit 2 Alternative Request RR-05-06 – Inspection Interval Extension for Steam Generator Pressure-Retaining Welds and Full-

Penetration Welded Nozzles," July 15, 2020, ADAMS Accession No. ML20198M682.

- Letter from James G. Danna, Chief (NRC) to Daniel G. Stoddard (Dominion Nuclear), "Millstone Power Station Unit 2 – Authorization and Safety Evaluation for Alternative Request No. RR-05-06 (EPID L-2020-LLR-0097)," July 16, 2021, ADAMS Accession No. ML21167A355.

The following is a list of approved Relief Requests related to inspections of SG MS and FW nozzles:

- Letter from J. W. Clifford (NRC) to S. E. Scace (Northeast Nuclear Energy Company), "Safety Evaluation of the Relief Requests Associated with the First and Second 10-Year Interval of the Inservice Inspection (ISI) Plan, Millstone Nuclear Power Station, Unit 3 (TAC No. MA 5446)," dated July 24, 2000, ADAMS Accession No. ML003730922.

- Letter from R. L. Emch (NRC) to J. B. Beasley, Jr. (SNOC), "Second 10-Year Interval Inservice Inspection Program Plan Requests for Relief 13, 14, 15, 21 and 33 for Vogtle Electric Generating Plant, Units 1 and 2 (TAC No. MB0603 and MB0604)," dated June 20, 2001, ADAMS Accession No. ML011640178.

- Letter from T. H. Boyce (NRC) to C. L. Burton (CP&L), "Shearon Harris Nuclear Power Plant Unit 1 – Request for Relief 2R1-019, 2R1-020, 2R1-021, 2R1-022, 2R2-009, 2R2-010, 2R2-011 for the Second Ten-Year Interval Inservice Inspection Program Plan (TAC Nos. ME0609, ME0610, ME0611, ME0612, ME0613, ME0614 and ME0615)," dated January 7, 2010, ADAMS Accession No. ML093561419.

- Letter from M, Khanna (NRC) to D. A. Heacock (Dominion Nuclear Connecticut Inc.), Millstone Power Plant Unit No. 2 – Issuance of Relief Requests RR-89-69 Through RR-89-78 Regarding Third 10-Year Interval Inservice Inspection Program Plan (TAC Nos. ME5998 Through ME6006)," dated March 12, 2012, ADAMS Accession No. ML120541062.

- Letter from R. J. Pascarelli (NRC) to E. D. Halpin (PG&E), "Diablo Canyon Power Plant, Units 1 and 2 – Relief Request NDE-SG-MS-IR, Main Steam Nozzle Inner Radius Examination Impracticality, Third 10-Year Interval, American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Inservice Inspection Program (CAC Nos. MF6646 and MF6647)," dated December 8, 2015, ADAMS Accession No. ML15337A021.

- Letter from Leonard N Olshan (NRC) to Dennis L Farrar (Commonwealth Edison Company), "Approval of Byron 2 Preservice Inspection Program," dated October 29, 1986, 8611070024.

In addition, there are precedents related to similar requests for relief for Class 1 nozzles:

- Based on studies presented in Reference [4], the NRC approved extending PWR reactor vessel nozzle-to-shell welds from 10 to 20 years in Reference [5].

- Based on work performed in BWRVIP-108 [6] and BWRVIP-241 [7], the NRC approved the reduction of BWR vessel feedwater nozzle-to-shell weld examinations (Item Number B3.90 for BWRs from 100% to a 25% sample of each nozzle type every 10 years) in References [8] and [9]. The work performed in BWRVIP-108 and BWRVIP-241 provided the technical basis for ASME Code Case N-702 [10], which has been conditionally approved by the NRC in Revision 19 of Regulatory Guide 1.147 [11].

Finally, there are precedents that used generic industry guidance in a similar approach to the approach requested in this submittal:

- Based on EPRI generic analysis, Exelon plants requested an alternative to the Reactor Pressure Vessel Threads in Flange examination requirements of ASME Section XI in Reference [12].

- NRC approval was granted for the Exelon request for alternative to the Reactor Pressure Vessel Threads in Flange examination requirements in the Reference [13] Safety Evaluation.

8 ACRONYMS:

ASME	American Society of Mechanical Engineers
B&W	Babcock and Wilcox
BWR	Boiling Water Reactor
BWRVIP	Boiling Water Reactor Vessel and Internals Program
CE	Combustion Engineering
CFR	Code of Federal Regulations
DFM	Deterministic fracture mechanics
EAF	Environmentally assisted fatigue
EPRI	Electric Power Research Institute
FAC	Flow accelerated corrosion
FEA	Finite element analysis
FW	Feedwater
ISI	Inservice Inspection
MIC	Microbiologically influenced corrosion
MS	Main Steam
NPS	Nominal pipe size
NRC	Nuclear Regulatory Commission
NSSS	Nuclear steam supply system
PFM	Probabilistic fracture mechanics
PWR	Pressurized Water Reactor
SCC	Stress corrosion cracking
SG	Steam Generator

9 REFERENCES:

1. *Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Inside Radius Sections*. EPRI, Palo Alto, CA: 2019. 3002014590, ADAMS Accession No. ML19347B107.
2. N. Palm (EPRI), BWR Vessel & Internals Project (BWRVIP) Memo No. 2019-016, “White Paper on Suggested Content for PFM Submittals to the NRC,” February 27, 2019, ADAMS Accession No. ML19241A545.
3. American Society of Mechanical Engineers, Risk-Based Inspection: Development of Guidelines, Volume 2-Part 1 and Volume 2-Part 2, Light Water Reactor (LWR) Nuclear Power Plant Components. CRTD-Vols. 20-2 and 20-4, ASME Research Task Force on Risk-Based Inspection Guidelines, Washington, D.C., 1992 and 1998.
4. B. A. Bishop, C. Boggess, N. Palm, “Risk-Informed extension of the Reactor Vessel In-Service Inspection Interval,” WCAP-16168-NP-A, Rev. 3, October 2011.
5. U.S. NRC, “Revised Final Safety Evaluation by the Office of Nuclear Reactor Regulation; Topical Report WCAP-16168-NP-A, Revision 2, ‘Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval,’ Pressurized Water Reactor Owners Group, Project No. 694,” July 26, 2011, ADAMS Accession No. ML111600303.
6. BWRVIP-108: BWR Vessels and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Shell Welds and Nozzle Blend Radii, EPRI, Palo Alto, CA 2002. 1003557.
7. BWRVIP-241: BWR Vessels and Internals Project, Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Shell Welds and Nozzle Blend Radii, EPRI, Palo Alto, CA 2010. 1021005.
8. U.S. NRC, Safety Evaluation of Proprietary EPRI Report, “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 Inner Radius (BWRVIP-108),” December 19, 2007, ADAMS Accession No. ML073600374.
9. U.S. NRC, Safety Evaluation of Proprietary EPRI Report, “BWR Vessel and Internals Project, Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Shell Welds and Nozzle Blend Radii (BWRVIP-241),” April 19, 2013, ADAMS Accession Nos. ML13071A240 and ML13071A233.
10. Code Case N-702, “Alternate Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds,” ASME Code, Section XI, Division 1, Approval Date: February 20, 2004.
11. U.S. NRC Regulatory Guide 1.147, Revision 19, “Inservice Inspection Code Case Acceptability, ASME Code Section XI, Division 1,” October 2019.
12. Exelon Generation Company, LLC, Letter RS-16-142, “Proposed Alternative for Examination of ASME Section XI, Examination Category B-G-1, Item Number

B6.40, Threads in Flange," October 31, 2016, ADAMS Accession No. ML16306A270.

13. David J. Wrona (NRC) to Bryan C. Hanson (Exelon), "Braidwood Station, Units 1 and 2; Byron Station, Unit Nos. 1 and 2; Calvert Cliffs Nuclear Power Plant, Units 1 and 2; Clinton Power Station, Unit No. 1; Dresden Nuclear Power Station, Units 2 and 3; Limerick Generating Station, Units 1 and 2; Nine Mile Point Nuclear Station, Units 1 and 2; Peach Bottom Atomic Power Station, Units 2 and 3; Quad Cities Nuclear Power Station, Units 1 and 2; R. E. Ginna Nuclear Power Plant; and Three Mile Island Nuclear Station, Unit 1 - Proposed Alternative to Eliminate Examination of Threads in Reactor Pressure Vessel Flange (CAC Nos. MF8712-MF8729 and MF9548)," June 26, 2017, ADAMS Accession No. ML17170A013.
14. U.S. NRC Regulatory Federal Register Volume 86, Issue 57, March 26, 2021.
15. American Society of Mechanical Engineers, Section XI "Rules for Inservice Inspection of Nuclear Power Plant Components", 2007 Edition with 2008 Addenda, 2013 Edition, and 2019 Edition.
16. U.S. NRC, "Vogtle Electric Generating Plant, Units 1 and 2 - Audit Report for the Promise Version 1.0 Probabilistic Fracture Mechanics Software Used in Relief Request VEGP-ISI-ALT-04-04 (EPID L-2019-LLR-0109)," December 10, 2020, ADAMS Accession No. ML20258A002.
17. Letter from Leonard N Olshan (NRC) to Dennis L Farrar (Commonwealth Edison Company), "Approval of Byron 2 Preservice Inspection Program," dated October 29, 1986, 8611070024.

APPENDIX A

PLANT SPECIFIC APPLICABILITY

Plant-Specific Applicability for Braidwood

Section 9 of Reference [A1] provides requirements that must be demonstrated in order to apply the representative stress and flaw tolerance analyses to a specific plant. Plant-specific evaluation of these requirements for Braidwood, Units 1 and 2, is provided in Table A1.

Table A1 indicates that all plant-specific requirements are met for Braidwood, Units 1 and 2. Therefore, the results and conclusions of the EPRI report are applicable to Braidwood, Units 1 and 2.

Table A1. Applicability of Reference [A1] Representative Analyses to Braidwood, Units 1 and 2

Category	Requirement from Reference [A1]	Applicability to Braidwood, Unit 1	Applicability to Braidwood, Unit 2
General Requirements	The nozzle-to-shell weld shall be one of the configurations shown in Figure 1-1 or Figure 1-2 of Reference [A1].	The Braidwood, Unit 1, FW nozzle configuration is shown in Figures A1 and A3, and is representative of the configuration shown in Figure 1-1 of Reference [A1].	The Braidwood, Unit 2, MS and FW nozzle configurations are shown in Figures A1, A2, and A4, and are representative of the configuration shown in Figure 1-2 of Reference [A1].

<table>
<tr><th>Category</th><th>Requirement from Reference [A1]</th><th>Applicability to Braidwood, Unit 1</th><th>Applicability to Braidwood, Unit 2</th></tr>
<tr><td rowspan="2"></td><td>The materials of the SG shell, FW nozzles, and MS nozzles must be low alloy ferritic steels which conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.</td><td>The Braidwood Station, Unit 1, FW nozzles, steam drum head, and shell cone are fabricated of SA-508, Class 3 material. The steam drum shell and secondary shell are fabricated from SA-533, Grade B, Class 1 material. These materials conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110. The MS nozzles are integrally forged into the steam drum head for Unit 1.</td><td>The Braidwood Station, Unit 2, FW and MS nozzles are fabricated of SA-508, Class 2a material. The upper head and barrels are fabricated from SA-533, Grade A, Class 2 material. Material SA-508, Class 2 conforms to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.

The material properties in SA-533, Grade A, Class 2 material are identical compared to the SA-533, Grade B Class 1 material used in the FEA in the EPRI report as is shown Table 5-2 of Reference [A1]. Appendix G, Table G-2110-1, of the 2019 Edition of ASME Section XI acknowledges that Figure G-2210-1 is applicable to material SA-533, Grade A, Class 2. Therefore, it can be concluded that the EPRI report is applicable to material SA-533, Grade A, Class 2.</td></tr>
<tr><td>The number of transients shown in Table 5-5 of Reference [A1] are bounding for application over a 60-year operating life.</td><td colspan="2">The transient cycles in Table 5-5 of Reference [A1] meet or exceed the 60-year projected cycles for Braidwood, Units 1 and 2, as shown in Tables A2 and A3 [A9-A12].</td></tr>
<tr><td rowspan="2">SG Feedwater Nozzle</td><td>The piping attached to the FW nozzle must be 14-inch to 18-inch NPS.</td><td colspan="2">The Braidwood, Units 1 and 2, FW piping lines are both 16-inch NPS.</td></tr>
<tr><td>The FW nozzle design must have an integrally attached thermal sleeve.</td><td colspan="2">The Braidwood, Units 1 and 2, FW nozzle configuration has an integrally attached thermal sleeve (References [A7] and [A8]).</td></tr>
</table>

Category	Requirement from Reference [A1]	Applicability to Braidwood, Unit 1	Applicability to Braidwood, Unit 2
SG Main Steam Nozzle (Unit 2 Only)	For Westinghouse and CE plants, the piping attached to the SG MS nozzle must be 28-inch to 36-inch NPS.	The MS nozzles for Braidwood Station, Unit 1 are integrally forged into the upper head and therefore are not a part of this proposed alternative.	Braidwood Station, Unit 2, is a Westinghouse 4-loop PWR. The Braidwood Station, Unit 2, MS piping lines are 30-inch NPS.
	The SG must have one main steam nozzle that exits the top dome of the SG.		As shown in Figure A4, Braidwood Station, Unit 2, has one MS nozzle per SG that exits the top dome of each SG.
	The main steam nozzle shall not significantly protrude into the SG (e.g., see Figure 4-7 of Reference [A1]) or have a unique nozzle weld configuration (e.g., see Figure 4-6 of Reference [A1]).		The Braidwood Station, Unit 2, MS nozzle configuration is shown in Figures A2 and A4, and does not protrude significantly into the SG. The Braidwood Station, Unit 2 MS nozzles are similar to Figure 4-5 of Reference [A1].

REFERENCES:

A1. *Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Inside Radius Sections*. EPRI, Palo Alto, CA: 2019. 3002014590, ADAMS Accession No. ML19347B107.

A2. Braidwood, Unit 1, Drawing 7720E038, *Section 'XI' Preservice U/T*.

A3. Braidwood, Unit 2, Drawing 1103J99, Sheet 1, General Arrangement - *Steam Generator Model "D5-3"*.

A4. Braidwood, Unit 2, Drawing 1106J25, Sheet 2, *Vertical Steam Generator Outline Model "D-5"*.

A5. Braidwood, Unit 1, Drawing 1SG-05, *Inservice Inspection Drawing for Replacement Steam Generator 1RC01BA – Loop 1, Unit 1*.

A6. Braidwood, Unit 2, Drawing 2SG-01, *Spec L2907, Inspection Identification Drawing for Inservice Inspection of Steam Generator 2RC01BA – Loop #1, Unit 2*.

A7. Braidwood, Unit 1, Design Specification 18-1229648, *Certified Design Specification for Replacement Steam Generator, Byron and Braidwood Stations Unit 1*.

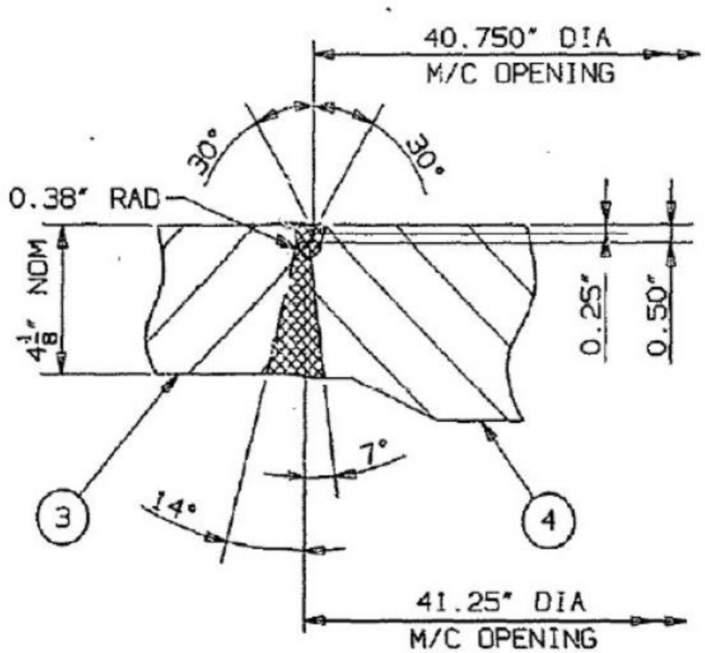
A8. Braidwood, Unit 2, Design Specification G-953431, *Reactor Coolant System* - *Model D-5 Steam Generator*.

A9. Braidwood, Unit 1, Work Order 5154527*, LR BWVP 850-7 Operational Transients* (Data through May 31, 2021).

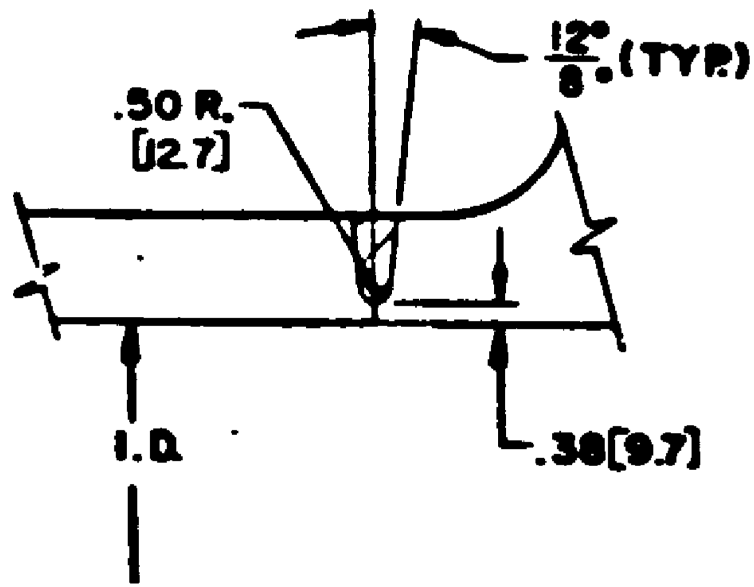
A10. Braidwood, Unit 1, Work Order 5064063, *LR Transient Monitoring Report – Semi-Annual Unit 1*.

A11. Braidwood, Unit 2, Work Order 5154525, *LR BWVP 850-7 Operational Transients* (Data through May 31, 2021).

A12. Braidwood, Unit 2, Work Order 5064062, *LR Transient Monitoring Report – Semi-Annual Unit 2*.

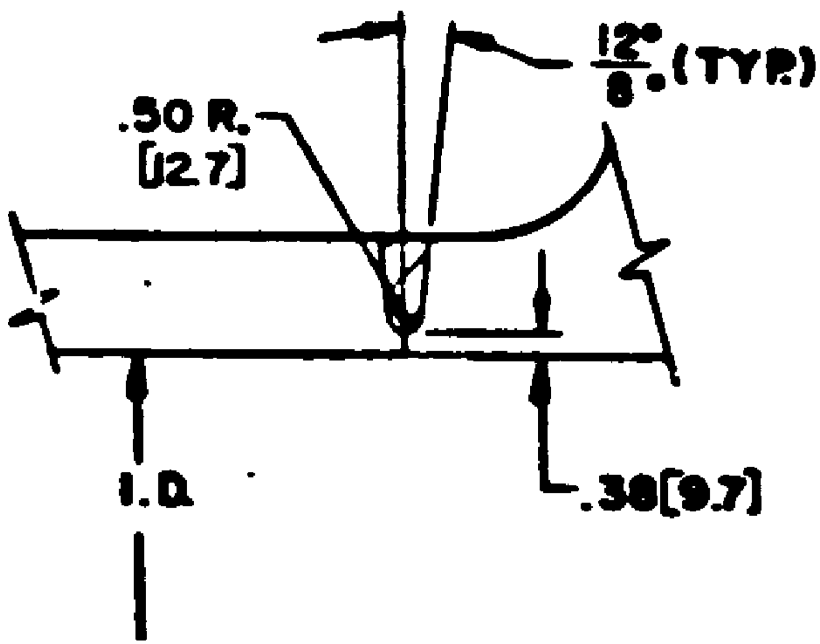


FEEDWATER NOZZLE TO STEAM DRUM W258
(Unit 1)



FEEDWATER NOZZLE TO STEAM DRUM
(Unit 2)

Figure A1. Braidwood, Units 1 and 2, SG Feedwater Nozzle Configuration



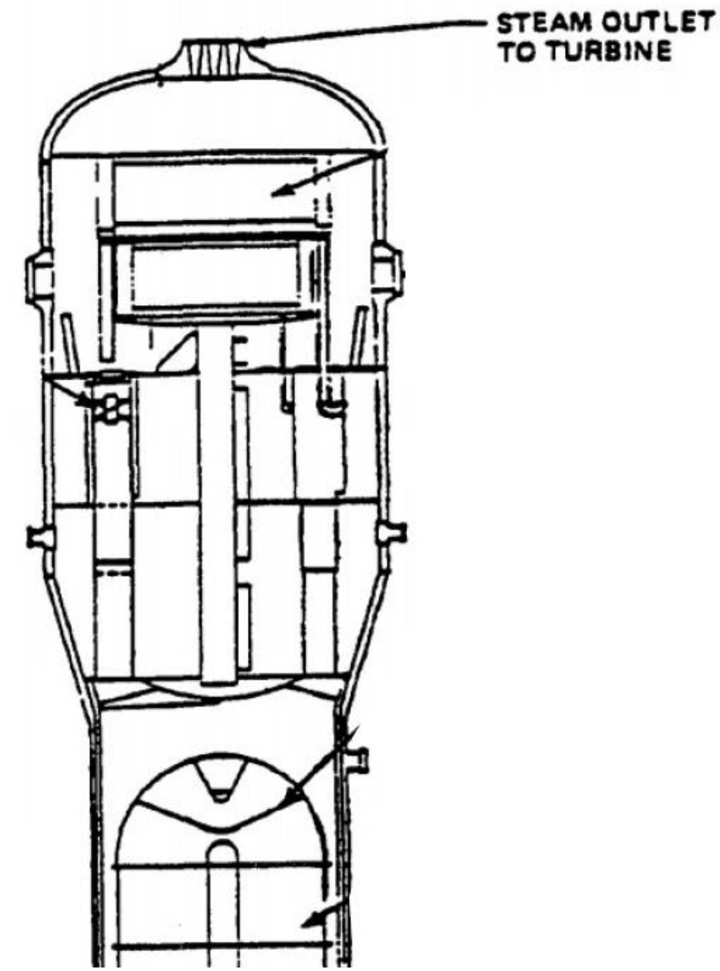


Figure A2. Braidwood, Unit 2, SG Main Steam Nozzle Configuration

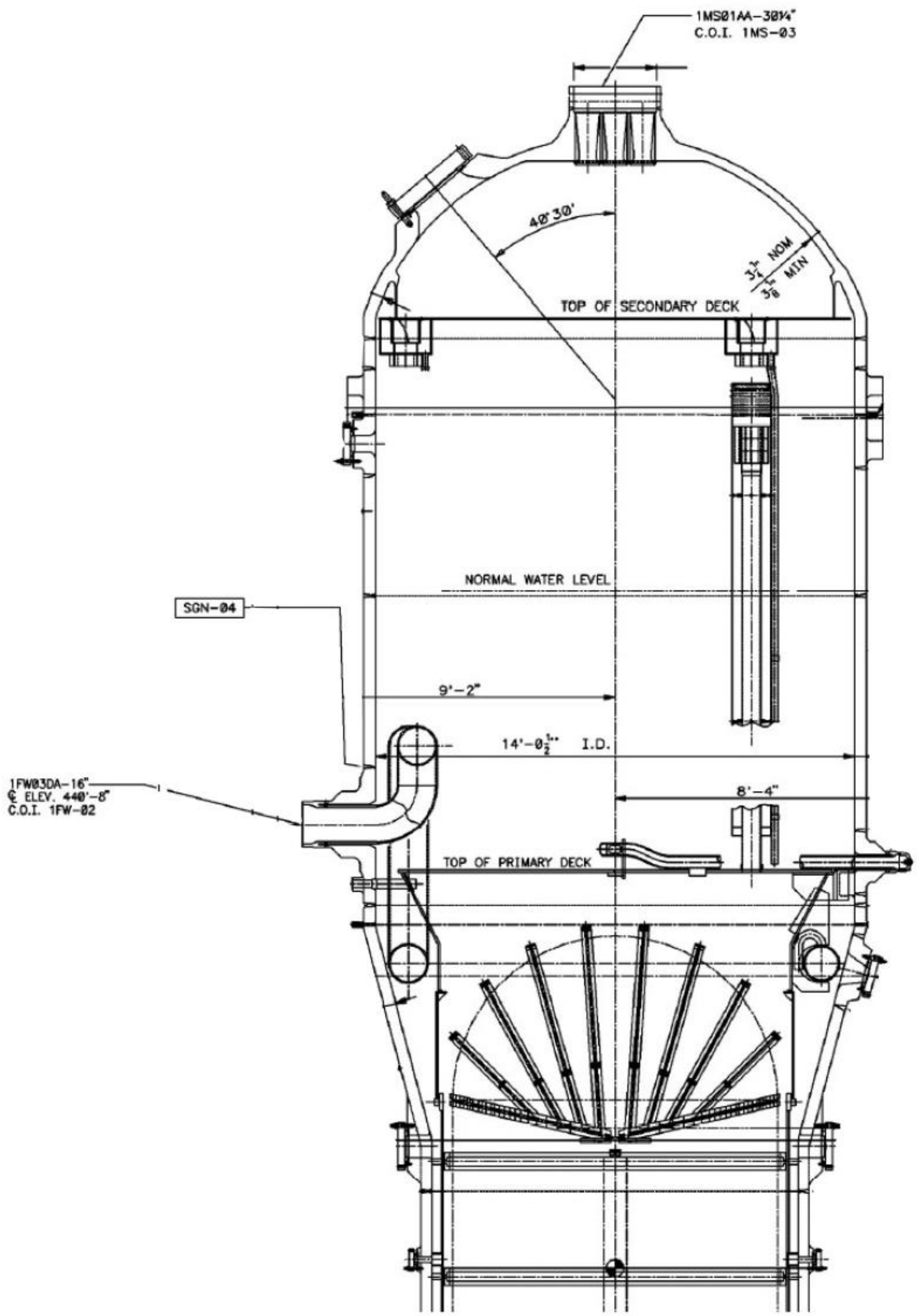


Figure A3. Braidwood, Unit 1, Steam Generator General Outline

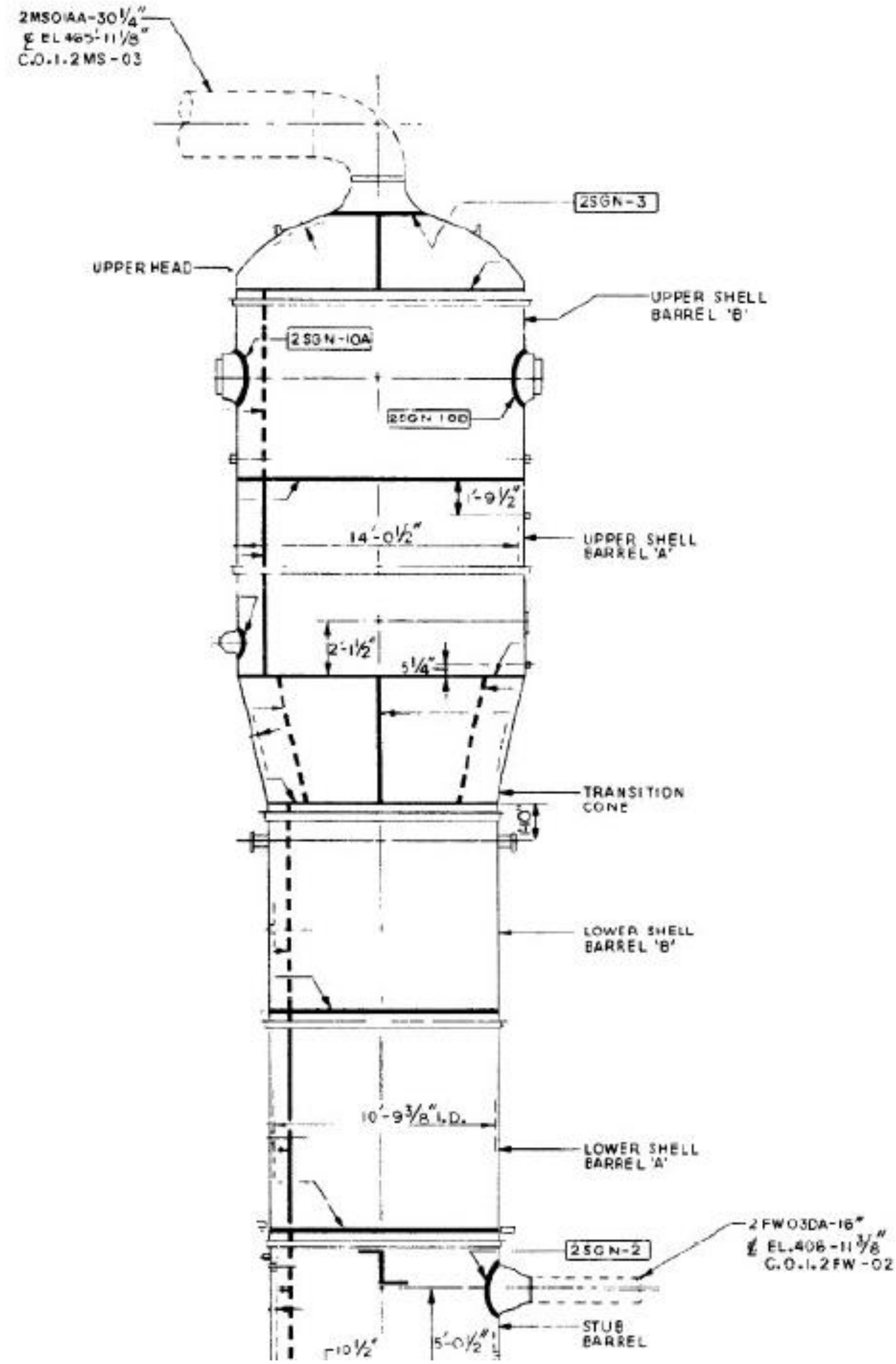


Figure A4. Braidwood, Unit 2, Steam Generator General Outline

Table A2. Transient Cycles for Braidwood, Unit 1, in Comparison to the Requirements in Reference [A1]

Transient	Cycles from Table 5-5 of EPRI Report 3002014590 [A1]	Unit 1 Cycles at 35.5 Years from Reference [A9]	Unit 1 60-Year Projected Cycles from Reference [A10]	Allowable Cycles from Reference [A9]
Heatup/Cooldown	300	41/40	72/70	200
Plant Loading at 5%/min	5000	150[1]	266[1]	13200
Plant Unloading at 5%/min	5000	122[2]	218[2]	12240
Loss of Load	360	1	2	80
Loss of Power	60	1	2	40

Note:
1. Cycle counts for Plant Loading conservatively includes all events for transients “Plant Loading at 5%/min”, “Plant Loading, 0% - 15%, Cold Turb/Gen”, and “Plant Loading, 0% - 15%, Hot Turb/Gen”.
2. Cycle counts for Plant Unloading conservatively includes all events for transients “Plant Unloading at 5%/min” and “Plant Unloading, 15% - 0%”.

Table A3. Transient Cycles for Braidwood, Unit 2, in Comparison to the Requirements in Reference [A1]

Transient	Cycles from Table 5-5 of EPRI Report 3002014590 [A1]	Unit 2 Cycles at 34 Years from Reference [A11]	Unit 2 60-Year Projected Cycles from Reference [A12]	Allowable Cycles from Reference [A11]
Heatup/Cooldown	300	43/51	79/94	200
Plant Loading at 5%/min	5000	121[1]	220[1]	13200
Plant Unloading at 5%/min	5000	102[2]	186[2]	13200
Loss of Load	360	1	2	80
Loss of Power	60	2	4	40

Note:
1. Cycle counts for Plant Loading conservatively includes all events for transients “Plant Loading at 5%/min” and “Plant Loading, 0% - 15%”.
2. Cycle counts for Plant Unloading conservatively includes all events for transients “Plant Unloading at 5%/min” and “Plant Unloading, 15% - 0%”.

Plant-Specific Applicability for Byron

Section 9 of Reference [A13] provides requirements that must be demonstrated in order to apply the representative stress and flaw tolerance analyses to a specific plant. Plant-specific evaluation of these requirements for Byron, Units 1 and 2, is provided in Table A4.

Table A4 indicates that all plant-specific requirements are met for Byron, Units 1 and 2. Therefore, the results and conclusions of the EPRI report are applicable to Byron, Units 1 and 2.

Table A4. Applicability of Reference [A13] Representative Analyses to Byron, Units 1 and 2

Category	Requirement from Reference [A13]	Applicability to Byron, Unit 1	Applicability to Byron, Unit 2
General Requirements	The nozzle-to-shell weld shall be one of the configurations shown in Figure 1-1 or Figure 1-2 of Reference [A13].	The Byron, Unit 1, FW nozzle configuration is shown in Figures A5 and A7, and is representative of the configuration shown in Figure 1-1 of Reference [A13].	The Byron, Unit 2, MS and FW nozzle configurations are shown in Figures A5, A6, and A8, and are representative of the configuration shown in Figure 1-2 of Reference [A13].

Category	Requirement from Reference [A13]	Applicability to Byron, Unit 1	Applicability to Byron, Unit 2
	The materials of the SG shell, FW nozzles, and MS nozzles must be low alloy ferritic steels which conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.	The Byron Station, Unit 1, FW nozzles, steam drum head, and shell cone are fabricated of SA-508, Class 3 material. The steam drum shell and secondary shell are fabricated from SA-533, Grade B, Class 1 material. These materials conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110. The MS nozzles are integrally forged into the steam drum head for Unit 1.	The Byron Station, Unit 2, FW and MS nozzles are fabricated of SA-508, Class 2 material. The upper head and barrels are fabricated from SA-533, Grade A, Class 2 material. Material SA-508, Class 2 conforms to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110. The material properties in SA-533, Grade A, Class 2 material are identical compared to the SA-533, Grade B Class 1 material used in the FEA in the EPRI report as is shown Table 5-2 of Reference [A13]. Appendix G, Table G-2110-1, of the 2019 Edition of ASME Section XI acknowledges that Figure G-2210-1 is applicable to material SA-533, Grade A, Class 2. Therefore, it can be concluded that the EPRI report is applicable to material SA-533, Grade A, Class 2.
	The number of transients shown in Table 5-5 of Reference [A13] are bounding for application over a 60-year operating life.	The transient cycles in Table 5-5 of Reference [A13] meet or exceed the 60-year projected cycles for Byron, Units 1 and 2, as shown in Tables A5 and A6 [A20]	
SG Feedwater Nozzle	The piping attached to the FW nozzle must be 14-inch to 18-inch NPS.	The Byron, Units 1 and 2, FW piping lines are both 16-inch NPS.	

<table>
<tr><th>Category</th><th>Requirement from Reference [A13]</th><th>Applicability to Byron, Unit 1</th><th>Applicability to Byron, Unit 2</th></tr>
<tr><td></td><td>The FW nozzle design must have an integrally attached thermal sleeve.</td><td colspan="2">The Byron, Units 1 and 2, FW nozzle configuration has an integrally attached thermal sleeve (References [A18] and [A19]).</td></tr>
<tr><td rowspan="3">SG Main Steam Nozzle

(Unit 2 Only)</td><td>For Westinghouse and CE plants, the piping attached to the SG MS nozzle must be 28-inch to 36-inch NPS.</td><td rowspan="3">The MS nozzles for Byron Station, Unit 1 are integrally forged into the upper head and therefore are not a part of this proposed alternative.</td><td>Byron Station, Unit 2, is a Westinghouse 4-loop PWR. The Byron Station, Unit 2, MS piping lines are 30-inch NPS.</td></tr>
<tr><td>The SG must have one main steam nozzle that exits the top dome of the SG.</td><td>As shown in Figure A8, Byron Station, Unit 2, has one MS nozzle per SG that exits the top dome of each SG.</td></tr>
<tr><td>The main steam nozzle shall not significantly protrude into the SG (e.g., see Figure 4-7 of Reference [A13]) or have a unique nozzle weld configuration (e.g., see Figure 4-6 of Reference [A13]).</td><td>The Byron Station, Unit 2, MS nozzle configuration is shown in Figure A6, and does not protrude significantly into the SG. The Byron Station, Unit 2 MS nozzles are similar to Figure 4-5 of Reference [A13].</td></tr>
</table>

REFERENCES:

A13. *Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Inside Radius Sections*. EPRI, Palo Alto, CA: 2019. 3002014590, ADAMS Accession No. ML19347B107.

A14. Byron, Unit 1, Drawing 7720E038, *Section 'XI' Preservice U/T*.

A15. Byron, Unit 2, Drawing 1103J99, Sheet 1, General Arrangement - Steam Generator Model "D5-3".

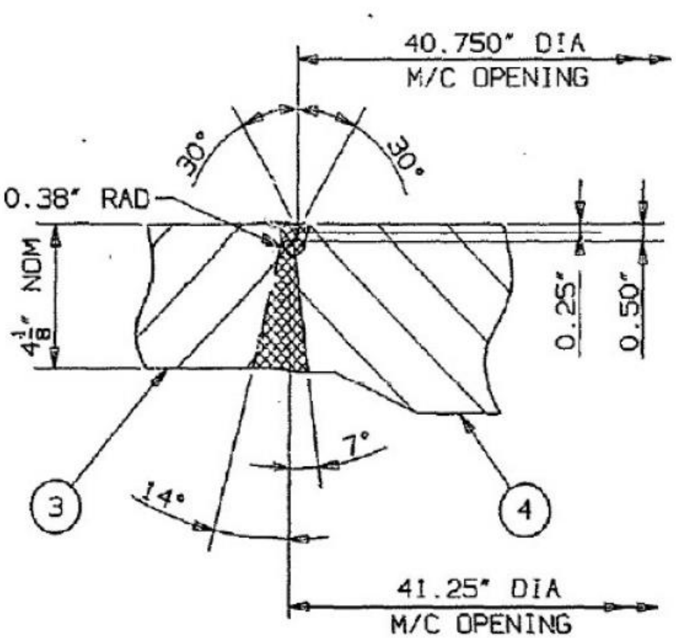
A16. Byron, Unit 1, Drawing 1SG-1-ISI, *Inspection Identification Drawing for Inservice Inspection for Replacement Steam Generator No. 1RC01BA*.

A17. Byron, Unit 2, Drawing 2SG-1-ISI, *Inspection Identification Drawing for Inservice Inspection of Steam Generator No. 2RC01BA*.

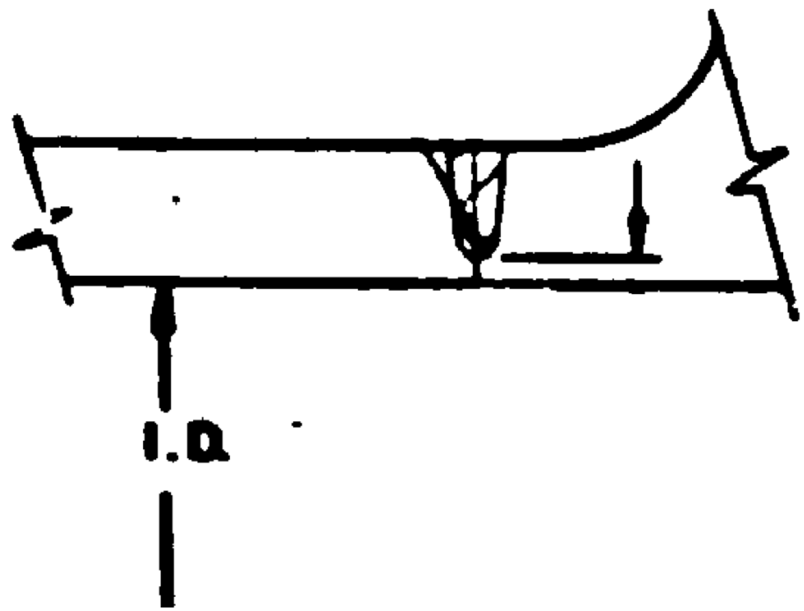
A18. Byron, Unit 1, Specification 18-1229648, *Certified Design Specification for Replacement Steam Generator, Byron and Braidwood Stations Unit 1*.

A19. Byron, Unit 2, Design Specification G-953431, *Reactor Coolant System - Model D-5 Steam Generator*.

A20. Byron Work Order 05140866-01, *LR-Cycle Counting* (Data through 3/31/2021).

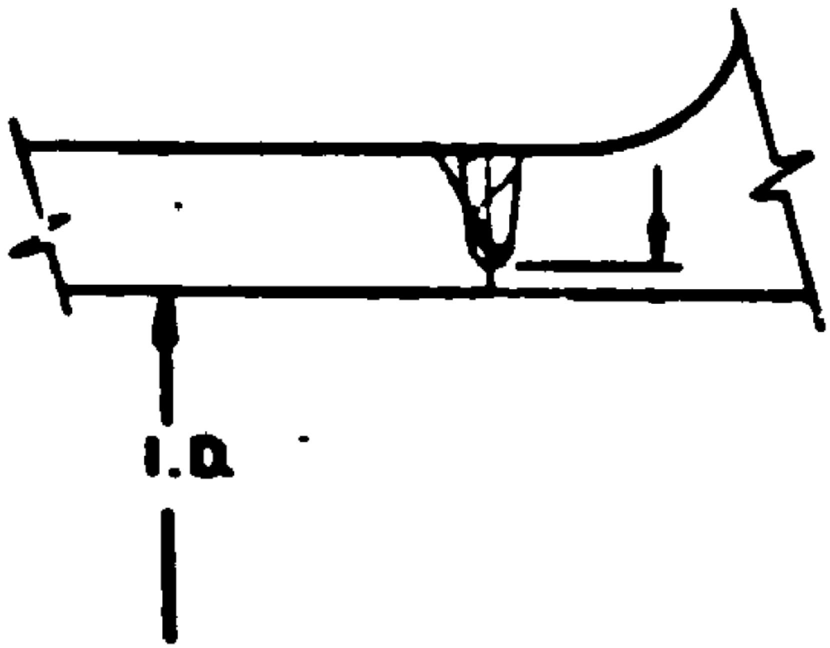


FEEDWATER NOZZLE TO
STEAM DRUM W258
(Unit 1)



FEEDWATER NOZZLE TO STEAM DRUM
(Unit 2)

Figure A5. Byron, Units 1 and 2, SG Feedwater Nozzle Configuration



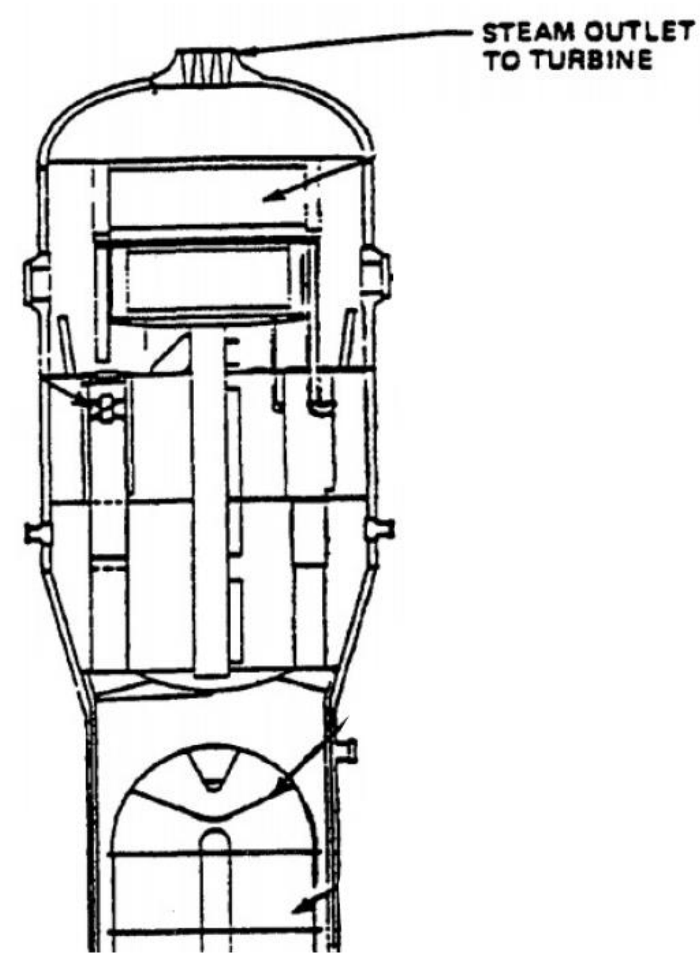


Figure A6. Byron, Unit 2, SG Main Steam Nozzle Configuration

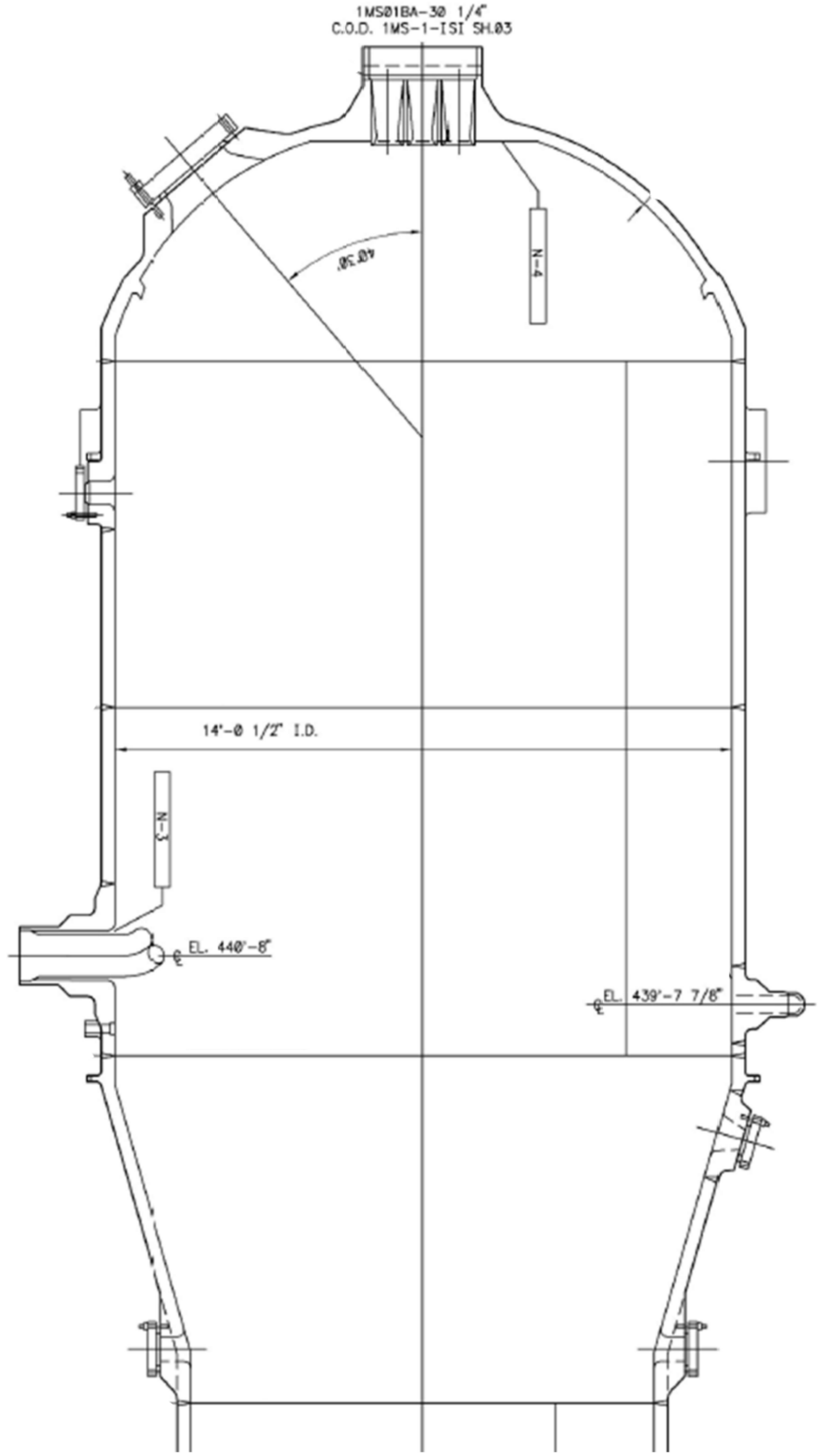


Figure A7. Byron, Unit 1, Steam Generator General Outline

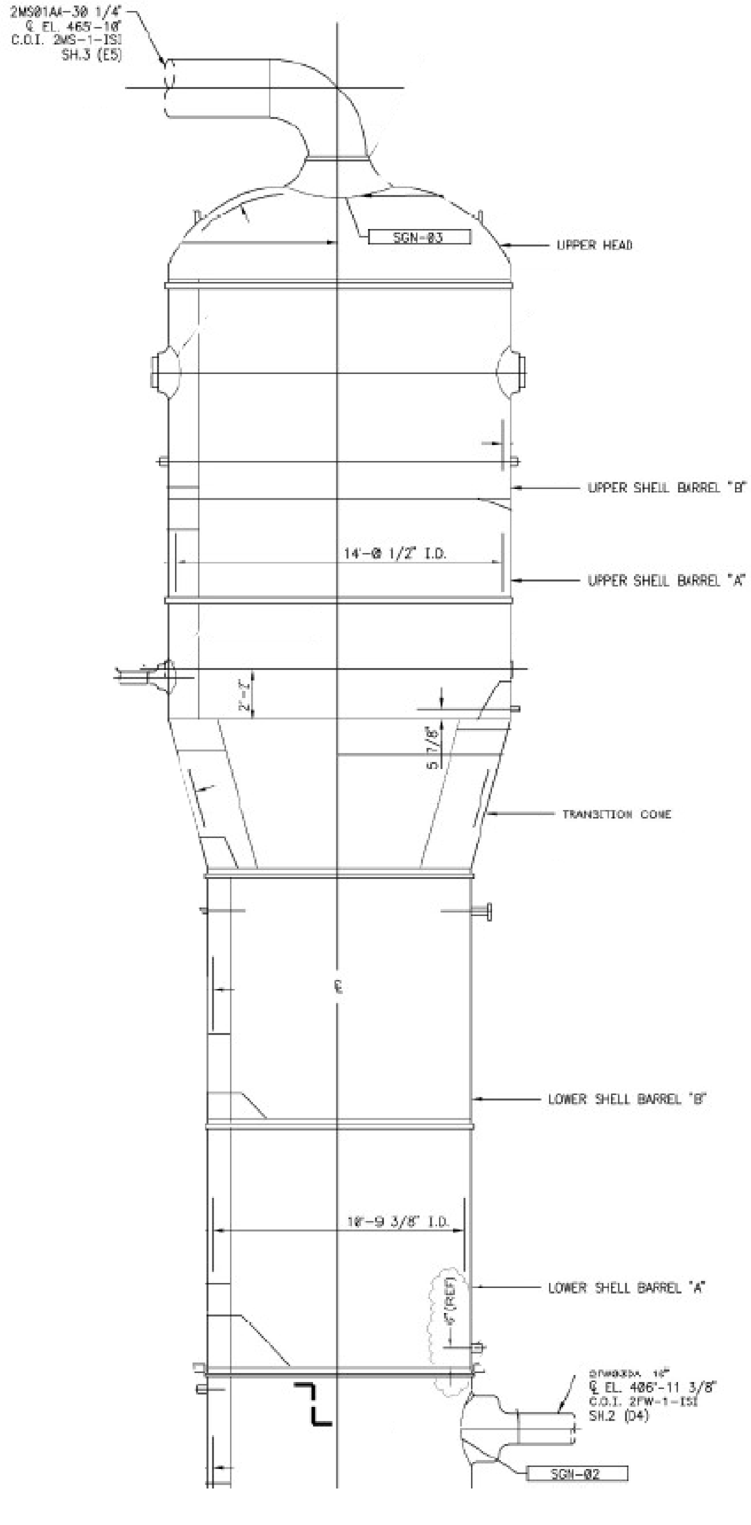


Figure A8. Byron, Unit 2, Steam Generator General Outline

Table A5. Transient Cycles for Byron, Unit 1, in Comparison to the Requirements in Reference [A13]

Transient	Cycles from Table 5-5 of EPRI Report 3002014590 [A13]	Unit 1 Cycles at 34 Years from Reference [A20]	Unit 1 60-Year Projected Cycles from Reference [A20]	Allowable Cycles from Reference [A20]
Heatup/Cooldown	300	38/38	65/65	200
Plant Loading at 5%/min	5000	107[1]	181[1]	13200
Plant Unloading at 5%/min	5000	111[2]	188[2]	12240
Loss of Load	360	3	6	80
Loss of Power	60	1	2	40

Note:
1. Cycle counts for Plant Loading conservatively includes all events for transients “Plant Loading at 5%/min”, “Plant Loading, 0% - 15%, Cold Turb/Gen”, and “Plant Loading, 0% - 15%, Hot Turb/Gen”.
2. Cycle counts for Plant Unloading conservatively includes all events for transients “Plant Unloading at 5%/min” and “Plant Unloading, 15% - 0%”.

Table A6. Transient Cycles for Byron, Unit 2, in Comparison to the Requirements in Reference [A13]

Transient	Cycles from Table 5-5 of EPRI Report 3002014590 [A13]	Unit 2 Cycles at 34 Years from Reference [A20]	Unit 2 60-Year Projected Cycles from Reference [A20]	Allowable Cycles from Reference [A20]
Heatup/Cooldown	300	34/34	60/60	200
Plant Loading at 5%/min	5000	101[1]	179[1]	13200
Plant Unloading at 5%/min	5000	101[2]	179[2]	13200
Loss of Load	360	1	2	80
Loss of Power	60	3	6	40

Note:
1. Cycle counts for Plant Loading conservatively includes all events for transients “Plant Loading at 5%/min” and “Plant Loading, 0% - 15%”.
2. Cycle counts for Plant Unloading conservatively includes all events for transients “Plant Unloading at 5%/min” and “Plant Unloading, 15% - 0%”.

Plant-Specific Applicability for Calvert Cliffs

Section 9 of Reference [A21] provides requirements that must be demonstrated in order to apply the representative stress and flaw tolerance analyses to a specific plant. Plant-specific evaluation of these requirements for Calvert Cliffs, Units 1 and 2, is provided in Table A7.

Table A7 indicates that all plant-specific requirements are met for Calvert Cliffs, Units 1 and 2. Therefore, the results and conclusions of the EPRI report are applicable to Calvert Cliffs, Units 1 and 2.

Table A7. Applicability of Reference [A21] Representative Analyses to Calvert Cliffs, Units 1 and 2

Category	Requirement from Reference [A21]	Applicability to Calvert Cliffs, Units 1 and 2
General Requirements	The nozzle-to-shell weld shall be one of the configurations shown in Figure 1-1 or Figure 1-2 of Reference [A21].	The Calvert Cliffs, Units 1 and 2, MS and FW nozzle configurations are shown in Figures A9 and A11, and are representative of the configuration shown in Figure 1-2 of Reference [A21].
	The materials of the SG shell, FW nozzles, and MS nozzles must be low alloy ferritic steels which conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.	The Calvert Cliffs, Units 1 and 2, FW nozzles are fabricated of SA-508, Class 1 material and the MS Nozzles are fabricated of SA-508, Class 2 material. The SG secondary head dome, and secondary shell and cone are fabricated from SA-533, Grade B, Class 1 material. These materials conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.
	The number of transients shown in Table 5-5 of Reference [A21] are bounding for application over a 60-year operating life.	The transient cycles in Table 5-5 of Reference [A21] meet or exceed the 60-year projected cycles for Calvert Cliffs, Units 1 and 2, as shown in Table A8 [A26]
SG Feedwater Nozzle	The piping attached to the FW nozzle must be 14-inch to 18-inch NPS.	The Calvert Cliffs, Units 1 and 2, FW piping lines are both 16-inch NPS.

Category	Requirement from Reference [A21]	Applicability to Calvert Cliffs, Units 1 and 2
	The FW nozzle design must have an integrally attached thermal sleeve.	The Calvert Cliffs, Units 1 and 2, FW nozzle configuration is shown in Figure A9 and an integrally attached thermal sleeve was installed during the steam drum modification per Figure A10.
SG Main Steam Nozzle	For Westinghouse and CE plants, the piping attached to the SG MS nozzle must be 28-inch to 36-inch NPS.	Calvert Cliffs, Units 1 and 2, are CE 2-loop PWRs. The Calvert Cliffs, Units 1 and 2, MS piping lines exiting the SG are both 36-inch NPS.
	The SG must have one main steam nozzle that exits the top dome of the SG.	As shown in Figure A12, Calvert Cliffs, Units 1 and 2, both have one MS nozzle per SG that exits the top dome of each SG.
	The main steam nozzle shall not significantly protrude into the SG (e.g., see Figure 4-7 of Reference [A21]) or have a unique nozzle weld configuration (e.g., see Figure 4-6 of Reference [A21]).	The Calvert Cliffs, Units 1 and 2, MS nozzle configuration is shown in Figures A11 and A12, and does not protrude significantly into the SG. The Calvert Cliffs, Units 1 and 2, MS nozzles are similar to Figure 4-5 of Reference [A21].

REFERENCES:

A21. *Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Inside Radius Sections*. EPRI, Palo Alto, CA: 2019. 3002014590, ADAMS Accession No. ML19347B107.

A22. Calvert Cliffs, Units 1 and 2, Drawing 12010-0011, *Nozzle Details Steam Generator*.

A23. Calvert Cliffs, Units 1 and 2, Drawing 12010A-0001SH0001, *General Arrangement*.

A24. Calvert Cliffs, Units 1 and 2, Drawing 12010A-0004SH0001, *Steam Drum Refurbishment*.

A25. Calvert Cliffs, Units 1 and 2, Specification 8067-31-2, *Engineering Specification for a Steam Generator Assembly for Baltimore Gas & Electric Company Calvert Cliffs Station*.

A26. Calvert Cliffs, Units 1 and 2, ECP-18-000545, *Fatigue Plant Transient Review (2018).*

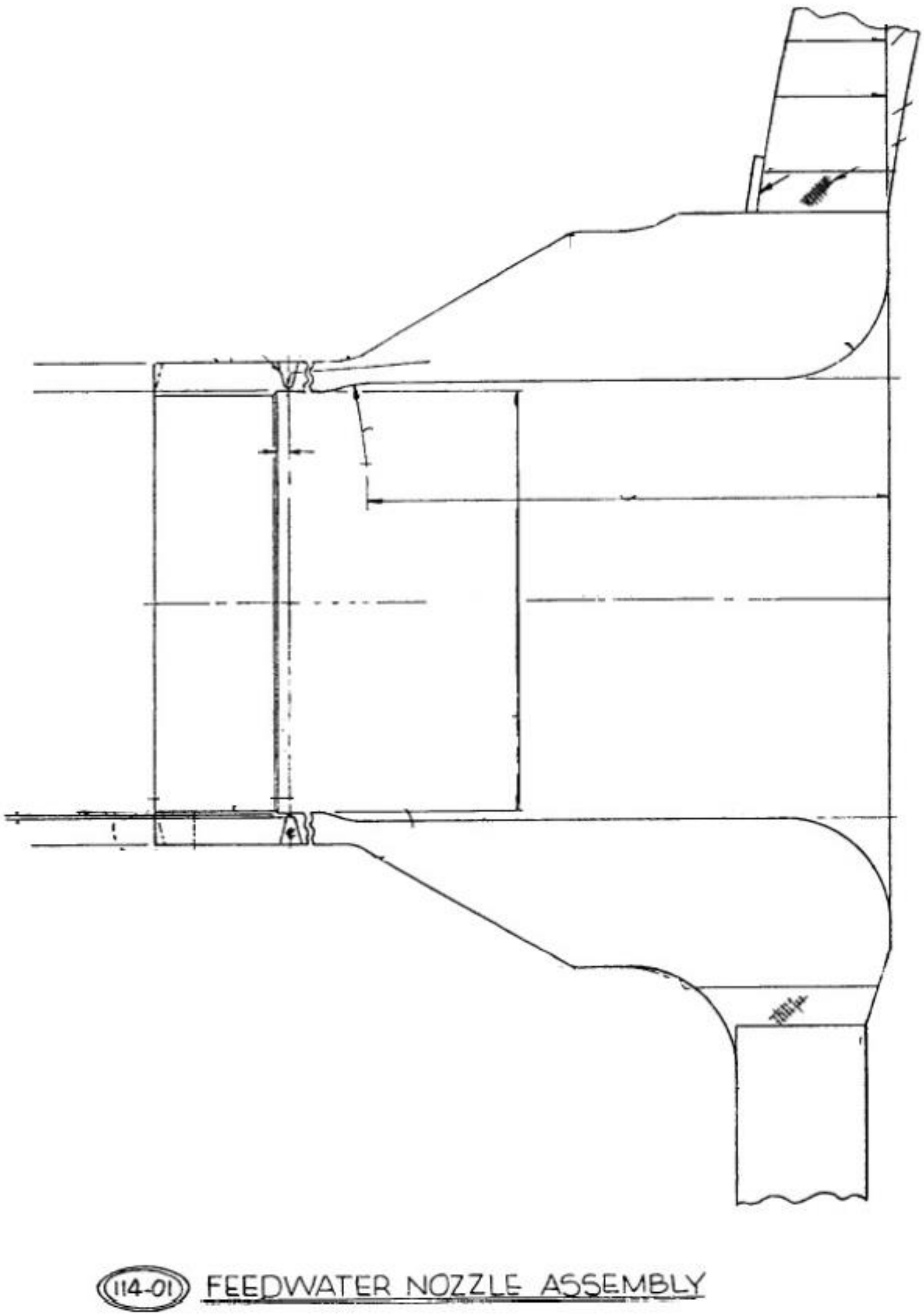


Figure A9. Calvert Cliffs, Units 1 and 2, SG Feedwater Nozzle Configuration

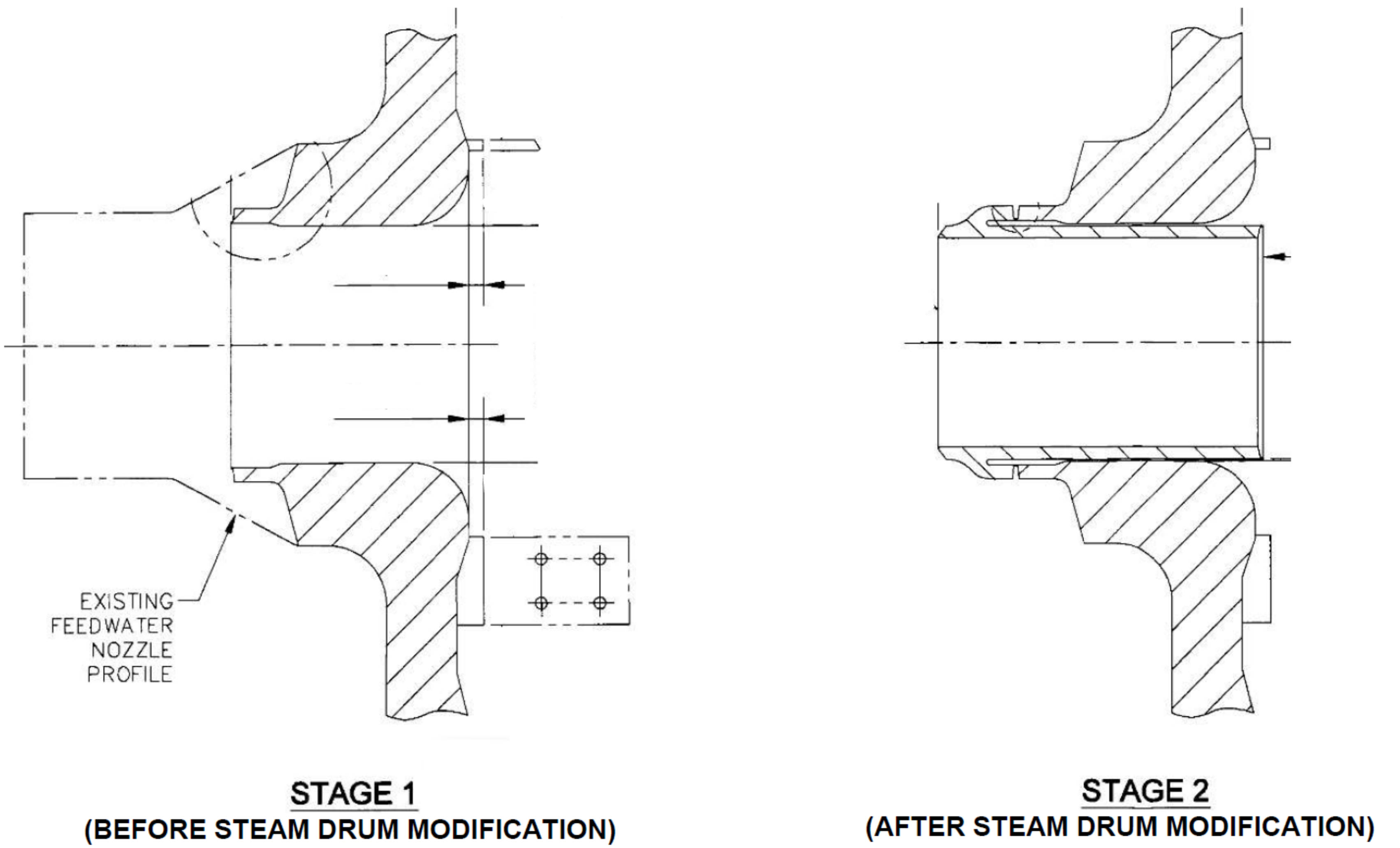


Figure A10. Calvert Cliffs, Units 1 and 2, Installation of Thermal Sleeve on Feedwater Nozzle During Steam Drum Modification

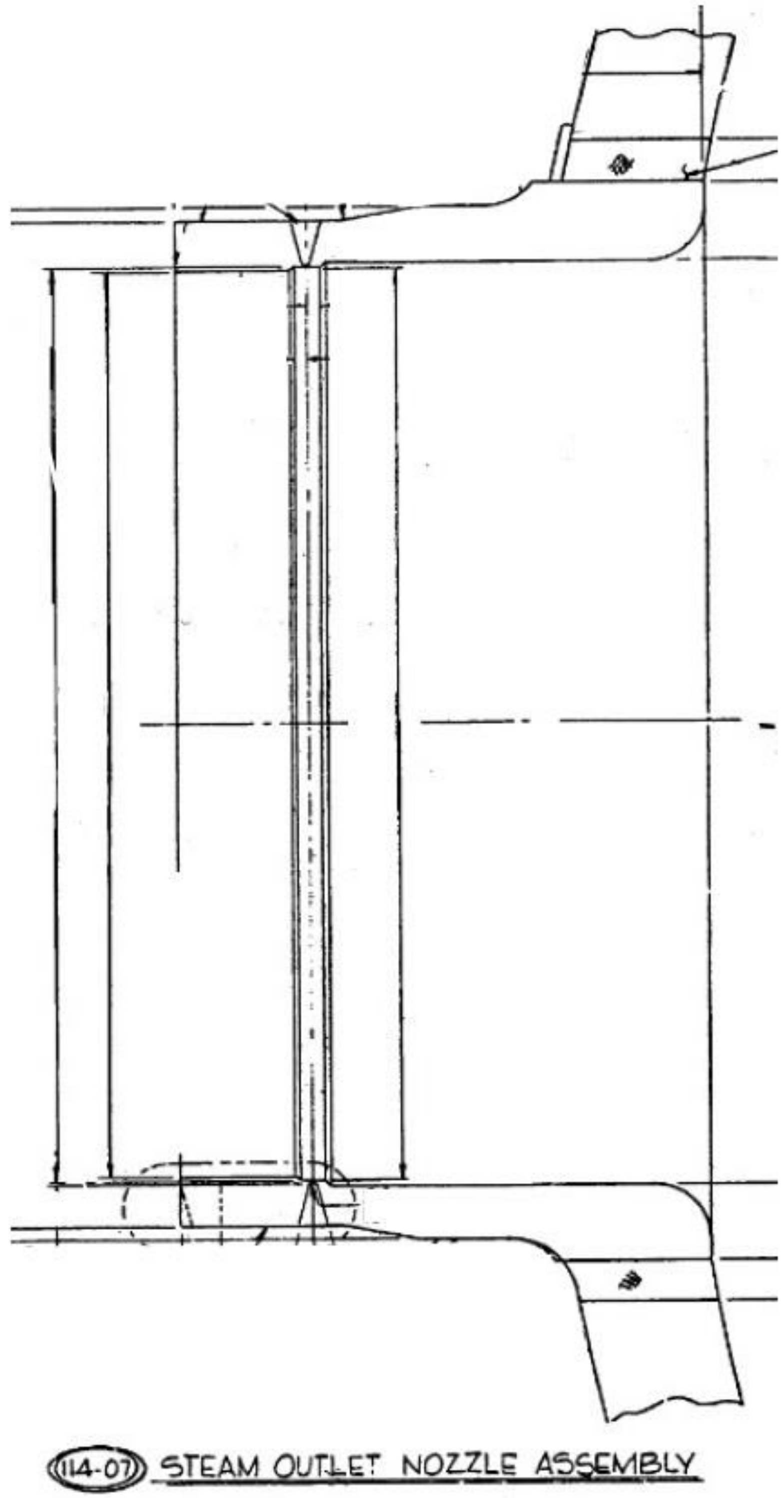


Figure A11. Calvert Cliffs, Units 1 and 2, SG Main Steam Nozzle Configuration

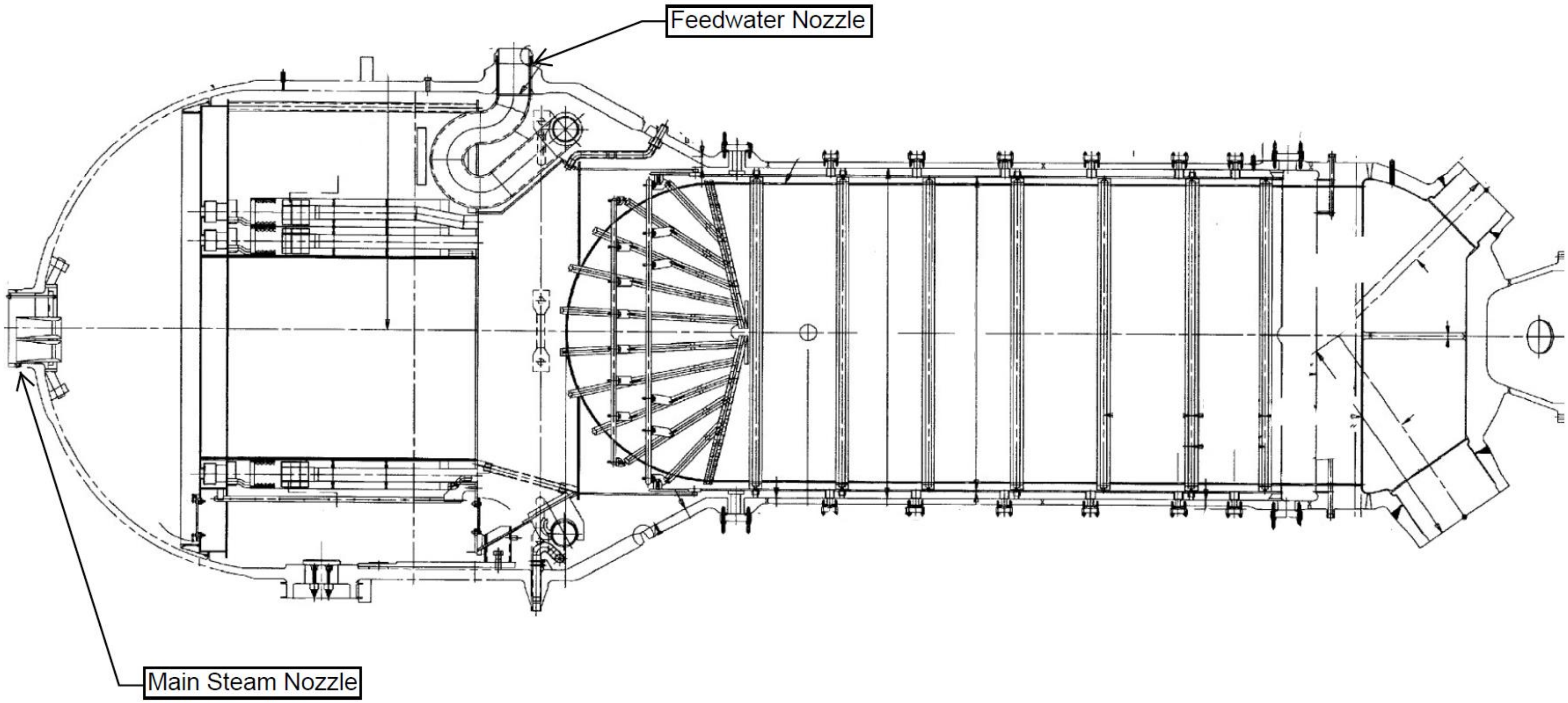


Figure A12. Calvert Cliffs, Units 1 and 2, Steam Generator General Outline

Table A8. Transient Cycles for Calvert Cliffs, Units 1 and 2, in Comparison to the Requirements in Reference [A21]

Transient	Cycles from Table 5-5 of EPRI Report 3002014590 [A21]	Unit 1 60-Year Projected Cycles from Reference [A26]	Unit 2 60-Year Projected Cycles from Reference [A26]	Allowable Cycles from Reference [A26]
Heatup/Cooldown	300	149/139	137/129	205
Plant Loading	5000	146	123	6150
Plant Unloading	5000	137	114	6150
Loss of Load	360	6	6	40
Loss of Power[1]	60	NA	NA	NA

Note:
1. Loss of Power is not part of the cycle tables calculated at Calvert Cliffs.

Plant-Specific Applicability for Ginna

Section 9 of Reference [A27] provides requirements that must be demonstrated in order to apply the representative stress and flaw tolerance analyses to a specific plant. Plant-specific evaluation of these requirements for Ginna is provided in Table A9.

Table A9 indicates that all plant-specific requirements are met for Ginna. Therefore, the results and conclusions of the EPRI report are applicable to Ginna.

Table A9. Applicability of Reference [A27] Representative Analyses to Ginna

Category	Requirement from Reference [A27]	Applicability to Ginna
General Requirements	The nozzle-to-shell weld shall be one of the configurations shown in Figure 1-1 or Figure 1-2 of Reference [A27].	The Ginna FW nozzle configuration is shown in Figure A13, and is representative of the configuration shown in Figure 1-1 of Reference [A27].
	The materials of the SG shell, FW nozzles, and MS nozzles must be low alloy ferritic steels which conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.	The Ginna FW nozzles, steam drum head, and shell cone are fabricated of SA-508, Class 3 material, and the secondary shell and steam drum shell are fabricated from SA-533, Grade B, Class 1 material. This material conforms to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.
	The number of transients shown in Table 5-5 of Reference [A27] are bounding for application over a 60-year operating life.	The transient cycles in Table 5-5 of Reference [A27] meet or exceed the 60-year projected cycles for Ginna as shown in Table A10 [A30 and A31]
SG Feedwater Nozzle	The piping attached to the FW nozzle must be 14-inch to 18-inch NPS.	The Ginna FW piping is 18-inch NPS.
	The FW nozzle design must have an integrally attached thermal sleeve	The Ginna FW nozzle configuration is shown in Figure A13 and has an integrally attached thermal sleeve.

Category	Requirement from Reference [A27]	Applicability to Ginna
SG Main Steam Nozzle	For Westinghouse and CE plants, the piping attached to the SG MS nozzle must be 28-inch to 36-inch NPS.	The MS nozzles for Ginna are integrally forged into the upper head and therefore are not a part of this proposed alternative.
	The SG must have one main steam nozzle that exits the top dome of the SG.	
	The main steam nozzle shall not significantly protrude into the SG (e.g., see Figure 4-7 of Reference [A27]) or have a unique nozzle weld configuration (e.g., see Figure 4-6 of Reference [A27]).	

REFERENCES:

A27. *Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Inside Radius Sections*. EPRI, Palo Alto, CA: 2019. 3002014590, ADAMS Accession No. ML19347B107.

A28. Ginna, Drawing 7705E001, *Steam Generator Arrangement - SGA*.

A29. Ginna, Drawing 7705E158, *Layout of Vessel Reference Points for Welds*.

A30. Design Specification 18-1224785-05, *Design Specification for Replacement Steam Generator*.

A31. Ginna, Fatigue Monitoring Program LR-FATM-PROGPLAN, Revision 6, *License Renewal Aging Management Program Basis Document,* December 20, 2017.

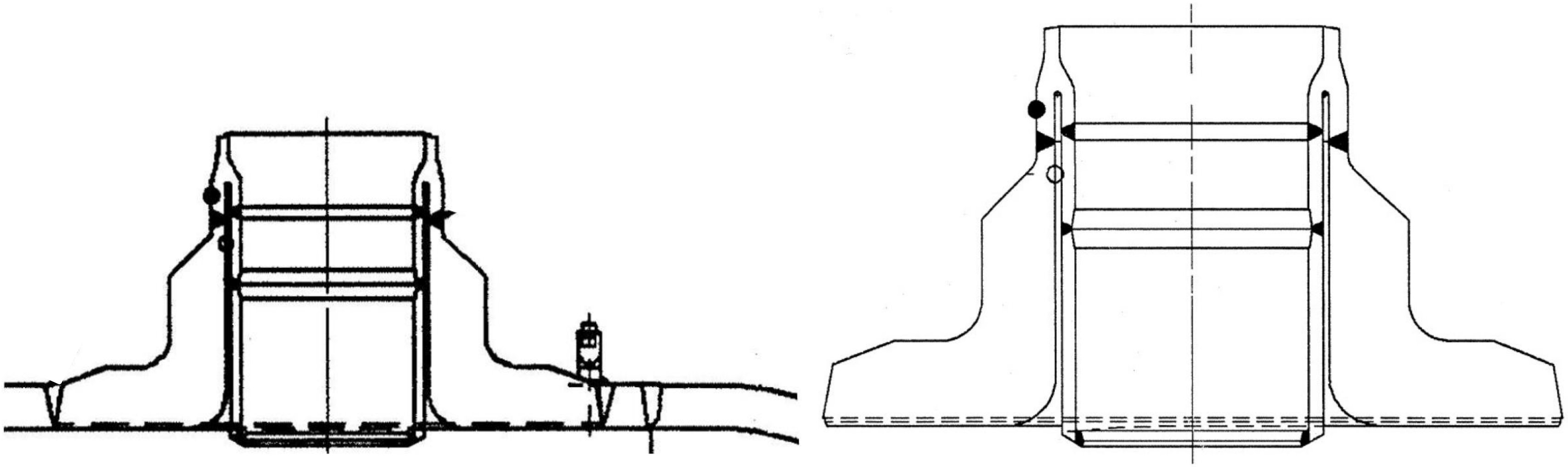


Figure A13. Ginna SG Feedwater Nozzle Configuration

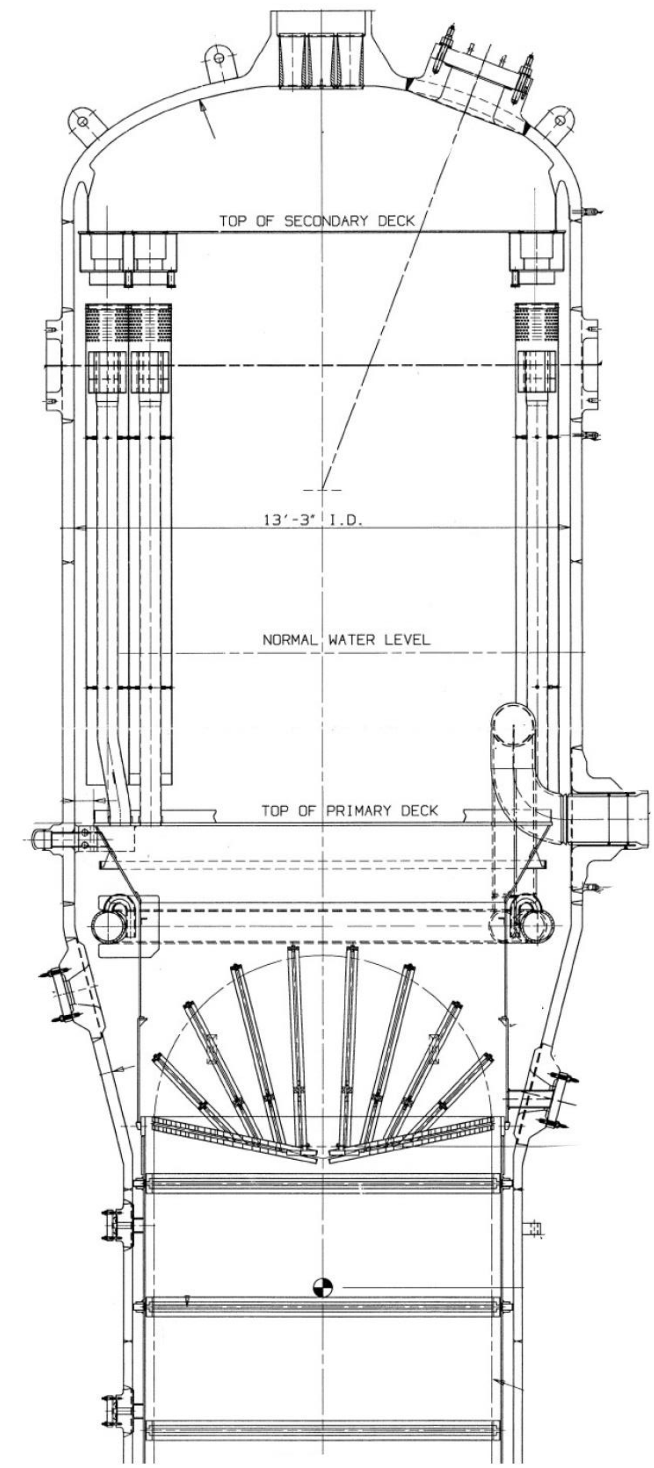


Figure A14. Ginna Steam Generator General Outline

Table A10. Transient Cycles for Ginna in Comparison to the Requirements in Reference [A27]

Transient	Cycles from Table 5-5 of EPRI Report 3002014590 [A27]	Unit 1 60-Year Projected Cycles from Table 4.2 of [A31]	Allowable Cycles from Table B-1 of [A30]
Heatup/Cooldown	300	153/153	200/200
Plant Loading	5000	1110	14500
Plant Unloading	5000	681	14500
Loss of Load	360	1	80
Loss of Power	60	12	40

APPENDIX B

PLANT SPECIFIC INSPECTION HISTORY

BRAIDWOOD, UNITS 1 AND 2, INSPECTION HISTORY

Currently, the scheduling of the MS and FW nozzle components for Braidwood, Units 1 and 2, satisfies the inspection requirements of ASME Code, Section XI, 2013 Edition. A summary of the inspection history for Item Numbers C2.21 and C2.22 for Braidwood, Units 1 and 2, is provided in Tables B1 through B3.

Table B1. MS Nozzle (C2.21)

Item Number	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
C2.21	2	4/1996 (A2R05)	1st Interval/ 3rd Period	2SG-03-SGN-03	NRI	100%
	2	5/2008 (A2R13)	2nd Interval/ 3rd Period	2SG-03-SGN-03	NRI	100%
	2	5/2014 (A2R17)	3rd Interval/ 2nd Period	2SG-03-SGN-03	NRI	100%

Table B2. FW Nozzle (C2.21)

Item Number	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
C2.21	1	4/2006 (A1R12)	2nd Interval/ 3rd Period	1SG-05-SGN-04	NRI	100%
	1	4/2015 (A1R18)	3rd Interval/ 2nd Period	1SG-05-SGN-04	NRI	100%
	2	4/1996 (A2R05)	1st Interval/ 3rd Period	2SG-03-SGN-02	NRI	100%
	2	5/2008 (A2R13)	2nd Interval/ 3rd Period	2SG-03-SGN-02	NRI	100%
	2	5/2017 (A2R19)	3rd Interval/ 3rd Period	2SG-03-SGN-02	NRI	100%

Table B3. FW Nozzle (C2.22)

Item Number	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
C2.22	1	4/2006 (A1R12)	2nd Interval/ 3rd Period	1SG-05-SGN-04 (NIR)	NRI	95%
	1	10/2010 (A1R15)	3rd Interval/ 1st Period	1SG-05-SGN-04 (NIR)	NRI	100%
	2	5/2008 (A2R13)	2nd Interval/ 3rd Period	2SG-03-SGN-02 (NIR)	NRI	100%
	2	5/2017 (A2R19)	3rd Interval/ 3rd Period	2SG-03-SGN-02 (NIR)	NRI	100%

BYRON, UNITS 1 AND 2, INSPECTION HISTORY

Currently, the scheduling of the MS and FW nozzle components for Byron, Units 1 and 2, satisfies the inspection requirements of ASME Code, Section XI, 2007 Edition with the 2008 Addenda. A summary of the inspection history for Item Numbers C2.21 and C2.22 for Byron, Units 1 and 2, is provided in Tables B4 through B6.

Table B4. MS Nozzle (C2.21)

Item Number	Unit	Date	Interval/ Period	Component ID	Exam Results	Coverage
C2.21	2	4/2013 (B2R17)	3rd Interval/ 3rd Period	2RC-01-BA/SGN-03	NRI[1]	100%
	2	8/1996 (B2R06)	1st Interval/ 3rd Period	2RC-01-BB/SGN-03	NRI	100%
	2	10/2005 (B2R12)	2nd Interval/ 3rd Period	2RC-01-BB/SGN-03	NRI	97% limited due to level taps and weld pads
	2	4/2013 (B2R17)	3rd Interval/ 3rd Period	2RC-01-BB/SGN-03	2 linear indications were identified with MT, 1 was acceptable and the other was ground to an acceptable size – both indications were determined to be acceptable to IWC-3511. No recordable indications were identified with UT	100%

Note:
1. Additional examination after indications were found on weld 2RC-01-BB/SGN-03.

Table B5. FW Nozzle (C2.21)

Item Number	Unit	Date	Interval/ Period	Component ID	Exam Results	Coverage
C2.21	1	3/2005 (B1R13)	2nd Interval/ 3rd Period	1RC-01-BB/N-3	NRI	97.2% Limited due to configuration and adjacent upper lateral restraint
	1	3/2014 (B1R19)	3rd Interval/ 3rd Period	1RC-01-BB/N-3	NRI	97.2% Limited due to configuration and adjacent upper lateral restraint
	2	3/1995 (B2R05)	1st Interval/ 3rd Period	2RC-01-BA/SGN-02	NRI	100%
	2	4/2007 (B2R13)	3rd Interval/ 1st Period	2RC-01-BA/SGN-02	NRI	100%
	2	4/2019 (B2R21)	4th Interval/ 1st Period	2RC-01-BA/SGN-02	NRI	100%
	2	1/1989 (B2R01)	1st Interval/ 1st Period	2RC-01-BC/SGN-02[1]	Recorded indication identified with the 45° shear during PSI. The indication was unacceptable to IWB-3511 during PSI.	90% Limited due to configuration
	2	9/1993 (B2R04)	1st Interval/ 2nd Period	2RC-01-BC/SGN-02[1]	Previously identified indication was recorded and determined essentially unchanged.	N/A

Note:

1. Weld was found to have an indication during the PSI examinations. Relief was requested to accept these flaws which was approved through an NRC safety evaluation on 10/29/1986. Successive examinations were performed for the weld and the indications were resized with no change.

Table B6. FW Nozzle (C2.22)

Item Number	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
C2.22	1	3/2005 (B1R13)	2nd Interval/ 3rd Period	1RC-01-BB/N-3-NIR	NRI	100%
	1	3/2008 (B1R15)	3rd Interval/ 1st Period	1RC-01-BB/N-3-NIR	NRI	100%
	1	3/2017 (B1R21)	4th Interval/ 1st Period	1RC-01-BB/N-3-NIR	NRI	100%
	2	3/1995 (B2R05)	1st Interval/ 3rd Period	2RC-01-BA/SGN-02-NIR	NRI	100%
	2	4/2007 (B2R13)	3rd Interval/ 1st Period	2RC-01-BA/SGN-02-NIR	NRI	100%
	2	4/2019 (B2R21)	4th Interval/ 1st Period	2RC-01-BA/SGN-02-NIR	NRI	100%

CALVERT CLIFFS, UNITS 1 AND 2, INSPECTION HISTORY

Currently, the scheduling of the MS and FW nozzle components for Calvert Cliffs, Units 1 and 2, satisfies the inspection requirements of ASME Code, Section XI, 2013 Edition. A summary of the inspection history for Item Numbers C2.21 and C2.22 for Calvert Cliffs, Units 1 and 2, is provided in Tables B7 through B10.

Table B7. MS Nozzle (C2.21)

Item Number	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
C2.21	1	3/2010 (1RFO19)	3rd Interval/ 3rd Period	SG-11-MS	NRI	100%
	1	3/2018 (1RFO23)	4th Interval/ 3rd Period	SG-11-MS	NRI	100%
	2	3/2009 (2RFO17)	3rd Interval/ 3rd Period	SG-21-MS	NRI	92%
	2	2/2019 (2RFO22)	4th Interval/ 3rd Period	SG-21-MS	NRI	100%

Table B8. MS Nozzle (C2.22)

Item Number	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
C2.22	1	3/2010 (1RFO19)	3rd Interval/ 3rd Period	SG-11-MS-IRS	NRI	100%
	1	3/2018 (1RFO23)	4th Interval/ 3rd Period	SG-11-MS-IRS	NRI	100%
	2	3/2009 (2RFO17)	3rd Interval/ 3rd Period	SG-21-MS-IRS	NRI	100%
	2	2/2019 (2RFO22)	4th Interval/ 3rd Period	SG-21-MS-IRS	NRI	100%

Table B9. FW Nozzle (C2.21)

Item Number	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
C2.21	1	3/2010 (1RFO19)	3rd Interval/ 3rd Period	SG-11-FW	NRI	93.8%
	1	2/2016 (1RFO22)	4th Interval/ 2nd Period	SG-11-FW	NRI	99.9%
	2	3/2007 (2RFO16)	3rd Interval/ 3rd Period	SG-22-FW	NRI	91.3%
	2	2/2015 (2RFO20)	4th Interval/ 2nd Period	SG-22-FW	NRI	99.1%

Table B10. FW Nozzle (C2.22)

Item Number	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
C2.22	1	3/2010 (1RFO19)	3rd Interval/ 3rd Period	SG-11-FW-IRS	NRI	100%
	1	2/2016 (1RFO22)	4th Interval/ 2nd Period	SG-11-FW-IRS	NRI	100%
	2	3/2007 (2RFO16)	3rd Interval/ 3rd Period	SG-22-FW-IRS	NRI	100%
	2	2/2015 (2RFO20)	4th Interval/ 2nd Period	SG-22-FW-IRS	NRI	100%

GINNA INSPECTION HISTORY

Currently, the scheduling of the MS and FW nozzle components for Ginna satisfies the inspection requirements of ASME Code, Section XI, 2013 Edition. A summary of the inspection history for Item Numbers C2.21 and C2.22 for Ginna is provided in Tables B11 through B12.

Table B11. FW Nozzle (C2.21)

Item Number	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
C2.21	1	1/1999 (O1999)	3rd Interval/ 3rd Period	FWIN-BR	NRI	>90%
	1	9/2009 (O2009)	4th Interval/ 3rd Period	FWIN-BR	NRI	100%
	1	4/2017 (O2017)	5th Interval/ 3rd Period	FWIN-BR	NRI	100%

Table B12. FW Nozzle (C2.22)

Item Number	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
C2.22	1	1/1999 (O1999)	3rd Interval/ 3rd Period	FWIN-BR	NRI	>90%
	1	9/2009 (O2009)	4th Interval/ 3rd Period	FWIN-BR	NRI	100%
	1	4/2017 (O2017)	5th Interval/ 3rd Period	FWIN-BR	NRI	100%

APPENDIX C

RESULTS OF INDUSTRY SURVEY

Overall Industry Inspection Summary

The results of an industry survey of past inspections of SG MS and FW nozzles are summarized in Section 3 of Reference [1]. Table C1 provides a summary of the combined survey results for Item Numbers C2.22, C2.21, and C2.32(1). The results identified that SG MS and FW nozzle-to-shell welds and nozzle inside radius section examinations adversely impact outage activities including worker exposure, personnel safety, and radwaste. A total of 74 domestic and international BWR and PWR units responded to the survey and provided information representing all PWR plant designs currently in operation in the U.S. This included 2-loop, 3-loop, and 4-loop PWR designs from each of the PWR nuclear steam supply system (NSSS) vendors (i.e., Babcock and Wilcox (B&W), Combustion Engineering (CE), and Westinghouse). A total of 727 examinations for Item Numbers C2.21, C2.22, and C2.32(1) components were conducted, with 563 of these specifically for PWR components. The majority of the PWR examinations were performed on SG MS and FW nozzles. Only one PWR examination identified two (2) flaws that exceeded ASME Code Section XI acceptance criteria. The flaws were linear indications of 0.3" and 0.5" in length, and were detected in a MS nozzle-to-shell weld using magnetic particle examination techniques. The indications were dispositioned by light grinding (ADAMS Accession No. ML13217A093).

Table C1. Summary of Survey Results

Plant Type	Number of Units	Number of Examinations	Number of Reportable Indications
BWR	27	164	0
PWR	47	563	2
Totals	**74**	**727**	**2**

1. Item Number C2.32 is similar to Item Number C2.21 and was evaluated in the Reference [1] technical basis and included in the industry survey. Item Number C2.32 is not applicable to the Steam Generators covered in this proposed alternative.