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10 CFR 50.55a

December 14, 2021

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 Categories B-B, B-D, and C-A Steam Generator Pressure Retaining Welds and Full Penetration Welded Nozzles

In accordance with 10 CFR 50.55a(z)(1), Exelon Generation Company, LLC (Exelon) 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). This proposed alternative requests to extend the inspection interval for the ASME Code, Section XI, Table IWB- 2500-1, Examination Categories B-B and B-D, and Table IWC-2500-1, Examination Category C-A steam generator examinations for the remainder of the currently licensed operating periods.

The proposed alternative is to extend the frequency of volumetric examinations of steam generator pressure retaining welds and full penetration welded nozzles 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, and is provided in the Attachment 1 to this letter.

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 September 30, 2022 to support the Braidwood Fall 2022 outage 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-17 for Braidwood Station, Units 1 and 2, Proposed Alternative I4R-23 for Byron Station, Units 1 and 2, Proposed Alternative ISI-05-018 for Calvert Cliffs Nuclear Power Plant, Units 1 and 2, and Proposed Alternative I6R-10 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
S. Seaman, State of Maryland

Attachment 1

Braidwood Station
Byron Station
Calvert Cliffs
Ginna

10 CFR 50.55a Proposed Alternative I4R-17 for Braidwood Station, Units 1 and 2,
Proposed Alternative I4R-23 for Byron Station, Units 1 and 2,
Proposed Alternative ISI-05-018 for Calvert Cliffs Nuclear Power Plant, Units 1 and 2, and
Proposed Alternative I6R-10 for Ginna,
Revision 0

Proposed Alternative for Examinations of Pressure Retaining Welds and Full Penetration Welded Nozzles (Examination Categories B-B, B-D, and C-A) In Accordance with 10 CFR 50.55a(z)(1)

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1 ASME CODE COMPONENTS AFFECTED:

Code Class: Class 1 and Class 2

Description: Steam Generator (SG) Pressure Retaining Welds and Full Penetration Welded Nozzles

Examination Categories: Category B-B, Pressure Retaining Welds in Vessels Other Than Reactor Vessels
Category B-D, Full Penetration Welded Nozzles in Vessels
Category C-A, Pressure Retaining Welds in Pressure Vessels

Item Numbers: B2.40 - Steam Generators (Primary Side), Tubesheet-to-Head Weld
B3.130 - Steam Generators (Primary Side), Nozzle-to-Vessel Welds
C1.10 - Shell Circumferential Welds
C1.20 - Head Circumferential Welds
C1.30 - Tubesheet-to-Shell Weld

Braidwood Component IDs:

Unit	SG	Component ID	Item Number	Description
1	A	1SG-05-SGC-01	B2.40	Primary Head - Tubesheet
1	A	1SG-05-SGC-05	C1.10	Upper Secondary Shell - Shell Cone
1	A	1SG-05-SGC-06	C1.10	Shell Cone - Steam Drum Lower Shell
1	A	1SG-05-SGC-08	C1.20	Steam Drum Upper Shell - Steam Drum Head
1	A	1SG-05-SGC-02	C1.30	Tubesheet - Lower Secondary Shell
1	B	1SG-06-SGC-01	B2.40	Primary Head - Tubesheet
1	B	1SG-06-SGC-05	C1.10	Upper Secondary Shell - Shell Cone
1	B	1SG-06-SGC-06	C1.10	Shell Cone - Steam Drum Lower Shell
1	B	1SG-06-SGC-08	C1.20	Steam Drum Upper Shell - Steam Drum Head
1	B	1SG-06-SGC-02	C1.30	Tubesheet - Lower Secondary Shell
1	C	1SG-07-SGC-01	B2.40	Primary Head - Tubesheet
1	C	1SG-07-SGC-05	C1.10	Upper Secondary Shell - Shell Cone
1	C	1SG-07-SGC-06	C1.10	Shell Cone - Steam Drum Lower Shell
1	C	1SG-07-SGC-08	C1.20	Steam Drum Upper Shell - Steam Drum Head
1	C	1SG-07-SGC-02	C1.30	Tubesheet - Lower Secondary Shell
1	D	1SG-08-SGC-01	B2.40	Primary Head - Tubesheet
1	D	1SG-08-SGC-05	C1.10	Upper Secondary Shell - Shell Cone
1	D	1SG-08-SGC-06	C1.10	Shell Cone - Steam Drum Lower Shell
1	D	1SG-08-SGC-08	C1.20	Steam Drum Upper Shell - Steam Drum Head
1	D	1SG-08-SGC-02	C1.30	Tubesheet - Lower Secondary Shell

Unit	SG	Component ID	Item Number	Description
2	A	2SG-01-SGC-01	B2.40	Channel Head - Tubesheet
2	A	2SG-01-SGC-03	C1.10	Stub Barrel - Lower Shell (A)
2	A	2SG-01-SGC-05	C1.10	Lower Shell (B) - Transition Cone
2	A	2SG-01-SGC-06	C1.10	Transition Cone - Upper Shell (A)
2	A	2SG-01-SGC-08	C1.20	Upper Shell (B) - Upper Head
2	A	2SG-01-SGC-02	C1.30	Tubesheet - Stub Barrel
2	B	2SG-02-SGC-01	B2.40	Channel Head - Tubesheet
2	B	2SG-02-SGC-03	C1.10	Stub Barrel - Lower Shell (A)
2	B	2SG-02-SGC-05	C1.10	Lower Shell (B) - Transition Cone
2	B	2SG-02-SGC-06	C1.10	Transition Cone - Upper Shell (A)
2	B	2SG-02-SGC-08	C1.20	Upper Shell (B) - Upper Head
2	B	2SG-02-SGC-02	C1.30	Tubesheet - Stub Barrel
2	C	2SG-03-SGC-01	B2.40	Channel Head - Tubesheet
2	C	2SG-03-SGC-03	C1.10	Stub Barrel - Lower Shell (A)
2	C	2SG-03-SGC-05	C1.10	Lower Shell (B) - Transition Cone
2	C	2SG-03-SGC-06	C1.10	Transition Cone - Upper Shell (A)
2	C	2SG-03-SGC-08	C1.20	Upper Shell (B) - Upper Head
2	C	2SG-03-SGC-02	C1.30	Tubesheet - Stub Barrel
2	D	2SG-04-SGC-01	B2.40	Channel Head - Tubesheet
2	D	2SG-04-SGC-03	C1.10	Stub Barrel - Lower Shell (A)
2	D	2SG-04-SGC-05	C1.10	Lower Shell (B) - Transition Cone
2	D	2SG-04-SGC-06	C1.10	Transition Cone - Upper Shell (A)
2	D	2SG-04-SGC-08	C1.20	Upper Shell (B) - Upper Head
2	D	2SG-04-SGC-02	C1.30	Tubesheet - Stub Barrel

*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/SGW-01	B2.40	Tubesheet - Primary Head
1	A	1RC-01-BA/SGW-05	C1.10	Secondary Shell - Shell Cone
1	A	1RC-01-BA/SGW-06	C1.10	Steam Drum - Shell Cone
1	A	1RC-01-BA/SGW-08	C1.20	Steam Drum Head - Steam Drum
1	A	1RC-01-BA/SGW-02	C1.30	Secondary Shell - Tubesheet
1	B	1RC-01-BB/SGW-01	B2.40	Tubesheet - Primary Head
1	B	1RC-01-BB/SGW-05	C1.10	Secondary Shell - Shell Cone
1	B	1RC-01-BB/SGW-06	C1.10	Steam Drum - Shell Cone
1	B	1RC-01-BB/SGW-08	C1.20	Steam Drum Head - Steam Drum
1	B	1RC-01-BB/SGW-02	C1.30	Secondary Shell - Tubesheet
1	C	1RC-01-BC/SGW-01	B2.40	Tubesheet - Primary Head
1	C	1RC-01-BC/SGW-05	C1.10	Secondary Shell - Shell Cone
1	C	1RC-01-BC/SGW-06	C1.10	Steam Drum - Shell Cone
1	C	1RC-01-BC/SGW-08	C1.20	Steam Drum Head - Steam Drum
1	C	1RC-01-BC/SGW-02	C1.30	Secondary Shell - Tubesheet
1	D	1RC-01-BD/SGW-01	B2.40	Tubesheet - Primary Head
1	D	1RC-01-BD/SGW-05	C1.10	Secondary Shell - Shell Cone
1	D	1RC-01-BD/SGW-06	C1.10	Steam Drum - Shell Cone
1	D	1RC-01-BD/SGW-08	C1.20	Steam Drum Head - Steam Drum
1	D	1RC-01-BD/SGW-02	C1.30	Secondary Shell - Tubesheet
2	A	2RC-01-BA/SGC-01	B2.40	Stub Barrel - Channel Head
2	A	2RC-01-BA/SGC-03	C1.10	Stub Barrel - Lower Barrel “A”
2	A	2RC-01-BA/SGC-05	C1.10	Lower Barrel “B” - Transition Cone
2	A	2RC-01-BA/SGC-06	C1.10	Transition Cone - Upper Barrel “B”
2	A	2RC-01-BA/SGC-08	C1.20	Upper Barrel “B” - Upper Head
2	A	2RC-01-BA/SGC-02	C1.30	Stub Barrel Weld
2	B	2RC-01-BB/SGC-01	B2.40	Stub Barrel - Channel Head
2	B	2RC-01-BB/SGC-03	C1.10	Stub Barrel - Lower Barrel “A”
2	B	2RC-01-BB/SGC-05	C1.10	Lower Barrel “B” - Transition Cone
2	B	2RC-01-BB/SGC-06	C1.10	Transition Cone - Upper Barrel “B”
2	B	2RC-01-BB/SGC-08	C1.20	Upper Barrel “B” - Upper Head
2	B	2RC-01-BB/SGC-02	C1.30	Stub Barrel Weld
2	C	2RC-01-BC/SGC-01	B2.40	Stub Barrel - Channel Head
2	C	2RC-01-BC/SGC-03	C1.10	Stub Barrel - Lower Barrel “A”
2	C	2RC-01-BC/SGC-05	C1.10	Lower Barrel “B” - Transition Cone
2	C	2RC-01-BC/SGC-06	C1.10	Transition Cone - Upper Barrel “B”
2	C	2RC-01-BC/SGC-08	C1.20	Upper Barrel “B” - Upper Head
2	C	2RC-01-BC/SGC-02	C1.30	Stub Barrel Weld
2	D	2RC-01-BD/SGC-01	B2.40	Stub Barrel - Channel Head
2	D	2RC-01-BD/SGC-03	C1.10	Stub Barrel - Lower Barrel “A”
2	D	2RC-01-BD/SGC-05	C1.10	Lower Barrel “B” - Transition Cone
2	D	2RC-01-BD/SGC-06	C1.10	Transition Cone - Upper Barrel “B”
2	D	2RC-01-BD/SGC-08	C1.20	Upper Barrel “B” - Upper Head
2	D	2RC-01-BD/SGC-02	C1.30	Stub Barrel Weld

*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	11-W22	B2.40	Tubesheet to Primary Head
1	11	SG-11- W5	B3.130	11 Primary Outlet Nozzle to Primary Head (Hot Leg)
1	11	SG-11- W6	B3.130	11A Primary Head to Cold Leg A Nozzle
1	11	SG-11- W7	B3.130	11B Primary Head to Cold Leg B Nozzle
1	11	SG-11-4	C1.10	Upper Assembly to Transition Assembly
1	11	SG-11-3	C1.20	Upper Head to Upper Assembly
1	11	SG-11 - W65	C1.30	Shell Can #1 to Tubesheet
1	12	12-W22	B2.40	Tubesheet to Primary Head
1	12	SG-12 - W5	B3.130	12 Primary Outlet Nozzle to Primary Head (Hot Leg)
1	12	SG-12 - W6	B3.130	12A Primary Head to Cold Leg A Nozzle
1	12	SG-12 - W7	B3.130	12B Primary Head to Cold Leg B Nozzle
1	12	SG-12-4	C1.10	Upper Assembly to Transition Assembly
1	12	SG-12-3	C1.20	Upper Head to Upper Assembly
1	12	SG-12 - W65	C1.30	Shell Can #1 to Tubesheet
2	21	21-W22	B2.40	Tubesheet to Primary Head
2	21	SG-21 - W5	B3.130	21 Primary Outlet Nozzle to Primary Head (Hot Leg)
2	21	SG-21 - W6	B3.130	21A Primary Head to Cold Leg A Nozzle
2	21	SG-21 - W7	B3.130	21B Primary Head to Cold Leg B Nozzle
2	21	SG-21-4	C1.10	Upper Assembly to Transition Assembly
2	21	SG-21-3	C1.20	Upper Head to Upper Assembly
2	21	SG-21- W65	C1.30	Shell Can #1 to Tubesheet
2	22	22-W22	B2.40	Tubesheet to Primary Head
2	22	SG-22 - W5	B3.130	22 Primary Outlet Nozzle to Primary Head (Hot Leg)
2	22	SG-22 - W6	B3.130	22A Primary Head to Cold Leg A Nozzle
2	22	SG-22 - W7	B3.130	22B Primary Head to Cold Leg B Nozzle
2	22	SG-22-4	C1.10	Upper Assembly to Transition Assembly
2	22	SG-22-3	C1.20	Upper Head to Upper Assembly
2	22	SG-22 -W65	C1.30	Shell Can #1 to Tubesheet

* Partial replacements of the Unit 1 Steam Generators were performed during the 2002 outage. Partial replacements of the U2 Steam Generators were performed during the 2003 outage. The steam drum, which includes the Upper Assembly to Transition Assembly and Upper Head to Upper Assembly welds were retained and refurbished as part of the replacements. Therefore, these welds were retained. Examination records prior to the Steam Generator replacements could not be located; however, two inspections intervals of examinations have been performed since the Steam Generator replacements with no unacceptable indications identified.

Ginna Component IDs:

Unit	SG	Component ID	Item Number	Description
1	A	LHTSW-AR	B2.40	Lower Head to Tubesheet Weld
1	A	LST-AR	C1.10	Lower Shell to Transition Circ Weld
1	A	TUS-AR	C1.10	Transition to Upper Shell Circ Weld
1	A	USH-AR	C1.20	Upper Shell to Head Circ Weld
1	A	TSSW-AR	C1.30	Tubesheet to Shell Weld
1	B	LHTSW-BR	B2.40	Lower Head to Tubesheet Weld
1	B	LST-BR	C1.10	Lower Shell to Transition Circ Weld
1	B	TUS-BR	C1.10	Transition to Upper Shell Circ Weld
1	B	USH-BR	C1.20	Upper Shell to Head Circ Weld
1	B	TSSW-BR	C1.30	Tubesheet to Shell Weld

*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)
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 Register, Vol. 86, No. 57. 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 REQUIREMENT:

The 2007 Edition with the 2008 Addenda and 2013 Edition of ASME Code, Section XI (ASME Section XI), IWB-2500(a), Table IWB-2500-1, Examination Categories B-B and B-D, and IWC-2500(a), Table IWC-2500-1, Examination Category C-A, require examination of the applicable Item Numbers:

Item Number B2.40 - Volumetric examination of essentially 100% of the weld length. The examination may be limited to one vessel among the group of vessels performing a similar function during successive inspection intervals. The examination volume is shown in Figure IWB-2500-6.

Item Number B3.130 - Volumetric examination of all nozzles during each inspection interval. The examination volume is shown in Figures IWB-2500-7(a), (b), (c) or (d).

Item Number C1.10 - Volumetric examination of essentially 100% of the weld length of the cylindrical-shell-to-conical shell-junction welds and shell (or head)-to-flange welds during each inspection interval. In the case of multiple vessels of similar design, size, and service (such as steam generators), the required examinations may be limited to one vessel or distributed among the vessels. The examination volume is shown in Figure IWC-2500-1.

Item Number C1.20 - Volumetric examination of essentially 100% of the weld length of the head-to-shell weld during each inspection interval. In the case of multiple vessels of similar design, size, and service (such as steam generators), the required examinations may be limited to one vessel or distributed among the vessels. The examination volume is shown in Figure IWC-2500-1.

Item Number C1.30 - Volumetric examination of essentially 100% of the weld length of the tubesheet-to-shell weld during each inspection interval. In the case of multiple vessels of similar design, size, and service (such as steam generators), the required examinations may be limited to one vessel or distributed among the vessels. The examination volume is shown in Figure IWC-2500-2.

4 REASON FOR REQUEST:

The Electric Power Research Institute (EPRI) performed assessments in Reference [1] of the basis for the ASME Section XI examination requirements specified for the above-listed ASME Section XI Examination Categories and Item Numbers for Steam Generator (SG) welds and components. The assessments include a survey of inspection results from 74 domestic and international nuclear units and 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 Reference [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, IWB-2500(a), Table IWB-2500-1, Examination Categories B-B and B-D, and IWC-2500(a), Table IWC-2500-1, Examination Category C-A. 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	B-B	B2.40	Steam Generator (Primary Side), Tubesheet-To-Head Weld	10/8/2010	10/17/2046	36.0
		C-A	C1.10	Steam Generator (Secondary Side), Shell Circumferential Welds	4/19/2018		28.5
		C-A	C1.20	Steam Generator (Secondary Side), Head Circumferential Welds	4/8/2015		31.5
		C-A	C1.30	Steam Generator (Secondary Side), Tubesheet-to-Shell Weld	10/8/2010		36.0
Braidwood	2	B-B	B2.40	Steam Generator (Primary Side), Tubesheet-To-Head Weld	10/26/2012	12/18/2047	35.2
		C-A	C1.10	Steam Generator (Secondary Side), Shell Circumferential Welds	10/28/2021		26.2
		C-A	C1.20	Steam Generator (Secondary Side), Head Circumferential Welds	5/10/2014		33.6
		C-A	C1.30	Steam Generator (Secondary Side), Tubesheet-to-Shell Weld	10/31/2018		29.2
Byron	1	B-B	B2.40	Steam Generator (Primary Side), Tubesheet-To-Head Weld	3/2/2017	10/31/2044	27.7
		C-A	C1.10	Steam Generator (Secondary Side), Shell Circumferential Welds	3/20/2014		30.6
		C-A	C1.20	Steam Generator (Secondary Side), Head Circumferential Welds	3/15/2020		24.6
		C-A	C1.30	Steam Generator (Secondary Side), Tubesheet-to-Shell Weld	9/13/2018		26.2
Byron	2	B-B	B2.40	Steam Generator (Primary Side), Tubesheet-To-Head Weld	4/13/2019	11/6/2046	27.6
		C-A	C1.10	Steam Generator (Secondary Side), Shell Circumferential Welds	4/14/2013		33.6
		C-A	C1.20	Steam Generator (Secondary Side), Head Circumferential Welds	4/18/2013		33.6
		C-A	C1.30	Steam Generator (Secondary Side), Tubesheet-to-Shell Weld	4/13/2019		27.6

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)
Calvert Cliffs	1	B-B	B2.40	Steam Generator (Primary Side), Tubesheet-To-Head Weld	2/25/2014	7/31/2034	20.4
		B-D	B3.130	Steam Generator (Primary Side), Nozzle-to-Vessel Welds	2/27/2018		16.4
		C-A	C1.10	Steam Generator (Secondary Side), Shell Circumferential Welds	3/4/2018		16.4
		C-A	C1.20	Steam Generator (Secondary Side), Head Circumferential Welds	3/3/2018		16.4
		C-A	C1.30	Steam Generator (Secondary Side), Tubesheet-to-Shell Weld	3/2/2018		16.4
Calvert Cliffs	2	B-B	B2.40	Steam Generator (Primary Side), Tubesheet-To-Head Weld	2/27/2013	8/13/2036	23.5
		B-D	B3.130	Steam Generator (Primary Side), Nozzle-to-Vessel Welds	3/1/2013		23.5
		C-A	C1.10	Steam Generator (Secondary Side), Shell Circumferential Welds	2/20/2011		25.5
		C-A	C1.20	Steam Generator (Secondary Side), Head Circumferential Welds	2/26/2019		17.5
		C-A	C1.30	Steam Generator (Secondary Side), Tubesheet-to-Shell Weld	2/20/2017		19.5
Ginna	1	B-B	B2.40	Steam Generator (Primary Side), Tubesheet-To-Head Weld	10/25/2015	9/18/2029	13.9
		C-A	C1.10	Steam Generator (Secondary Side), Shell Circumferential Welds	5/1/2017		12.4
		C-A	C1.20	Steam Generator (Secondary Side), Head Circumferential Welds	10/31/2012		16.9
		C-A	C1.30	Steam Generator (Secondary Side), Tubesheet-to-Shell Weld	10/25/2015		13.9

As indicated in Table 1, the proposed alternative results in a maximum effective operating period of 36 years from the last inspection for Item Numbers B2.40 and C1.30 for Braidwood Unit 1 included in this Proposed Alternative. As summarized in Attachment 1, the EPRI report demonstrates that after preservice inspection (PSI) and two 10-year intervals, 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 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 pressure retaining welds and full penetration welded nozzles was performed in Reference [1]. Evaluated mechanisms included 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 pressure retaining welds and full penetration welded nozzles.

Stress Analysis

Finite element analysis (FEA) was performed in Reference [1] to determine the stresses in the SG pressure retaining welds and full penetration welded nozzles. The analysis was performed using representative pressurized water reactor (PWR) geometries, bounding transients, and typical material properties.

As stated in Section 9.0 of Reference [1], the operating transients and associated cycles evaluated are typical of historical PWR operations, although they can vary depending on the design type as well as within each design type. In the fracture mechanics evaluations performed in Reference [1], the transients that contribute the most to crack growth are the heatup and cooldown events. A total of 300 such events were evaluated during a 60-year plant life. Reference [1] also used a conservative rate of 200°F per hour which would exceed the limits allowed by the Braidwood, Byron, Calvert Cliffs, and Ginna technical specifications. The sensitivity study performed in Section 8.0 of Reference [1] shows that, for the realistic ISI scenario of PSI followed by ISI at 20, 40, and 60 years, a stress multiplier of 1.9 can accommodate small variations in the pressures and temperatures of the evaluated transients. Due to the small variations from the values from Reference [1], the conservative definition of the heatup and cooldown transient, and the low number of transient cycles predicted in Appendix A over the life of the plant vs. the number of cycles evaluated in Reference [1], it is determined that the stress analysis performed in Reference [1] is applicable to Braidwood, Units 1 and 2, Byron, Units 1 and 2, Calvert Cliffs, Units 1 and 2, and Ginna.

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 PSI followed by two 10-year intervals, 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 two 10-year interval inspections for welds W5, W6, W7, W22, and W65. For Calvert Cliffs, Units 1 and 2, PSI has been followed by the performance of four 10-year interval inspections for welds W3 and W4, but records were only found for the Third and Fourth Interval examinations. 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 10-year intervals, 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 "PSI + 20 + 40 + 60" case has similar Probability of Failure results, as shown in Table 8-10 of the EPRI report. The Probability of Failure results would be larger for the limiting "PSI + 10 + 20 + 50" case than the "PSI + 20 + 40 + 60" case, but they would be of the same magnitude. The Probability of Failure results in Table 8-9 for the "PSI only" case and Table 8-10 for the "PSI + 20 + 40 + 60" case 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 8.3.4.3 of the EPRI report. Therefore, results of the sensitivity studies should be applied to the limiting "PSI + 10 + 20 + 50" case requested in this proposed alternative.

A sensitivity study was performed on the effects of various 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. and the stress multiplier was changed to 1.9. The limiting case for the sensitivity study, that is applicable to this relief request, was found to be the primary tubesheet-to-shell weld with a probability of rupture of $7.33x10^{-7}$, for the "PSI + 20 + 40 + 60" case, which is below the 1x10-6 per year acceptance criteria. All other locations reviewed by the sensitivity study, applicable to this relief request, were below the $1x10^{-6}$ per year acceptance criteria.

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 Steam Generator welds 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, Braidwood and Calvert Cliffs recorded examinations with limited coverage for the applicable components. The lowest recorded coverage for Braidwood was 88.4% for the Item Number C1.30 (tubesheet-to-shell welds), and for Calvert Cliffs the lowest coverage was 73.8% for the Item Number B3.130 (nozzle-to-vessel welds). Based on the sensitivity study results presented in Table 8-33 of Reference [1], these coverage values meet the Probability of Rupture acceptance criterion for all locations when the life of the component reaches 60 years. Also, as shown in Appendix B, Byron was the only plant to identify recordable indications that exceeded the ASME Section XI acceptance standards. Welds 2RC-01-BB/SGC-03, 2RC-01-BC/SGC-05, and 2RC-01-BD/SGC-06 were found to have embedded indications during the PSI examinations exceeding the ASME Section XI acceptance standards. Relief to accept these flaws was approved through an NRC safety evaluation on 10/29/1986. Successive examinations were performed for each weld and the embedded indications were either found to have no change in sizing or were found to be non-recordable due to improved inspection methods during later ISI Intervals. Weld 2RC-01-BA/SGC-02 was also found to have 4 embedded indications during the 1st ISI Interval examination. Successive examinations were performed and the embedded indications were either found to be non-relevant or did not change in size during the latest 4th ISI Interval examination.

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 pressure retaining welds and full penetration welded nozzles 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 10^{-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 Categories B-B, B-D, and C-A, Item Numbers B2.40, B3.130, C1.10, C1.20, and C1.30 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 110 inspections of SG pressure retaining welds and full penetration welded nozzles at Braidwood, Units 1 and 2, Byron, Units 1 and 2, Calvert Cliffs, Units 1 and 2, and Ginna. The flaws exceeding the ASME Section XI acceptance standards were fabrication flaws detected during PSI and are documented in Appendix B. Successive examinations have been performed for all recordable indications and there has been no increase in flaw sizing or the flaws are no longer recordable due to improved inspection methods. The lowest inspection coverage listed in Appendix B was 73.8%. Section 8.3.5 of Reference [1] discusses limited coverage and determines that the conclusions of the report are applicable to components with limited coverage greater than 50%. In addition, it is important to note all other inspection activities, including the system leakage test (Examination Categories B-P and C-H) conducted each outage or each ISI period, as applicable, will continue to be performed providing further assurance of safety.

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 PRECEDENT:

To-date, there has been one previous submittal that request relief from the ASME Examination Categories B-B, B-D, and C-A, Item Numbers B2.31, B2.32, B2.40, B3.130, C1.10, C1.20, and C1.30 volumetric examinations on the basis of the Reference [1] technical basis:

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

The following is a list of approved Request for Alternatives related to inspections of SG welds and components:

- Letter from N. L. Salgado (NRC) to B. C. Hanson (Exelon Generation Company, LLC), “Braidwood Station, Units 1 and 2 – Relief from the Requirements of the ASME Code (EPID L-2019-LLR-0081),” May 14, 2020, ADAMS Accession No. ML20133K093.

- Letter from J. G. Danna (NRC) to D. P. Rhoades (Exelon Generation Company, LLC), “Calvert Cliffs Nuclear Power Plant, Units 1 and 2 – Relief from the Requirements of the ASME Code Concerning Volumetric or Surface Examination Coverage for the Subject Welds (EPID L-2020-LLR-0089),” January 15, 2021, ADAMS Accession No. ML20356A121.

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

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 Class 1 Nozzle-to-Vessel Welds and Class 1 and Class 2 Vessel Head, Shell, Tubesheet-to-Head and Tubesheet-to-Shell Welds*. EPRI, Palo Alto, CA: 2019. 3002015906, ADAMS Accession No. ML20225A141.
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. 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.
8. *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.
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," March 2020.

12. 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.

13. 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.

14. U.S. NRC Regulatory Federal Register Volume 86, Issue 57, March 26, 2021.

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 Tables A1 through A4.

Tables A1 through A4 indicate 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
<u>Items Number B2.40 (SG Primary Side Shell Welds)</u>

<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>General Requirements</td><td>The Loss of Power transient (involving unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of portion of the vessel) is not considered in this evaluation due to its rarity. In the event that such a significant thermal event occurs at a plant, its impact on the K_{IC} (material fracture toughness) value may require more frequent examinations and other plant actions outside the scope of this report's guidance.</td><td colspan="2">Braidwood Station, Units 1 and 2 has not experienced a loss of power transient resulting in unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of portion of the vessel [A8].</td></tr>
</table>

Category	Requirement from Reference [A1]	Applicability to Braidwood, Unit 1	Applicability to Braidwood, Unit 2
	The materials of the SG shell must be low alloy ferritic steels which conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G- 2110.	The Braidwood Station, Unit 1, SG vessel primary head and tubesheet are fabricated of SA-508, Class 3 material. This material conforms to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.	The Braidwood Station, Unit 2 SG channel head is fabricated of SA-216, Grade WCC material, and the SG tube plate is fabricated from SA-508, Class 2a material. Material SA-508, Class 2a conforms to the requirements of ASME Code, Section XI, Appendix G, Paragraph G- 2110. SA-216 Gr. WCC material is not specifically listed in ASME Code, Section XI, Appendix G. However, SA-216 Grade WCC material has similar toughness and chemical composition to SA-533, Grade B, Class 1, SA-508-1, SA-508-2, and SA-508-3, it has a specified minimum yield strength at room temperature of 40 ksi (which is less than 50 ksi), and the RT_{NDT} values for the Braidwood Unit 2 steam generators channel head material was required to be 60°F or less (so the RT_{NDT} of 60°F used in the EPRI report is bounding). Also, thermal transient stresses in SA-216 Grade WCC material are lower compared to the SA-533 Grade B Class 1 material used in the FEA in the EPRI report. Therefore, by comparison, SA-216 Gr. WCC material is consistent with the requirements of ASME Code, Section XI, Appendix G and satisfies the requirements for application of the EPRI report.

Category	Requirement from Reference [A1]	Applicability to Braidwood, Unit 1	Applicability to Braidwood, Unit 2
Specific Requirements	The weld configurations must conform to those shown in Figures 1-1 and Figure 1-2 of Reference [A1].	The Braidwood Station, Unit 1 weld configuration is shown in Figure A1 and conforms to Figure 1-2 of Reference [A1].	The Braidwood Station, Unit 2 weld configuration is shown in Figure A1 and conforms to Figure 1-2 of Reference [A1].
	The SG vessel dimensions must be within 10% percent of the upper and lower bounds of the values provided in the table in Section 9.4.3 of Reference [A1].	The Braidwood Station, Unit 1 SG vessel dimensions are as follows: • SG Lower Head diameter = 136.88 inches • SG Upper Shell diameter = 176.75 inches These dimensions are within 10% of those specified in Table 9-2 in Section 9.4.3 of Reference [A1] for Westinghouse plants.	The Braidwood Station, Unit 2 SG vessel dimensions are as follows: • SG Lower Head diameter = 135.38 inches • SG Upper Shell diameter = 176.25 inches These dimensions are within 10% of those specified in Table 9-2 in Section 9.4.3 of Reference [A1] for Westinghouse plants.
	The component must experience transients and cycles bounded by those shown in Table 5- 7 of Reference [A1] over a 60-year operating life.	As shown in Table A3, the Braidwood Station, Units 1 and 2 transients and number of cycles projected to occur over a 60-year life are bounded by those shown in Table 5-7 of Reference [A1] for Westinghouse plants.	

Table A2. Applicability of Reference [A1] Representative Analyses to Braidwood Station, Units 1 and 2

Items Numbers C1.10, C1.20 and C1.30 (SG Secondary Side Shell Welds)

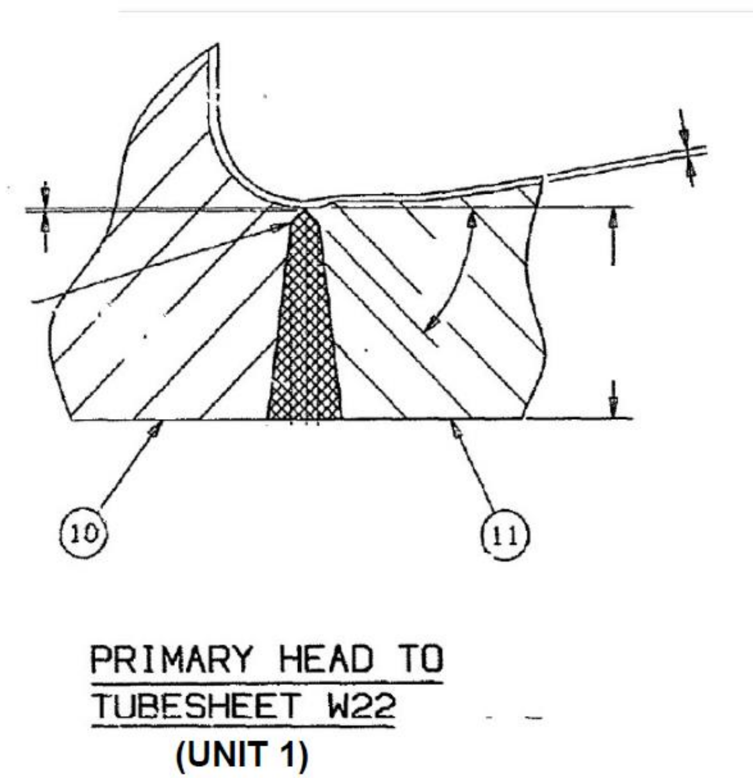
Category	Requirement from Reference [A1]	Applicability to Braidwood, Unit 1	Applicability to Braidwood, Unit 2
General Requirements	The Loss of Power transient (involving unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of portion of the vessel) is not considered in this evaluation due to its rarity. In the event that such a significant thermal event occurs at a plant, its impact on the K_{IC} *(material fracture toughness)* value may require more frequent examinations and other plant actions outside the scope of this report's guidance.	Braidwood Station, Units 1 and 2 has not experienced a loss of power transient resulting in unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of portion of the vessel.	

Category	Requirement from Reference [A1]	Applicability to Braidwood, Unit 1	Applicability to Braidwood, 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 Braidwood Station, Unit 1 SG tubesheet, shell cone, and steam drum head are fabricated of SA-508, Class 3 material, and the SG 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 Braidwood Station, Unit 2, SG tube plate is fabricated from SA-508, Class 2a material and the SG barrels, transition cone, and upper head are fabricated from SA-533, Grade A, Class 2. Material SA-508, Class 2a 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.

Category	Requirement from Reference [A1]	Applicability to Braidwood, Unit 1	Applicability to Braidwood, Unit 2
Specific Requirements	The weld configurations must conform to those shown in Figure 1-7 and Figure 1-8 of Reference [A1].	The Braidwood Station, Unit 1 weld configurations are shown in Figure A2 and conform to Figures 1-7 and 1-8 of Reference [A1].	The Braidwood Station, Unit 2 weld configurations are shown in Figure A3 and conforms to Figures 1-7 and 1-8 of Reference [A1].
	The SG vessel dimensions must be within 10% percent of the upper and lower bounds of the values provided in the table in Section 9.4.4 of Reference [A1].	The Braidwood Station, Unit 1, SG vessel dimensions are as follows: • SG Lower Head diameter = 136.88 inches • SG Upper Shell diameter = 176.75 inches These dimensions are within 10% of those specified in Table 9-3 in Section 9.4.4 of Reference [A1] for Westinghouse plants.	The Braidwood Station, Unit 2 SG vessel dimensions are as follows: • SG Lower Head diameter = 135.38 inches • SG Upper Shell diameter = 176.25 inches These dimensions are within 10% of those specified in Table 9-3 in Section 9.4.4 of Reference [A1] for Westinghouse plants.
	The component must experience transients and cycles bounded by those shown in Table 5-9 of Reference [A1] over a 60-year operating life.	As shown in Table A4, the Braidwood Station, Units 1 and 2, transients and number of cycles projected to occur over a 60-year life are bounded by those shown in Table 5-9 of Reference [A1].	

REFERENCES:

A1. *Technical Bases for Inspection Requirements for PWR Steam Generator Class 1 Nozzle-to-Vessel Welds and Class 1 and Class 2 Vessel Head, Shell, Tubesheet-to-Head and Tubesheet-to-Shell Welds*. EPRI, Palo Alto, CA: 2019. 3002015906, ADAMS Accession No. ML20225A141.
A2. Braidwood, Unit 1, Drawing 7720E038, *Section 'XI' Preservice U/T*.
A3. Braidwood, Unit 2, Drawing 1103J99, Sheet 1, *Steam Generator General Arrangement D5-3*.
A4. Braidwood, Unit 1, Drawing 1SG-05, *Inservice Inspection Drawing for Replacement Steam Generator 1RC01BA – Loop 1, Unit 1*.
A5. Braidwood, Unit 2, Drawing 2SG-01, *Spec L2907, Inspection Identification Drawing for Inservice Inspection of Steam Generator 2RC01BA – Loop #1, Unit 2*.
A6. Braidwood, Unit 1, Design Specification 18-1229648, *Certified Design Specification for Replacement Steam Generator, Byron and Braidwood Stations Unit 1*.
A7. Braidwood, Unit 2, Design Specification G-953431, *Reactor Coolant System - Model D-5 Steam Generator*.
A8. AREVA Specification 18-1229648-009 (MUR), Certified Design Specification for Replacement Steam Generator Byron and Braidwood Stations Unit 1.
A9. Framatone 32-500-7271-00, *Byron/Braidwood RSG Design Transients – Power Uprate*.
A10. WCAP-11388, Volume 2, Byron Units 1 and 2 and Braidwood Units 1 and 2, *T-Hot Reduction Program Engineering Report*.
A11. Design Specification 953442, Rev 8, *Model D5 Steam Generator (SG) Stress Report*.
A12. CN-OPES(99)-061, Revision 0, Power Uprate – *Design Transient Revisions for BY/BR Uprating*.



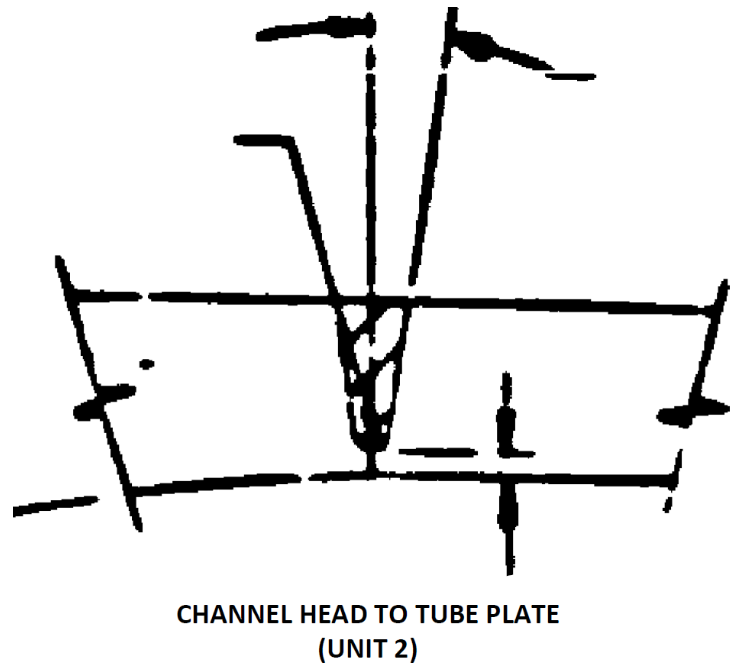
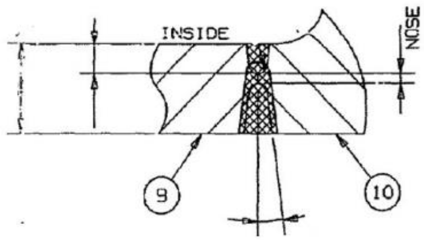
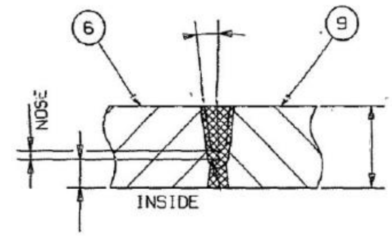


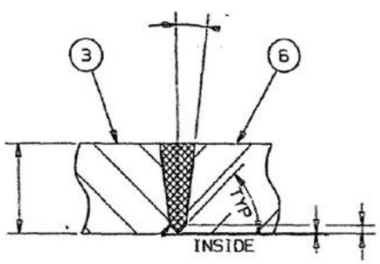
Figure A1. Braidwood Station, Units 1 and 2 SG Primary Side Head to Tubesheet Weld Configuration



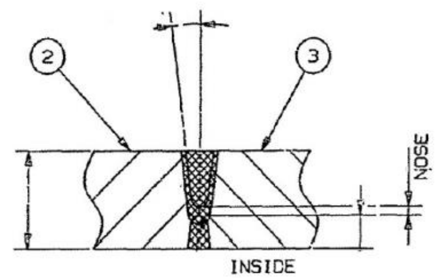
SECONDARY SHELL TO
TUBESHEET W65



SECONDARY SHELL TO
SHELL CONE W80

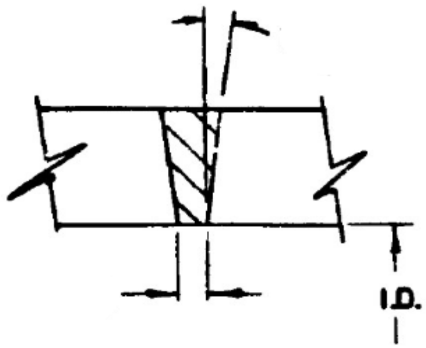


STEAM DRUM TO
SHELL CONE W142

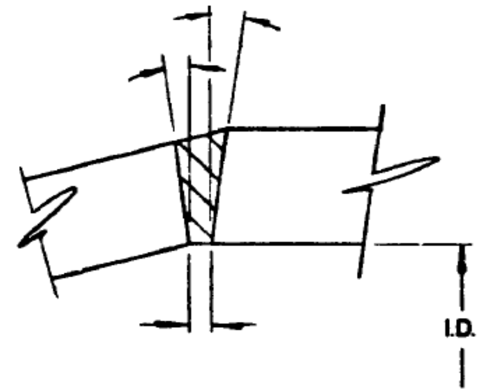


STEAM DRUM HEAD TO
STEAM DRUM W144

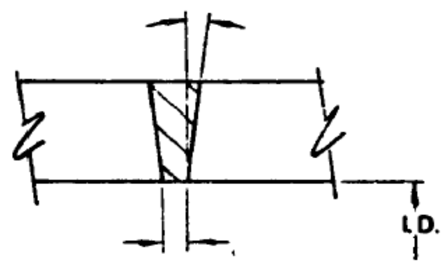
Figure A2. Braidwood Station, Unit 1, SG Secondary Side Shell Weld Configurations



TUBE PLATE TO STUB BARREL,
STUB BARREL TO LOWER SHELL BARREL, AND
LOWER SHELL BARREL TO TRANSITION CONE



TRANSITION CONE TO UPPER SHELL BARREL



UPPER STEAM BARREL SHELL TO UPPER HEAD

Figure A3. Braidwood Station, Unit 2 SG Secondary Side Shell Weld Configurations

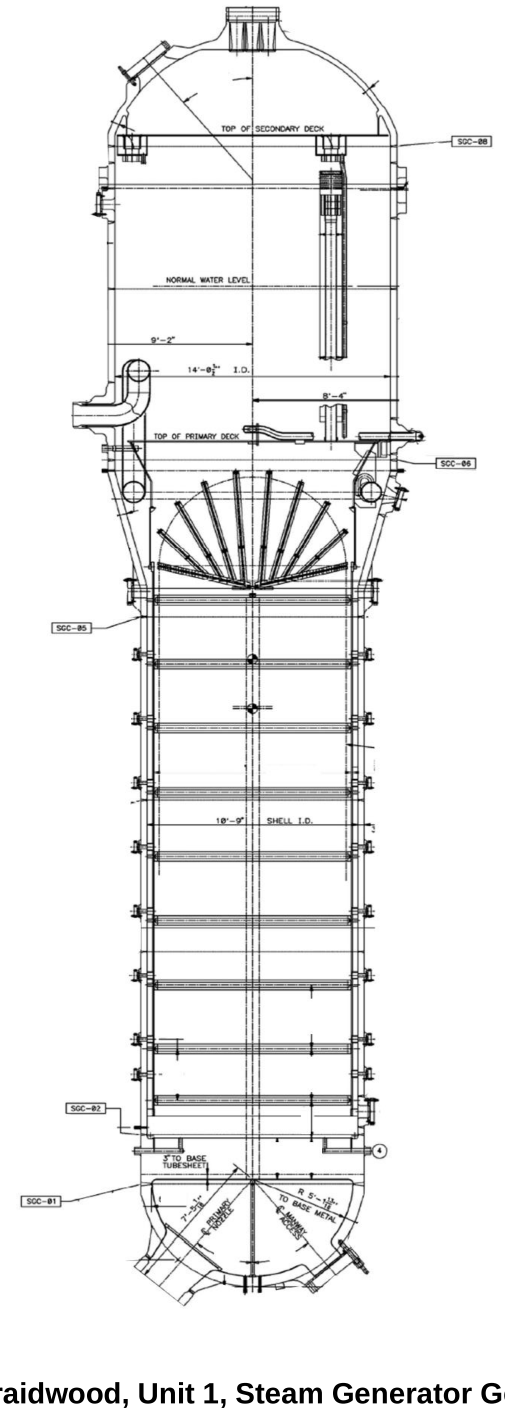


Figure A4. Braidwood, Unit 1, Steam Generator General Outline

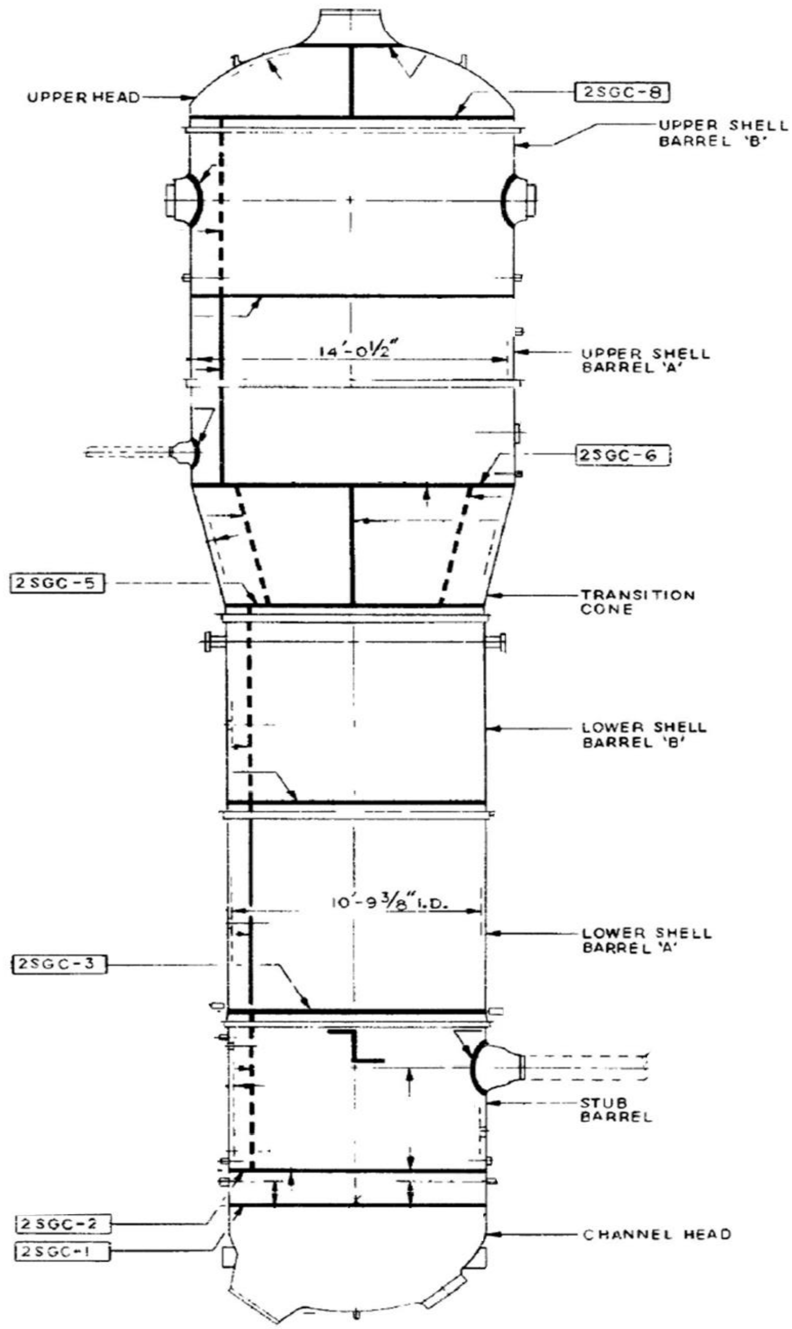


Figure A5. Braidwood, Unit 2, Steam Generator General Outline

Table A3. Transient Cycles for PWR SG Primary-Side Head Welds for Braidwood Station, Units 1 and 2 in Comparison to Table 5-7 of Reference [A1]

Transient	Maximum T_{HOT} (°F)	Minimum T_{HOT} (°F)	Maximum T_{COLD} (°F)	Minimum T_{COLD} (°F)	Maximum Pressure (psig)	Minimum Pressure (psig)	60-Year Cycles
Heatup/Cooldown From Reference [A1]	545	70	545	70	2235	0	300
Heatup/Cooldown for Braidwood Station, Units 1 and 2 [A10, A12]	557[3]	120	557[3]	120	2250[3]	467	72/70 [Unit 1] 79/94 [Unit 2]
Plant Loading/ Unloading from Reference [A1]	610	550	550	545	2300	2300	5000
Plant Loading/ Unloading for Braidwood Station, Units 1 and 2 [A10, A12]	618.4[3]	566.5	558.1[3]	538.2[3]	2370[3]	2295[3]	266[1]/218[2] [Unit 1] 220[1]/186[2] [Unit 2]
Reactor Trip from Reference [A1]	615	530	565	530	2435	1700	360
Reactor Trip for Braidwood Station, Units 1 and 2 [A10, A12]	618.4[3]	435[3]	555.2	538.2	2250	1582[3]	11 [Unit 1] 69 [Unit 2]

Note:

1. 60-Year 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. 60-Year Cycle counts for Plant Unloading conservatively includes all events for transients “Plant Unloading at 5%/min” and “Plant Unloading, 15% - 0%”.
3. See the Stress Analysis discussion in Section 5 for a basis to the applicability of the EPRI report for transients having a maximum pressure/temperature higher than what is listed in the EPRI Report or a minimum pressure/temperature lower than what is listed in the EPRI Report.

Table A4. Transient Cycles for PWR SG Secondary-Side Vessel Welds for Braidwood Station, Units 1 and 2 in Comparison to Table 5-9 of Reference [A1]

Transient	Maximum T_{ss} (°F)	Minimum T_{ss} (°F)	Maximum Pressure (psig)	Minimum Pressure (psig)	60-Year Cycles
Heatup/Cooldown from Reference [A1]	545	70	1000	0	300
Heatup/Cooldown for Braidwood Station, Units 1 and 2 [A8, A10]	557[3]	120	1077[3]	0	72/70 [Unit 1] 79/94 [Unit 2]
Plant Loading/ Unloading from Reference [A1]	545	540	1000	1000	5000
Plant Loading/ Unloading for Braidwood Station, Units 1 and 2 [A8, A9]	555.5 [Unit 1][3] 554.3 [Unit 2][3]	528.8 [Unit 1][3] 524.4 [Unit 2][3]	1077 [Unit 1][3,4] 1077 [Unit 2][3,4]	1077 [Unit 1][4] 1077 [Unit 2][4]	266[1]/218[2] [Unit 1] 220[1]/186[2] [Unit 2]
Reactor Trip from Reference [A1]	555	530	1130	1000	360
Reactor Trip for Braidwood Station, Units 1 and 2 [A8, A9]	557.9 [Unit 1][3] 553.8 [Unit 2]	460.9 [Unit 1][3] 458.8 [Unit 2][3]	1092 [Unit 1] 1092 [Unit 2][5]	995 [Unit 1][3,6] 860 [Unit 2][3,6]	11 [Unit 1] 69 [Unit 2]

Note:

1. 60-Year Cycle counts for Plant Loading conservatively includes all events for transients “Plant Loading at 5%/min” and “Plant Loading, 0% - 15%”.
2. 60-Year Cycle counts for Plant Unloading conservatively includes all events for transients “Plant Unloading at 5%/min” and “Plant Unloading, 15% - 0%”.
3. See the Stress Analysis discussion in Section 5 for a basis to the applicability of the EPRI report for transients having a maximum pressure/temperature higher than what is listed in the EPRI Report or a minimum pressure/temperature lower than what is listed in the EPRI Report.
4. Maximum and Minimum Pressure values for the Plant Loading/Unloading Transient could not be located in design documentation for the Secondary Side of the Vessel. Since the Maximum and Minimum T_{SS} for the Heatup/Cooldown Transient are similar to the Loading/Unloading Transient, the same pressure was assumed. This is the basis for the 1077 psig values listed in the table.
5. Maximum Pressure for the Reactor Trip transient for Unit 2 could not be located in design documentation. Due to the similar Maximum T_{ss} values for the Reactor Trip Transient between Units 1 and 2, the same Maximum Pressure was assumed for Unit 2.
6. Minimum Pressure for the Reactor Trip transient for Unit 1 and 2 could not be located in design documentation. Plant operating data for Byron and Braidwood was reviewed, and the minimum pressure corresponding to 100% reactor power was used. This pressure corresponds to the minimum pressure when a reactor trip transient would initiate at 100% power. Due to similarities in SG design and operational strategies between the stations, the same values were used for Braidwood Unit 1 and Byron Unit 1 and also for Braidwood Unit 2 and Byron Unit 2.

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 Tables A5 through A8.

Tables A5 through A8 indicate 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 A5. Applicability of Reference [A13] Representative Analyses to Byron, Units 1 and 2
<u>Items Number B2.40 (SG Primary Side Shell Welds)</u>

<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>General Requirements</td><td>The Loss of Power transient (involving unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of portion of the vessel) is not considered in this evaluation due to its rarity. In the event that such a significant thermal event occurs at a plant, its impact on the K_{IC} (material fracture toughness) value may require more frequent examinations and other plant actions outside the scope of this report's guidance.</td><td colspan="2">Byron Station, Units 1 and 2 has not experienced a loss of power transient resulting in unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of portion of the vessel.</td></tr>
</table>

	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, SG vessel primary head and tubesheet are fabricated of SA-508, Class 3 material. This material conforms to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.	The Byron Station, Unit 2 SG channel head is fabricated of SA-216, Grade WCC, and the SG tube plate is fabricated from SA-508, Class 2a material. Material SA-508, Class 2a conforms to the requirements of ASME Code, Section XI, Appendix G, Paragraph G- 2110. SA-216 Gr. WCC material is not specifically listed in ASME Code, Section XI, Appendix G. However, SA-216 Grade WCC material has similar toughness and chemical composition to SA-533, Grade B, Class 1, SA-508-1, SA-508-2, and SA-508-3, it has a specified minimum yield strength at room temperature of 40 ksi (which is less than 50 ksi), and the RT_{NDT} values for the Byron Unit 2 steam generators channel head material was required to be 60°F or less (so the RT_{NDT} of 60°F used in the EPRI report is bounding). Also, thermal transient stresses in SA-216 Grade WCC material are lower compared to the SA-533 Grade B Class 1 material used in the FEA in the EPRI report. Therefore, by comparison, SA-216 Gr. WCC material is consistent with the requirements of ASME Code, Section XI, Appendix G and satisfies the requirements for application of the EPRI report.
Specific Requirements	The weld configurations must	The Byron Station, Unit 1 weld	The Byron Station, Unit 2 weld configuration is

Category	Requirement from Reference [A13]	Applicability to Byron, Unit 1	Applicability to Byron, Unit 2
	conform to those shown in Figures 1-1 and Figure 1-2 of Reference [A13].	configuration is shown in Figure A6 and conforms to Figure 1-2 of Reference [A13].	shown in Figure A6 and conforms to Figure 1-2 of Reference [A13].
	The SG vessel dimensions must be within 10% percent of the upper and lower bounds of the values provided in the table in Section 9.4.3 of Reference [A13].	The Byron Station, Unit 1 SG vessel dimensions are as follows: • SG Lower Head diameter = 136.88 inches • SG Upper Shell diameter = 176.75 inches These dimensions are within 10% of those specified in Table 9-2 in Section 9.4.3 of Reference [A13] for Westinghouse plants.	The Byron Station, Unit 2 SG vessel dimensions are as follows: • SG Lower Head diameter = 135.58 inches • SG Upper Shell diameter = 176.5 inches These dimensions are within 10% of those specified in Table 9-2 in Section 9.4.3 of Reference [A13] for Westinghouse plants.
	The component must experience transients and cycles bounded by those shown in Table 5- 7 of Reference [A13] over a 60-year operating life.	As shown in Table A7, the Byron Station, Units 1 and 2 transients and number of cycles projected to occur over a 60-year life are bounded by those shown in Table 5-7 of Reference [A13] for Westinghouse plants.	

Table A6. Applicability of Reference [A13] Representative Analyses to Byron, Units 1 and 2
Items Numbers C1.10, C1.20 and C1.30 (SG Secondary Side Shell Welds)

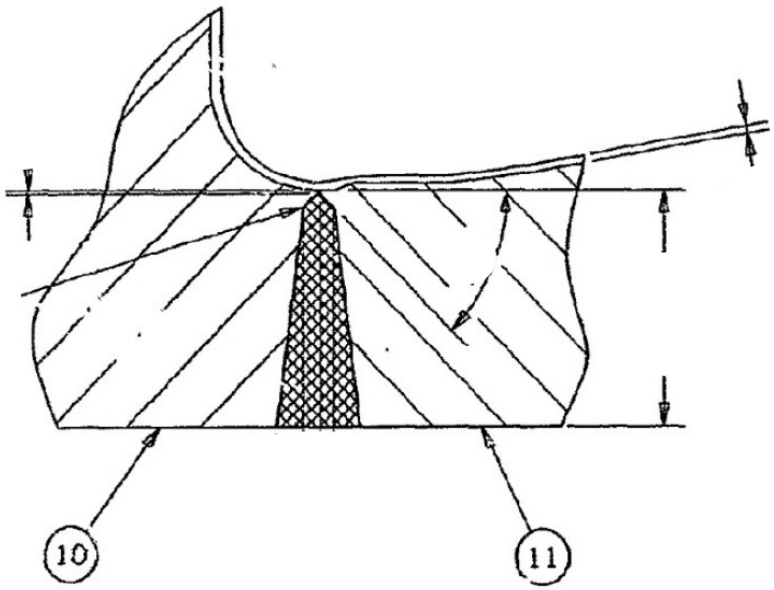
Category	Requirement from Reference [A13]	Applicability to Byron, Unit 1	Applicability to Byron, Unit 2
General Requirements	The Loss of Power transient (involving unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of portion of the vessel) is not considered in this evaluation due to its rarity. In the event that such a significant thermal event occurs at a plant, its impact on the K_{IC} *(material fracture toughness)* value may require more frequent examinations and other plant actions outside the scope of this report's guidance.	Byron Station, Units 1 and 2 has not experienced a loss of power transient resulting in unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of portion of the vessel.	

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 SG tubesheet, shell cone, and steam drum head are fabricated of SA-508, Class 3 material, and the SG 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 Byron Station, Unit 2, SG tube plate is fabricated from SA-508, Class 2a material and the SG barrels, transition cone, and upper head are fabricated from SA-533, Grade A, Class 2. Material SA-508, Class 2a 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.
Specific Requirements	The weld configurations must conform to those shown in Figure 1-7 and Figure 1-8 of Reference [A13].	The Byron Station, Unit 1 weld configurations are shown in Figure A7 and conform to Figures 1-7 and 1-8 of Reference [A13].	The Byron Station, Unit 2 weld configurations are shown in Figure A8 and conforms to Figures 1-7 and 1-8 of Reference [A13].

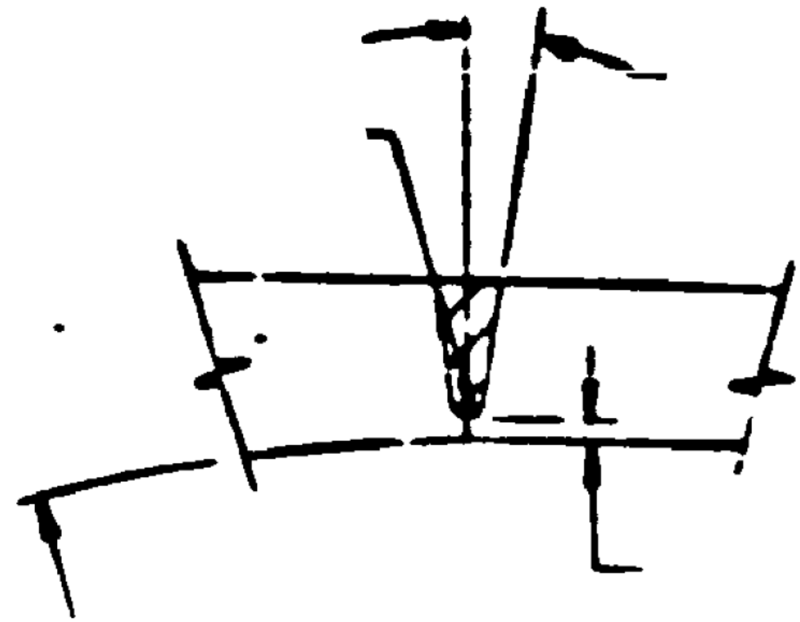
Category	Requirement from Reference [A13]	Applicability to Byron, Unit 1	Applicability to Byron, Unit 2
	The SG vessel dimensions must be within 10% percent of the upper and lower bounds of the values provided in the table in Section 9.4.4 of Reference [A13].	The Byron Station, Unit 1 SG vessel dimensions are as follows: • SG Lower Head diameter = 136.88 inches • SG Upper Shell diameter = 176.75 inches These dimensions are within 10% of those specified in Table 9-3 in Section 9.4.4 of Reference [A13] for Westinghouse plants.	The Byron Station, Unit 2 SG vessel dimensions are as follows: • SG Lower Head diameter = 135.58 inches • SG Upper Shell diameter = 176.5 inches These dimensions are within 10% of those specified in Table 9-3 in Section 9.4.4 of Reference [A13] for Westinghouse plants.
	The component must experience transients and cycles bounded by those shown in Table 5-9 of Reference [A13] over a 60-year operating life.	As shown in Table A8, the Byron Station, Units 1 and 2 transients and number of cycles projected to occur over a 60-year life are bounded by those shown in Table 5-9 of Reference [A13].	

REFERENCES:

A13. *Technical Bases for Inspection Requirements for PWR Steam Generator Class 1 Nozzle-to-Vessel Welds and Class 1 and Class 2 Vessel Head, Shell, Tubesheet-to-Head and Tubesheet-to-Shell Welds*. EPRI, Palo Alto, CA: 2019. 3002015906, ADAMS Accession No. ML20225A141.
A14. Byron, Unit 1, Drawing 7720E038, *Section 'XI' Preservice U/T*.
A15. Byron, Unit 2, Drawing 1512E55, Sheet 1, *Steam Generator Model "D5" As Built Dimensions*.
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)
A21. AREVA Specification 18-1229648-009 (MUR), Certified Design Specification for Replacement Steam Generator Byron and Braidwood Stations Unit 1.
A22. Framatone 32-500-7271-00, *Byron/Braidwood RSG Design Transients – Power Uprate*.
A23. WCAP-11388, Volume 2, Byron Units 1 and 2 and Braidwood Units 1 and 2, *T-Hot Reduction Program Engineering Report*.
A24. Design Specification 953442, Rev 8, *Model D5 Steam Generator (SG) Stress Report*.
A25. CN-OPES(99)-061, Revision 0, Power Uprate – *Design Transient Revisions for BY/BR Uprating*.

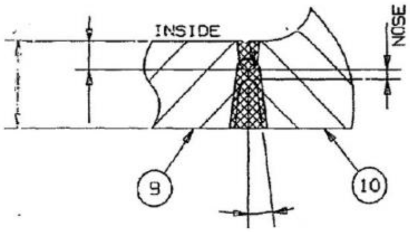


PRIMARY HEAD TO TUBESHEET W22
(UNIT 1)

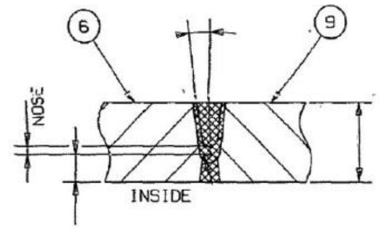


CHANNEL HEAD TO TUBE PLATE
(UNIT 2)

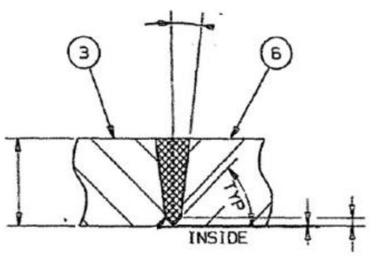
Figure A6. Byron Station, Units 1 and 2 SG Primary Side Head to Tubesheet Weld Configuration



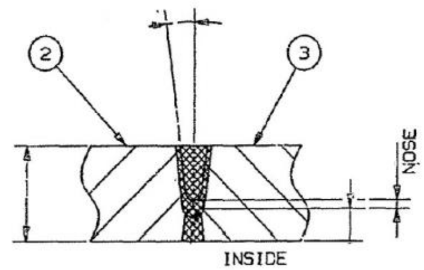
SECONDARY SHELL TO
TUBESHEET W65



SECONDARY SHELL TO
SHELL CONE W80

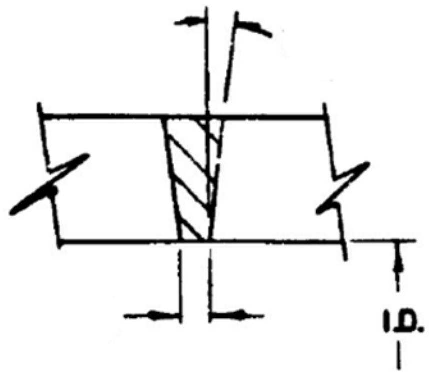


STEAM DRUM TO
SHELL CONE W142

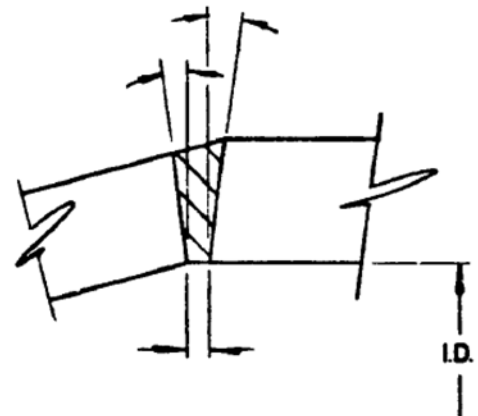


STEAM DRUM HEAD TO
STEAM DRUM W144

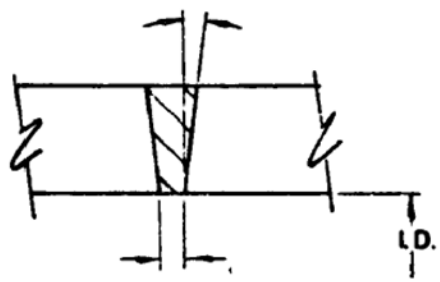
Figure A7. Byron, Unit 1, SG Secondary Side Shell Weld Configurations



TUBE PLATE TO STUB BARREL,
STUB BARREL TO LOWER SHELL BARREL, AND
LOWER SHELL BARREL TO TRANSITION CONE



TRANSITION CONE TO UPPER SHELL BARREL



UPPER STEAM BARREL SHELL TO UPPER HEAD

Figure A8. Byron, Unit 2 SG Secondary Side Shell Weld Configurations

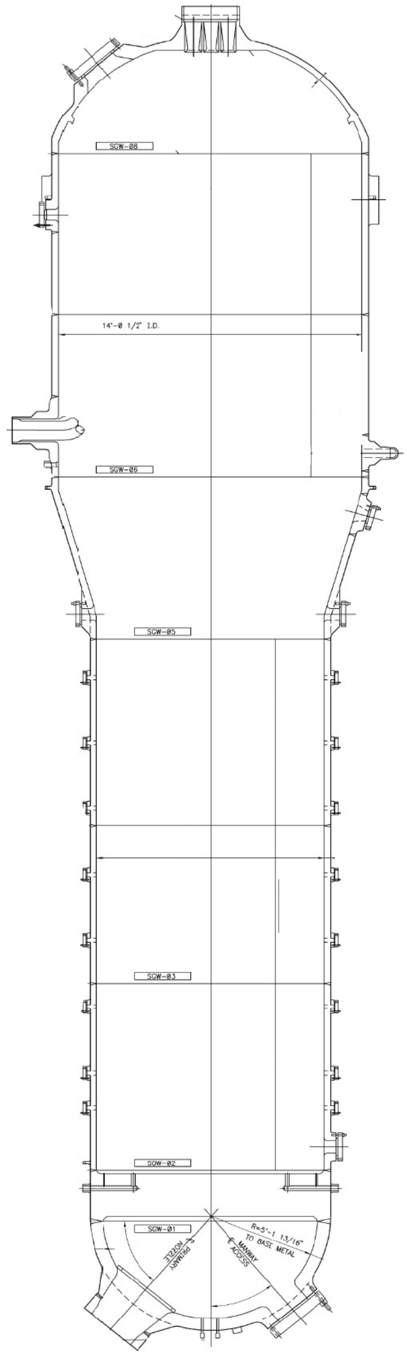


Figure A9. Byron, Unit 1, Steam Generator General Outline

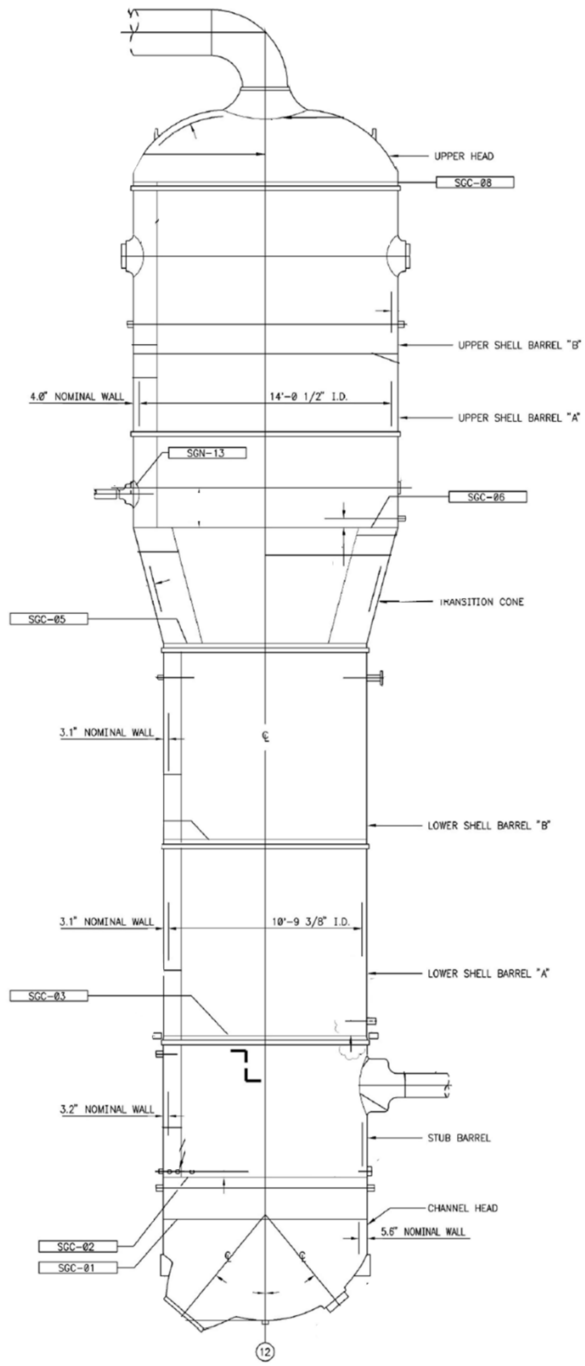


Figure A10. Byron, Unit 2, Steam Generator General Outline

Table A7. Transient Cycles for PWR SG Primary-Side Head Welds for Byron, Units 1 and 2 in Comparison to Table 5-7 of Reference [A13]

Transient	Maximum T_{HOT} (°F)	Minimum T_{HOT} (°F)	Maximum T_{COLD} (°F)	Minimum T_{COLD} (°F)	Maximum Pressure (psig)	Minimum Pressure (psig)	60-Year Cycles
Heatup/Cooldown From Reference [A13]	545	70	545	70	2235	0	300
Heatup/Cooldown for Byron Station, Units 1 and 2 [A20, A23, A25]	557[3]	120	557[3]	120	2250[3]	467	66/66 [Unit 1] 62/62 [Unit 2]
Plant Loading/ Unloading from Reference [A13]	610	550	550	545	2300	2300	5000
Plant Loading/ Unloading for Byron Station, Units 1 and 2 [A20, A23, A25]	618.4[3]	566.5	558.1[3]	538.2[3]	2370[3]	2295[3]	181[1]/188[2] [Unit 1] 179[1]/179[2] [Unit 2]
Reactor Trip from Reference [A13]	615	530	565	530	2435	1700	360
Reactor Trip for Byron Station, Units 1 and 2 [A20, A23, A25]	618.4[3]	435[3]	555.2	538.2	2250	1582[3]	9 [Unit 1] 13 [Unit 2]

Note:

1. 60-Year 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. 60-Year Cycle counts for Plant Unloading conservatively includes all events for transients “Plant Unloading at 5%/min” and “Plant Unloading, 15% - 0%”.
3. See the Stress Analysis discussion in Section 5 for a basis to the applicability of the EPRI report for transients having a maximum pressure/temperature higher than what is listed in the EPRI Report or a minimum pressure/temperature lower than what is listed in the EPRI Report.

Table A8. Transient Cycles for PWR SG Secondary-Side Vessel Welds for Byron, Units 1 and 2 in Comparison to Table 5-9 of Reference [A13]

Transient	Maximum T_{ss} (°F)	Minimum T_{ss} (°F)	Maximum Pressure (psig)	Minimum Pressure (psig)	60-Year Cycles
Heatup/Cooldown from Reference [A13]	545	70	1000	0	300
Heatup/Cooldown for Byron Station, Units 1 and 2 [A20, A21, A23]	557[3]	120	1077[3]	0	66/66 [Unit 1] 62/62 [Unit 2]
Plant Loading/ Unloading from Reference [A13]	545	540	1000	1000	5000
Plant Loading/ Unloading for Byron Station, Units 1 and 2 [A20, A21, A22]	555.5 [Unit 1][3] 554.3 [Unit 2][3]	525.8 [Unit 1][3] 524.4 [Unit 2][3]	1077 [Unit 1][3,4] 1077 [Unit 2][3,4]	1077 [Unit 1][4] 1077 [Unit 2][4]	181[1]/188[2] [Unit 1] 179[1]/179[2] [Unit 2]
Reactor Trip from Reference [A13]	555	530	1130	1000	360
Reactor Trip for Byron Station, Units 1 and 2 [A20, A21, A22]	557.9 [Unit 1][3] 553.8 [Unit 2]	460.9 [Unit 1] 458.8 [Unit 2]	1092 [Unit 1] 1092 [Unit 2][5]	995 [Unit 1][3,6] 860 [Unit 2][3,6]	9 [Unit 1] 13 [Unit 2]

Note:

1. 60-Year Cycle counts for Plant Loading conservatively includes all events for transients “Plant Loading at 5%/min” and “Plant Loading, 0% - 15%”.
2. 60-Year Cycle counts for Plant Unloading conservatively includes all events for transients “Plant Unloading at 5%/min” and “Plant Unloading, 15% - 0%”.
3. See the Stress Analysis discussion in Section 5 for a basis to the applicability of the EPRI report for transients having a maximum pressure/temperature higher than what is listed in the EPRI Report or a minimum pressure/temperature lower than what is listed in the EPRI Report.
4. Maximum and Minimum Pressure values for the Plant Loading/Unloading Transient could not be located in design documentation for the Secondary Side of the Vessel. Since the Maximum and Minimum T_{SS} for the Heatup/Cooldown Transient are similar to the Loading/Unloading Transient, the same pressure was assumed. This is the basis for the listed 1077 psig values listed in the table.
5. Maximum Pressure for the Reactor Trip transient for Unit 2 could not be located in design documentation. Due to the similar Maximum T_{ss} values for the Reactor Trip Transient between Units 1 and 2, the same Maximum Pressure was assumed for Unit 2.
6. Minimum Pressure for the Reactor Trip transient for Unit 1 and 2 could not be located in design documentation. Plant operating data for Byron and Braidwood was reviewed, and the minimum pressure corresponding to 100% reactor power was used. This pressure corresponds to the minimum pressure when a reactor trip transient would initiate at 100% power. Due to similarities in SG design and operational strategies between the stations, the same values were used for Braidwood Unit 1 and Byron Unit 1 and also for Braidwood Unit 2 and Byron Unit 2.

Plant-Specific Applicability for Calvert Cliffs

Section 9 of Reference [A26] 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 Tables A9 through A13.

Tables A9 through A13 indicate 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 A9. Applicability of Reference [A26] Representative Analyses to Calvert Cliffs, Units 1 and 2
<u>Items Number B2.40 (SG Primary Side Shell Welds)</u>

Category	Requirement from Reference [A26]	Applicability to Calvert Cliffs, Units 1 and 2
General Requirements	The Loss of Power transient (involving unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of portion of the vessel) is not considered in this evaluation due to its rarity. In the event that such a significant thermal event occurs at a plant, its impact on the K_{IC} *(material fracture toughness)* value may require more frequent examinations and other plant actions outside the scope of this report's guidance.	Calvert Cliffs, Units 1 and 2 has not experienced a loss of power transient resulting in unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of portion of the vessel.
	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 SG vessel primary head and tubesheet are fabricated of SA-508, Class 3a material. This material conforms to the requirements of ASME Code, Section XI, Appendix G, Paragraph G- 2110.
Specific Requirements	The weld configurations must conform to those shown in Figures 1-1 and Figure 1-2 of Reference [A26].	The Calvert Cliffs, Units 1 and 2 weld configurations are shown in Figure A11 and conform to Figure 1-2 of Reference [A26].

Category	Requirement from Reference [A26]	Applicability to Calvert Cliffs, Units 1 and 2
	The SG vessel dimensions must be within 10% percent of the upper and lower bounds of the values provided in the table in Section 9.4.3 of Reference [A26].	The Calvert Cliffs, Units 1 and 2 SG vessel dimensions are as follows: • SG Lower Head diameter = 169 inches • SG Upper Shell diameter = 239.63 inches These dimensions are within 10% of those specified in Table 9-2 in Section 9.4.3 of Reference [A26] for CE plants.
	The component must experience transients and cycles bounded by those shown in Table 5- 7 of Reference [A26] over a 60-year operating life.	As shown in Table A12, the Calvert Cliffs, Units 1 and 2 transients and number of cycles projected to occur over a 60-year life are bounded by those shown in Table 5-7 of Reference [A26] for CE plants.

Table A10. Applicability of Reference [A26] Representative Analyses to Calvert Cliffs, Units 1 and 2
Item Number B3.130 (SG Primary Inlet/Outlet Nozzles)

Category	Requirement from Reference [A26]	Applicability to Calvert Cliffs, Units 1 and 2
General Requirements	The Loss of Power transient (involving unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of portion of the vessel) is not considered in this evaluation due to its rarity. In the event that such a significant thermal event occurs at a plant, its impact on the K_{IC} *(material fracture toughness)* value may require more frequent examinations and other plant actions outside the scope of this report's guidance.	Calvert Cliffs, Units 1 and 2 has not experienced a loss of power transient resulting in unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of portion of the vessel.
	The materials of the SG shell and 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 SG vessel primary head and primary inlet and outlet nozzles are fabricated of SA- 508, Class 3a material. This material conforms to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.
Specific Requirements	The weld configurations must conform to those shown in Figures 1-3 through 1-5 of Reference [A26].	The Calvert Cliffs, Units 1 and 2 weld configurations are shown in Figure A12 and conform to Figure 1-5 of Reference [A26].

Category	Requirement from Reference [A26]	Applicability to Calvert Cliffs, Units 1 and 2
	The piping attached to the primary inlet and outlet nozzles (RCS piping) for the various designs must be within 10% percent of the values provided in the table in Section 9.4.2 of Reference [A26].	The Calvert Cliffs, Units 1 and 2 primary inlet/outlet nozzle and piping dimensions are as follows: • SG primary side inlet nozzle= 42 inches NPS • SG primary side outlet nozzle= 30 inches NPS These dimensions are within 10% of those specified in Table 9-1 in Section 9.4.2 of Reference [A26] for CE plants.
	The component must experience transients and cycles bounded by those shown in Table 5-7 of Reference [A26] over a 60-year operating life.	As shown in Table A12, the Calvert Cliffs, Units 1 and 2 transients and number of cycles projected to occur over a 60-year life are bounded by those shown in Table 5-7 of Reference [A26] for CE plants.

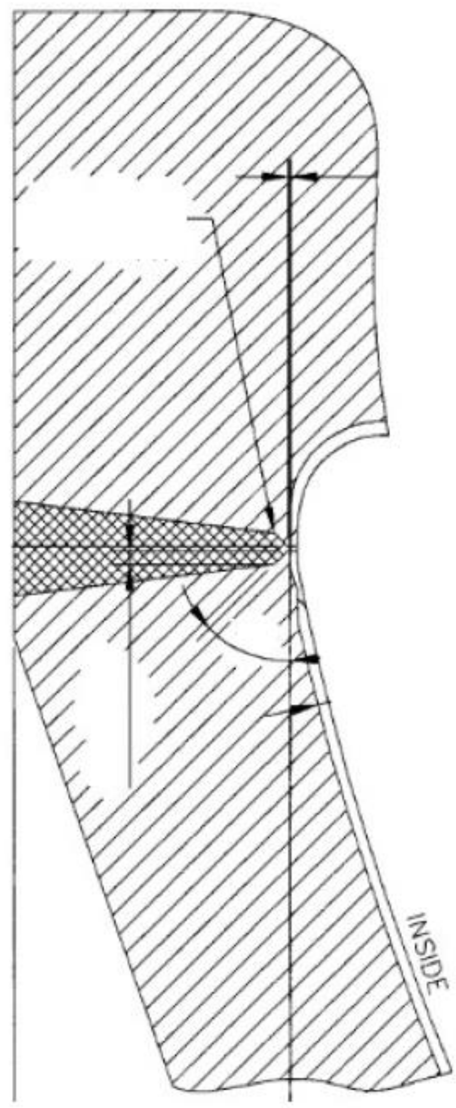
Table A11. Applicability of Reference [A26] Representative Analyses to Calvert Cliffs, Units 1 and 2
<u>Items Numbers C1.10, C1.20 and C1.30 (SG Secondary Side Shell Welds)</u>

Category	Requirement from Reference [A26]	Applicability to Calvert Cliffs, Units 1 and 2
General Requirements	The Loss of Power transient (involving unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of portion of the vessel) is not considered in this evaluation due to its rarity. In the event that such a significant thermal event occurs at a plant, its impact on the K_{IC} *(material fracture toughness)* value may require more frequent examinations and other plant actions outside the scope of this report's guidance.	Calvert Cliffs, Units 1 and 2 has not experienced a loss of power transient resulting in unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of portion of the vessel.
	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, SG vessel original upper head, cone segment and steam drum shell are fabricated from SA-533, Grade B, Class 1 material, and the SG replaced shell cone, secondary shell assembly, and tubesheet are fabricated from SA-508, Class 3a. These materials conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.
Specific Requirements	The weld configurations must conform to those shown in Figure 1-7 and Figure 1-8 of Reference [A26].	The Calvert Cliffs, Units 1 and 2 weld configurations are shown in Figure A13, and conform to Figures 1-7 and 1-8 of Reference [A26].

Category	Requirement from Reference [A26]	Applicability to Calvert Cliffs, Units 1 and 2
	The SG vessel dimensions must be within 10% percent of the upper and lower bounds of the values provided in the table in Section 9.4.4 of Reference [A26].	The Calvert Cliffs, Units 1 and 2 SG vessel dimensions are as follows: • SG Lower Head diameter = 169 inches • SG Upper Shell diameter • = 239.63 inches These dimensions are within10% of those specified in Table 9-3 in Section 9.4.4 of Reference [A26] for CE plants.
	The component must experience transients and cycles bounded by those shown in Table 5-9 of Reference [A26] over a 60-year operating life.	As shown in Table A13, the Calvert Cliffs, Units 1 and 2 transients and number of cycles projected to occur over a 60-year life are bounded by those shown in Table 5-9 of Reference [A26].

REFERENCES:

A26. *Technical Bases for Inspection Requirements for PWR Steam Generator Class 1 Nozzle-to-Vessel Welds and Class 1 and Class 2 Vessel Head, Shell, Tubesheet-to-Head and Tubesheet-to-Shell Welds*. EPRI, Palo Alto, CA: 2019. 3002015906, ADAMS Accession No. ML20225A141.
A27. Calvert Cliffs, Units 1 and 2, Drawing 12010-0007, *Upper Shell Details and Assembly Steam Generator*.
A28. Calvert Cliffs, Units 1 and 2, Drawing 12010A-0001SH0001, *General Arrangement*.
A29. Calvert Cliffs, Units 1 and 2, Drawing 12010A-0015SH0001, *Section 'XI' Pre-Service NDE Examination*.
A30. Cliffs, Units 1 and 2, Figure A-4, *S.G. Tube Sheet & Lower Head*.
A31. Cliffs, Units 1 and 2, Figure B-1, *Steam Generator Outline*.
A32. 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*.
A33. ECP-18-000545, *Fatigue Plant Transient Data Review Report for the Year of 2018*.
A34. Design Specification SP-0811, *Replacement Steam Generator Design Specification*.



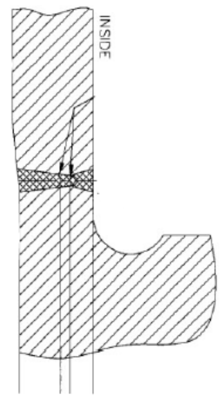
TUBESHEET TO PRIMARY HEAD W22

Figure A11. Calvert Cliffs, Units 1 and 2, SG Primary Side Head to Tubesheet Weld Configuration

PRIMARY INLET NOZZLE
TO PRIMARY HEAD W5

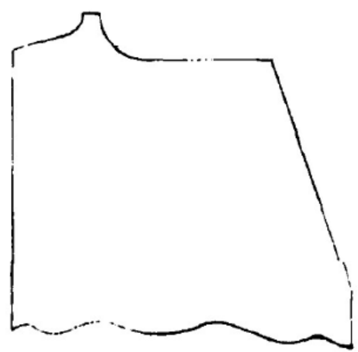
PRIMARY OUTLET NOZZLE
TO PRIMARY HEAD W6, W7

Figure A12. Calvert Cliffs, Units 1 and 2, SG Primary Side Inlet/Outlet Nozzle Configuration



SECONDARY SHELL ASSEMBLY
TO TUBESHEET W65

TRANSITION CONE TO STEAM DRUM



STEAM DRUM TO UPPER HEAD

Figure A13. Calvert Cliffs, Units 1 and 2, SG Secondary Side Shell Welds Configuration

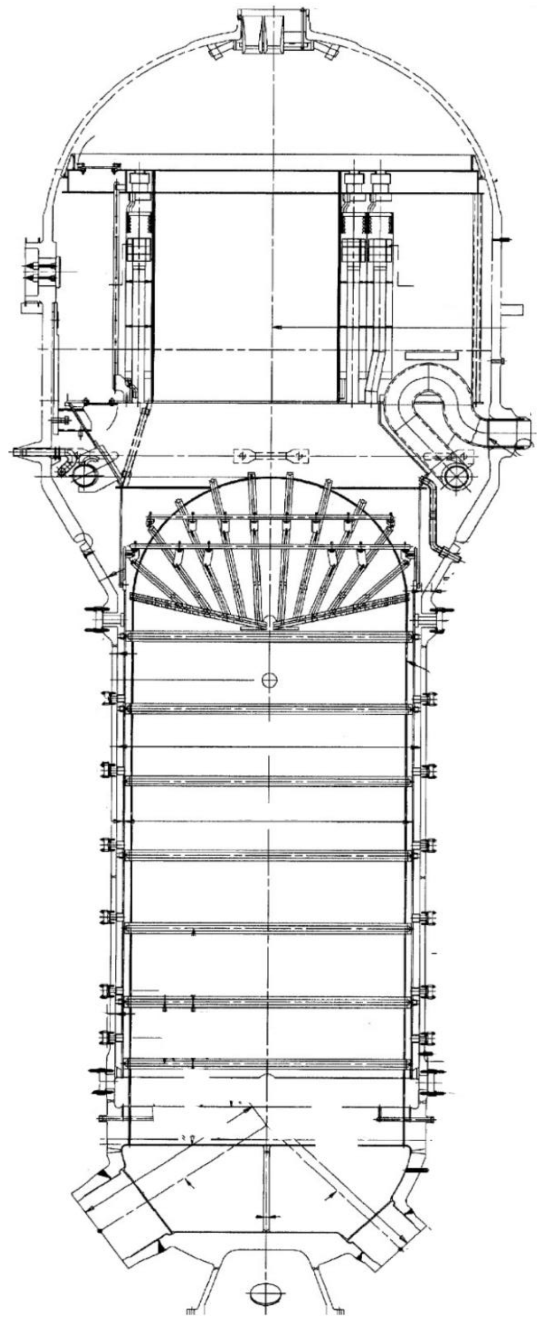


Figure A14. Calvert Cliffs, Units 1 and 2, Steam Generator General Outline

Table A12. Transient Cycles for PWR SG Primary-Side Head Welds for Calvert Cliffs, Units 1 and 2 in Comparison to Table 5-7 of Reference [A26]

Transient	Maximum T_{HOT} (°F)	Minimum T_{HOT} (°F)	Maximum T_{COLD} (°F)	Minimum T_{COLD} (°F)	Maximum Pressure (psig)	Minimum Pressure (psig)	60-Year Cycles
Heatup/Cooldown From Reference [A26]	545	70	545	70	2235	0	300
Heatup/Cooldown for Calvert Cliffs Station, Units 1 and 2 [A34]	535	70	535	70	2235	100	149/139 [Unit 1] 137/129 [Unit 2]
Plant Loading/ Unloading from Reference [A26]	610	550	550	545	2300	2300	5000
Plant Loading/ Unloading for Calvert Cliffs Station, Units 1 and 2 [A34]	605	548[1]	550	533[1]	2298	2235[1]	146/137 [Unit 1] 123/114 [Unit 2]
Reactor Trip from Reference [A26]	615	530	565	530	2435	1700	360
Reactor Trip for Calvert Cliffs Station, Units 1 and 2 [A34]	604	537	557	534	2240	1785	171 [Unit 1] 147 [Unit 2]

Note:
1. See the Stress Analysis discussion in Section 5 for a basis to the applicability of the EPRI report for transients having a maximum pressure/temperature higher than what is listed in the EPRI Report or a minimum pressure/temperature lower than what is listed in the EPRI Report.

Table A13. Transient Cycles for PWR SG Secondary-Side Vessel Welds for Calvert Cliffs, Units 1 and 2 in Comparison to Table 5-9 of Reference [A26]

Transient	Maximum T_{ss} (°F)	Minimum T_{ss} (°F)	Maximum Pressure (psig)	Minimum Pressure (psig)	60-Year Cycles
Heatup/Cooldown from Reference [A26]	545	70	1000	0	300
Heatup/Cooldown for Calvert Cliffs Station, Units 1 and 2 [A32]	531	70	870	58	149/139 [Unit 1] 137/129 [Unit 2]
Plant Loading/ Unloading from Reference [A26]	545	540	1000	1000	5000
Plant Loading/ Unloading for Calvert Cliffs Station, Units 1 and 2 [A32]	532	526	890	845	146/137 [Unit 1] 123/114 [Unit 2]
Reactor Trip from Reference [A26]	555	530	1130	1000	360
Reactor Trip for Calvert Cliffs Station, Units 1 and 2 [A32]	540	532	942	885[1]	171[1] [Unit 1] 147 [Unit 2]

Note:
1. See the Stress Analysis discussion in Section 5 for a basis to the applicability of the EPRI report for transients having a maximum pressure/temperature higher than what is listed in the EPRI Report or a minimum pressure/temperature lower than what is listed in the EPRI Report.

Plant-Specific Applicability for Ginna

Section 9 of Reference [A35] 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 Tables A14 through A17.

Tables A14 through A17 indicate that all plant-specific requirements are met for Ginna. Therefore, the results and conclusions of the EPRI report are applicable to Ginna.

Table A14. Applicability of Reference [A35] Representative Analyses to Ginna
Items Number B2.40 (SG Primary Side Shell Welds)

Category	Requirement from Reference [A35]	Applicability to Ginna
General Requirements	The Loss of Power transient (involving unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of portion of the vessel) is not considered in this evaluation due to its rarity. In the event that such a significant thermal event occurs at a plant, its impact on the K_{IC} *(material fracture toughness)* value may require more frequent examinations and other plant actions outside the scope of this report's guidance.	Ginna has not experienced a loss of power transient resulting in unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of portion of the vessel.
	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 SG vessel primary head and tubesheet are fabricated of SA-508, Class 3 material. This material conforms to the requirements of ASME Code, Section XI, Appendix G, Paragraph G- 2110.
Specific Requirements	The weld configurations must conform to those shown in Figures 1-1 and Figure 1-2 of Reference [A35].	The Ginna weld configuration is shown in Figure A15 and conforms to Figure 1-2 of Reference [A35].

Category	Requirement from Reference [A35]	Applicability to Ginna
	The SG vessel dimensions must be within 10% percent of the upper and lower bounds of the values provided in the table in Section 9.4.3 of Reference [A35].	The Ginna SG vessel dimensions are as follows: • SG Lower Head diameter = 130.63 inches • SG Upper Shell diameter =166.38 inches These dimensions are within 10% of those specified in Table 9-2 in Section 9.4.3 of Reference [A35] for Westinghouse plants.
	The component must experience transients and cycles bounded by those shown in Table 5- 7 of Reference [A35] over a 60-year operating life.	As shown in Table A16, the Ginna transients and number of cycles projected to occur over a 60-year life are bounded by those shown in Table 5-7 of Reference [A35] for Westinghouse plants.

Table A15. Applicability of Reference [A35] Representative Analyses to Ginna
<u>Items Numbers C1.10, C1.20 and C1.30 (SG Secondary Side Shell Welds)</u>

Category	Requirement from Reference [A35]	Applicability to Ginna
General Requirements	The Loss of Power transient (involving unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of portion of the vessel) is not considered in this evaluation due to its rarity. In the event that such a significant thermal event occurs at a plant, its impact on the K_{IC} *(material fracture toughness)* value may require more frequent examinations and other plant actions outside the scope of this report's guidance.	Ginna has not experienced a loss of power transient resulting in unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of portion of the vessel.
	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 SG vessel tubesheet, shell cone, and steam drum head are fabricated of SA-508, Class 3 material, and the secondary shell and steam drum shell are fabricated of SA-533, Type B, Class 1 material. These materials conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.
Specific Requirements	The weld configurations must conform to those shown in Figure 1-7 and Figure 1-8 of Reference [A35].	The Ginna weld configurations are shown in Figure A16, and conform to Figures 1-7 and 1-8 of Reference [A35].

Category	Requirement from Reference [A35]	Applicability to Ginna
	The SG vessel dimensions must be within 10% percent of the upper and lower bounds of the values provided in the table in Section 9.4.4 of Reference [A35].	The Ginna SG vessel dimensions are as follows: • SG Lower Head diameter = 130.63 inches • SG Upper Shell diameter =166.38 inches These dimensions are within 10% of those specified in Table 9-3 in Section 9.4.4 of Reference [A35] for Westinghouse plants.
	The component must experience transients and cycles bounded by those shown in Table 5-9 of Reference [A35] over a 60-year operating life.	As shown in Table A17, the Ginna transients and number of cycles projected to occur over a 60-year life are bounded by those shown in Table 5-9 of Reference [A35].

REFERENCES:

A35. *Technical Bases for Inspection Requirements for PWR Steam Generator Class 1 Nozzle-to-Vessel Welds and Class 1 and Class 2 Vessel Head, Shell, Tubesheet-to-Head and Tubesheet-to-Shell Welds*. EPRI, Palo Alto, CA: 2019. 3002015906, ADAMS Accession No. ML20225A141.

A36. Ginna, Drawing 7705E001, *Steam Generator Arrangement - SGA*.

A37. Ginna, Drawing 7705E158, *Layout of Vessel Reference Points for Welds*.

A38. Ginna, Drawing 7705E205, *Pressure Boundary Welds Prep & Weld No's*.

A39. Design Specification 18-1224785-05, *Design Specification for Replacement Steam Generator*.

A40. Ginna, Fatigue Monitoring Program LR-FATM-PROGPLAN, Revision 6, *License Renewal Aging Management Program Basis Document,* December 20, 2017.

A41. Ginna, Calculation BWC-143O-B1, Revision 0, *R.E. Ginna – Power Uprate versus Non Power Uprate Transient Comparison*, December 10, 2004.

A42. Westinghouse Document Number CN-SCS-04-060, Revision 0, *RGE Secondary Side Design Transients for 19.5% Uprated Conditions*.

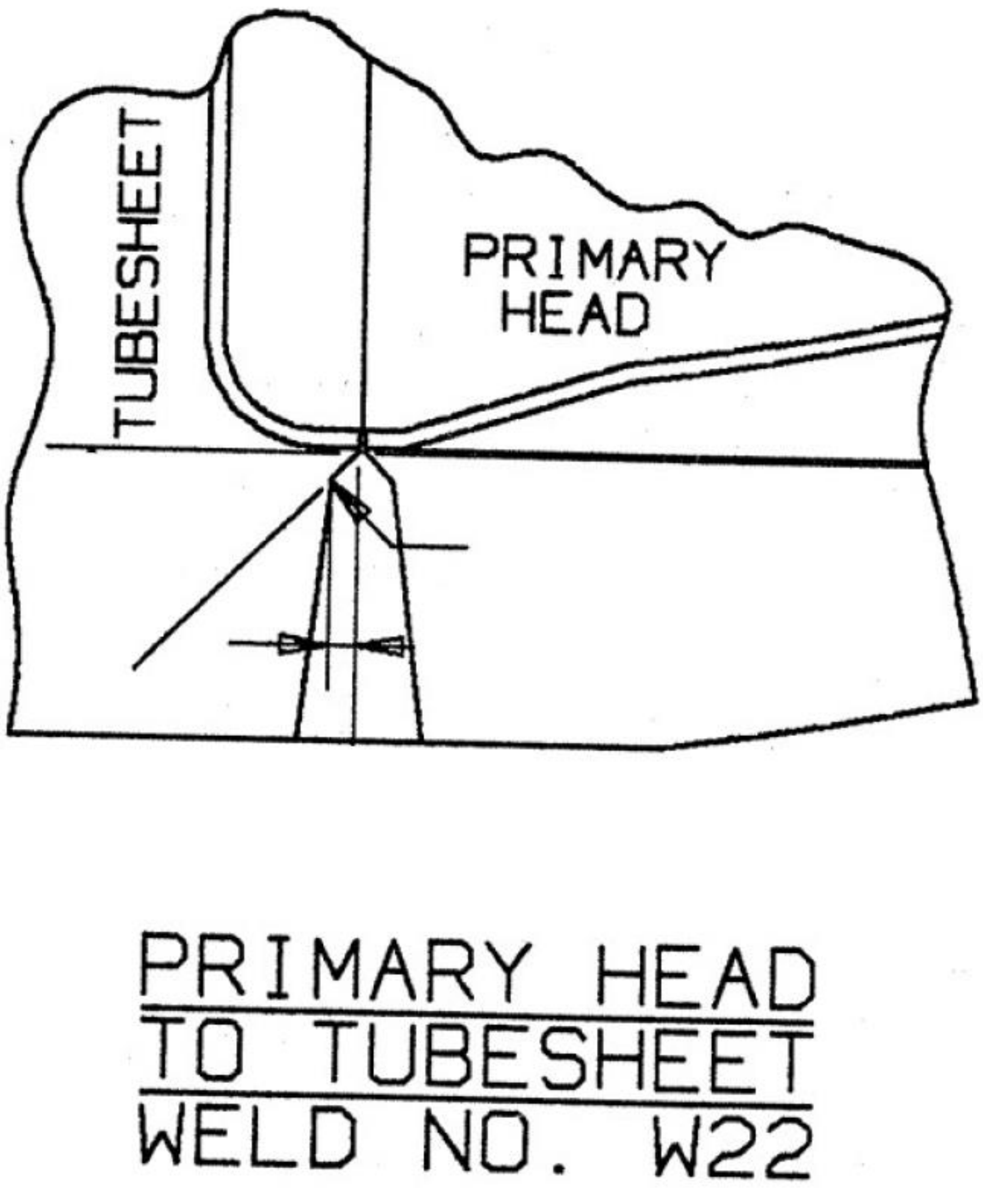


Figure A15. Ginna Primary Side Head to Tubesheet Weld Configuration

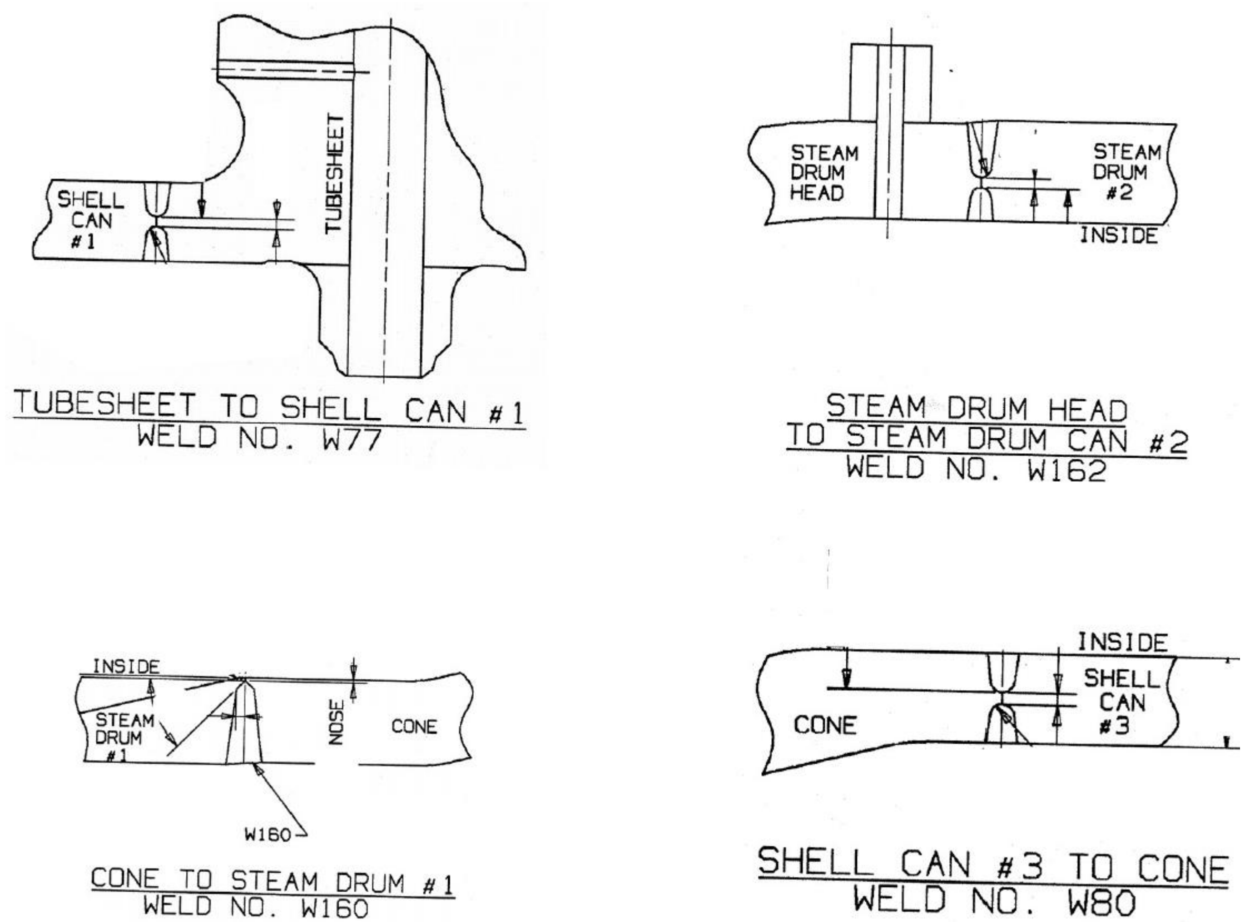


Figure A16. Ginna SG Secondary Side Shell Welds Configuration

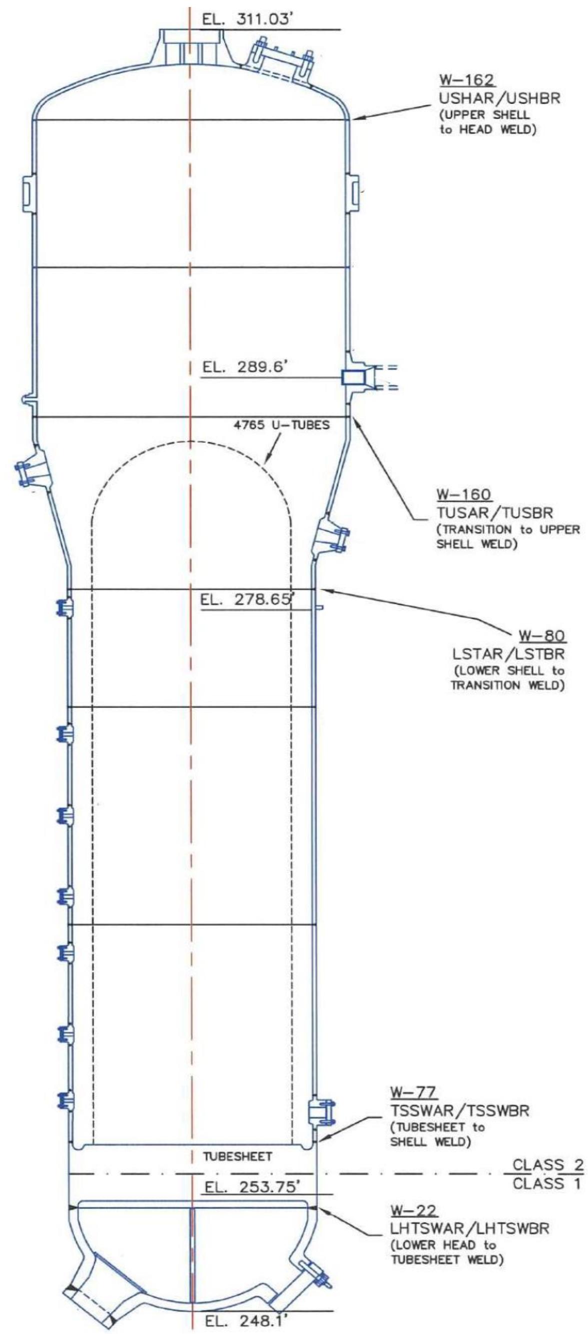


Figure A17. Ginna Steam Generator General Outline

Table A16. Transient Cycles for PWR SG Primary-Side Head Welds for Ginna in Comparison to Table 5-7 of Reference [A35]

Transient	Maximum T_{HOT} (°F)	Minimum T_{HOT} (°F)	Maximum T_{COLD} (°F)	Minimum T_{COLD} (°F)	Maximum Pressure (psig)	Minimum Pressure (psig)	60-Year Cycles
Heatup/Cooldown From Reference [A35]	545	70	545	70	2235	0	300
Heatup/Cooldown for Ginna [A39, A40]	547[2]	100	547[2]	100	2235	0	153
Plant Loading/ Unloading from Reference [A35]	610	550	550	545	2300	2300	5000
Plant Loading/ Unloading for Ginna	Not typical operation, not counted - See Note 1						
Reactor Trip from Reference [A35]	615	530	565	530	2435	1700	360
Reactor Trip for Ginna [A40, A41]	611.8	547	585.9[2]	539.9	2638[2]	1700	138

Note:

1. Ginna is operated as a base load plant. Typically, it is continuously operated at 100% (nominal) steady-state power. The only exceptions to this are refueling outages, certain equipment tests or maintenance requiring down power, and unplanned transients.
2. See the Stress Analysis discussion in Section 5 for a basis to the applicability of the EPRI report for transients having a maximum pressure/temperature higher than what is listed in the EPRI Report or a minimum pressure/temperature lower than what is listed in the EPRI Report.

Table A17. Transient Cycles for PWR SG Secondary-Side Vessel Welds for Ginna in Comparison to Table 5-9 of Reference [A35]

Transient	Maximum T_{ss} (°F)	Minimum T_{ss} (°F)	Maximum Pressure (psig)	Minimum Pressure (psig)	60-Year Cycles
Heatup/Cooldown from Reference [A35]	545	70	1000	0	300
Heatup/Cooldown for Ginna [A40, A42]	547[2]	100	1004.8[2]	0	153
Plant Loading/ Unloading from Reference [A35]	545	540	1000	1000	5000
Plant Loading/ Unloading for Ginna	Not typical Operation, not counted - See Note 1				
Reactor Trip from Reference [A35]	555	530	1130	1000	360
Reactor Trip for Ginna [A40, A41]	565[2]	515.4[2]	1179[2]	781[2]	138

Note:

1. Ginna is operated as a base load plant. Typically, it is continuously operated at 100% (nominal) steady-state power. The only exceptions to this are refueling outages, certain equipment tests or maintenance requiring down power, and unplanned transients.
2. See the Stress Analysis discussion in Section 5 for a basis to the applicability of the EPRI report for transients having a maximum pressure/temperature higher than what is listed in the EPRI Report or a minimum pressure/temperature lower than what is listed in the EPRI Report.

APPENDIX B

PLANT SPECIFIC INSPECTION HISTORY

BRAIDWOOD, UNITS 1 AND 2, INSPECTION HISTORY

Currently, the SG components for Braidwood, Units 1 and 2, satisfy all of the inspection requirements of ASME Code, Section XI, 2013 Edition. A summary of the inspection history for Item Numbers B2.40, C1.10, C1.20, and C1.30 for Braidwood, Units 1 and 2, is provided in Tables B1 through B4.

Table B1. Braidwood SG Primary Side Tubesheet-to-Shell Welds (B2.40)

Item No.	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
B2.40	1	9/2001 (A1R09)	2nd Interval/ 2nd Period	1SG-05-SGC-01	NRI	100%
	1	10/2010 (A1R15)	3rd Interval/ 1st Period	1SG-05-SGC-01	NRI	100%
	2	10/1997 (A2R06)	1st Interval/ 3rd Period	2SG-04-SGC-01	NRI[1]	99.2%
	2	4/1990 (A2R06)	1st Interval/ 3rd Period	2SG-01-SGC-01	NRI[1]	100%
	2	10/1994 (A1R04)	1st Interval/ 2nd Period	2SG-02-SGC-01	NRI[1]	100%
	2	11/2003 (A2R10)	2nd Interval/ 2nd Period	2SG-02-SGC-01	NRI[1]	100%
	2	10/2012 (A2R16)	3rd Interval/ 2nd Period	2SG-02-SGC-01	NRI[1]	99.7%
	2	10/1997 (A2R06)	1st Interval/ 3rd Period	2SG-03-SGC-01	NRI[1]	99.2%

Note:

1. ID Geometry (45 Degree) noted for welds 2SG-04-SGC-01, 2SG-01-SGC-01, 2SG-02-SGC-01, and 2SG-03-SGC-01.

Table B2. Braidwood SG Secondary Side Shell Welds (C1.10)

Item No.	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
C1.10	1	4/2006 (A1R12)	2nd Interval/ 3rd Period	1SG-05-SGC-06	NRI	100%
	1	4/2018 (A1R20)	3rd Interval/ 3rd Period	1SG-05-SGC-06	NRI	100%
	2	4/1996 (A2R05)	1st Interval/ 3rd Period	2SG-03-SGC-03	NRI[1]	100%
	2	5/2008 (A2R13)	2nd Interval/ 3rd Period	2SG-03-SGC-03	NRI	99.9%
	2	5/2017 (A2R19)	3rd Interval/ 3rd Period	2SG-03-SGC-03	NRI[1]	100%
	2	3/1993 (A2R03)	1st Interval/ 2nd Period	2SG-03-SGC-05	NRI	100%
	2	4/2002 (A2R09)	2nd Interval/ 2nd Period	2SG-03-SGC-05	NRI	100%
	2	4/2011 (A2R15)	3rd Interval/ 1st Period	2SG-03-SGC-05	NRI	100%
	2	3/1993 (A2R03)	1st Interval/ 2nd Period	2SG-03-SGC-06	NRI[2]	100%
	2	4/2002 (A2R09)	2nd Interval/ 2nd Period	2SG-03-SGC-06	NRI[2]	100%
	2	4/2011 (A2R15)	3rd Interval/ 1st Period	2SG-03-SGC-06	NRI[2]	100%
	2	10/2021 (A2R22)	4th Interval/ 1st Period	2SG-03-SGC-06	NRI[2]	100%

Note:

1. Suspected slag noted (45 Degree) for 2SG-03-SGC-03 (A2R05) and 2SG-03-SGC-03 (A2R19).
2. ID Geometry noted (45 Degree) for weld 2SG-03-SGC-06.
3. Examination of the upper secondary shell - shell cone welds (1SG-05-SGC-05, 1SG-06-SGC-05, 1SG-07-SGC-05, and 1SG-08-SGC-05) were not examined during the Second or Third Intervals, due to the basis that these welds were not located at gross structural discontinuities as defined in ASME Section III NB-3213.2. The welds were removed from the Examination Category C-A Item Number C1.10 examination population as permitted by Table IWC-2500-1 Examination Category C-A Note 2.

Table B3. Braidwood SG Secondary Side Head Welds (C1.20)

Item No.	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
C1.20	1	9/2001 (A1R09)	2nd Interval/ 2nd Period	1SG-05-SGC-08	NRI	100%
	1	4/2015 (A1R18)	3rd Interval/ 2nd Period	1SG-05-SGC-08	NRI	100%
	2	4/1996 (A2R05)	1st Interval/ 3rd Period	2SG-03-SGC-08	NRI[1]	100%
	2	5/2008 (A2R13)	2nd Interval/ 3rd Period	2SG-03-SGC-08	NRI[1]	100%
	2	5/2014 (A2R17)	3rd Interval/ 2nd Period	2SG-03-SGC-08	NRI[1]	100%

Note:
1. ID Geometry noted (45 Degree) for weld 2SG-03-SGC-08.

Table B4. Braidwood SG Secondary Side Tubesheet-to-Shell Welds (C1.30)

Item No.	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
C1.30	1	9/2001 (A1R09)	2nd Interval/ 2nd Period	1SG-05-SGC-02	NRI	99%
	1	10/2010 (A1R15)	3rd Interval/ 1st Period	1SG-05-SGC-02	NRI	100%
	2	4/1990 (A2R01)	1st Interval/ 1st Period	2SG-01-SGC-02	NRI[1]	88.4%
	2	5/2008 (A2R13)	2nd Interval/ 3rd Period	2SG-01-SGC-02	NRI	88.4%
	2	10/2018 (A2R20)	3rd Interval/ 3rd Period	2SG-01-SGC-02	NRI	88.4%

Note:
1. ID Geometry noted (45 Degree) for weld 2SG-01-SGC-02 (A2R01).

BYRON, UNITS 1 AND 2, INSPECTION HISTORY

Currently, the SG components for Byron, Units 1 and 2, satisfy all of the inspection requirements of ASME Code, Section XI, 2007 Edition with the 2008 Addenda. A summary of the inspection history for Item Numbers B2.40, C1.10, C1.20, and C1.30 for Byron, Units 1 and 2, is provided in Tables B5 through B8.

Table B5. Byron SG Primary Side Tubesheet-to-Shell Welds (B2.40)

Item No.	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
B2.40	1	4/1999 (B1R09)	2nd Interval/ 1st Period	1RC-01-BB/SGW-01	NRI	100%
	1	9/2006 (B1R14)	3rd Interval/ 1st Period	1RC-01-BB/SGW-01	NRI	100%
	1	3/2017 (B1R21)	4th Interval/ 1st Period	1RC-01-BB/SGW-01	NRI	100%
	2	9/1996 (B2R06)	1st Interval/ 3rd Period	2RC-01-BC/SGC-01	NRI[1]	100%
	2	9/1990 (B2R02)	1st Interval/ 1st Period	2RC-01-BA/SGC-01	NRI[1]	90% - limited due to weld pads
	2	10/1999 (B2R08)	2nd Interval/ 1st Period	2RC-01-BA/SGC-01	NRI[1]	99% - limited due to weld pads
	2	10/2008 (B2R14)	3rd Interval/ 1st Period	2RC-01-BA/SGC-01	NRI[1]	98% - limited due to weld pads
	2	4/2019 (B2R21)	4th Interval/ 1st Period	2RC-01-BA/SGC-01	NRI[1]	98.76% - limited due to weld pads
	2	10/1993 (B2R04)	1st Interval/ 2nd Period	2RC-01-BB/SGC-01	NRI[1]	90% - limited due to weld pads
	2	8/1996 (B2R06)	1st Interval/ 3rd Period	2RC-01-BD/SGC-01	NRI[1]	100%

Note:
1. Noted ID Geometry.

Table B6. Byron SG Secondary Side Shell Welds (C1.10)

Item No.	Unit	Date	Interval/ Period	Component ID	Exam Results	Coverage
C1.10	1	3/2005 (B1R13)	2nd Interval/ 3rd Period	1RC-01-BB/SGW-06	NRI	99.5% - limited due to adjacent nozzles
	1	3/2014 (B1R19)	3rd Interval/ 3rd Period	1RC-01-BB/SGW-06	NRI	100%
	2	10/1990 (B2R02)	1st Interval/ 1st Period	2RC-01-BB/SGC-03[1]	Recorded 3 embedded indications that were identified with the 60° shear during PSI – 2 indications were below recordable levels; therefore, they were not required to be compared to code standards. The original Indication #1 from PSI was unacceptable based on IWB-3511.	100%
	2	9/1993 (B2R04)	1st Interval/ 2nd Period	2RC-01-BB/SGC-03[1]	Resize of 3 embedded indications recorded with 60° shear during PSI – no change.	N/A
	2	11/1999 (B2R08)	2nd Interval/ 1st Period	2RC-01-BB/SGC-03[1]	Recorded the same embedded indications identified in PSI and ID Geometry with no change.	97.5% - limited due to weld pads
	2	10/2008 (B2R14)	3rd Interval/ 1st Period	2RC-01-BB/SGC-03[1]	Recorded the same embedded indications identified in PSI and ID Geometry with no change.	97.5% - limited due to weld pads
	2	10/1990 (B2R02)	1st Interval/ 1st Period	2RC-01-BB/SGC-05	Recorded 6 embedded reflectors (3 identified in PSI). Acceptable in accordance with IWB-3511.	100%
	2	9/2002 (B2R10)	2nd Interval/ 2nd Period	2RC-01-BB/SGC-05	Recorded 3 embedded reflectors with no change.	100%
	2	4/2013 (B2R17)	3rd Interval/ 3rd Period	2RC-01-BB/SGC-05	NRI	100%

Item No.	Unit	Date	Interval/ Period	Component ID	Exam Results	Coverage
	2	10/1990 (B2R02)	1st Interval/ 1st Period	2RC-01-BC/SGC-05[1]	Resize of 2 embedded indications recorded with 45° shear during PSI with no significant changes. Both indications were unacceptable based on IWB-3511 during PSI.	N/A
	2	9/1993 (B2R04)	1st Interval/ 2nd Period	2RC-01-BC/SGC-05[1]	Resize of 2 embedded indications recorded with 45° shear during PSI with no changes.	N/A
	2	1/1989 (B2R01)	1st Interval/ 1st Period	2RC-01-BD/SGC-06[1]	Recorded 5 embedded reflectors. 1 indication was unacceptable based on IWB-3511 during PSI	90% Limited due to weld pads and adjacent nozzles
	2	10/1993 (B2R04)	1st Interval/ 2nd Period	2RC-01-BD/SGC-06[1]	Resize of 1 embedded indication previously recorded with 60° shear during PSI and B2R01 with no change	N/A
	2	10/2005 (B2R12)	2nd Interval/ 3rd Period	2RC-01-BD/SGC-06[1]	NRI	97% limited due to level taps and weld pads
	2	9/2011 (B2R16)	3rd Interval/ 2nd Period	2RC-01-BD/SGC-06[1]	NRI	97% limited due to level taps and weld pads

Note:

1. Welds were found to have embedded indications 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 the welds and the embedded indications were either found to have no change in sizing or were found to be non-recordable due to improved inspection methods during later ISI Intervals.
2. Examination of the upper secondary shell - shell cone welds (1RC-01-BA/SGW-05, 1RC-01-BB/SGW-05, 1RC-01-BC/SGW-05, and 1RC-01-BD/SGW-05) were not examined during the Second or Third Intervals, due to the basis that these welds were not located at gross structural discontinuities as defined in ASME Section III NB-3213.2. The welds were removed from the Examination Category C-A Item Number C1.10 examination population as permitted by Table IWC-2500-1 Examination Category C-A Note 2.

Table B7. Byron SG Secondary Side Head Welds (C1.20)

Item No.	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
C1.20	1	3/2005 (B1R13)	2nd Interval/ 3rd Period	1RC-01-BA/SGW-08	NRI	100%
	1	9/2012 (B1R18)	3rd Interval/ 2nd Period	1RC-01-BA/SGW-08	NRI	100%
	1	3/2020 (B1R23)	4th Interval/ 2nd Period	1RC-01-BA/SGW-08	NRI	100%
	2	3/1992 (B2R03)	1st Interval/ 2nd Period	2RC-01-BA/SGC-08	NRI	100%
	2	3/2004 (B2R11)	2nd Interval/ 2nd Period	2RC-01-BA/SGC-08	NRI	100%
	2	4/2013 (B2R17)	3rd Interval/ 3rd Period	2RC-01-BA/SGC-08	Recorded a spot indication with no through wall dimension. Acceptable in accordance with IWC-3510	100%

Table B8. Byron SG Secondary Side Tubesheet-to-Shell Welds (C1.30)

Item No.	Unit	Date	Interval/ Period	Component ID	Exam Results	Coverage
C1.30	1	3/2002 (B1R11)	2nd Interval/ 2nd Period	1RC-01-BA/SGW-02	NRI[2]	100%
	1	9/2009 (B1R16)	3rd Interval/ 1st Period	1RC-01-BA/SGW-02	NRI[2]	100%
	1	9/2018 (B1R22)	4th Interval/ 1st Period	1RC-01-BA/SGW-02	NRI[2]	100%
	2	9/1990 (B2R02)	1st Interval/ 1st Period	2RC-01-BA/SGC-02[1]	Recorded 4 embedded reflectors and 1 was unacceptable based on IWC-3510	90% - limited due to weld pads and adjacent nozzles
	2	9/1993 (B2R04)	1st Interval/ 2nd Period	2RC-01-BA/SGC-02	Resize of 4 embedded indications identified in B2R02 – No change	N/A
	2	8/1996 (B2R06)	1st Interval/ 3rd Period	2RC-01-BA/SGC-02	Resize of 4 embedded indications identified in B2R02 – No change	N/A
	2	10/1999 (B2R08)	2nd Interval/ 1st Period	2RC-01-BA/SGC-02	Recorded 4 embedded indications previously identified in B2R02 – No change	96.4% - limited due to weld pads and adjacent nozzles
	2	10/2008 (B2R14)	3rd Interval/ 1st Period	2RC-01-BA/SGC-02	Recorded 4 embedded indications previously identified in B2R02 – No change	96.4% - limited due to weld pads and adjacent nozzles
	2	4/2019 (B2R21)	4th Interval/ 1st Period	2RC-01-BA/SGC-02	Recorded ID Geometry and 3 embedded reflectors that were determined to be non-relevant.	95.9% limited due to adjacent nozzles
	2	1/1989 (B2R01)	1st Interval/ 1st Period	2RC-01-BC/SGC-02	NRI	100%
	2	9/1993 (B2R04)	1st Interval/ 2nd Period	2RC-01-BC/SGC-02	NRI	100%
	2	2/1989 (B2R01)	1st Interval/ 1st Period	2RC-01-BD/SGC-02	NRI	100%
	2	10/1993 (B2R04)	1st Interval/ 2nd Period	2RC-01-BD/SGC-02	NRI	100%

Note:

1. Weld 2RC-01-BA/SGC-02 was found to have 4 embedded indications during the 1st ISI Interval examination. Successive examinations were performed and the embedded indications were either found to be non-relevant or did not change in size during the latest 4th ISI Interval examination.
2. Noted ID geometry.

CALVERT CLIFFS, UNITS 1 AND 2, INSPECTION HISTORY

Currently, the SG components for Calvert Cliffs, Units 1 and 2, satisfy all of the inspection requirements of ASME Code, Section XI, 2013 Edition. A summary of the inspection history for Item Numbers B2.40, B3.130, C1.10, C1.20, and C1.30 for Calvert Cliffs, Units 1 and 2, is provided in Tables B9 through B13.

Table B9. Calvert Cliffs SG Primary Side Tubesheet-to-Shell Welds (B2.40)

Item No.	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
B2.40	1	3/2010 (1RFO19)	3rd Interval/ 3rd Period	11-W22	NRI	97.7%
	1	2/2014 (1RFO21)	4th Interval/ 2nd Period	11-W22	NRI	97.6%
	2	3/2007 (2RFO16)	3rd Interval/ 3rd Period	21-W22	NRI	99.5%
	2	2/2013 (2RFO19)	4th Interval/ 2nd Period	21-W22	NRI	97.0%

Table B10. Calvert Cliffs SG Primary Side Nozzle-to-Vessel Welds (B3.130)

Item No.	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
B3.130	1	2/2006 (1RFO17)	3rd Interval/ 3rd Period	SG-11- W5	NRI	84%
	1	2/2014 (1RFO21)	4th Interval/ 2nd Period	SG-11- W5	NRI	81.6%
	1	3/2010 (1RFO19)	3rd Interval/ 3rd Period	SG-11- W6	NRI	74.2%
	1	2/2018 (1RFO23)	4th Interval/ 3rd Period	SG-11- W6	NRI	73.8%
	1	3/2010 (1RFO19)	4th Interval/ 1st Period	SG-11- W7	NRI	74.2%
	1	2/2006 (1RFO17)	3rd Interval/ 3rd Period	SG-12 - W5	NRI	84%
	1	2/2010 (1RFO19)	4th Interval/ 1st Period	SG-12 - W5	NRI	84.3%
	1	2/2010 (1RFO19)	4th Interval/ 1st Period	SG-12 - W6	NRI	74.2%
	1	2/2010 (1RFO19)	4th Interval/ 1st Period	SG-12 - W7	NRI	74.2%
	2	2/2013 (2RFO19)	4th Interval/ 2nd Period	SG-21 - W5	NRI	80.2%
	2	2/2013 (2RFO19)	4th Interval/ 2nd Period	SG-21 - W6	NRI	73.8%
	2	3/2011 (2RFO18)	4th Interval/ 1st Period	SG-21 - W7	NRI	78.0%
	2	3/2009 (2RFO17)	3rd Interval/ 3rd Period	SG-22 - W5	NRI	100%
	2	3/2013 (2RFO19)	4th Interval/ 2nd Period	SG-22 - W5	NRI	80.2%
	2	3/2009 (2RFO17)	3rd Interval/ 3rd Period	SG-22 - W6	NRI	100%
	2	3/2013 (2RFO19)	4th Interval/ 2nd Period	SG-22 - W6	NRI	73.8%
	2	3/2013 (2RFO19)	4th Interval/ 2nd Period	SG-22 - W7	NRI	73.8%

Note:

1. Since the steam generator replacement was performed during the first period of the 3rd Interval, Paragraph IWB-2412(b) was utilized which only required 25% of the population to be inspected in each of the Second and Third Periods, for a total of 50% inspected by end of Third Period. This allowed Calvert Cliffs to forego the inspection of weld W7 for both Units.

Table B11. Calvert Cliffs SG Secondary Side Shell Welds (C1.10)

Item No.	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
C1.10	1	3/2010 (1RFO19)	3rd Interval/ 3rd Period	SG-11-4	NRI	93.1%
	1	3/2018 (1RFO23)	4th Interval/ 3rd Period	SG-11-4	NRI	100%
	2	3/2005 (2RFO15)	3rd Interval/ 2nd Period	SG-22-4	NRI	99.0%
	2	2/2011 (2RFO18)	4th Interval/ 1st Period	SG-22-4	NRI	99.0%

Table B12. Calvert Cliffs SG Secondary Side Head Welds (C1.20)

Item No.	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
C1.20	1	3/2010 (1RFO19)	3rd Interval/ 3rd Period	SG-11-3	NRI	100%
	1	3/2018 (1RFO23)	4th Interval/ 3rd Period	SG-11-3	Unchanged ID Geometry Indication	100%
	2	3/2009 (2RFO17)	3rd Interval/ 3rd Period	SG-21-3	Base material laminar indication	100%
	2	2/2019 (2RFO22)	4th Interval/ 3rd Period	SG-21-3	NRI	100%

Table B13. Calvert Cliffs SG Secondary Side Tubesheet-to-Shell Welds (C1.30)

Item No.	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
C1.30	1	3/2010 (1RFO19)	3rd Interval/ 3rd Period	SG-11 - W65	NRI	98.6%
	1	3/2018 (1RFO23)	4th Interval/ 3rd Period	SG-11 - W65	NRI	100%
	2	3/2007 (2RFO16)	3rd Interval/ 3rd Period	SG-21- W65	NRI	100%
	2	2/2017 (2RFO21)	4th Interval/ 3rd Period	SG-21- W65	NRI	100%

GINNA INSPECTION HISTORY

Currently, the SG components for Ginna satisfy all of the inspection requirements of ASME Code, Section XI, 2013 Edition. A summary of the inspection history for Item Numbers B2.40, C1.10, C1.20, and C1.30 for Ginna is provided in Tables B14 through B17.

Table B14. Ginna SG Primary Side Tubesheet-to-Shell Welds (B2.40)

Item No.	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
B2.40	1	4/2008 (O2008)	4th Interval/ 3rd Period	LHTSW-AR	NRI	99%
	1	10/2015 (O2015)	5th Interval/ 2nd Period	LHTSW-AR	NRI	96.7%

Table B15. Ginna SG Secondary Side Shell Welds (C1.10)

Item No.	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
C1.10	1	9/2009 (O2009)	4th Interval/ 3rd Period	TUS-BR	NRI	98%
	1	5/2017 (O2017)	5th Interval/ 3rd Period	TUS-BR	NRI	99.2%
	1	1/2002 (O2002)	4th Interval/ 1st Period	LST-AR	NRI	>90%
	1	11/2012 (O2012)	5th Interval/ 1st Period	LST-AR	NRI	100%

Table B16. Ginna SG Secondary Side Head Welds (C1.20)

Item No.	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
C1.20	1	1/2002 (O2002)	4th Interval/ 1st Period	USH-AR	NRI	>90%
	1	10/2012 (O2012)	5th Interval/ 1st Period	USH-AR	NRI	99.7%

Table B17. Ginna SG Secondary Side Tubesheet-to-Shell Welds (C1.30)

Item No.	Unit	Date	Interval/Period	Component ID	Exam Results	Coverage
C1.30	1	10/2015 (O2015)	5th Interval/ 2nd Period	TSSW-AR	NRI	99.49%
	1	10/2006 (O2006)	4th Interval/ 2nd Period	TSSW-BR	NRI	99.6%

APPENDIX C

RESULTS OF INDUSTRY SURVEY

Overall Industry Inspection Summary for Code Item Numbers B2.31, B2.32, B2.40, B3.130, C1.10, C1.20, and C1.30

The results of an industry survey of past inspections of SG nozzle-to-shell welds, inside radius sections and shell welds are summarized in Reference [1]. Table C1 provides a summary of the combined survey results for Item Numbers B2.31, B2.32, B2.40, B3.130, C1.10, C1.20, and C1.30. The results of the industry survey identified numerous steam generator (SG) examinations being performed with no service-induced flaws being detected. Performing these examinations adversely impact outage activities including worker exposure, personnel safety, and radwaste. A total of 74 domestic and international boiling water reactor (BWR) and pressurized water reactor (PWR) units responded to the survey and provided information representing all PWR plant designs currently in operation in the United States. 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 1,324 examinations for the components of the affected Item Numbers were conducted, with 1,098 of these specifically for PWR components. The majority of the PWR examinations were performed on SG welds.

A relatively small number of flaws were identified during these examinations which required flaw evaluation. None of these flaws were found to be service-induced. For Item Number B2.40, examinations at two units at a single plant site identified multiple flaws exceeding the acceptance criteria of ASME Code Section XI; however, these were determined to be subsurface-embedded fabrication flaws and not service-induced (see Note 1 for Table 1). For Item Number C1.20, two PWR units reported flaws exceeding the acceptance criteria of ASME Code, Section XI. In the first unit, a single flaw was identified, and was evaluated as an inner diameter surface imperfection. Reference [2] indicates that this was a spot indication with no measurable through-wall depth. This indication is therefore not considered to be service-induced but rather fabrication-related. A flaw evaluation per IWC-3600 was performed for this flaw and it was found to be acceptable for continued operation. In the second unit, multiple flaws were identified (see Note 2 for Table 1). As discussed in References [3] and [4], these flaws were most likely subsurface weld defects typical of thick vessel welds and not service- induced. A flaw evaluation per IWC-3600 was performed for these flaws and they were found to be acceptable for continued operation.

References:

1. *Technical Bases for Inspection Requirements for PWR Steam Generator Class 1 Nozzle-to-Vessel Welds and Class 1 and Class 2 Vessel Head, Shell, Tubesheet-to-Head and Tubesheet-to-Shell Welds*. EPRI, Palo Alto, CA: 2019. 3002015906, ADAMS Accession No. ML20225A141.
2. Letter from F. A. Kearney (Exelon) to U.S. NRC, "Byron Station Unit 2 90-Day Inservice Inspection Report for Interval 3, Period 3, (B2R17)," dated July 29, 2013, Docket No. 50-455, ADAMS Accession Number ML13217A093.
3. Letter from J. M. Sorensen (NMC) to U.S. NRC, "Unit 1 Inservice Inspection Summary Report,

Interval 3, Period 3 Refueling Outage Dates 1-19-2001 to 2-25-2001 Cycle 20 / 05-26-99 to 02-25-2001," May 29, 2001, Docket Nos. 50-282 and 50-306, ADAMS Accession Number ML011550346.

4. Letter from J. P. Solymossy (NMC) to U. S. NRC, "Response to Opportunity for Comment on Task Interface Agreement (TIA) 2003-01, 'Application of ASME Code Section XI, IWB-2430 Requirements Associated with Scope of Volumetric Weld Expansion at the Prairie Island Nuclear Generating Plant' (Tac Nos. MB7294 and MB7295)," April 4, 2003, Docket Nos. 50-282 and 50-306, ADAMS Accession Number ML031040553.

Table C1. Summary of Survey Results

Item Number	Number of Examinations			Number of Reportable Indications		
	BWR	PWR	Total	BWR	PWR	Total
B2.31	0	30	30	0	0	0
B2.32	0	13	13	0	0	0
B2.40	0	183	183	0	Note 1	Note 1
B3.130	0	135	135	0	0	0
C1.10	140	305	445	0	0	0
C1.20	54	319	373	0	Note 2	Note 2
C1.30	32	113	145	0	0	0
Totals	**226**	**1,098**	**1,324**	**0**	**---**	**---**

Note:
1. Two PWR W-2 Loop units at a single plant reported multiple subsurface embedded fabrication flaws.
2. A single PWR W-2 Loop unit reported multiple flaws [3, 4].