



Energy Harbor Nuclear Corp.
Davis-Besse Nuclear Power Station.
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Terry J. Brown
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419-321-7676

September 13, 2021
L-21-214

10 CFR 50.55a

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

Subject:
Davis-Besse Nuclear Power Station
Docket No. 50-346, License No. NPF-3
Proposed Inservice Inspection Alternative RR-A2

In accordance with 10 CFR 50.55a(z)(1), Energy Harbor Nuclear Corp. hereby requests Nuclear Regulatory Commission (NRC) approval of a proposed inservice inspection (ISI) alternative to the American Society of Mechanical Engineers (ASME) Section XI, Table IWB-2500-1, Examination Category B-B, and Table IWC-2500-1, Examination Category C-A and C-B for use at Davis-Besse Nuclear Power Station. The proposed alternative is enclosed and requests to increase the inspection interval for the items from 10 years to 30 years.

NRC staff review and approval of the proposed ISI alternative is respectfully requested by March 1, 2022 to allow for application of the alternative during the spring 2022 refueling outage.

There are no regulatory commitments contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Phil H. Lashley, Manager - Fleet Licensing, at (330) 696-7208.

Sincerely,

A handwritten signature in black ink, appearing to read "Terry J. Brown".

Terry J. Brown

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cc: NRC Region III Administrator
NRC Resident Inspector
NRC Project Manager
Utility Radiological Safety Board

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Proposed Alternative
in Accordance with 10 CFR 50.55a(z)(1)

-- Alternative Provides Acceptable Level of Quality and Safety --
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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 (nozzle-to-shell welds and inside radius sections)
 Examination Category: Class 1, Category B-B, Pressure Retaining Welds in Vessels Other Than Reactor Vessels
 Class 2, Category C-A, Pressure Retaining Welds in Pressure Vessels
 Class 2, Category C-B, Pressure Retaining Nozzle Welds in Vessels
 Item Numbers: B2.40 – Steam Generators (Primary Side), Tubesheet-To-Head Weld
 C1.30 – Tubesheet-to-Shell Weld
 C2.21 – Nozzle-to-Shell (Nozzle to Head or Nozzle to Nozzle) Weld
 C2.22 – Nozzle Inside Radius Section

Component IDs:

Davis-Besse Nuclear Power Station (Davis-Besse)		
Component ID	Component Description	ASME Item No.
RC-SG-1-1-W23	Upper Tubesheet to Upper Primary Head Weld	B2.40
RC-SG-1-2-W23	Upper Tubesheet to Upper Primary Head Weld	B2.40
RC-SG-1-1-W22	Lower Tubesheet to Lower Primary Head Weld	B2.40
RC-SG-1-2-W22	Lower Tubesheet to Lower Primary Head Weld	B2.40
SP-SG-1-1-W65	Shell to Lower Tubesheet Weld	C1.30
SP-SG-1-1-W69	Upper Tubesheet to Shell Weld	C1.30
SP-SG-1-2-W65	Shell to Lower Tubesheet Weld	C1.30
SP-SG-1-2-W69	Upper Tubesheet to Shell Weld	C1.30
SP-SG-1-1-W127-X/Y	24 in. X/Y Axis Steam Outlet Nozzle to Shell Weld	C2.21
SP-SG-1-1-W128-W/X	24 in. W/X Axis Steam Outlet Nozzle to Shell Weld	C2.21
SP-SG-1-2-W127-X/Y	24 in. X/Y Axis Steam Outlet Nozzle to Shell Weld	C2.21
SP-SG-1-2-W128-W/X	24 in. W/X Axis Steam Outlet Nozzle to Shell Weld	C2.21
SP-SG-1-1-W127-X/Y-IR	24 in. X/Y Axis Steam Outlet Nozzle Inside Radius	C2.22
SP-SG-1-1-W128-W/X-IR	24 in. W/X Axis Steam Outlet Nozzle Inside Radius	C2.22
SP-SG-1-2-W127-X/Y-IR	24 in. X/Y Axis Steam Outlet Nozzle Inside Radius	C2.22
SP-SG-1-2-W128-W/X-IR	24 in. W/X Axis Steam Outlet Nozzle Inside Radius	C2.22

2. Applicable Code Edition and Addenda

The fourth 10-year inservice inspection (ISI) interval Code of record for Davis-Besse is the 2007 Edition through 2008 Addenda of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

3. Applicable Code Requirement

ASME Section XI IWB-2500(a), Table IWB-2500-1, Examination Category B-B, and IWC-2500(a), Table IWC-2500-1, Examination Categories C-A and C-B, require examination of the following Item Nos.:

Item No. B2.40 – Volumetric examination of essentially 100 percent of the weld length of all welds during the first Section XI inspection interval. For successive inspection intervals, the examination may be limited to one vessel among the group of vessels performing a similar function. The examination volume is shown in Figure IWB-2500-6.

Item No. C1.30 – Volumetric examination of essentially 100 percent of the weld length of the tubesheet-to-shell welds during each Section XI inspection interval. In the case of multiple vessels of similar design, size, and service (such as steam generators, heat exchangers), the required examinations may be limited to one vessel or distributed among the vessels. The examination volume is shown in Figure IWC-2500-2.

Item No. C2.21 – Volumetric and surface examination of all nozzle welds at terminal ends of piping runs during each Section XI inspection interval. In the case of multiple vessels of similar design, size, and service (such as steam generators, heat exchangers), the required examinations may be limited to one vessel or distributed among the vessels. The examination area and volume are shown in Figures IWC-2500-4(a), (b), or (d).

Item No. C2.22 – Volumetric examination of all nozzle inside radius sections at terminal ends of piping runs during each Section XI inspection interval. In the case of multiple vessels of similar design, size, and service (such as steam generators, heat exchangers), the required examinations may be limited to one vessel or distributed among the vessels. The examination volume is shown in Figures IWC-2500-4(a), (b), or (d).

4. **Reason for Request**

The Electric Power Research Institute (EPRI) performed assessments in References [E-1] and [E-2] of the basis for the ASME Section XI examination requirements specified for the above listed ASME Section XI, Division 1 examination categories for steam generator 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 [E-1] and [E-2] reports concluded that the current ASME Code Section XI inspection interval of 10 years can be increased significantly with no impact to plant safety. Based on the conclusions of the two EPRI reports, supplemented by plant-specific evaluations contained herein, Energy Harbor Nuclear Corp. is requesting an alternate inspection interval for the subject welds. The Reference [E-1] and [E-2] reports were developed consistent with the recommendations provided in EPRI's White Paper on PFM [E-12].

5. **Proposed Alternative and Basis for Use**

Energy Harbor Nuclear Corp. is requesting an inspection alternative to the examination requirements of ASME Section XI, Tables IWB-2500-1 and IWC-2500-1, for the following examination categories and item numbers:

ASME Category	Item No.	Description
B-B	B2.40	Steam generators (primary side), tubesheet-to-head weld
C-A	C1.30	Tubesheet-to-shell weld
C-B	C2.21	Nozzle-to-shell (nozzle to head or nozzle to nozzle) weld
C-B	C2.22	Nozzle inside radius section

In 2014 (first period of the fourth inspection interval), both Davis-Besse SGs were replaced. The new SG welds and components received the required preservice inspection (PSI) examinations prior to service followed by ISI examinations through the second period of the current fourth inspection interval.

The proposed alternative is to increase the inspection interval for these item numbers for the replacement steam generators at Davis-Besse to 30 years (from the current ASME Code, Section XI Division 1 10-year requirement) for the remainder of the fourth 10-year inspection interval and through the sixth 10-year inspection interval, which is currently scheduled to end on September 20, 2042.

Technical Basis

A summary of the key aspects of the technical basis for this request is provided below. The applicability of the technical basis to Davis-Besse is shown in Attachment 1.

Degradation Mechanism Evaluation

An evaluation of degradation mechanisms that could potentially impact the reliability of the SG welds and components was performed in References [E-1] and [E-2]. The degradation mechanisms that were evaluated 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 welds and components covered in this request. Therefore, these fatigue-related mechanisms were considered in the PFM and DFM evaluations in References [E-1] and [E-2].

Stress Analysis

Finite element analyses (FEA) were performed in References [E-1] and [E-2] to determine the stresses in the SG welds and components covered in this request. The analyses were performed using representative pressurized water reactor (PWR) geometries, bounding transients, and typical material properties. The results of the stress analyses were used in a flaw tolerance evaluation. The applicability of the FEA analysis to Davis-Besse is demonstrated in Attachment 1 and confirms that all plant-specific requirements are met. Therefore, the evaluation results and conclusions of References [E-1] and [E-2] are applicable to Davis-Besse.

In the selection of the transients in Section 5 of References [E-1] and [E-2] and the subsequent stress analyses in Section 7, test conditions beyond a system leakage test were not considered since pressure tests at Davis-Besse are performed at normal operating conditions. No system hydrostatic testing has been performed at Davis-Besse since the plant went into operation.

Flaw Tolerance Evaluation

Flaw tolerance evaluations were performed in References [E-1] and [E-2] consisting of PFM evaluations and confirmatory DFM evaluations. The results of the PFM analyses indicate that, after a PSI followed by subsequent ISIs, the Nuclear Regulatory Commission's (NRC's) safety goal of 10^{-6} failures per year is met. The PFM analysis in Reference [E-1] was performed using the **PR**obabilistic **OptiM**ization of **InSpE**ction (**PROMISE**) Version 1.0 software, developed by

Structural Integrity Associates. As part of the NRC's review of an alternative request submitted by Southern Nuclear Company (SNC), the NRC performed an audit of the **PROMISE** Version 1.0 software as discussed in the NRC's audit plan dated May 14, 2020 (ADAMS Accession No. ML20128J311) and the audit summary report issued by letter dated December 10, 2020 (ADAMS Accession No. ML20258A002). The PFM analysis in Reference [E-2] was performed using the **PROMISE** Version 2.0 software, which has not been audited by the NRC. The main difference between the two versions is that in **PROMISE** Version 1.0, a single, user-specified examination coverage value is applied to all inspections assumed over the component evaluation time period, whereas in **PROMISE** Version 2.0, a unique, user-specified examination coverage value can be applied to each inspection assumed over the component evaluation period. In both Versions 1.0 and 2.0, the software assumes 100 percent coverage for the PSI examination.

In Section 8.2.2.2 of Reference [E-1] and Section 8.3.2.2 of Reference [E-2], the number of fabrication flaws for the nozzle-to-vessel weld was assumed to be 1.0 per nozzle. In Section 8.2.2.2 of Reference [E-1], a nozzle flaw density of 0.001 flaws per nozzle was assumed for the nozzle inside radius sections. In the safety evaluation (SE) for Vogtle Electric Generating Plant, Units 1 and 2 (Reference [E-13]), the NRC indicated that a nozzle flaw density of 0.1 flaws per nozzle is the acceptable number at the nozzle inside radius. Sensitivity studies performed in Section 8.2.4.3.4 in Reference [E-1] indicated that by changing the number of flaws in the nozzle inside radius sections from 0.001 to 0.1, the probabilities of leak and rupture increased by two orders of magnitude but were still significantly below the acceptance criterion of 1×10^{-6} per year. Since the sensitivity studies performed in References [E-1] and [E-2] involve PSI/ISI scenarios that are different from those at Davis-Besse, supplemental analyses were performed for the plant-specific inspection scenarios at Davis-Besse as detailed below.

For the Davis-Besse replacement SGs, PSI examinations have been performed followed by ISI examinations in the subsequent two periods following SG replacement. Plant-specific evaluations were performed assuming PSI examinations only (since the ISI examinations for the current interval have not been completed, credit was not taken for these examinations). The PSI/ISI scenario considered is therefore PSI to be followed by a 30-year ISI examination (PSI+30).

First, evaluations were performed for the critical nozzle inside radius section and nozzle-to-shell weld locations identified in Reference [E-1]. Davis-Besse does not have feedwater nozzle Item Nos. C2.21 and C2.22; therefore, the evaluation was performed for the main steam nozzle. From Reference [E-1], the critical Case ID for the main steam nozzle inside radius section is SGB-P1N. An evaluation similar to that shown in Table 8-28 of Reference [E-1] was performed for this location assuming a nozzle flaw density of 0.1, a stress multiplier of 1.5, a fracture toughness of 200 ksi $\sqrt{\text{in}}$ and a standard deviation 5 ksi $\sqrt{\text{in}}$ as described by the

NRC in Reference [E-13]. The results of the evaluation, using **PROMISE** Version 1.0, are summarized in Table E-1 and show that after 80 years of plant operation from the last completed 10-year ISI interval examinations, the probabilities of rupture and leakage are well below the acceptance criterion of 1.0×10^{-6} by at least three orders of magnitude. The results indicate that a much higher stress multiplier than 1.5 could have been used, and the acceptance criteria would still be met.

Table E-1

Sensitivity to Combined Effects of Fracture Toughness, Stress, and Nozzle Flaw Density for 80 Years for the Davis-Besse Main Steam Nozzle Inside Radius Section (Case ID SGB-P1N from Reference [E-1])

Time (yr)	Probability per Year for Combined Case $K_{IC} = 200 \text{ ksi}\sqrt{\text{in.}}$ $SD = 5 \text{ ksi}\sqrt{\text{in.}}$ Stress Multiplier = 1.5 Nozzle Flaw Density = 0.1 PSI+30	
	Rupture	Leak
10	1.00E-09	1.00E-09
20	5.00E-10	5.00E-10
30	3.33E-10	3.33E-10
40	2.50E-10	2.50E-10
50	2.00E-10	2.00E-10
60	1.67E-10	1.67E-10
70	1.43E-10	5.71E-10
80	1.25E-10	2.13E-09

For the main steam nozzle-to-shell weld, Table 8-15 of Reference [E-1] indicates that the critical Case ID is SGB-P3A. For the evaluation, a flaw density of 1.0 flaw per weld was assumed, consistent with the evaluations in Reference [E-1]. A fracture toughness of 200 ksi $\sqrt{\text{in}}$ and standard deviation of 5 ksi $\sqrt{\text{in}}$ were also used with a stress multiplier of 1.9. (This stress multiplier was chosen to result in probability of rupture or probability of leakage close to the acceptance criteria after 80 years.) The results of the evaluation, using **PROMISE** Version 1.0, are summarized in Table E-2 and show that after 60 years of plant operation the probabilities of rupture and leakage are well below the acceptance criterion of 1.0×10^{-6} . After 80 years of plant operation the probability of leakage is still below the acceptance criteria. The probability of rupture after 80 years is just above the acceptance criteria, which should be acceptable since a very high stress multiplier of 1.9 was conservatively used in the evaluation. A slightly lower stress multiplier would have resulted in an acceptable probability of rupture after 80 years.

Table E-2

Sensitivity to Combined Effects of Fracture Toughness, Stress, and Weld Flaw Density for 80 Years for the Davis-Besse Main Steam Nozzle-to-Shell Weld (Case ID SGB-P3A from Reference [E-1])

Time (yr)	Probability per Year for Combined Case K _{IC} = 200 ksi $\sqrt{\text{in}}$. SD = 5 ksi $\sqrt{\text{in}}$. Stress Multiplier = 1.9 Nozzle Flaw Density = 1 PSI+30	
	Rupture	Leak
10	1.00E-08	1.00E-08
20	3.50E-08	5.00E-09
30	6.83E-07	3.33E-09
40	5.15E-07	2.50E-09
50	4.38E-07	2.00E-09
60	4.48E-07	1.67E-09
70	8.64E-07	1.43E-09
80	2.95E-06	1.25E-09

For the remaining SG welds, Table 8-32 of Reference [E-2] indicates that the critical Case ID is SGPTH-P4A. This case was evaluated for the inspection scenario of PSI+30, a flaw density of 1.0 flaw per weld, a fracture toughness of 200 ksi√in and a standard deviation 5 ksi√in with a stress multiplier of 1.6. (This stress multiplier was chosen to result in probability of rupture or probability of leakage close to the acceptance criteria after 80 years.) The results of the evaluation, using **PROMISE** Version 2.0, are summarized in Table E-3 and show that after 80 years of plant operation the probabilities of rupture and leakage are well below the acceptance criterion of 1.0×10^{-6} .

Table E-3

Sensitivity to Combined Effects of Fracture Toughness, Stress, and Weld Flaw Density for 80 Years for the Remaining Davis-Besse SG Welds (Case ID SGPTH-P4A from Reference [E-2])

Time (yr)	Probability per Year for Combined Case $K_{IC} = 200 \text{ ksi}\sqrt{\text{in.}}$ $SD = 5 \text{ ksi}\sqrt{\text{in.}}$ Stress Multiplier = 1.6 Nozzle Flaw Density = 1 PSI+30	
	Rupture	Leak
10	1.00E-08	1.00E-08
20	1.00E-08	5.00E-09
30	2.57E-07	3.33E-09
40	1.98E-07	2.50E-09
50	1.82E-07	2.00E-09
60	1.93E-07	1.67E-09
70	2.26E-07	1.43E-09
80	3.10E-07	1.25E-09

The above evaluations indicate that for Davis-Besse, the acceptance criterion is met with significant margins when conservative assumptions for the key variables are assumed for the plant-specific inspection scenario. The difference between the geometrical parameters evaluated in References [E-1] and [E-2] and those at Davis-Besse are summarized in Table E-4. This table shows that the largest variation of the R/t ratio between the geometry evaluated in Reference [E-1] and that at Davis-Besse is 28 percent, which is lower than the stress multipliers applied in the sensitivity studies in Tables E-1 through E-3.

Table E-4

Comparison of Model Geometry in Reference [E-1] with Davis-Besse

Component	Parameter	Modeled in EPRI Report	Davis-Besse	% Variation
SG Shell	OD (in)	151.125	148.125	1.99
	ID (in)	137.875	137.875	0
	(R/t) _{mean}	10.91	13.95	27.93
MS Nozzle	OD (in)	31.3125	27.38	12.57
	ID (in)	22.25	20.38	8.40
	(R/t) _{mean}	2.96	3.41	15.51

The ASME Code, Section XI ISI Code of record for Davis-Besse is the 2007 Edition through the 2008 Addenda. ASME Code, Section XI, Mandatory Appendix I, I-2120, “*Other Vessels*,” indicates that ultrasonic examination of all other vessels greater than 2 inches in thickness shall be conducted in accordance with Section V, Article 4. However, I-2600, “*Mandatory Appendix VIII Examination*,” states that, for components to which Appendix VIII is not applicable, the examination procedures, personnel and equipment qualified in accordance with Appendix VIII may be applied, provided each of the components, materials, sizes and shapes are within the scope of the qualified procedures.

The EPRI reports documented in Reference [E-1] and Reference [E-2] used a Section XI, Appendix VIII-based probability of detection (POD) curve in the PFM evaluation because most ISI examinations of major plant Class 1 and Class 2 components are performed using Appendix VIII procedures. However, for Class 2 components, the use of Appendix VIII procedures is plant-specific. Many plants adopt and use their Appendix VIII procedures for major Class 2 components (such as SGs) for consistency across all their examinations. In the case of Davis-Besse, Energy Harbor Nuclear Corp. does not use Appendix VIII procedures for all the examination categories included in the request for alternative, so use of the Appendix VIII POD curve may not be appropriate for all of the items. Despite this, the evaluation contained in the EPRI reports, as documented in Reference [E-1]

and Reference [E-2], demonstrates that the 30-year interval is supported for these welds, regardless of the POD curve used.

The plant-specific PFM evaluations presented in Tables E-1 through E-4 indicate that with conservative inputs of the critical parameters, the probabilities of rupture and leakage are below the acceptance criterion of 1.0×10^{-6} . The analyses involve conservative assumptions with regards to the PSI/ISI scenarios. No credit was taken for examinations performed in the current inspection interval for the plant since the examinations have not been completed. Furthermore, the evaluation was performed for 80 years, which is longer than the extension being sought by Energy Harbor Nuclear Corp. in this request for alternative.

The DFM evaluations in Table 8-31 of Reference [E-1] and Table 8-3 of Reference [E-2] provide verification of the above PFM results for Davis-Besse by demonstrating that it takes approximately 80 years for a postulated flaw with an initial depth equal to ASME Code, Section XI acceptance standards to grow to a depth where the maximum stress intensity factor (K) exceeds the ASME Code, Section XI allowable fracture toughness.

Inspection History

Inspection history for Davis-Besse (including examinations performed to date, examination findings, inspection coverage, and relief requests) is presented in Attachment 2. As shown in the attachment, all welds/components have examinations coverage greater than 90 percent (essentially 100 percent). Also, as shown in Attachment 2, no flaws that exceeded the ASME Code, Section XI acceptance standards were identified during any examinations.

Industry Survey

The inspection history for these components as obtained from an industry survey is presented in Attachment 3. The results of the survey indicate that these components are very flaw tolerant.

Conclusion

The SG welds and nozzles considered in this request for alternative are very flaw tolerant. PFM and DFM evaluations performed as part of the technical basis reports [E-1] and [E-2], as supplemented by plant-specific evaluations performed as part of this request for alternative, demonstrate that using conservative PSI/ISI inspection scenarios for Davis-Besse, supports the NRC safety goal of 10^{-6} failures per reactor year is met with considerable margins. Plant-specific applicability of the technical basis to Davis-Besse is demonstrated in Attachment 1. The requested inspection interval provides an acceptable level of quality and safety in lieu of the current ASME Section XI 10-year inspection frequency.

Operating and examination experience demonstrates that these components have performed with very high reliability, mainly due to their robust design. Attachment 2 shows the examination history for the SG welds examined in the two most recent 10-year inspection intervals (the third interval plus first and second periods of the fourth interval).

In addition to the required PSI examinations for these SG welds and components, Energy Harbor Nuclear Corp. has performed examinations through three complete 10-year intervals for the original SGs and through the second period of the current fourth inspection interval for the replacement SGs.

No flaws that exceeded the ASME Code, Section XI acceptance standards were identified during any examinations, as shown in Attachment 2.

In addition, all other inspection activities, including the system leakage test (Examination Categories B-P and C-H) will continue to be performed consistent with this request for alternative and in accordance with all other ASME Section XI requirements, providing further assurance of safety.

Finally, as discussed in Reference [E-3], for situations where no active degradation mechanism is present, subsequent ISI examinations do not provide additional value after PSI has been performed and the inspection volumes have been confirmed to be free of defects.

Therefore, Energy Harbor Nuclear Corp. requests the NRC grant this proposed alternative in accordance with 10 CFR 50.55a(z)(1).

6. Duration of Proposed Alternative

The proposed alternative is requested for the remainder of the fourth 10-year inspection interval and through the sixth 10-year inspection interval for Davis-Besse. The sixth 10-year inspection interval is currently scheduled to end on September 20, 2042, recognizing that the existing 60-year license expires April 22, 2037.

7. Precedents

Relief from the ASME Section XI Examination Category C-B (Item Nos. C2.21 and C2.22) surface and volumetric examinations based on the Reference [E-1] technical basis report was granted for SNC in January 2021.

- Letter from M. T. Markley (NRC) to C. A. Gayheart (SNC), "Vogtle Electric Generating Plant, Units 1 and 2 – Relief Request for Proposed Inservice Inspection Alternative VEGP-ISI-ALT-04-04 to the Requirements of ASME Code (EPID L-2020-LLR-0109)," dated January 11, 2021 (ADAMS Accession No. ML20352A155), Reference [E-13].

Relief from ASME Section XI Examination Category B-B (Item Nos. B2.31 and B2.40), Examination Category B-D (Item No. B3.130), Examination Category C-A (Item Nos. C1.10, C1.20, and C1.30), and Category C-B (Item Nos. C2.21 and C2.22) surface and volumetric examinations based on the Reference [E-1] and [E-2] technical basis reports was granted for Dominion Energy Nuclear Connecticut, Inc. (Dominion) in July 2021.

- SE from J. G. Danna (NRC) to D. G. Stoddard (Dominion), "Millstone Power Station Unit 2 – Authorization and Safety Evaluation for Alternative Request No. RR-05-06 (EPID L-2020-LLR-0097)," dated July 16, 2021 (ADAMS Accession No. ML21167A355), Reference [E-14].

In addition, the following is a list of approved actions (including relief requests and topical reports) related to inspections of SG welds and components:

- Letter from J. W. Clifford (NRC) to S. E. Scace (Northeast Nuclear Energy Company), "Safety Evaluation of Relief Requests Associated with the First and Second 10-Year Interval of the Inservice Inspection (ISI) Plan, Millstone Nuclear Power Station, Unit No. 3 (TAC No. MA5446)," dated July 24, 2000 (ADAMS Accession No. ML003730922).
- Letter from R. L. Emch (NRC) to J. B. Beasley, Jr. (Southern Nuclear Operating Company, Inc.), "Second 10-Year Interval Inservice Inspection Program Plan Requests for Relief 13, 14, 15, 21 and 33 for Vogtle Electric Generating Plant, Units 1 and 2 (TAC No. MB0603 and MB0604)," dated June 20, 2001 (ADAMS Accession No. ML011640178).
- Letter from T. H. Boyce (NRC) to C. L. Burton (Carolina Power & Light Company), "Shearon Harris Nuclear Power Plant, Unit 1 – Requests for Relief 2R1-019, 2R1-020, 2R1-021, 2R1-022, 2R2-009, 2R2-010, and 2R2-011 for the Second 10-Year Interval Inservice Inspection Program Plan (TAC Nos. ME0609, ME0610, ME0611, ME0612, ME0613, ME0614, and ME0615)," dated January 7, 2010 (ADAMS Accession No. ML093561419).
- Letter from M. Khanna (NRC) to D. A. Heacock (Dominion), "Millstone Power Station, Unit No. 2 – Issuance of Relief Requests RR-89-69 Through RR-89-78 Regarding Third 10-Year Interval Inservice Inspection Plan (TAC Nos. ME5998 Through ME6006)," dated March 12, 2012 (ADAMS Accession No. ML120541062).
- Letter from R. J. Pascarelli (NRC) to E. D. Halpin (Pacific Gas and Electric Company), "Diablo Canyon Power Plant, Unit Nos. 1 and 2 – Relief Request NDE-SG-MS-IR, Main Steam Nozzle Inner Radius Examination Impracticality, Third 10-Year Interval American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Inservice

Inspection Program (CAC Nos. MF6646 and MF6647),” dated December 8, 2015 (ADAMS Accession No. ML15337A021).

There are also precedents related to similar topical reports that justify relief for Class 1 nozzles:

- Based on studies presented in Reference [E-4], the NRC approved extending PWR reactor vessel nozzle-to-shell welds from 10 to 20 years in Reference [E-5].
- Based on work performed in BWRVIP-108 [E-6] and BWRVIP-241 [E-8], the NRC approved the reduction of BWR vessel feedwater nozzle-to-shell weld examinations (Item No. B3.90 for BWRs from 100 percent to a 25 percent sample of each nozzle type every 10 years) in References [E-7] and [E-9]. The work performed in BWRVIP-108 and BWRVIP-241 provided the technical basis for ASME Code Case N-702 [E-10], which has been conditionally approved by the NRC in Revision 19 of Regulatory Guide 1.147 [E-11].

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
O.D.	outside diameter
PFM	probabilistic fracture mechanics
POD	probability of detection
PSI	preservice inspection
PWR	pressurized water reactor
SCC	stress corrosion cracking
SE	safety evaluation

SG steam generator
WEC Westinghouse Electric Company

9. References

- E-1. *Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Inside Radius Sections*. EPRI, Palo Alto, CA: 2019. 3002014590 (ADAMS Accession No. ML19347B107).
- E-2. *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).
- E-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.
- E-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.
- E-5. NRC, "Revised 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).
- E-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.
- E-7. 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).
- E-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.

- E-9. 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).
- E-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.
- E-11. NRC Regulatory Guide 1.147, Revision 19, "Inservice Inspection Code Case Acceptability, ASME Code Section XI, Division 1," dated October 2019.
- E-12. 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).
- E-13. Letter from M. T. Markley (NRC) to C. A. Gayheart (SNC), "Vogtle Electric Generating Plant, Units 1 and 2 – Relief Request for Proposed Inservice Inspection Alternative VEGP-ISI-ALT-04-04 to the Requirements of ASME Code (EPID L-2020-LLR-0109)," dated January 11, 2021 (ADAMS Accession No. ML20352A155).
- E-14. SE from J. G. Danna (NRC) to D. G. Stoddard (Dominion), "Millstone Power Station Unit 2 – Authorization and Safety Evaluation for Alternative Request No. RR-05-06 (EPID L-2020-LLR-0097)," dated July 16, 2021 (ADAMS Accession No. ML21167A355).

ATTACHMENT 1

PLANT-SPECIFIC APPLICABILITY

DAVIS-BESSE

Section 9 of References [1-1] and [1-2] provide 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 Davis-Besse is provided in Table 1-1 and indicates that all plant-specific requirements are met. Therefore, the results and conclusions of the EPRI reports are applicable to Davis-Besse.

Table 1-1

Applicability of References [1-1] and [1-2] Representative Analyses to Davis-Besse

Item No. B2.40 (SG Primary Side Shell Welds)

Category	Requirement from Reference [1-1]	Applicability to Davis-Besse
General Requirements	<p>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.</p>	<p>For the replacement SGs that were installed in 2014 and are currently in service, Davis-Besse 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 any portion of the vessel.</p>
	<p>The materials of the SG vessel heads and tubesheet must be low alloy ferritic steels that conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.</p>	<p>The Davis-Besse SG vessel heads and tubesheet are fabricated of SA-508, Gr. 3 Class 2 material (Reference [1-3] and Table A-5 of Reference [1-4]). The RT_{NDT} values for the Davis-Besse SG vessel heads and tubesheet materials are 0°F or less (so the RT_{NDT} of 60°F used in the EPRI report is bounding).</p> <p>This material is a low alloy ferritic steel that conforms to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.</p>
Specific Requirements	<p>The weld configurations must conform to those shown in Figure 1-1 and Figure 1-2 of Reference [1-1].</p>	<p>The Davis-Besse tubesheet-to-shell weld configuration is shown in Figure 1-2 below and conforms to Figure 1-2 of Reference [1-1].</p>
	<p>The SG vessel dimensions must be within 10% of the upper and lower bounds of the values</p>	<p>The Davis-Besse SG vessel dimensions are as follows:</p>

Category	Requirement from Reference [1-1]	Applicability to Davis-Besse
	<p>provided in the table in Section 9.4.3 of Reference [1-1].</p>	<ul style="list-style-type: none"> • SG Lower Head diameter = 131.2" • SG Upper Shell diameter = 151.125" <p>The dimension of the upper shell is within 10% of that specified in Table 9-2 in Section 9.4.3 of Reference [1-1] for B&W plants (Reference [1-5]).</p> <p>The dimension of the lower head is inconsistent with the 149" diameter given in the table. This diameter was assumed the same as the upper shell but did not account for the reduction in diameter of the head. Upon comparison with Figure 4-3 of Reference [1-1], it can be seen that the head dimension is consistent with that of the B&W design evaluated and is therefore deemed to be within acceptable geometrical tolerances.</p>
	<p>The component must experience transients and cycles bounded by those shown in Table 5-7 of Reference [1-1] over a 60-year operating life.</p>	<p>As shown in Table 1-2, there are slight variations on some temperature values between Davis-Besse and the Reference [1-1] values. However, the Davis-Besse number of cycles projected to occur over a 60-year life are significantly lower than those shown in Table 5-7 of Reference [1-1] for B&W plants.</p>

Item No. C1.30 (SG Secondary Side Shell Welds)

Category	Requirement from Reference [1-1]	Applicability to Davis-Besse
<p>General Requirements</p>	<p>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</p>	<p>For the replacement SGs that were installed in 2014 and are currently in service, Davis-Besse 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</p>

Category	Requirement from Reference [1-1]	Applicability to Davis-Besse
	require more frequent examinations and other plant actions outside the scope of this report's guidance.	any portion of the vessel.
	The materials of the SG vessel shell and tubesheet must be low alloy ferritic steels that conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.	<p>The Davis-Besse SG vessel shell and tubesheet are fabricated of SA-508, Gr. 3 Class 2 material (Reference [1-3] and Table A-5 of Reference [1-4]). The RT_{NDT} values for the Davis-Besse SG vessel shell material is 0°F or less (so the RT_{NDT} of 60°F used in the EPRI report is bounding).</p> <p>This material is a low alloy ferritic steel that conforms to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.</p>
Specific Requirements	The weld configurations must conform to those shown in Figure 1-7 and Figure 1-8 of Reference [1-1].	The Davis-Besse weld configuration is shown in Figure 1-3 and conforms to Figure 1-8 of Reference [1-1].
	The SG vessel dimensions must be within 10% of the upper and lower bounds of the values provided in the table in Section 9.4.4 of Reference [1-1].	<p>The Davis-Besse SG vessel dimensions are as follows:</p> <ul style="list-style-type: none"> • SG Lower Head diameter = 131.2" • SG Upper Shell diameter = 151.125" <p>These dimensions are within 10% of those specified in Table 9-3 in Section 9.4.4 of Reference [1-1] for B&W plants (Reference [1-5]).</p> <p>The dimension of the lower head is inconsistent with the 149" diameter given in the table. This diameter was assumed the same as the upper shell but did not account for the reduction in diameter of the head. Upon comparison with Figure 4-3 of Reference [1-1], it can be seen that the head dimension is consistent with that of the B&W design evaluated and is therefore deemed to be within acceptable geometrical tolerances.</p>

Category	Requirement from Reference [1-1]	Applicability to Davis-Besse
	The component must experience transients and cycles bounded by those shown in Table 5-9 of Reference [1-1] over a 60-year operating life.	As shown in Table 1-3, there are slight variations on some temperature values between Davis-Besse and the Reference [1-1] values. However, the Davis-Besse number of cycles projected to occur over a 60-year life are significantly lower than those shown in Table 5-9 of Reference [1-1] for B&W plants.

Item Nos. C2.21 and C2.22 (MS Nozzle to Shell Welds and Inside Radius Sections)

Category	Requirement from Reference [1-2]	Applicability to Davis-Besse
General Requirements	The nozzle-to-shell weld shall be one of the configurations shown in Figure 1-1 or Figure 1-2 of Reference [1-2].	<p>The Davis-Besse MS nozzle-to-shell weld is shown in Figure 1-4 below and is representative of the configuration shown in Figure 1-2 of Reference [1-2].</p> <p>Per Section 4.3.1.3, Item 3 of Reference [1-2], B&W plants (like Davis-Besse) do not have FW nozzles welded into the SG shells (the nozzle is actually a bolted joint) and have multiple penetrations in the shell that riser pipes enter to provide feedwater flow to the feedwater ring inside the SG. There are therefore no Item Nos. C2.21 or C2.22 components for the FW nozzle.</p>
	The materials of the SG shell, FW nozzles, and MS nozzles must be low alloy ferritic steels that conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.	<p>The Davis-Besse SG side shell, and MS nozzles are fabricated of SA-508, Gr. 3 Class 2 material (Reference [1-3] and Table A-5 of Reference [1-4]). The RT_{NDT} value for the material of Davis-Besse SG nozzle-to-shell welds is 0°F or less (so the RT_{NDT} of 60°F used in the EPRI report is bounding).</p> <p>This material is a low alloy ferritic steel that conforms to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.</p>

Category	Requirement from Reference [1-2]	Applicability to Davis-Besse
		Per above, there are no Item Nos. C2.21 or C2.22 components for the FW nozzle.
	The SG must not experience more than the number of all transients shown in Table 5-5 of Reference [1-2] over a 60-year operating life.	As shown in Table 1-4, the Davis-Besse SGs are not projected to experience more than the number of transients shown in Table 5-5 of Reference [1-2].
SG Feedwater Nozzle	The piping attached to the FW nozzle must be 14-inch to 18-inch NPS.	Per above, there are no Item Nos. C2.21 or C2.22 components for the FW nozzle. The header piping attached to the FW risers is 14-inch NPS per Reference [1-5].
	The FW nozzle design must have an integrally attached thermal sleeve.	Per above, there are no Item Nos. C2.21 or C2.22 components for the FW nozzle.
	Auxiliary feedwater nozzles connected directly to the SG are not covered in this evaluation.	N/A for Davis-Besse.
SG Main Steam Nozzle	For Westinghouse and CE SGs, the piping attached to the SG main steam nozzle must be 28-inch to 36-inch NPS.	N/A for Davis-Besse (B&W design).
	For B&W SGs, the piping attached to the main steam nozzle must be 22-inch to 26-inch NPS.	The piping attached to the Davis-Besse MS nozzle is 24" Sch. 60 per Reference [1-5].
	The SG must have one main steam nozzle that exits the top dome of the SG. For B&W plants, there may be more than one main steam nozzle; it will exit the side of the SG.	Davis-Besse is a B&W design, with the main steam nozzles exiting the side of the SG (as shown in Figure 1-1).
	The main steam nozzle shall not significantly protrude into the SG (for example, see Figure 4-7 of Reference [1-2]) or have a unique nozzle weld configuration (for example, see Figure 4-6	Figure 4-7 of Reference [1-2] is a CE System 80 design. Figure 4-6 of Reference [1-2] is a Westinghouse two-loop design. Davis-Besse is a B&W design, so these figures do not

Category	Requirement from Reference [1-2]	Applicability to Davis-Besse
	of Reference [1-2]).	apply. As shown in Figure 1-4, the Davis-Besse MS nozzle configuration does not protrude significantly into the SG as shown in Figure 4-7 of Reference [1-2] and does not have a unique weld configuration as shown in Figure 4-6 of Reference [1-2] (Reference [1-3] and [1-5]).

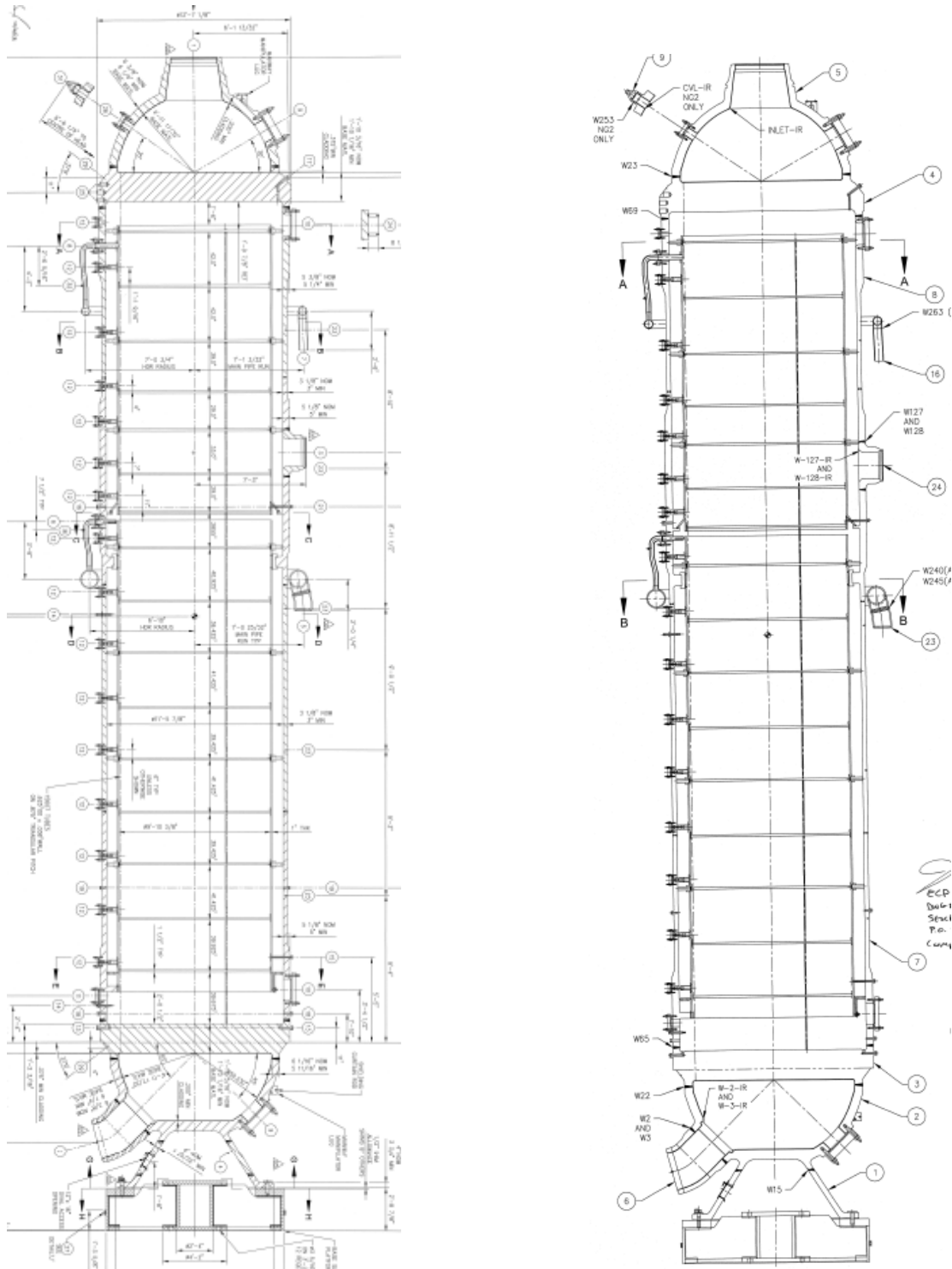
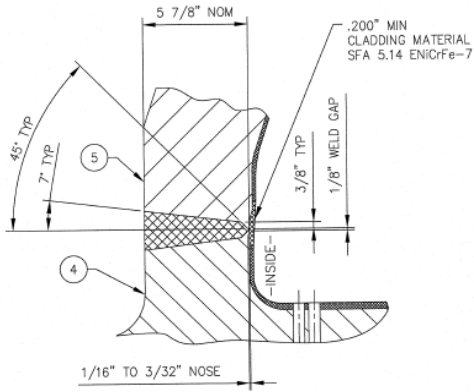
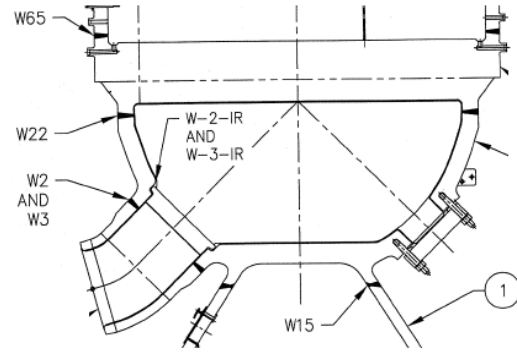
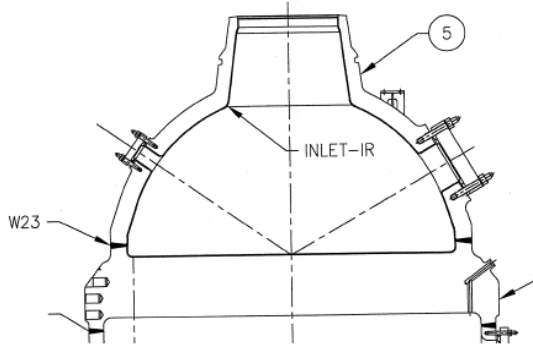
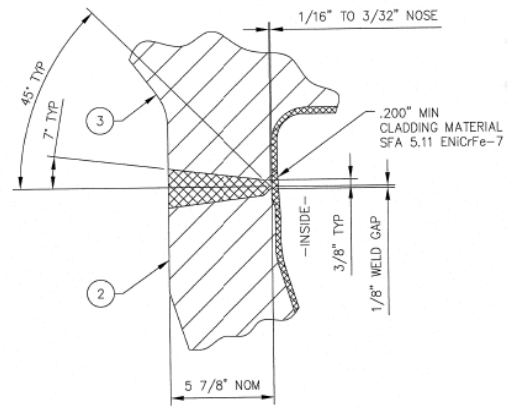


Figure 1-1
Davis-Besse Steam Generator Layout [1-3, 1-5]



UPPER PRIMARY HEAD TO
UPPER TUBESHEET (W23)
SCALE: 3"=1'-0"



LOWER PRIMARY HEAD TO
LOWER TUBESHEET (W22)
SCALE: 3"=1'-0"

Figure 1-2
Davis-Besse Item No. B2.40 Weld Configuration [1-3]

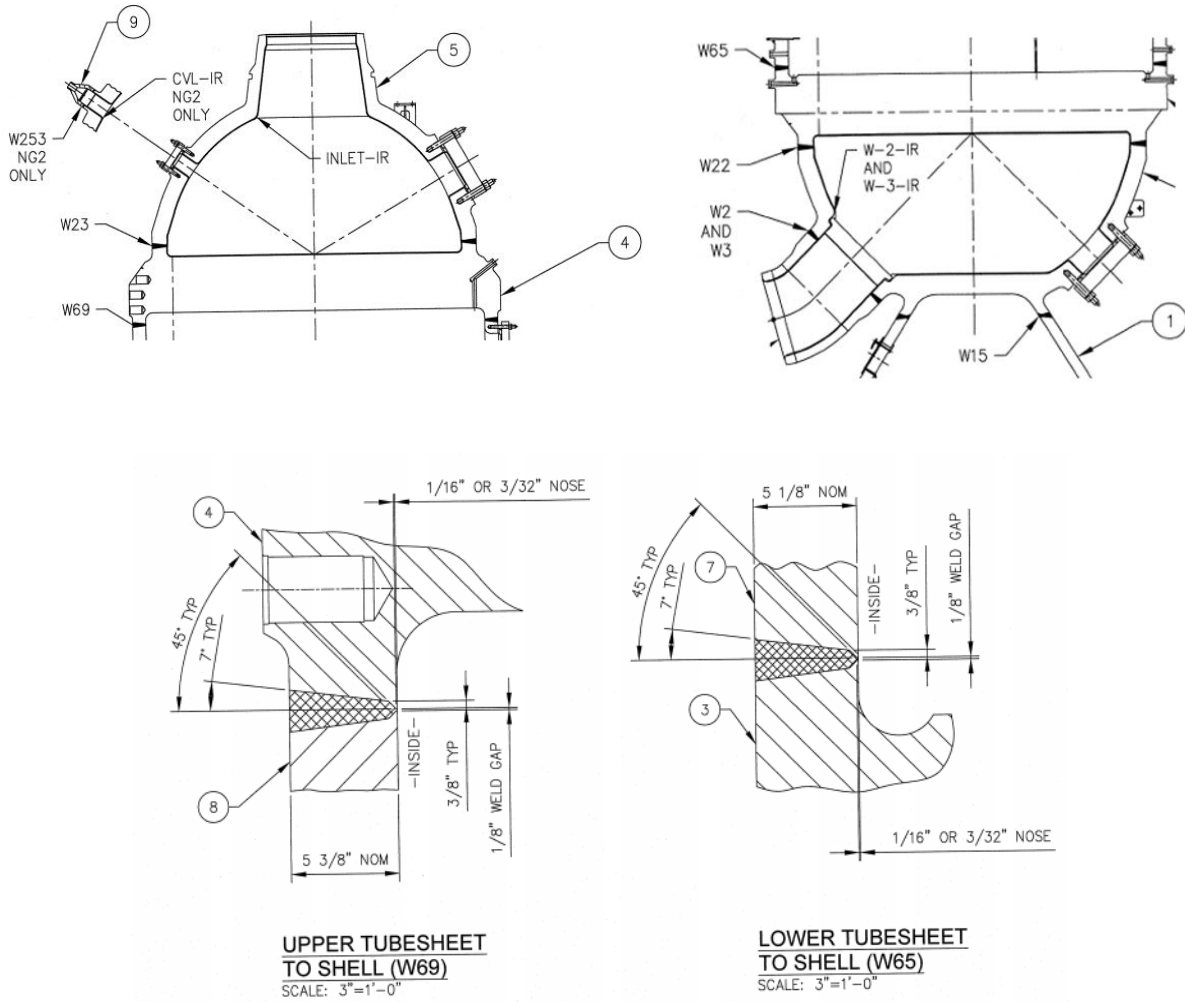
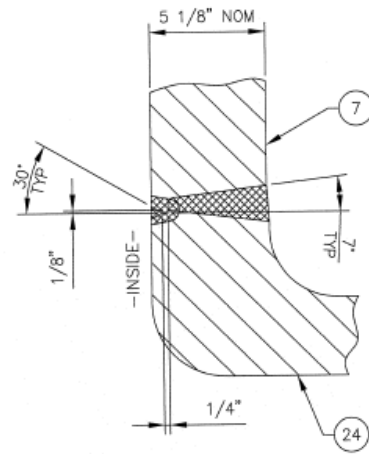
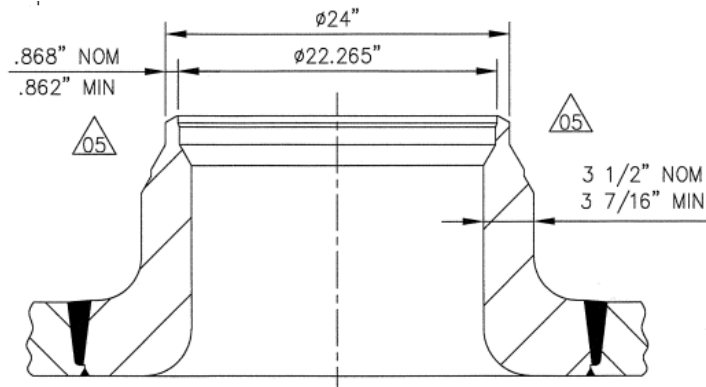


Figure 1-3
Davis-Besse Item No. C1.30 Weld Configuration [1-3]



**STEAM OUTLET NOZZLE
TO SHELL (W127, W128)
(W-127-IR, W-128-IR)**
SCALE: 3"=1'-0"



3 STEAM OUTLET NOZZLE
SCALE: 1 1/2"=1'-0"
TYPICAL (2) LOCATIONS

Figure 1-4
Davis-Besse Main Steam Nozzle Configuration [1-3, 1-5]

Table 1-2

Davis-Besse Data for Thermal Transients for Stress Analysis of the PWR SG Primary-Side Head Welds (Comparison to Table 5-7 of Reference [1-1])

Transient	Max Thot °F	Min Thot °F	Max Tcold °F	Min Tcold °F	Max Press PSIG	Min Press PSIG	60-Year Projected Cycles
Heatup/Cooldown EPRI Report 3002015906	545	70	545	70	2235	0	300
Heatup/Cooldown Davis-Besse ^(1,2)	561	70	557	70	2235	0	128
Plant Loading / Unloading EPRI Report 3002015906	610	550	550	545	2300	2300	5000
Plant Loading / Unloading Davis- Besse ^(1,3)	608	561	578	556	2235	2135	1800
Reactor Trip EPRI Report 3002015906	615	530	565	530	2435	1700	360
Reactor Trip Davis-Besse ^(1,4)	644	547	590	547	2615	1695	187

Notes:

1. Davis-Besse's Replacement Once Through Steam Generators (ROTSGs) were replaced in 2014 (1st period of the 4th ISI Interval). Since the ROTSGs were replaced late in original 40-year licensed life, the Certified Design Specification only went out to 40-years for the transients discussed above (per Table A-1, item number 38 of TS-3985, Certified Design Specification). The 60-year projected cycles were determined as part of license renewal and are identified in EN-DP-00355, Determination of Allowable Operating Transient Cycles.
2. Heatup/Cooldown = Transients #1A and #1B of Reference [1-6]. Max T_{Hot}, Max T_{Cold}, Max Pressure, Min T_{hot} and Min T_{cold} taken from Document 18-1149327-005, Functional Specification for Reactor Coolant System for Davis-Besse. The Functional Specification is the basis for Figure A1 of Calc. No. 205S-B6, "Davis-Besse ROTSG Transient Load Summary" used to evaluate the ROTSGs. EPRI report assumed a bounding ramp rate of 200°F/hour. Davis-Besse heatup is limited to 50°F in any one hour period, and a maximum cooldown of 100°F in any one-hour period with Cold Leg temperature greater than or equal to 270°F and a maximum cooldown of 50°F in any one hour period with Cold Leg temperature less than 270°F, in accordance with the Pressure and Temperature Limits Report (PTLR).
3. Plant Loading/Unloading = Transients #3 and 4 of Reference [1-6].
4. Reactor Trip = Transients #8A, 8B, and 8C of Reference [1-6]. Transient 8A is a Reactor Trip with Loss of Flow and maximum temperature occurs at the reactor vessel. Figure B2 of Calc. No. 205S-B6 is for Reactor Trip Type A corresponding to loss of RC flow (Transient #8A), Figure B3 is for Reactor Trip Type B corresponding to a control system malfunction (Transient #8B), and Figure B4 is for Reactor Trip Type C corresponding to a loss of MFW flow (Transient #8C). The values for Max T_{Hot}, Min T_{hot}, Max T_{Cold}, and Min T_{cold}, Max Pressure and Min Pressure were obtained from each figure and the bounding value was selected. Transient #8D (Other trips) is bounded by the other Reactor Trips (Transients #8A, 8B, and 8C). Transient #8E applies to the reactor vessel head vent line only.

Table 1-3

Davis-Besse Data for Thermal Transients for Stress Analysis of the PWR SG Secondary-Side Vessel Welds (Comparison to Table 5-9 of Reference [1-1])

Transient	Max Tss °F	Min Tss °F	Max Press PSIG	Min Press PSIG	60-Year Projected Cycles
Heatup/Cooldown EPRI Report 3002015906	545	70	1000	0	300
Heatup/Cooldown Davis-Besse ^(1,2)	561	70	1035	0	128
Plant Loading / Unloading EPRI Report 3002015906	545	540	1000	1000	5000
Plant Loading / Unloading Davis- Besse ^(1,3)	591	532	941	885	1800
Reactor Trip EPRI Report 3002015906	555	530	1130	1000	360
Reactor Trip Davis-Besse ^(1,4)	613	538	1135	810	187

Notes:

1. Davis-Besse's Replacement Once Through Steam Generators (ROTSGs) were replaced in 2014 (1st period of the 4th ISI Interval). Since the ROTSGs were replaced late in original 40-year licensed life, the Certified Design Specification only went out to 40-years for the transients discussed above (per Table A-1, item number 38 of TS-3985, Certified Design Specification). The 60-year projected cycles were determined as part of license renewal and are identified in EN-DP-00355, Determination of Allowable Operating Transient Cycles.
2. Heatup/Cooldown = Transients #1A and #1B of Reference [1-6].
3. Plant Loading/Unloading = Transients #3 and 4 of Reference [1-6].
4. Reactor Trip = Transients #8A, 8B, and 8C of Reference [1-6]. Transient #8D (Other trips) is bounded by the other Reactor Trips (Transients #8A, 8B, and 8C). Transient #8E applies to the reactor vessel head vent line only.

Table 1-4

Davis-Besse Data for Thermal Transients Applicable to PWR SG Feedwater and Main Steam Nozzles (Comparison to Table 5-5 of Reference [1-2])

Transient	60-Year Allowable Cycles from Table 5-5 of EPRI Report 3002014590 [1-2]	60-Year Projected Cycles Davis-Besse
Heatup/Cooldown ^(1,2)	300	128
Plant Loading ^(1,3)	5000	1800
Plant Unloading ^(1,3)	5000	1800
Loss of Load ^(1,4)	360	187
Loss of Power ^(1,5)	60	6

Notes:

1. Davis-Besse's Replacement Once Through Steam Generators (ROTSGs) were replaced in 2014 (1st period of the 4th ISI Interval). Since the ROTSGs were replaced late in original 40-year licensed life, the Certified Design Specification only went out to 40-years for the transients discussed above (per Table A-1, item number 38 of TS-3985, Certified Design Specification). The 60-year projected cycles were determined as part of license renewal and are identified in EN-DP-00355, Determination of Allowable Operating Transient Cycles.
2. Heatup/Cooldown = Transients #1A and #1B of Reference [1-6].
3. Plant Loading/Unloading = Transients #3 and 4 of Reference [1-6].
4. Loss of Load = Reactor Trip = Transients #8A, 8B, and 8C of Reference [1-6]. Transient #8D (Other trips) is bounded by the other Reactor Trips (Transients #8A, 8B, and 8C). Transient #8E applies to the reactor vessel head vent line only.
5. Loss of Power = Transient #15 of Reference [1-6]. Projected cycles also obtained from [1-6].

Attachment 1 References

- 1-1. *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).
- 1-2. *Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Inside Radius Sections*. EPRI, Palo Alto, CA: 2019. 3002014590 (ADAMS Accession No. ML19347B107).
- 1-3. Drawing M-506-00190, "Section 'XI' Pre-Service NDE Examination," Revision 01 (Vendor Drawing No. 205SE007, Revision 02).
- 1-4. Technical Specification(s) Document No. TS-3985, "Certified Design Specification," Revision 03.
- 1-5. Drawing M-506-00188, "Davis Besse ROTSG General Arrangement," Revision 02 (Vendor Drawing No. 205SE001, Revision 05).
- 1-6. Procedure No. EN-DP-00355, "Determination of Allowable Operating Transient Cycles," Revision 10.

ATTACHMENT 2

INSPECTION HISTORY

DAVIS-BESSE INSPECTION HISTORY

SG Primary Side Welds

Item No.	Examination Date	Interval/Period (Outage)	Component ID	Examination Results	Coverage	Relief Request	Original (O) or Replacement (R) Generator
B2.40	8/21/2013	4th Interval / 1st Period (PSI for 18R)	RC-SG-1-1-W23	Acceptable	97.0%	N/A	R
B2.40	8/21/2013	4th Interval / 1st Period (PSI for 18R)	RC-SG-1-2-W23	Acceptable	97.0%	N/A	R
B2.40	8/21/2013	4th Interval / 1st Period (PSI for 18R)	RC-SG-1-1-W22	Acceptable	99.0%	N/A	R
B2.40	8/21/2013	4th Interval / 1st Period (PSI for 18R)	RC-SG-1-2-W22	Acceptable	99.0%	N/A	R
B2.40	3/19/2018	4th Interval / 2nd Period (20R)	RC-SG-1-2-W22	Acceptable	96.6%	N/A	R
B2.40	4/10/2006	3rd Interval / 1st Period (14R)	RC-SG-1-1-WG-58-1	Acceptable	91.5%	N/A	O
B2.40	5/18/2010	3rd Interval / 3rd Period (16R)	RC-SG-1-1-WG-58-2	Acceptable	> 90%	N/A	O

SG Secondary Side Shell Welds

Item No.	Examination Date	Interval/Period (Outage)	Component ID	Examination Results	Coverage	Relief Request	Original (O) or Replacement (R) Generator
C1.30	8/22/2013	4th Interval / 1st Period (PSI for 18R)	SP-SG-1-1-W65	Acceptable	100.0%	N/A	R
C1.30	8/21/2013	4th Interval / 1st Period (PSI for 18R)	SP-SG-1-1-W69	Acceptable	99.8%	N/A	R
C1.30	8/7/2013	4th Interval / 1st Period (PSI for 18R)	SP-SG-1-2-W65	Acceptable	99.0%	N/A	R
C1.30	3/19/2018	4th Interval / 2nd Period (20R)	SP-SG-1-2-W65	Acceptable	100.0%	N/A	R
C1.30	8/7/2013	4th Interval / 1st Period (PSI for 18R)	SP-SG-1-2-W69	Acceptable	96.0%	N/A	R
C1.30	5/17/2010	3rd Interval / 3rd Period (16R)	SP-SG-1-1-WG-60	Acceptable	94.9%	N/A	O
C1.30	3/20/2002	3rd Interval / 1st Period (13R)	SP-SG-1-2-WG-59	Acceptable	99.7%	N/A	O

SG Secondary Side Nozzle Welds

Item No.	Examination Date	Interval/Period (Outage)	Component ID	Examination Results	Coverage	Relief Request	Original (O) or Replacement (R) Generator
C2.21	8/21/2013	4th Interval / 1st Period (PSI for 18R)	SP-SG-1-1-W127-X/Y	Acceptable	100.0%	N/A	R
C2.21	8/21/2013	4th Interval / 1st Period (PSI for 18R)	SP-SG-1-1-W128-W/X	Acceptable	100.0%	N/A	R
C2.21	3/13/2020	4th Interval / 3rd Period (21R)	SP-SG-1-1-W128-W/X	Acceptable	100.0%	N/A	R
C2.21	8/21/2013	4th Interval / 1st Period (PSI for 18R)	SP-SG-1-2-W127-X/Y	Acceptable	100.0%	N/A	R
C2.21	3/17/2018	4th Interval / 2nd Period (20R)	SP-SG-1-2-W127-X/Y	Acceptable	100.0%	N/A	R
C2.21	8/21/2013	4th Interval / 1st Period (PSI for 18R)	SP-SG-1-2-W128-W/X	Acceptable	100.0%	N/A	R
C2.21	1/21/2008	3rd Interval / 2nd Period (15R)	SP-SG-1-2-WG-23-X/Y	Acceptable	99.9%	N/A	O
C2.21	10/20/2011	3rd Interval / 3rd Period (17R)	SP-SG-1-2-WG-23-W/X	Acceptable	100.0%	N/A	O
C2.22	8/20/2013	4th Interval / 1st Period (PSI for 18R)	SP-SG-1-1-W127-X/Y-IR	Acceptable	100.0%*	N/A	R
C2.22	8/20/2013	4th Interval / 1st Period (PSI for 18R)	SP-SG-1-1-W128-W/X-IR	Acceptable	100.0%*	N/A	R
C2.22	3/7/2020	4th Interval / 3rd Period (21R)	SP-SG-1-1-W128-W/X-IR	Acceptable	100.0%*	N/A	R
C2.22	7/31/2013	4th Interval / 1st Period (PSI for 18R)	SP-SG-1-2-W127-X/Y-IR	Acceptable	100.0%*	N/A	R
C2.22	3/19/2018	4th Interval / 2nd Period (20R)	SP-SG-1-2-W127-X/Y-IR	Acceptable	100.0%*	N/A	R
C2.22	7/31/2013	4th Interval / 1st Period (PSI for 18R)	SP-SG-1-2-W128-W/X-IR	Acceptable	100.0%*	N/A	R
C2.22	1/23/2008	3rd Interval / 2nd Period (15R)	SP-SG-1-2-WG-23-X/Y-IR	Acceptable	100%	N/A	O
C2.22	10/20/2011	3rd Interval / 3rd Period (17R)	SP-SG-1-2-WG-23-W/X-IR	Acceptable	> 90%	N/A	O

*100% of area defined in EPRI Report IR-2011-426 "Davis-Besse ROTSG Nozzle Examination"

ATTACHMENT 3

RESULTS OF INDUSTRY SURVEY

Overall Industry Inspection Summary for Code Items 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 [3-1]. Table 3-1 provides a summary of the combined survey results for Item Nos. B2.31, B2.32 (see Table 3-1, Note 3), B2.40, B3.130, C1.10, C1.20, and C1.30. The results of the industry survey identified numerous steam generator (SG) examinations are being performed with no service-induced flaws being detected. Performing these examinations adversely impacts 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 (that is, Babcock and Wilcox (B&W), Combustion Engineering (CE), and Westinghouse). A total of 1324 examinations for the components of the affected Item Nos. were conducted, with 1098 of these specifically for PWR components. The majority of 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 No. 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 non-service-induced (see Table 3-1, Note 1). For Item No. 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 [3-3] 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 Table 3-1, Note 2). As discussed in References [3-4] and [3-5], these flaws were most likely subsurface weld defects typical of thick vessel welds and not service-induced. A flaw evaluation for IWC-3600 was performed for these flaws, and they were found to be acceptable for continued operation.

Table 3-1

**Summary of Survey Results for SG Nozzle-to-Shell, Inside Radius Section,
and Shell Weld Components**

Item No.	No. of Examinations			No. of Reportable Indications		
	BWR	PWR	Total	BWR	PWR	Total
B2.31	0	30	30	0	0	0
B2.32 (Note 3)	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	1098	1324	0	Notes 1 and 2	Notes 1 and 2

Notes:

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, 3-5].
3. Item No. B2.32 was evaluated in the Reference [3-1] technical basis and included in the industry survey but is not contained in the scope of this alternative request.

Overall Industry Inspection Summary for Code Items C2.21, C2.22, and C2.32

The results of an industry survey of past inspections of SG main steam (MS) and feedwater (FW) nozzles are summarized in Reference [3-2]. Table 3-2 provides a summary of the combined survey results for Item Nos. C2.21, C2.22, and C2.32 (see Table 3-2, Note 1). The results identify that SG MS and FW Nozzle-to-Shell Welds and Nozzle Inside Radius Section examinations are 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 BWR and 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 NSSS vendors (that is, B&W, CE, and Westinghouse). A total of 727 examinations for Item Nos. C2.21, C2.22, and C2.32 (see Table 3-2, Note 1) components were conducted, with 563 of these specifically for PWR components. The majority of the PWR examinations were performed on SG MS and FW nozzles. Only one PWR examination identified two (2) flaws that exceeded ASME Code, Section XI acceptance criteria. The flaws were linear indications of 0.3" and 0.5" in length and were detected in a MS nozzle-to-shell weld using magnetic particle examination techniques. The indications were dispositioned by light grinding (ADAMS Accession No. ML13217A093).

Table 3-2

Summary of Survey Results for SG Main Steam and Feedwater Nozzle Components

Plant Type	Number of Units	Number of Examinations	Number of Reportable Indications
BWR	27	164	0
PWR	47	563	2
Totals	74	727 (Note 1)	2

Notes:

1. Item No. C2.32 was evaluated in the Reference [3-2] technical basis and included in the industry survey but is not contained in the scope of this alternative request.

Attachment 3 References

- 3-1. *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).
- 3-2. *Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Inside Radius Sections*. EPRI, Palo Alto, CA: 2019. 3002014590 (ADAMS Accession No. ML19347B107).
- 3-3. Letter from F. A. Kearney (Exelon Generation) to 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 No. ML13217A093).
- 3-4. Letter from J. P. Sorensen (Nuclear Management Company, LLC) to 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," dated May 29, 2001, Docket Nos. 50-282 and 50-306 (ADAMS Accession No. ML011550346).
- 3-5. Letter from J. M. Solymossy (Nuclear Management Company, LLC) to 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)," dated April 4, 2003, Docket Nos. 50-282 and 50-306 (ADAMS Accession No. ML031040553).