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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

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NRA/ENC:	R0
Docket No.:	50-395
License No.:	NPF-12

DOMINION ENERGY SOUTH CAROLINA, INC. (DESC)
VIRGIL C. SUMMER NUCLEAR STATION UNIT 1
ALTERNATIVE REQUEST TO DEFER ASME CODE SECTION XI INSERVICE INSPECTION EXAMINATIONS FOR PRESSURIZER AND STEAM GENERATOR PRESSURE-RETAINING WELDS AND FULL PENETRATION WELDED NOZZLES

In accordance with 10 CFR 50.55a, "Codes and Standards," paragraph (z)(1), Dominion Energy South Carolina (DESC) requests Nuclear Regulatory Commission (NRC) approval of a proposed alternative to the inservice inspection (ISI) requirements for American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI, Table IWB-2500-1, Examination Category B-B and B-D and Table IWC-2500-1, Examination Category C-A and C-B, component examinations for Virgil C. Summer Nuclear Station Unit 1 (VCSNS). Specifically, DESC requests to defer examinations for the remainder of the current fourth 10-year ISI interval through the fifth 10-year ISI interval ending on December 31, 2033. The proposed alternative is requested on the basis that it provides an acceptable level of quality and safety in lieu of the current ASME Code, Section XI 10-year inspection frequency requirement.

The proposed alternative, which includes a summary of the technical basis for the request, is provided in Attachment 1. The plant-specific applicability of the technical basis to VCSNS is provided in Attachments 2 and 3. The VCSNS pressurizer and steam generators inservice inspection history and the inspection history for the applicable components, as obtained from an industry survey, are presented in Attachments 4 and 5, respectively.

Pursuant to 10 CFR 50.55a(z), the proposed alternative requires NRC review and approval before implementation. DESC requests NRC approval of this request by April 1, 2023, to support the VCSNS Spring 2023 refueling outage.

If you have any questions or require additional information, please contact Ms. Erica N. Combs at (804)-273-3386.

Sincerely,

Douglas C. Lawrence
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Dominion Energy South Carolina

Attachments:

1. Proposed Alternative to ASME Code Section XI Requirements for Inservice Inspection of Pressurizer and Steam Generator Pressure-Retaining Welds and Full Penetration Welded Nozzles
2. Plant-Specific Applicability
3. Comparison of Insurge/Outsurge Transients
4. Inspection History
5. Results of Industry Survey

Commitments made in this letter: None

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ATTACHMENT 1

Proposed Alternative to ASME Code Section XI Requirements for Inservice Inspection of Pressurizer and Steam Generator Pressure–Retaining Welds and Full Penetration Welded Nozzles

VIRGIL C. SUMMER NUCLEAR STATION UNIT 1 (VCSNS)
DOMINION ENERGY SOUTH CAROLINA, INC. (DESC)

Proposed Alternative to ASME Code Section XI Requirements for Inservice Inspection of Pressurizer and Steam Generator Pressure–Retaining Welds and Full Penetration Welded Nozzles

In accordance with 10 CFR 50.55a(z)(1), the proposed alternative provides an acceptable level of quality and safety.

1.0 <u>American Society of Mechanical Engineers (ASME) Code Components Affected</u>

The ASME Code components affected are Class 1 and 2 pressurizer (PZR) vessel shell-to-head welds and full penetration welded nozzles, and steam generator (SG) pressure-retaining welds and full penetration welded nozzles listed in Table 1. The affected components are identified in Table 2.

Table 1. ASME Code components affected

<table>
<tr><td>Code Class</td><td colspan="2">Class 1 and Class 2</td></tr>
<tr><td rowspan="2">Description</td><td colspan="2">PZR vessel shell-to-head welds and full penetration welded nozzles</td></tr>
<tr><td colspan="2">SG pressure-retaining vessel welds and full penetration welded nozzles</td></tr>
<tr><td rowspan="4">Examination Categories</td><td colspan="2">Class 1, Category B-B, pressure-retaining welds in vessels other than reactor vessels</td></tr>
<tr><td colspan="2">Class 1, Category B-D, full penetration welded nozzles in vessels</td></tr>
<tr><td colspan="2">Class 2, Category C-A, pressure-retaining welds in pressure vessels</td></tr>
<tr><td colspan="2">Class 2, Category C-B, pressure-retaining nozzle welds in vessels</td></tr>
<tr><td rowspan="9">Item Numbers</td><td>B2.11</td><td>PZR, shell-to-head welds, circumferential</td></tr>
<tr><td>B2.12</td><td>PZR, shell-to-head welds, longitudinal</td></tr>
<tr><td>B2.40</td><td>SG (primary side), tubesheet-to-head weld</td></tr>
<tr><td>B3.110</td><td>PZR, nozzle-to-vessel welds</td></tr>
<tr><td>C1.10</td><td>Shell circumferential welds</td></tr>
<tr><td>C1.20</td><td>Head circumferential welds</td></tr>
<tr><td>C1.30</td><td>Tubesheet-to-shell weld</td></tr>
<tr><td>C2.21</td><td>Nozzle-to-shell (nozzle-to-head/nozzle-to-nozzle) welds</td></tr>
<tr><td>C2.22</td><td>Nozzle inside radius sections</td></tr>
</table>

Table 2. Affected component IDs

ASME Category	ASME Item No.	Component ID	Component Description
Pressurizer			
B-B	B2.11	1-2100A-1	PZR Shell to Lower Head
B-B	B2.11	1-2100A-4	PZR Shell to Upper Head
B-B	B2.12	1-2100A-5	PZR Shell Longitudinal Weld - Lower
B-B	B2.12	1-2100A-7	PZR Shell Longitudinal Weld - Upper
B-D	B3.110	1-2100A-8	PZR Surge Line Nozzle to Vessel Weld
B-D	B3.110	1-2100B-9	PZR Spray Line Nozzle to Vessel Weld
B-D	B3.110	1-2100B-10	PZR 'A' Safety Line Nozzle to Vessel Weld
B-D	B3.110	1-2100B-11	PZR 'B' Safety Line Nozzle to Vessel Weld
B-D	B3.110	1-2100B-12	PZR 'C' Safety Line Nozzle to Vessel Weld
B-D	B3.110	1-2100B-13	PZR Relief Line Nozzle to Vessel Weld
Steam Generator 'A'			
B-B	B2.40	1-3100-14A	SG Primary Head to Tubesheet
C-A	C1.10	2-1100-17A	SG Shell to Lower Transition Cone
C-A	C1.10	2-1100-18A	SG Shell to Upper Transition Cone
C-A	C1.20	2-1100-20A	SG Shell to Upper Head
C-A	C1.30	2-1100-15A	SG Shell to Tubesheet
C-B	C2.21	2-1100-23A	SG Shell to Feedwater Nozzle
C-B	C2.22	2-1100-23IR-A	SG Shell to Feedwater Nozzle Inner Radius
Steam Generator 'B'			
B-B	B2.40	1-3100-14B	SG Primary Head to Tubesheet
C-A	C1.10	2-1100-17B	SG Shell to Lower Transition Cone
C-A	C1.10	2-1100-18B	SG Shell to Upper Transition Cone
C-A	C1.20	2-1100-20B	SG Shell to Upper Head
C-A	C1.30	2-1100-15B	SG Shell to Tubesheet
C-B	C2.21	2-1100-23B	SG Shell to Feedwater Nozzle
C-B	C2.22	2-1100-23IR-B	SG Shell to Feedwater Nozzle Inner Radius
Steam Generator 'C'			
B-B	B2.40	1-3100-14C	SG Primary Head to Tubesheet
C-A	C1.10	2-1100-17C	SG Shell to Lower Transition Cone
C-A	C1.10	2-1100-18C	SG Shell to Upper Transition Cone
C-A	C1.20	2-1100-20C	SG Shell to Upper Head
C-A	C1.30	2-1100-15C	SG Shell to Tubesheet
C-B	C2.21	2-1100-23C	SG Shell to Feedwater Nozzle
C-B	C2.22	2-1100-23IR-C	SG Shell to Feedwater Nozzle Inner Radius

2.0 Applicable Code Edition and Addenda

The Code of record for the Virgil C. Summer Nuclear Station Unit 1 (VCSNS) fourth 10-year inservice inspection (ISI) interval is the ASME Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2007 Edition with 2008 Addenda [1-1]. The VCSNS fourth 10-year ISI interval started on January 1, 2014, and ends on December 31, 2023.

3.0 Applicable Code Requirement

The ASME Code, 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 Categories C-A and C-B require examination of the following Item Nos.:

B2.11 — Volumetric examination of essentially 100% of the weld length for both circumferential shell-to-head welds during each inspection interval. The examination volume is shown in Figure IWB-2500-1.

B2.12 — Volumetric examination of one (1) foot of all longitudinal shell-to-head welds during the first inspection interval and one (1) foot of one (1) weld per head during successive intervals. The examination volume is shown in Figure IWB-2500-2.

B2.40 — Volumetric examination of essentially 100% 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 (1) vessel among the group of vessels performing a similar function. The examination volume is shown in Figure IWB-2500-6.

B3.110 — Volumetric examination of all full penetration nozzle-to-vessel welds during each inspection interval. The examination volume is shown in Figures IWB-2500-7(a), (b), and (c).

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 Section XI inspection interval. In the case of multiple vessels of similar design, size, and service (such as SGs, heat exchangers), the required examinations may be limited to one (1) vessel or distributed among the vessels. The examination volume is shown in Figure IWC-2500-1.

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

C1.30 Volumetric examination of essentially 100% 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 SGs, heat exchangers), the required examinations may be limited to one (1) vessel or distributed among the vessels. The examination volume is shown in Figure IWC-2500-2.

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 SGs, heat exchangers), the required examinations may be limited to one (1) vessel or distributed among the vessels. The examination area and volume are shown in Figures IWC-2500-4(a), (b), or (d).

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 SGs, heat exchangers), the required examinations may be limited to one (1) vessel or distributed among the vessels. The examination volume is shown in Figures IWC-2500-4(a), (b), or (d).

4.0 Reason for Request

Electric Power Research Institute (EPRI) performed assessments in References [1-2], [1-3], and [1-4] of the basis for the ASME Code, Section XI examination requirements specified for the above listed ASME Code, Section XI, Division 1 examination categories for PZR and 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 reports in References [1-2], [1-3], and [1-4], developed consistent with the recommendations provided in EPRI's White Paper on PFM [1-5], concluded that the current ASME Code, Section XI inspection interval of ten (10) years can be increased significantly with no impact to plant safety. Based on the conclusions of the three EPRI reports, Dominion Energy South Carolina, Inc. (DESC) is requesting an alternative to the 10-year inspection interval for the subject welds.

5.0 Proposed Alternative and Basis for Use

DESC is requesting an alternative to the ASME Code, Section XI Examination Requirements in Tables IWB-2500-1 and IWC-2500-1 for the following Examination Categories and Item Nos.:

ASME Category	Item No.	Description
B-B	B2.11	PZR, shell-to-head welds, circumferential
B-B	B2.12	PZR, shell-to-head welds, longitudinal
B-B	B2.40	SG (primary side), tubesheet-to-head weld
B-D	B3.110	PZR, nozzle-to-vessel welds
C-A	C1.10	Shell circumferential welds
C-A	C1.20	Head circumferential welds
C-A	C1.30	Tubesheet-to-shell weld
C-B	C2.21	Nozzle-to-shell (nozzle-to-head or nozzle-to-nozzle) welds
C-B	C2.22	Nozzle inside radius sections

VCSNS is currently in the third period of the fourth 10-year ISI interval. Some welds have not yet received an ISI examination for the fourth interval. The proposed alternative increases the inspection interval for these examination items from the current ASME Code, Section XI 10-year requirement thereby deferring examinations for the remainder of the current fourth 10-year ISI interval through the fifth 10-year ISI interval ending on December 31, 2033. During the sixth 10-year ISI interval starting on January 1, 2034, examinations will be done in accordance with ASME Code, Section XI ISI 10-year requirements. The sixth 10-year ISI interval ends on December 31, 2043.

A summary of the technical basis for this request is provided below. The applicability of the technical basis to VCSNS is demonstrated in Attachments 2, 3, and 4.

A. Degradation Mechanism Evaluation

An evaluation of degradation mechanisms that could potentially impact the reliability of the PZR and SG welds and components was performed in References [1-2], [1-3], and [1-4]. 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 PZR and SG welds and components covered in this request. Therefore, only those fatigue-related mechanisms considered in the PFM and DFM evaluations in References [1-2], [1-3], and [1-4] are applicable to the components in this request.

B. Stress Analysis

Finite element analyses (FEA) were performed in References [1-2], [1-3], and [1-4] to determine the stresses in the PZR and SG welds and components covered in this request. The finite element models used in References [1-2], [1-3], and [1-4] are consistent with the configurations for VCSNS, therefore no new FEA model is required for the stress analysis of VCSNS. 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 VCSNS is demonstrated in Attachments 2 and 3 and confirms that all plant-specific requirements are met. Therefore, the evaluation results and conclusions contained in References [1-2], [1-3], and [1-4] are applicable to VCSNS. In particular, the key geometric parameters used in the stress analyses in References [1-2], [1-3], and [1-4] are compared to those of VCSNS, in Tables 3 and 4 for the PZR and Tables 5 and 6 for the SGs.

Table 3. PZR shell dimensions

	Shell Inside Diameter (ID) (inches)	**Shell / Clad Thickness** (inches)	**Shell R_o/t**	**Shell R_i/t**
EPRI Report (Table 4-4 of [1-2])	84 (1)	3.75 / 0.063 (1)	12.2 (1)	11.2
VCSNS	84	3.75 / 0.19	12.2	11.2

(1) Westinghouse PZR dimensions associated with model for lower head.

Table 4. PZR nozzle dimensions

	Surge Nozzle			Safety / Relief Nozzle		
	ID (inches)	Thickness (inches)	R_i/t	ID (inches)	Thickness (inches)	R_i/t
EPRI Report (Table 4-5 of [1-2])	12.44 (1)	3.27 (1)	1.9 (1)	5.625 (2)	1.19 (2)	2.363 (2)
VCSNS	12.44	3.28	1.9	5.89	2.57	1.15

(1) Westinghouse PZR nozzle dimensions associated with model for lower head.
(2) Combustion Engineering (CE) PZR nozzle dimensions associated with model for upper head.

As noted by the Nuclear Regulatory Commission (NRC) in the Safety Evaluation (SE)[1] [1-6] for Salem Generating Station (Salem), Units 1 and 2, the dominant stress is pressure stress. Therefore, the variation in the R_i/t ratio determined in Tables 3 and 4 can be used to scale up the stresses in Reference [1-2] to obtain the plant-specific stresses for each unit and component.

In the selection of the transients in Section 5 of Reference [1-2], test conditions beyond a system leakage test were not considered since pressure tests for VCSNS are performed at normal operating conditions. No hydrostatic testing of the VCSNS PZR or SGs has been performed since the plant began operation.

Table 5. SG vessel dimensions

	Primary Lower Head ID (inches)	**Primary Lower Head Thickness / Clad** (inches)	**Primary Lower Head R_i/t**	**Secondary Lower Shell ID** (inches)	**Secondary Lower Shell Thickness** (inches)	**Secondary Lower Shell R_i/t**
EPRI Report (Table 4-2 of [1-3])	155.87	6.94 / 0.27	11.2	162.45	3.65	22.3
VCSNS	125.57	5.26 / 0.22	11.9	128.80	3.35	19.22

1 Section 5.1, page 7, fourth paragraph

Table 6. SG nozzle dimensions

	FW Nozzle ID (inches)	FW Nozzle Thickness (inches)	FW Nozzle R_i/t
EPRI Report (Figure 4-10 of [1-4])	16.5	6	1.38
VCSNS	16.5	4.75	1.74

As discussed in Sections 4.3.3 and 4.6 of Reference [1-4] and noted by the NRC in the SE[2] [1-7] for Vogtle Electric Generating Plant (Vogtle), Units 1 and 2, the dominant stress is pressure stress. Therefore, the variation in the R_i/t ratio determined in Tables 5 and 6 can be used to scale up the stresses in References [1-3] and [1-4] to obtain the plant-specific stresses for each unit and component.

In the selection of the transients in Section 5 of References [1-3] and [1-4] and the subsequent stress analyses in Section 7, test conditions beyond a system leakage test were not considered since pressure tests for VCSNS are performed at normal operating conditions. No hydrostatic testing of the VCSNS PZR or SGs has been performed since the plant began operation.

In Reference [1-3], clad residual stress was not considered for the primary side welds. This was noted by the NRC in a RAI for Millstone Power Station, Unit 2 (MPS2). In response to the RAI [1-8], an evaluation was performed that showed the clad residual stress has no significant impact on the conclusions of Reference [1-3].[3] This was found acceptable by the NRC in Section 5.3 of the SE [1-9] for MPS2.

C. Flaw Tolerance Evaluation

Flaw tolerance evaluations were performed in References [1-2], [1-3], and [1-4] consisting of PFM evaluations and confirmatory DFM evaluations. The results of the PFM analyses indicate that, after a preservice inspection (PSI) followed by subsequent ISI, the NRC's safety goal of 1.0×10^{-6} failures per year is met. The PFM analysis in Reference [1-4] was performed using the PRobabilistic OptiMization of InSpEction (PROMISE), Version 1.0 software developed by Structural Integrity Associates. As part of the NRC's review of Southern Nuclear Operating Company, Inc.'s Alternative Request [1-10] for Vogtle, the NRC performed an audit [1-22] of the PROMISE, Version 1.0 software. The PFM

[2] Section 3.8.3.1, page 9, third paragraph
[3] RAI Response No. 3c

analysis in References [1-2] and [1-3] was performed using the PROMISE, Version 2.0 software, which has not been audited by the NRC. The only technical difference between Version 1.0 and Version 2.0 of the PROMISE software is that the user-specified examination coverage is applied to all inspections in Version 1.0, whereas the examination coverage can be specified by the user uniquely for each inspection in Version 2.0. In both versions of the software, 100% coverage for the PSI examination is assumed. The NRC staff found the use of the PROMISE, Version 2.0 acceptable as approved in the Salem SE[4] [1-6].

A comparison of the PSI/ISI scenarios used in the sensitivity studies performed in Reference [1-2] to those for the VCSNS PZR is provided below.

For the VCSNS PZR, PSI examinations have been performed followed by ISI examinations over three (3) complete 10-year intervals. The inspection schedule scenario for these welds is PSI plus three (3) 10-year ISI examinations (PSI + 10 + 20 + 30 Inspection Scenario). Most of the required fourth interval examinations have been completed at the time of this request. The analyses involve conservative assumptions with regards to the PSI/ISI scenarios. Furthermore, the evaluation was performed for 80 years, which is longer than the alternative proposed by DESC for VCSNS in this request.

In the PFM evaluations in Reference [1-2], the Pressure Vessel Research Facility User's Facility (PVRUF) initial flaw size distribution was used. This distribution is applicable to thick vessels and not to relatively thin vessels like PZRs. In a RAI [1-11], the NRC staff asked PSEG Nuclear, LLC., (PSEG) to justify its application of this distribution to the Salem PZR vessel lower head shell welds. In response to the RAI [1-12], PSEG used various initial flaw size distributions in a sensitivity study which showed that regardless of which distribution was used, the conclusions of Reference [1-2] remain the same.[5] The NRC determined this conclusion was acceptable in its SE [1-6] for Salem, dated April 12, 2021.

The results of the PFM analyses indicate that, after a PSI, no other inspections are required for up to 80 years of plant operation to meet the NRC's safety goal of 1.0×10^{-6} failures per year. For the specific case of VCSNS, where PSI followed by at least two (2) 10-year interval inspections have been performed, Table 8-10 of References [1-3] and [1-4] and Table 8-12 of Reference [1-2] indicates that if the inspection interval is increased to 30 years after these previous inspections, the NRC safety goal is met (with considerable margin) for up to 80 years of plant operation. The DFM evaluations provide verification of the PFM results by demonstrating that it takes approximately 80 years for a postulated flaw with an

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[4] Section 3.1, page 5, fourth paragraph
[5] Section 9.1, page 15, last paragraph

initial depth equal to the 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.

In Section 8.2.2.2 of Reference [1-4] and Section 8.3.2.2 of Reference [1-3], a nozzle flaw density of 0.001 flaws per nozzle was assumed for the nozzle inside radius sections. In Section 3.8.5 of the SE [1-7] for Vogtle, the NRC indicated that a nozzle flaw density of 0.1 flaws per nozzle should have been used. Sensitivity studies performed in Section 8.2.4.3.4 of Reference [1-3] 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 (2) orders of magnitude but were still significantly below the acceptance criterion of $1.0\text{x}10^{-6}$ failures per year. A comparison of the PSI/ISI scenarios used in the sensitivity studies performed in References [1-3] and [1-4] to those for the VCSNS SGs is provided below.

In 1994, all three (3) VCSNS SGs were replaced. For the replaced SGs, PSI examinations have been performed in the first period of the second 10-year interval followed by ISI examinations for the second and third 10-year ISI intervals. The inspection scenario for these welds and components is PSI plus two (2) complete 10-year ISI examinations for the replaced SGs (PSI + 10 + 20 Inspection Scenario). It should be noted that most of the required fourth interval examinations have been completed at the time of this request. The analyses involve conservative assumptions with regards to the PSI/ISI scenarios. Furthermore, the evaluation was performed for 80 years, which is longer than the alternative proposed by DESC for VCSNS in this request.

The PFM evaluations documented in References [1-2], [1-3], and [1-4] used an ASME Code, Section XI, Appendix VIII-based probability of detection (POD) curve in the PFM evaluation because most ISI examinations of major Class 1 and Class 2 plant components are performed using Appendix VIII procedures. However, for some PZR components, the use of Appendix VIII procedures is plant specific. Many plants adopt and use their Appendix VIII procedures for major Class 1 components (such as PZRs) for consistency across all their examinations. In the case of VCSNS, the Section V procedures are used for the PZR and SG ultrasonic examinations. As stated in the NRC SEs for Salem[6] [1-6] and Vogtle[7] [1-7], the use of the ASME Code, Section XI, Appendix VIII-based POD curve for inspections based on Section V procedures would have minimal impact on the PFM results

[6] Section 9.2, page 15
[7] Section 3.8.8.2, page 21

since the POD curve is not one of the parameters that significantly affects the PFM results.

The DFM evaluations in Table 8-4 of Reference [1-2], Table 8-3 of Reference [1-3], and Table 8-31 of Reference [1-4] provide verification of the above PFM results for VCSNS by demonstrating that it takes approximately 80 years for a postulated flaw with an initial depth equal to the 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.

D. Inspection History

As described in Section 8.3.4.1 of Reference [1-2], Section 8.3.4.1 of Reference [1-3], and Section 8.2.4.1.1 of Reference [1-4], PSI refers to the collective examinations required by the ASME Code, Section III during fabrication and any Section XI examinations performed prior to service. The Section III fabrication examinations required for these components were robust and any Section XI PSI examinations further contributed to thorough initial examinations.

The inspection history for VCSNS (including examinations performed to-date, examination findings, examination coverage, and relief requests) is provided in Attachment 4.

As shown in the attachment, some of the welds/components have limited examination coverage, however all coverage is greater than 50%. This examination coverage was determined to be acceptable by the NRC in Section 10 of the SE [1-6] for Salem. Also, as shown in Attachment 4, no flaws that exceeded the ASME Code, Section XI acceptance standards were identified during any examinations.

E. Industry Survey

The inspection history for these components (as obtained from an industry survey) is presented in Attachment 5. The results of the survey indicate that these components are very flaw tolerant.

F. Conclusion

It is concluded that the PZR and SG pressure-retaining welds and full penetration welded nozzles are very flaw tolerant. PFM and DFM evaluations performed as part of the technical basis reports [1-2], [1-3], and [1-4] demonstrate that using conservative PSI/ISI inspection scenarios for all plants, the NRC safety goal of 1.0×10^{-6} failures per reactor year is met with considerable margins. Plant-specific applicability of the technical basis to VCSNS is demonstrated in Attachments 2 and 3. The requested inspection interval provides an acceptable level of quality and safety in lieu of the current ASME Code, Section XI 10-year inspection frequency.

Operating and examination history demonstrates that these components have performed with very high reliability, mainly due to their robust design. Attachment 4 shows the examination history for the PZR and SG welds examined during each of the 10-year inspection intervals to date. All three (3) SGs were replaced during the second ISI interval in 1994. The new welds in the SGs received the required fabrication acceptance and PSI examinations followed by the required scheduled ISI examinations.

In addition to the required fabrication and PSI examinations, ISI examinations have been performed during the first three (3) inspection intervals for the subject PZR and SG welds and components at VCSNS, as shown in Attachment 4. No flaws that exceeded the ASME Code, Section XI acceptance standards were identified during any examinations. It is important to note that all other inspection activities, including the system leakage test (Examination Categories B-P and C-H) will continue to be performed in accordance with the Section XI requirements providing further assurance of safety.

Finally, as discussed in Reference [1-13], for situations where no active degradation mechanism is present, it was concluded that subsequent ISI examinations do not provide additional value after PSI has been performed and the inspection volumes examined have been confirmed to be free of defects.

Therefore, DESC requests the NRC grant this proposed alternative for VCSNS in accordance with 10 CFR 50.55a(z)(1).

6.0 Duration of Proposed Alternative

Approval of the proposed alternative is requested by April 1, 2023, to support the VCSNS Spring 2023 refueling outage (RFO). Upon approval, the proposed alternative will cover the remainder of the current fourth 10-year ISI interval through the end of the fifth 10-year ISI interval ending on December 31, 2033.

7.0 Precedent

The following is a list of approved actions (including relief requests and topical reports) related to inspections of PZR and SG welds and components:

- Letter from J. G. Danna (NRC) to E. Carr (PSEG Nuclear, LLC), "Salem Generating Station Unit Nos. 1 and 2 – Authorization and Safety Evaluation for Alternative Request No. SC-I4R-200 (EPID L-2020-LLR-0103)," dated August 5, 2020. (ADAMS Accession No. ML21145A189)
- Letter from M. T. Markley (NRC) to C. A. Gayheart (Southern Nuclear Operating Company, Inc.), "Vogtle Electric Generating Plant, Units 1 and 2 – Relief Request for Proposed Inservice Inspection Alternative VEGPISI-ALT-04-04 to the Requirements of the ASME Code (EPID L-2020-LLR-0109)," dated January 11, 2021. (ADAMS Accession No. ML20352A155)
- Letter from J. G. Danna (NRC) to D. G. Stoddard (Dominion Energy Nuclear Connecticut, Inc.), "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)
- Letter from J. W. Clifford (NRC) to S. E. Scace (Northeast Nuclear Energy Company), "Safety Evaluation of the Relief Request Associated with the First and Second 10-Year Interval of the Inservice Inspection (ISI) Plan, Millstone Nuclear Power Station, Unit 3 (TAC No. MA 5446)," dated July 24, 2000. (ADAMS Accession No. ML003730922)
- Letter from R. L. Emch (NRC) to J. B. Beasley, Jr. (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 – Request for Relief 2R1-019, 2R1-020, 2R1-021, 2R1-022, 2R2-009, 2R2-010, 2R2-011 for the Second Ten-Year Interval Inservice Inspection Program Plan (TAC Nos. ME0609, ME0610, ME0611, ME0612, ME0613, ME0614 and ME0615)," dated January 7, 2010. (ADAMS Accession No. ML093561419)
- Letter from M. Khanna (NRC) to D. A. Heacock (Dominion Energy Nuclear Connecticut, Inc.), "Millstone Power Plant Unit No. 2 – Issuance of Relief Requests RR-89-69 Through RR-89-78 Regarding Third 10-Year Interval Inservice Inspection 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 & Electric Company), "Diablo Canyon Plant, Units 1 and 2 – Relief Request; NDE SG-MS-IR, Main Steam Nozzle Inner Radius Examination Impracticality, Third 10-Year Interval, American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Inservice Inspection Program (CAC Nos. MF6646 and MF6647)," dated December 8, 2015. (ADAMS Accession No. ML15337A021)

In addition, there are precedents related to similar topical reports that justify relief for Class 1 nozzles:

- Based on studies presented in Reference [1-14], the NRC approved extending PWR reactor vessel nozzle-to-shell welds from 10 to 20 years in Reference [1-15].
- Based on work performed in Boiling Water Reactor Vessel and Internals Program (BWRVIP)-108 [1-16] and BWRVIP-241 [1-17], the NRC approved the reduction of boiling water reactor (BWR) vessel feedwater nozzle-to-shell weld examinations (Item No. B3.90 for BWRs from 100% to a 25% sample of each nozzle type every 10 years) in References [1-18] and [1-19]. The work performed in BWRVIP-108 and BWRVIP-241 provided the technical basis for ASME Code Case N-702 [1-20], which has been conditionally approved by the NRC in Revision 20 of Regulatory Guide 1.147 [1-21].

REFERENCES

1-1 The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2007 Edition with 2008 Addenda.

1-2 EPRI Technical Report 3002015905: *Technical Bases for Inspection Requirements for PWR Pressurizer Vessel Head, Shell-to-Head and Nozzle-to-Vessel Welds*. Palo Alto, California, 2019.

1-3 EPRI Technical Report 3002015906: *Technical Bases for Inspection Requirements for PWR Steam Generator Class 1 Nozzle-to-Vessel Welds and Class 1 and Class 2 Vessel Head, Shell, Tubesheet-to-Head and Tubesheet-to-Shell Welds*. Palo Alto, California, 2019.

1-4 EPRI Technical Report 3002014590: *Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Inside Radius Sections*. Palo Alto, California, 2019.

1-5 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)

1-6 Letter from James G. Danna (NRC) to Eric Carr (PSEG Nuclear, LLC), "Salem Generating Station Unit Nos. 1 and 2 – Authorization and Safety Evaluation for Alternative Request No. SC-I4R-200 (EPID L-2020-LLR-0103)," dated April 12, 2021. (ADAMS Accession No. ML20218A587)

1-7 Letter from Michael T. Markley (NRC) to Cheryl A. Gayheart (Southern Nuclear Operating Company, Inc.), "Vogtle Electric Generating Plant, Units 1 & 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)

1-8 Letter from Gerald T. Bischof (Dominion Energy) to the NRC, "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," dated March 19, 2021. (ADAMS Accession No. ML21034A576)

1-9 Letter from James G. Danna (NRC) to Daniel G. Stoddard (Dominion Energy), "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)

1-10 Letter from C. A. Gayheart (Southern Nuclear Operating Company, Inc.) to the NRC, "Vogtle Electric Generating Plant, Units 1 & 2 Proposed Inservice Inspection Alternative VEGP-ISI-ALT-04-04 Version 2.0," dated September 9, 2020. (ADAMS Accession No. ML20253A311)

1-11 Email Letter from J. Kim (NRC) to P. R. Duke (PSEG Nuclear, LLC), "Salem Generating Station Units 1 and 2.– Final Request for Additional Information Regarding Alternative for Examination of ASME Section XI, Category B-B, Item Number B2.11 and B2.12 (L-2020-LRR-0103)," dated February 11, 2021. (ADAMS Accession No. ML21043A144)

1-12 Letter from P. R. Duke, Jr. (PSEG Nuclear, LLC) to the NRC, "Response to Request for Additional for Proposed Alternative for ASME Section XI, Category B-B, Item Number B2.11 and B2.12," dated April 12, 2021. (ADAMS Accession No. ML21102A024)

1-13 ASME, 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.

1-14 B. A. Bishop, C. Boggess, N. Palm, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," WCAP-16168-NP-A, Revision 3, October 2011.

1-15 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)

1-16 EPRI Technical Report 1003557: *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*. Palo Alto, California, 2002.

1-17 EPRI Technical Report 1021005: *BWRVIP-241: BWR Vessels and Internals Project, Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Shell Welds and Nozzle Blend Radii*. Palo Alto, California, 2010.

1-18 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)

1-19 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)

1-20 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.

1-21 NRC Regulatory Guide 1.147, Revision 20, "Inservice Inspection Code Case Acceptability, ASME Code Section XI, Division 1," dated December 2021.

1-22 Letter from John G. Lamb (NRC) to C. A. Gayheart (Southern Nuclear Operating Company, Inc.), "Vogtle Electric Generating Plant, Units 1 & 2 Audit Plan for Relief Request Inservice Inspection Alternative VEGP-ISI-ALT-04-04 (EPID L-2019-LLR-0109)," dated May 14, 2020. (ADAMS Accession No. ML20128J311)

ATTACHMENT 2

Plant-Specific Applicability

VIRGIL C. SUMMER NUCLEAR STATION UNIT 1 (VCSNS)
DOMINION ENERGY SOUTH CAROLINA, INC. (DESC)

Plant-Specific Applicability

Section 9 of References [2-1], [2-2], and [2-3] 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 VCSNS is provided in Tables A1 through A7.

Tables A1 through A7 indicate that all plant-specific requirements are met for VCSNS. Therefore, the results and conclusions of the Electric Power Research Institute (EPRI) reports are applicable to VCSNS. Figures A1 and A2 show the layout of the pressurizer (PZR) and steam generators (SGs).

Table A1. Plant-specific applicability of References [2-1], [2-2], and [2-3] representative analyses to the VCSNS PZR and SG components

PZR Shell-to-Head Welds (Circumferential and Longitudinal) and Nozzle-to-Shell Welds ITEM NOS. B2.11, B2.12, AND B3.110		
Category	**Requirement from Reference [2-1]**	**Applicability to VCSNS**
General Requirements	The plant-specific PZR general transients and cycles must be bounded by those shown in Table 5-6 for a 60- year operating life. It should be noted that the number of cycles were extrapolated to 80 years in the evaluations.	The number and type of the VCSNS general transients are compared to the transients listed in Table 5-6 of Reference [2-1]. As shown in Tables A2 and A3, the VCSNS transients are bounded by the transients listed in Table 5-6 of Reference [2-1].
	The materials of the PZR 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 VCSNS PZR upper and lower heads and shells are fabricated of SA-533, grade A, class 2 material. The nozzle forgings are fabricated of SA-508, class 2 material. The materials for the PZR conform to the requirements of ASME Code, Section XI, Division 1, Appendix G, Paragraph G-2110.

Category	Requirement from Reference [2-1]	Applicability to VCSNS
Specific Requirements	The plant-specific PZR upper head and bottom head weld configurations must conform to those shown in Figure 1-1 (Item No. B2.11), Figure 1-2 (Item No. B2.12) and Figures 1-4 and 1-5 (Item No. B3.110) of Reference [2-1].	The VCSNS PZR upper head and bottom head weld configurations conform to those shown in Figure 1-1 (Item No. B2.11), Figure 1-2 (Item No. B2.12) and Figures 1-4 and 1-5 (Item No. B3.110) of Reference [2-1].
	The plant-specific dimensions of the PZR upper head and nozzles, shell, lower head, and the surge nozzle must be within the range of values listed in Table 9-1 of Reference [2-1].	The comparison of the VCSNS PZR dimensions with those in Table 9-1 of Reference [2-1] is provided in Table A4. The comparison shows that the VCSNS configurations are within the range of values shown in Table 9-1 of Reference [2-1].
	The plant-specific Insurge/Outsurge transient definitions (temperature difference between the PZR shell and the PZR surge nozzle fluid temperature and associated number of cycles) must be bounded by those shown in Table 5-10 for a Westinghouse/CE plant, of Reference [2-1].	In Attachment 3 of this request, the VCSNS Insurge/Outsurge transients are compared to the number and type of transients listed in Table 5-10 of Reference [2-1]. As can be seen from Attachment 3, Table A9, the VCSNS transients are bounded by those transients listed in Table 5-10 of Reference [2-1].

<table>
<tr><th colspan="3">SG Primary Side Tubesheet-To-Head Welds
ITEM NO. B2.40</th></tr>
<tr><th>Category</th><th>Requirement from Reference [2-2]</th><th>Applicability to VCSNS</th></tr>
<tr><td rowspan="2">General Requirements</td><td>The Loss of Power transient (involving unheated auxiliary feedwater (AFW) 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. If 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>VCSNS has not experienced a loss of power transient resulting in unheated AFW being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of portion of the vessel.</td></tr>
<tr><td>The materials of the SG vessel heads and tubesheet must be low alloy ferritic steels which conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.</td><td>The VCSNS SG vessel heads, tubesheet, shell, and 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.</td></tr>
<tr><td rowspan="2">Specific Requirements</td><td>The weld configurations must conform to those shown in Figures 1-1 and Figure 1-2 of Reference [2-2]</td><td>The VCSNS weld configurations conform to Figure 1-1 and Figure 1-2 of Reference [2-2].</td></tr>
<tr><td>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 [2-2].</td><td>The VCSNS SG vessel dimensions are as follows:<table><tr><th></th><th>Diameter</th></tr><tr><td>SG Lower Head</td><td>136.08 inches</td></tr><tr><td>SG Upper Shell</td><td>176.26 inches</td></tr></table>These dimensions are within 10 percent of those specified in Table 9-2 in Section 9.4.3 of Reference [2-2].</td></tr>
</table>

Category	Requirement from Reference [2-2]	Applicability to VCSNS
Specific Requirements	The component must experience transients and cycles bounded by those shown in Table 5-7 of Reference [2-2] over a 60-year operating life.	As shown in Table A5, the VCSNS transients and number of cycles projected to occur over a 60-year life are bound by those shown in Table 5-7 of Reference [2-2]. There is slight variation in some of the temperature and pressure parameters between VCSNS and Table 5-7 of Reference [2-2]. Due to the conservative Heatup and Cooldown rates used in Reference [2-2], the slight variation in the Heatup and Cooldown temperatures is not a concern. In addition, the VCSNS 60-year projected Heatup and Cooldown cycles is 114 which is significantly less than the 300 cycles evaluated in Reference [2-2]. Likewise, the evaluation performed in Reference [2-2] uses conservative rates of change in temperature and pressure for reactor trip transients, and the slight variation in the reactor trip parameters between VCSNS and Table 5-7 of Reference [2-2] is not a concern. Further, the VCSNS 60-year projected reactor trip cycles is 147 which is less than half of the 360 cycles evaluated in Reference [2-2]. VCSNS considers three (3) reactor trip transients including: • Case A – No Cooldown; • Case B – Cooldown, no SI; and • Case C – Cooldown, with SI. As seen in Table A5, Case C is the worse-case transient, and the VCSNS 60-year projected reactor trip for Case C cycles is only 5.

<table>
<tr><th colspan="3">SG Secondary Side Shell Welds
ITEMS NOS. C1.10, C1.20 AND C1.30</th></tr>
<tr><th>Category</th><th>Requirement from Reference [2-1]</th><th>Applicability to VCSNS</th></tr>
<tr><td rowspan="2">General Requirements</td><td>The Loss of Power transient (involving unheated AFW 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. If 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>VCSNS has not experienced a loss of power transient resulting in unheated AFW being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of portion of the vessel.</td></tr>
<tr><td>The materials of the SG shell, feedwater (FW) nozzles, and main steam (MS) nozzles must be low alloy ferritic steels which conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.</td><td>The VCSNS SG vessel heads, tubesheet, shell, and 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.</td></tr>
<tr><td rowspan="2">Specific Requirements</td><td>The weld configurations must conform to those shown in Figure 1-7 and Figure 1-8 of Reference [2-2].</td><td>The VCSNS weld configurations conform to Figure 1-7 and Figure 1-8 of Reference [2-2].</td></tr>
<tr><td>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 [2-2].</td><td>The VCSNS SG vessel dimensions are as follows:
<table><tr><th></th><th>Diameter</th></tr><tr><td>SG Lower Head</td><td>136.08 inches</td></tr><tr><td>SG Upper Shell</td><td>176.26 inches</td></tr></table>These dimensions are within 10 percent of those specified in Table 9-3 in Section 9.4.4 of Reference [2-2].</td></tr>
</table>

Category	Requirement from Reference [2-1]	Applicability to VCSNS
	The component must experience transients and cycles bounded by those shown in Table 5-9 of Reference [2-2] over a 60-year operating life.	As shown in Table A6, the VCSNS transients and number of cycles projected to occur over a 60-year life are bound by those shown in Table 5-9 of Reference [2-2]. There is slight variation in some of the temperature parameters between VCSNS and Table 5-9 of Reference [2-2]. Due to the conservative Heatup and Cooldown rates used in Reference [2-2], the slight variation in the Heatup and Cooldown temperatures is not a concern. In addition, the VCSNS 60-year projected Heatup and Cooldown cycles is 114 which is significantly less than the 300 cycles evaluated in Reference [2-2]. Likewise, the evaluation performed in Reference [2-2] uses conservative rates of change in temperature and pressure for reactor trip transients, and the slight variation in the reactor trip temperatures between VCSNS and Table 5-9 of Reference [2-2] is not a concern. Further, the VCSNS 60-year projected reactor trip cycles is 147 which is less than half of the 360 cycles evaluated in Reference [2-2]. VCSNS considers three (3) reactor trip transients including: • Case A – No Cooldown; • Case B – Cooldown, no SI; and • Case C – Cooldown, with SI. The reactor trip temperature and pressure parameters in Table A6 are for Case C which is the worse-case transient, and the VCSNS 60-year projected reactor trip for Case C cycles is only 5.

SG FW Nozzle-to-Shell Welds and Inside Radius Sections ITEMS NOS. C2.21 AND C2.22		
Category	**Requirement from Reference [2-3]**	**Applicability to VCSNS**
General Requirements	The nozzle-to-shell weld shall be one of the configurations shown in Figure 1-1 or Figure 1-2 of Reference [2-3].	The VCSNS FW nozzles are representative of the configuration shown in Figure 1-2 of Reference [2-3].
	The materials of the SG shell and FW nozzles must be low alloy ferritic steels which conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.	The VCSNS FW nozzles and vessel shell are fabricated of SA-508, class 3a material. This material conforms to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.
	The SG must not experience more than the number of all transients shown in Table 5-5 of Reference [2-3] over a 60-year operating life.	As shown in Table A7, the VCSNS SG is not projected to experience more than the number of the transients shown in Table 5-5 of Reference [2-3].
SG FW Nozzle	The piping attached to the FW nozzle must be 14-inch to 18-inch NPS.	The VCSNS FW piping lines are 16 inches.
	The FW nozzle design must have an integrally attached thermal sleeve.	The VCSNS FW nozzle configuration has an integrally attached thermal sleeve.
	AFW nozzles connected directly to the SG are not covered in this evaluation.	N/A

Table A2. VCSNS PZR general transients applicable to this request

Transient Name	VCSNS		
	Up to 2018	60-Year Projected	Maximum Cycles (Controlling Limit)
Heatup @ <100°F/hr	68	114	200
Cooldown @ <100°F/hr	66	110	200
Reactor Trips (1)	88	147	400
50% Step Load Decrease w/ Steam Dump	58	97	200
Loss of Load	17	29	80
Loss of Flow in One RC Loop Only	2	4	80
Loss of Offsite AC Power	7	12	40

(1) Reactor Trip Transients include Case A – No Cooldown; Case B – Cooldown with no SI; and Case C – Cooldown with SI.

Table A3. Comparison of VCSNS PZR general transients to the transients evaluated in Reference [2-1]

Transient	Number of Cycles for 60 Years from Reference [2-1]	VCSNS 60-Year Projections
Heatup/Cooldown	300	114
Loss of Load (1)	360	289

(1) Sum of Reactor Trips, 50% Step Load Decrease with Steam Dump, Loss of Load, Loss of Flow in One RC Loop Only, and Loss of Offsite AC Power Events.

Table A4. Range of PZR geometric parameters for which the evaluation is applicable in comparison with VCSNS

Component	**Geometric Parameter** *(inches)*	**Westinghouse Plant** *(inches)*	**VCSNS Dimensions** *(inches)*
PZR Shell	Inside diameter	Must be between 80 and 88	84
Surge Nozzle	NPS of piping or component (e.g., reducer) attached to nozzle safe-end	Must be between 12 and 18	14
Safety/Relief Nozzle	NPS of piping or component (e.g., reducer) attached to nozzle safe-end	Must be between 4 and 8	6
Spray Nozzle	NPS of piping or component (e.g., reducer) attached to nozzle safe-end	Must be between 4 and 6	4

Table A5. VCSNS data for thermal transients for stress analysis of the PWR SG primary-side head welds (Comparison to Table 5-7 of Reference [2-2])

Transient		**Max T_{hot}** *(°F)*	**Min T_{hot}** *(°F)*	**Max T_{cold}** *(°F)*	**Min T_{cold}** *(°F)*	**Max Press** *(PSIG)*	**Min Press** *(PSIG)*	**60-Year Cycles**
Heatup/Cooldown	Reference [2-2]	545	70	545	70	2235	0	300
	VCSNS	557	70	557	70	2235	0	200
Plant Loading / Unloading	Reference [2-2]	610	550	550	545	2300	2300	5000
	VCSNS	Not typical operation, not counted [(1)]						
Reactor Trip	Reference [2-2]	615	530	565	530	2435	1700	360
Reactor Trip [(2)] – Case A	VCSNS	621.9	563.9	552.6	557.6	2235	1960	230
Reactor Trip [(2)] – Case B	VCSNS	621.9	531.9	552.6	517.6	2235	1600	160
Reactor Trip [(2)] – Case C	VCSNS	621.9	457.9	552.6	452.6	2235	1485	10

(1) Load following operation is not typical.

(2) Reactor Trip Transients include Case A – No Cooldown; Case B – Cooldown with no SI; and Case C – Cooldown, with SI. Transient is assumed to begin at 100% full power.

Table A6. VCSNS data for thermal transients for stress analysis of the PWR SG secondary-side vessel welds (Comparison to Table 5-9 of Reference [2-2])

Transient		Max T_{ss} (°F)	Min T_{ss} (°F)	Max Press (PSIG)	Min Press (PSIG)	60-Year Cycles
Heatup/Cooldown	Reference [2-2]	545	70	1000	0	300
	VCSNS	557	70	951	0	200
Plant Loading/Unloading	Reference [2-2]	545	540	1000	1000	5000
	VCSNS	Not typical operation, not counted (1)				
Reactor Trip	Reference [2-2]	555	530	1130	1000	360
Reactor Trip (2)	VCSNS	550.4	460.4	951 (3)	951 (3)	400

(1) Load following operation not typical.
(2) Reactor Trip Transient Case C – Cooldown, with SI, is considered since it is the worse-case transient.
(3) SG secondary-side pressure is not an analyzed design transient parameter for Reactor Trip Transients.

Table A7. VCSNS data for thermal transients applicable to pressurized water reactor SG FW (Comparison to Table 5-5 of Reference [2-3])

Transient	Cycles from Table 5-5 of Reference [2-3]	VCSNS 60-year Projected Cycles	VCSNS 60-year Allowable Cycles
Heatup/Cooldown	300	58	200
Plant Loading	5000	Not typical operation (1)	
Plant Unloading			
Loss of Load	360	N/A (2)	N/A (2)
Loss of Power	60	N/A (2)	N/A (2)

(1) Load following operation not typical.

(2) Loss of Load and Loss of Power Transients are not evaluated. Upon Loss of Load and Loss of Power Transients, flow to the FW nozzle is at nominal temperature before 60 seconds. After 60 seconds, flow is to the emergency FW nozzle. Therefore, VCSNS will not experience Loss of Load or Loss of Power Transients on the FW nozzle.

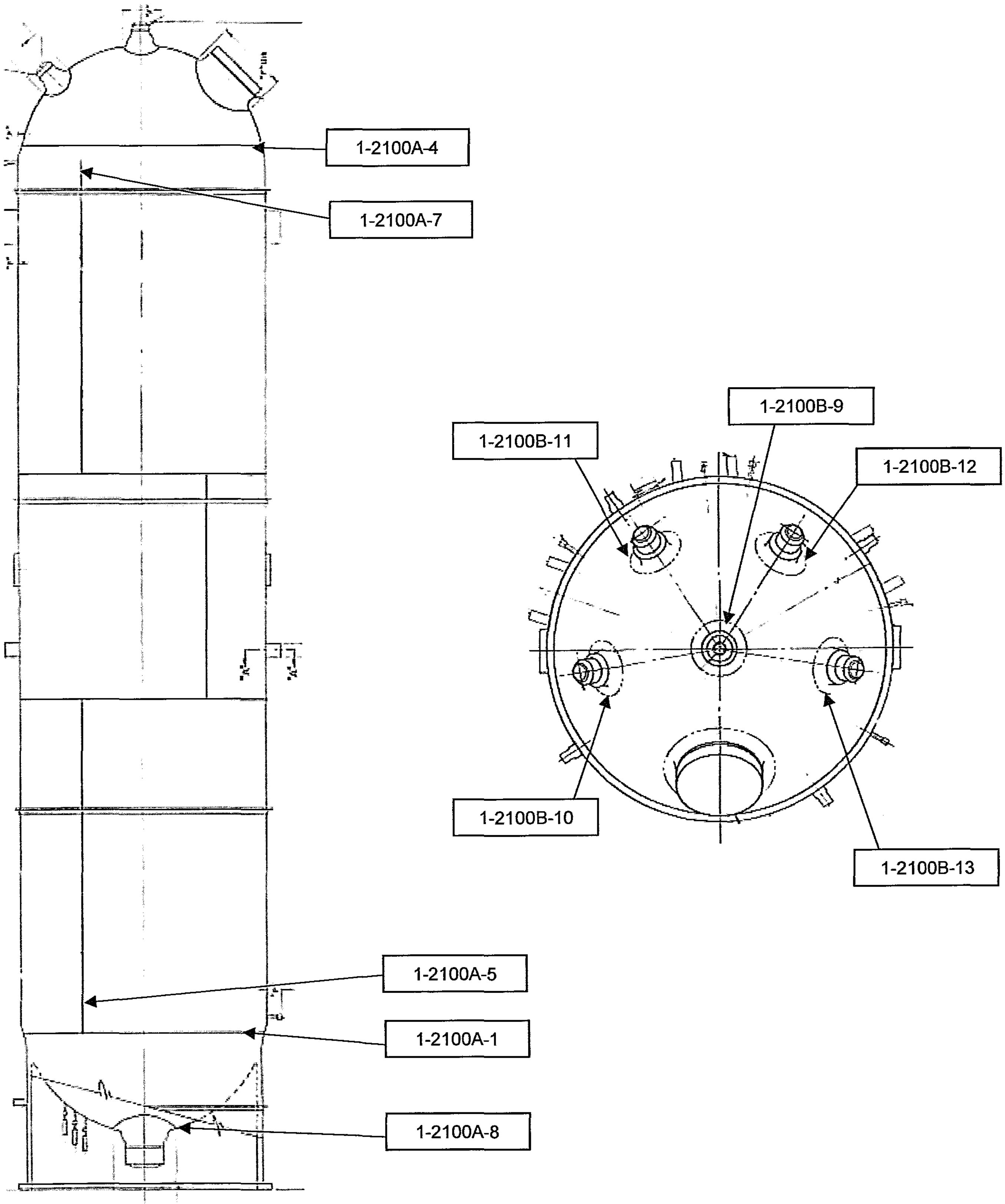

Figure A1. VCSNS Pressurizer Layout

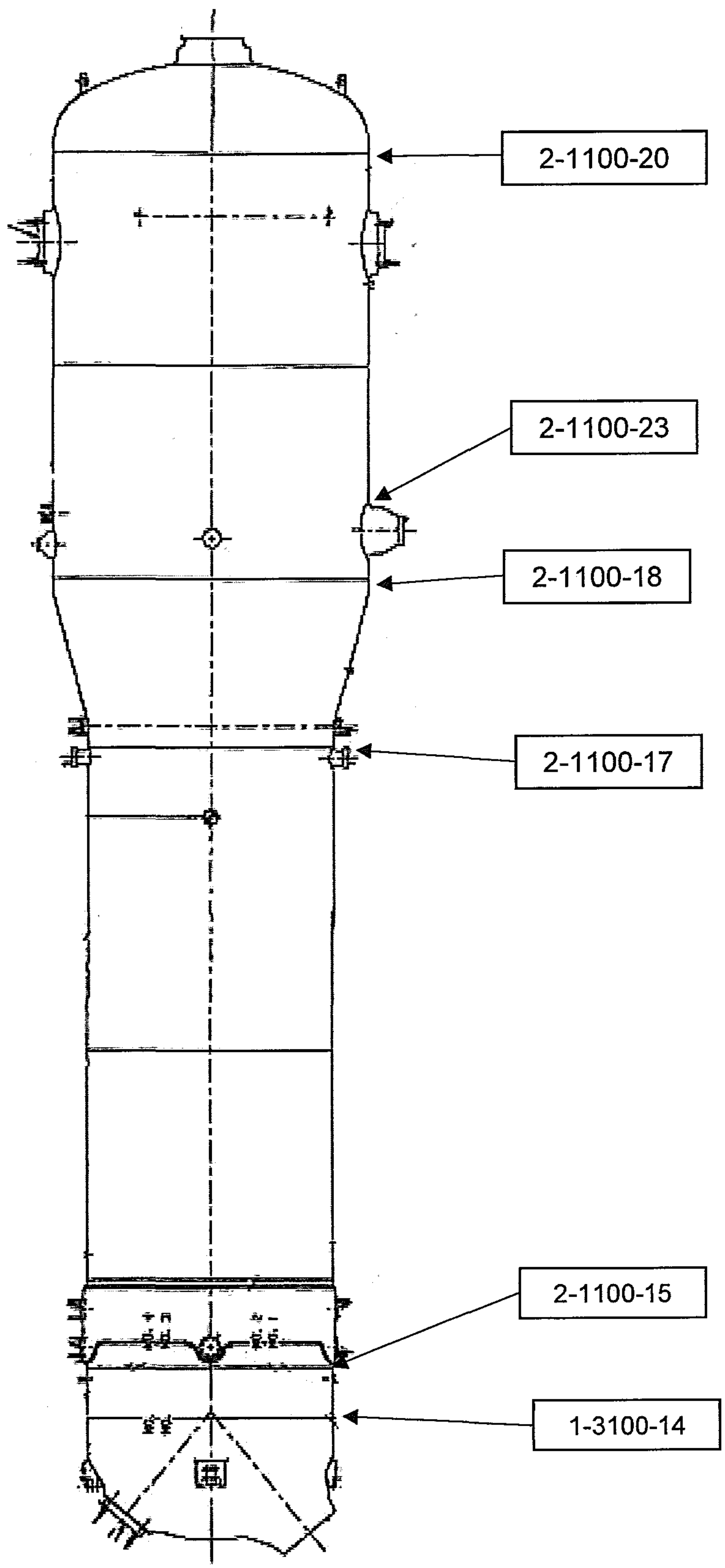

Figure A2. VCSNS Steam Generator Layout

REFERENCES

2-1 EPRI Technical Report 3002015905: *Technical Bases for Inspection Requirements for PWR Pressurizer Vessel Head, Shell-to-Head and Nozzle-to-Vessel Welds*. Palo Alto, California, 2019.

2-2 EPRI Technical Report 3002015906: *Technical Bases for Inspection Requirements for PWR Steam Generator Class 1 Nozzle-to-Vessel Welds and Class 1 and Class 2 Vessel Head, Shell, Tubesheet-to-Head and Tubesheet-to-Shell Welds*. Palo Alto, California, 2019.

2-3 EPRI Technical Report 3002014590: *Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Inside Radius Sections*. Palo Alto, California, 2019.

ATTACHMENT 3

Comparison of Insurge/Outsurge Transients

VIRGIL C. SUMMER NUCLEAR STATION UNIT 1 (VCSNS)
DOMINION ENERGY SOUTH CAROLINA, INC. (DESC)

Comparison of Insurge/Outsurge Transients

VCSNS Insurge/Outsurge (I/O) transients are provided in Table A8. The temperature differences (ΔTs) identified in Table A8 are combined conservatively by summing all the events into the 320°F ΔT bin. It should be noted that the transients in Table A8 reflect 35 years of operation. For comparison with Table 5-10 of Reference [3-1], they are extrapolated to 60 years by multiplying by 1.7. With this conservative treatment of the I/O transients, the comparison of VCSNS I/O transients to the requirements in Reference [3-1] is shown in Table A9. The results of Table A9 indicate that the VCSNS I/O transients are bounded by those in Reference [3-1].

Table A8. 35-year I/O transients for VCSNS

Number	Transient Name (1)	35-Year Cycles
1	HU340	0
2	HU330	0
3	HU320	0
4	HU310	0
5	HU300	2
6	HU280	7
7	HU260	37
8	HU240	8
9	CD340	1
10	CD330	0
11	CD320	0
12	CD310	1
13	CD300	3
14	CD280	13
15	CD260	32
16	CD240	3

(1) The Transient Name is XXnnn, where XX = HU for Insurge/Outsurge transients that occur during Heatup events, or CD for I/O transients that occur during Cooldown events, and nnn = the temperature difference, ΔT, between the PZR fluid temperature and the fluid temperature in the surge nozzle.

Table A9. Comparison of VCSNS I/O transient temperature differences and numbers of cycles with the I/O transient date from Reference [3-1]

ΔT (°F) [(1)]	60-Year Number of Cycles From Reference [3-1]	VCSNS Cycles Projected to 60 Years of Operation
330	600	0
320	3,000	182 [(2)(3)]
103	1,500	0

(1) ΔT is the temperature difference between the PZR fluid temperature and the fluid temperature in the surge nozzle.

(2) The number of cycles is conservatively equal to the sum of all events in Table A8 analyzed for 35 years (107 cycles), increased by a factor of 1.7 to reflect 60 years of operation.

(3) Transient CD340 was projected under the 320°F ΔT bin. A 340°F ΔT is an anomaly and is not expected to occur in the future.

REFERENCES

3-1 EPRI Technical Report 3002015905: *Technical Bases for Inspection Requirements for PWR Pressurizer Vessel Head, Shell-to-Head and Nozzle-to-Vessel Welds*. Palo Alto, California, 2019.

ATTACHMENT 4

Inspection History

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Inspection History

Table A10 provides the inspection history for the VCSNS pressurizer (PZR) and steam generator (SG) welds and components for the pre-service inspection (PSI) and the first, second, third, and fourth 10-year inservice inspection (ISI) intervals. Several of the older examination reports that were used to populate Table A10 did not record the examination coverage. The examination coverages documented in Table A10 show that the coverages are consistent for the examinations performed throughout the examination history for that weld or component. Therefore, the coverages for the more recent examinations can be used to determine the examination coverage for the earlier examination where coverage is not documented.

Table A10. PZR and SG inspection history

Pressurizer Inspection History						
Item No.	**Component ID**	**Examination Date**	**Interval / Period**	**Examination Results**	**Examination Coverage**	**Relief Request**
B2.11	1-2100A-1	1982	PSI	Acceptable	100%	N/A
	1-2100A-1	4 - 1990	I1 / P2	Acceptable	100%	N/A
	1-2100A-1	9 - 1994	I2 / P1	Acceptable	98.3%	N/A
	1-2100A-1	4 - 2011	I3 / P3	Acceptable	98.3%	N/A
	1-2100A-4	1982	PSI	Acceptable	100%	N/A
	1-2100A-4	10 - 1991	I1 / P3	Acceptable	87%	N/A
	1-2100A-4	10 - 1997	I2 / P1	Acceptable	98%	N/A
	1-2100A-4	5 - 2005	I3 / P1	Acceptable	98%	N/A
	1-2100A-4	4 - 2014	I4 / P1	Acceptable	97%	N/A
B2.12	1-2100A-5	1982	PSI	Acceptable	100%	N/A
	1-2100A-5	4 - 1990	I1 / P2	Acceptable	100%	N/A
	1-2100A-5	5 - 2002	I2 / P3	Acceptable	100%	N/A
	1-2100A-5	4 - 2011	I3 / P3	Acceptable	100%	N/A
	1-2100A-7	1982	PSI	Acceptable	100%	N/A
	1-2100A-7	10 - 1991	I1 / P3	Acceptable	100%	N/A
	1-2100A-7	10 - 1997	I2 / P1	Acceptable	100%	N/A
	1-2100A-7	5 - 2005	I3 / P1	Acceptable	100%	N/A
	1-2100A-7	4 - 2014	I4 / P1	Acceptable	100%	N/A

Pressurizer Inspection History						
Item No.	**Component ID**	**Examination Date**	**Interval / Period**	**Examination Results**	**Examination Coverage**	**Relief Request**
B3.110	1-2100A-8	1982	PSI	Acceptable	Note 1	N/A
	1-2100A-8	10 -1985	I1 / P1	Acceptable	Note 1	N/A
	1-2100A-8	10 -1997	I2 / P1	Acceptable	Note 1	N/A
	1-2100A-8	10 -2012	I3 / P3	Acceptable	75.5%	RR-III-10
	1-2100B-9	1982	PSI	Acceptable	Note 1	N/A
	1-2100B-9	10 -1985	I1 / P1	Acceptable	Note 1	N/A
	1-2100B-9	4 -1996	I2 / P1	Acceptable	Note 1	N/A
	1-2100B-9	5 -2005	I3 / P1	Acceptable	92%	N/A
	1-2100B-9	4 - 2014	I4 / P1	Acceptable	70.2%	TBD
	1-2100B-10	1982	PSI	Acceptable	Note 1	N/A
	1-2100B-10	10 - 1991	I1 / P3	Acceptable	Note 1	N/A
	1-2100B-10	4 - 1996	I2 / P1	Acceptable	Note 1	N/A
	1-2100B-10	5 - 2005	I3 / P1	Acceptable	92%	N/A
	1-2100B-10	4 - 2014	I4 / P1	Acceptable	72.3%	TBD
	1-2100B-11	1982	PSI	Acceptable	Note 1	N/A
	1-2100B-11	10 - 1991	I1 / P3	Acceptable	Note 1	N/A
	1-2100B-11	4 - 1996	I2 / P1	Acceptable	Note 1	N/A
	1-2100B-11	5 - 2005	I3 / P1	Acceptable	92%	N/A
	1-2100B-11	4 - 2014	I4 / P1	Acceptable	72.3%	TBD
	1-2100B-12	1982	PSI	Acceptable	Note 1	N/A
	1-2100B-12	10 - 1991	I1 / P3	Acceptable	Note 1	N/A
	1-2100B-12	4 - 1996	I2 / P1	Acceptable	Note 1	N/A
	1-2100B-12	5 - 2005	I3 / P1	Acceptable	92%	N/A

Pressurizer Inspection History						
Item No.	Component ID	Examination Date	Interval / Period	Examination Results	Examination Coverage	Relief Request
	1-2100B-12	4 - 2014	I4 / P1	Acceptable	72.3%	TBD
	1-2100B-13	1982	PSI	Acceptable	Note 1	N/A
	1-2100B-13	10 -1985	I1 / P1	Acceptable	Note 1	N/A
	1-2100B-13	4 - 1996	I2 / P1	Acceptable	Note 1	N/A
	1-2100B-13	5 - 2005	I3 / P1	Acceptable	92%	N/A
	1-2100B-13	4 - 2014	I4 / P1	Acceptable	72.3%	TBD

Note 1: Coverage was not documented.

Steam Generator Inspection History						
Item No.	**Component ID**	**Examination Date**	**Interval / Period**	**Examination Results**	**Examination Coverage**	**Relief Request**
SG Primary Side Shell Welds						
B2.40	1-3100-14A	7 - 1994	I2 / P1 PSI	Acceptable	100%	N/A
	1-3100-14B	7 - 1994	I2 / P1 PSI	Acceptable	100%	N/A
	1-3100-14C	8 - 1994	I2 / P1 PSI	Acceptable	100%	N/A
	1-3100-14C	11 - 2003	I2 / P3	Acceptable	100%	N/A
	1-3100-14C	11 - 2009	I3 / P2	Acceptable	100%	N/A
	1-3100-14C	4 - 2020	I4 / P2	Acceptable	100%	N/A
SG Secondary Side Shell Welds						
C1.10	2-1100-17A	7 - 1994	I2 / P1 PSI	Acceptable	Note 1	N/A
	2-1100-17B	7 - 1994	I2 / P1 PSI	Acceptable	Note 1	N/A
	2-1100-17C	7 - 1994	I2 / P1 PSI	Acceptable	Note 1	N/A
	2-1100-17C	4 - 1999	I2 / P2	Acceptable	Note 1	N/A
	2-1100-17C	11 - 2009	I3 / P2	Acceptable	98%	N/A
	2-1100-18A	7 - 1994	I2 / P1 PSI	Acceptable	100%	N/A
	2-1100-18B	7 - 1994	I2 / P1 PSI	Acceptable	100%	N/A
	2-1100-18C	7 - 1994	I2 / P1 PSI	Acceptable	100%	N/A
	2-1100-18C	4 - 1999	I2 / P2	Acceptable	100%	N/A
	2-1100-18C	11 - 2009	I3 / P2	Acceptable	100%	N/A
	2-1100-18C	4 - 2020	I4 / P3	Acceptable	100%	N/A

Steam Generator Inspection History						
Item No.	**Component ID**	**Examination Date**	**Interval / Period**	**Examination Results**	**Examination Coverage**	**Relief Request**
C1.20	2-1100-20A	7 - 1994	I2 / P1 PSI	Acceptable	100%	N/A
	2-1100-20B	7 - 1994	I2 / P1 PSI	Acceptable	100%	N/A
	2-1100-20C	7 - 1994	I2 / P1 PSI	Acceptable	100%	N/A
	2-1100-20C	5 - 2002	I2 / P3	Acceptable	100%	N/A
	2-1100-20C	10 - 2012	I3 / P3	Acceptable	100%	N/A
	2-1100-20C	4 - 2020	I4 / P3	Acceptable	99.5%	N/A
C1.30	2-1100-15A	7 - 1994	I2 / P1 PSI	Acceptable	Note 1	N/A
	2-1100-15B	7 - 1994	I2 / P1 PSI	Acceptable	Note 1	N/A
	2-1100-15C	7 - 1994	I2 / P1 PSI	Acceptable	Note 1	N/A
	2-1100-15C	11 - 2003	I2 / P3	Acceptable	94.4%	N/A
	2-1100-15C	10 - 2012	I3 / P3	Acceptable	94.4%	N/A
SG Secondary Side Nozzles						
C2.21	2-1100-23A	7 - 1994	I2 / P1 PSI	Acceptable	Note 1	N/A
	2-1100-23B	7 - 1994	I2 / P1 PSI	Acceptable	Note 1	N/A
	2-1100-23C	7 - 1994	I2 / P1 PSI	Acceptable	Note 1	N/A
	2-1100-23C	10 - 2003	I2 / P3	Acceptable	100%	N/A
	2-1100-23C	10 - 2012	I3 / P3	Acceptable	100%	N/A
C2.22	2-1100-23IR-A	7 - 1994	I2 / P1 PSI	Acceptable	100%	N/A
	2-1100-23IR-B	7 - 1994	I2 / P1 PSI	Acceptable	100%	N/A
	2-1100-23IR-C	7 - 1994	I2 / P1 PSI	Acceptable	100%	N/A
	2-1100-23IR-C	10 - 2003	I2 / P3	Acceptable	100%	N/A
	2-1100-23IR-C	10 - 2012	I3 / P3	Acceptable	100%	N/A

Note 1: Coverage was not documented

ATTACHMENT 5

Results of Industry Survey

VIRGIL C. SUMMER NUCLEAR STATION UNIT 1 (VCSNS)
DOMINION ENERGY SOUTH CAROLINA, INC. (DESC)

Overall Industry Inspection Summary for Pressurizer Code Item Nos. B2.11, B2.12, B2.21, B2.22 and B3.110

The results of an industry survey of past pressurizer (PZR) weld inspections are summarized in Electric Power Research Institute's (EPRI's) Technical Report 3002015905, *"Technical Bases for Inspection Requirements for PWR Pressurizer Vessel Head, Shell-to-Head and Nozzle-to-Vessel Welds"* [5-1]. A total of 47 domestic and international pressurized water reactor (PWR) units responded to the survey. The survey represented all PWR plant designs currently in operation in the United States, including two (2)-loop, three (3)-loop, and four (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).

The combined survey results for Item Nos. B2.11, B2.12, B2.21, B2.22, and B3.110 are summarized in Table A11 below. A total of 1,162 examinations of PWR PZR components were reported by the survey for the affected Item Nos. Of the 1,169 total examinations, only four (4) examinations identified flaws exceeding the acceptance criteria of ASME Code, Section XI. All four (4) flaw indications for Item No. B2.11 occurred at two (2) units of a single plant site. None of these flaws were found to be service induced. Flaw evaluations were performed to show acceptability of these indications and follow-on examinations showed no change in flaw sizes since the original inspections. No other indications were identified in any in-scope components.

The results in Table A11 indicate that the number of reportable indications resulting from examinations of the PWR PZR components for the affected Item Nos. is negligible. Therefore, the increase in worker radiation exposure, risk to personnel safety, and production of radwaste resulting from these examinations adversely impacts outage-related activities without a corresponding increase in the level of quality or safety.

Table A11. Summary of survey results for Item Nos. B2.11, B2.12, B2.21, B2.22 and B3.110

Item No.	Number of Examinations	Number of Reportable Indications
B2.11	269	4 (1)
B2.12	269	0
B2.21	4	0
B2.22	30	0
B3.110	590	0

(1) Flaw evaluations were performed to show acceptability of these indications and follow-on examinations showed no change in flaw sizes since the original inspections.

Overall Industry Inspection Summary for Steam Generator 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 steam generator (SG) nozzle-to-shell welds, inside radius sections, and shell welds are summarized in EPRI's Technical Report 3002015906, *"Technical Bases for Inspection Requirements for PWR Steam Generator Class 1 Nozzle-to-Vessel Welds and Class 1 and Class 2 Vessel Head, Shell, Tubesheet-to-Head and Tubesheet-to-Shell Welds"* [5-2]. A total of 74 domestic and international BWR and PWR units responded to the survey. The survey represented all PWR plant designs currently in operation in the United States including two (2)-loop, three (3)-loop, and four (4)-loop PWR designs from each of the PWR NSSS vendors (i.e., B&W, CE, and Westinghouse).

The combined survey results for Item Nos. B2.31, B2.32 (see Table Note 3), B2.40, B3.130, C1.10, C1.20, and C1.30 are summarized in Table A12 below. A total of 1,324 examinations were reported by the survey for the components of the affected Item Nos., with 1,098 of these specifically for PWR components. The majority of the PWR examinations were performed on SG welds.

A 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 (2) 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 Table Note 1). For Item No. C1.20, two (2) PWR units reported flaws exceeding the acceptance criteria of ASME Code, Section XI. In the first unit, a single flaw was identified and evaluated as an inner diameter surface imperfection. Reference [5-2] 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 performed in accordance with IWC-3600 determined this indication was acceptable for continued operation. In the second unit, multiple flaws were identified (see Table Note 2). As discussed in References [5-5] and [5-6], these flaws were most likely subsurface weld defects typical of thick vessel welds and not service-induced. A flaw evaluation performed in accordance with IWC-3600 determined these flaws to be acceptable for continued operation.

The results of the industry survey identified numerous SG examinations being performed with no service-induced flaws being detected. The results in Table A12 indicate that the number of reportable indications resulting from examinations of the PWR SG components for the affected Item Nos. is negligible. Therefore, the increase in worker radiation exposure, risk to personnel safety, and production of radwaste resulting from these examinations adversely impacts outage-related activities without a corresponding increase in level of quality or safety.

Table A12. Summary of survey results for Item Nos. B2.31, B2.32, B2.40, B3.130, C1.10, C1.20, and C1.30

Item No.	Number of Examinations			Number of Reportable Indications		
	BWR	PWR	Total	BWR	PWR	Total
B2.31	0	30	30	0	0	0
B2.32 (3)	0	13	13	0	0	0
B2.40	0	183	183	0	(1)	(1)
B3.130	0	135	135	0	0	0
C1.10	140	305	445	0	0	0
C1.20	54	319	373	0	(2)	(2)
C1.30	32	113	145	0	0	0
Totals	**226**	**1098**	**1324**	**0**	(1) (2)	(1) (2)

(1) Two (2) 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 [5-4, 5-5].
(3) Item No. B2.32 was evaluated in the Reference [5-1] technical basis and included in the industry survey but is not contained in the scope of this alternative request.

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

The results of an industry survey of past inspections of SG feedwater (FW) and main steam (MS) nozzles are summarized in EPRI's Technical Report 3002014590, "*Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Inside Radius Sections*" [5-3]. A total of 74 domestic and international BWR and PWR units responded to the survey. The survey represented all PWR plant designs currently in operation in the United States including two (2)-loop, three (3)-loop, and four (4)-loop PWR designs from each of the PWR NSSS vendors (i.e., B&W, CE, and Westinghouse).

The combined survey results for Item Nos. C2.21, C2.22, and C2.32 are summarized in Table A13 below (see Table Note 1). A total of 727 examinations for Item Nos. C2.21, C2.22, and C2.32 (see Table Note 1) components were conducted, with 563 of these specifically for PWR components. The majority of the PWR examinations were performed on SG FW and MS nozzles. Only one (1) PWR examination identified two (2) flaws that exceeded the ASME Code, Section XI acceptance criteria. The flaws were linear indications of 0.3 inches and 0.5 inches in length and were detected in a MS nozzle-to-shell weld using magnetic particle examination techniques. The indications were dispositioned by light grinding [5-4].

The results of the industry survey identified numerous SG FW and MS nozzle-to-shell welds and nozzle inside radius section examinations being performed with no service-induced flaws being detected. The results in Table A13 indicate that the number of reportable indications resulting from examinations of the SG FW and MS nozzles for the affected Item Nos. is negligible. Therefore, the increase in worker radiation exposure, risk to personnel safety, and production of radwaste resulting from these examinations adversely impacts outage-related activities without a corresponding increase in level of quality or safety.

Table A13. Summary of survey results for Item Nos. C2.21, C2.22, and C2.32

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

(1) Item No. C2.32 was evaluated in the Reference [5-2] technical basis and included in the industry survey but is not contained in the scope of this alternative request.

REFERENCES

5-1 EPRI Technical Report 3002015905: *Technical Bases for Inspection Requirements for PWR Pressurizer Vessel Head, Shell-to-Head and Nozzle-to-Vessel Welds*. Palo Alto, California, 2019.

5-2 EPRI Technical Report 3002015906: *Technical Bases for Inspection Requirements for PWR Steam Generator Class 1 Nozzle-to-Vessel Welds and Class 1 and Class 2 Vessel Head, Shell, Tubesheet-to-Head and Tubesheet-to-Shell Welds*. Palo Alto, California, 2019.

5-3 EPRI Technical Report 3002014590: *Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Inside Radius Sections*. Palo Alto, California, 2019.

5-4 Letter from F. A. Kearney (Exelon) to NRC, "Byron Station Unit 2 90-Day Inservice Inspection Report for Interval 3, Period 3, (B2R17)," dated July 29, 2013. (ADAMS Accession Number ML13217A093)

5-5 Letter from J. M. 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. (ADAMS Accession Number ML011550346)

5-6 Letter from J. P. 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. (ADAMS Accession Number ML031040553)