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0CAN062402 10 CFR 50.55a

June 6, 2024

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

Subject: Proposed Alternatives ANO1-ISI-24-01 and ANO2-ISI-24-01 for Examinations of Steam Generator Pressure-Retaining Welds and Full Penetration Welded Nozzles

Arkansas Nuclear One – Units 1 and 2
NRC Docket Nos. 50-313 and 50-368
Renewed Facility Operating License Nos. DPR-51 and NPF-6

In accordance with 10 CFR 50.55a(z)(1), Entergy Operations, Inc. (Entergy) hereby requests Nuclear Regulatory Commission (NRC) approval of proposed alternatives for Arkansas Nuclear One, Units 1 and 2 (ANO-1 and ANO-2, respectively).

The proposed alternatives are to defer the In-Service Inspection (ISI) examinations for select examination categories and item numbers for the steam generators at ANO-1 and ANO-2 from the current American Society of Mechanical Engineers (ASME) Code, Section XI, Division 1 10-year requirements to the end of currently licensed operating life, which is scheduled to end on May 20, 2034, and July 17, 2038, respectively. This equates to an extension of 16 years, 11 months, 20 days from the end of the fourth ISI interval (May 30, 2017) for ANO-1 and an extension of 17 years, 3 months, 23 days from the end of the fourth ISI interval (March 25, 2021) for ANO-2 at which time all ASME Code, Section XI, Division 1 requirements were satisfied. Entergy requests authorization to use the proposed alternatives pursuant to 10 CFR 50.55a(z)(1) on the basis that the alternative provides an acceptable level of quality and safety.

The Enclosure to this letter provides the Alternative Requests ANO1-ISI-24-01 and ANO2-ISI-24-01 with Enclosure, Attachment 1 providing the Plant-Specific Applicability ANO-1, Enclosure, Attachment 2 providing the Plant-Specific Applicability ANO-2, and Enclosure, Attachment 3 providing the Results of Industry Survey.

Entergy requests approval of the proposed Alternative Requests by May 30, 2025. This date will support the conclusion of the second period, fifth ISI interval for ANO-1.

This letter contains no new regulatory commitments.

If there are any questions or if additional information is needed, please contact Riley Keele, Manager, Regulatory Assurance, Arkansas Nuclear One, at 479-858-7826.

Respectfully,

Philip Couture

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PC/rwc

Enclosure: Alternative Requests ANO1-ISI-24-01 and ANO2-ISI-24-01

Attachments to Enclosure:

1. Plant-Specific Applicability ANO-1
2. Plant-Specific Applicability ANO-2
3. Results of Industry Survey

cc: NRC Region IV Regional Administrator
NRC Senior Resident Inspector – Arkansas Nuclear One
NRC Project Manager – Arkansas Nuclear One

ENCLOSURE

0CAN062402

ALTERNATIVE REQUESTS ANO1-ISI-24-01 AND ANO2-ISI-24-01

ALTERNATIVE REQUESTS ANO1-ISI-24-01 AND ANO2-ISI-24-01

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 1, Category B-D, full penetration welded nozzles in vessels
	Class 2, Category C-A, pressure-retaining welds in pressure vessels
	Class 2, Category C-B (Pressure Retaining Nozzle Welds in Pressure Vessels, Section XI, Division 1)
Item Numbers:	B2.31 – Steam Generator (Primary side), head welds, circumferential
	B2.40 – Steam Generators (primary side), tube sheet-to-head weld
	B3.130 – Steam Generators (primary side), nozzle-to-vessel welds
	C1.20 - Head Circumferential Welds
	C1.30 – Tube sheet-to-shell weld
	C2.21 - Nozzle-to-shell (nozzle-to-head or nozzle-to-nozzle) welds
	C2.22 - Nozzle inside radius sections

ANO Unit 1 (ANO-1)			
ASME Category	**ASME Item No.**	**Component ID**	**Component Description**
B-B	B2.40	03-103	Tube sheet-to-head weld
B-B	B2.40	03-109	Tube sheet-to-head weld
B-B	B2.40	04-103	Tube sheet-to-head weld
B-B	B2.40	04-109	Tube sheet-to-head weld
B-B	B2.31	03-102	Head weld circumferential
B-B	B2.31	03-110	Head weld circumferential
B-B	B2.31	04-102	Head weld circumferential
B-B	B2.31	04-110	Head weld circumferential
C-A	C1.30	03-104	Tube sheet-to-shell welds
C-A	C1.30	03-108	Tube sheet-to-shell welds
C-A	C1.30	04-104	Tube sheet-to-shell welds
C-A	C1.30	04-108	Tube sheet-to-shell welds
C-B	C2.21	03-117	Nozzle-to-shell welds Main Steam (MS)
C-B	C2.21	03-118	Nozzle-to-shell welds (MS)
C-B	C2.21	04-117	Nozzle-to-shell welds (MS)
C-B	C2.21	04-118	Nozzle-to-shell welds (MS)
C-B	C2.22	03-115IR	Nozzle inside radius sections (MS)
C-B	C2.22	03-116IR	Nozzle inside radius sections (MS)
C-B	C2.22	04-115IR	Nozzle inside radius sections (MS)
C-B	C2.22	04-116IR	Nozzle inside radius sections (MS)

ANO Unit 2 (ANO-2)			
ASME Category	**ASME Item No.**	**Component ID**	**Component Description**
B-B	B2.40	03-004	Tube sheet-to-head welds
B-B	B2.40	04-004	Tube sheet-to-head welds
B-D	B3.130	03-005	Nozzle-to-vessel welds
B-D	B3.130	04-005	Nozzle-to-vessel welds
B-D	B3.130	03-006	Nozzle-to-vessel welds
B-D	B3.130	04-006	Nozzle-to-vessel welds
B-D	B3.130	03-007	Nozzle-to-vessel welds
B-D	B3.130	04-007	Nozzle-to-vessel welds
C-A	C1.20	03-001	Head circumferential welds
C-A	C1.20	04-001	Head circumferential welds
C-A	C1.30	03-003	Tube sheet-to-shell welds
C-A	C1.30	04-003	Tube sheet-to-shell welds
C-B	C2.21	03-002	Nozzle-to-shell welds Feedwater (FW)
C-B	C2.21	04-002	Nozzle-to-shell welds (FW)
C-B	C2.22	03-002IR	Nozzle inside radius sections (FW)
C-B	C2.22	04-002IR	Nozzle inside radius sections (FW)

APPLICABLE CODE EDITION AND ADDENDA:

ANO-1

The fifth 10-year in-service inspection (ISI) interval Code of record for ANO-1 is the 2007 Edition of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI through the 2008 Addenda, "Rules for Inservice Inspection of Nuclear Power Plant Components."

ANO-2

The fifth 10-year ISI interval Code of record for ANO-2 is the 2007 Edition of the ASME B&PV Code, Section XI through the 2008 Addenda, "Rules for Inservice Inspection of Nuclear Power Plant Components."

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. B2.31 Volumetric of all nozzles during the first Section XI inspection interval and one weld per head during successive intervals. The examination areas are shown in Figures IWB-2500-3.

Item No. 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 vessel among the group of vessels performing a similar function. The examination volume is shown in Figure IWB-2500-6.

Item No. B3.130 Volumetric examination of all nozzles during each Section XI inspection interval. The examination areas for are shown in Figures IWB-2500-7(a), (b), (c) and (d).

Item No. C1.20 Volumetric examination of essentially 100% of the weld length of the head-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-2500-1.

Item No. C1.30 Volumetric examination of essentially 100% of the weld length of the tube sheet-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 nozzles 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).

REASON FOR REQUEST:

The Electric Power Research Institute (EPRI) performed assessments in References [9.1] and [9.2] of the bases for the ASME Code, Section XI examination requirements specified for the above listed ASME Code, Section XI, Division 1 examination categories for SG welds and components. The assessments include a survey of inspection results from 74 domestic and international nuclear units and flaw tolerance evaluations using probabilistic fracture mechanics (PFM) and deterministic fracture mechanics (DFM). The Reference [9.1] and [9.2] reports concluded that the current ASME Code, Section XI ISI examinations can be deferred for some time with no impact to plant safety. Based on the conclusions of the two EPRI reports supplemented by plant-specific evaluations contained herein, Entergy is requesting an ISI examination deferral for the subject welds. The Reference [9.1] and [9.2] reports were developed consistent with the recommendations provided in EPRI's White Paper on suggested content for PFM submittals [9.12] and NRC Regulatory Guide 1.245 for PFM submittals and associated technical basis [9.13, 9.14].

PROPOSED ALTERNATIVE AND BASIS FOR USE:

ANO-1

For ANO-1, Entergy is requesting an inspection alternative to the examination requirements of ASME Code, 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	B2.31	Steam generators (primary side), head welds, circumferential
B-B	B2.40	Steam generators (primary side), tube sheet-to-head weld
C-A	C1.30	Tube sheet-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

In 2005 (third period of the third inspection, interval) the ANO-1 SGs were replaced. The new SG welds and components received the required pre-service inspection (PSI) examinations followed by ISI examinations through the second period of the current fifth inspection interval.

The proposed alternative is to defer the ISI examinations for these Item Nos. for the SGs at ANO-1 from the current ASME Code, Section XI, Division 1 10-year requirement to the end of currently licensed operating life, which is currently scheduled to end on May 20, 2034. This equates to an extension of 16 years, 11 months, 20 days from the end of the fourth ISI interval (May 30, 2017) at which time all ASME Code, Section XI, Division 1 requirements were satisfied.

ANO-2

For ANO-2, Entergy is requesting an inspection alternative to the examination requirements of ASME Code, 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), tube sheet-to-head weld
B-D	B3.130	Steam generators (primary side), nozzle-to-vessel welds
C-A	C1.20	Head circumferential weld
C-A	C1.30	Tube sheet-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

In 2000 (first period of the third inspection interval), the ANO-2 SGs were replaced. The new SG welds and components received the required PSI examinations followed by ISI examinations through the first period of the current fifth inspection interval.

The proposed alternative is to defer the ISI examinations for these Item Nos. for the SGs at ANO-2 from the current ASME Code, Section XI, Division 1 10-year requirement to the end of currently licensed operating life, which is currently scheduled to end on July 17, 2038. This equates to an extension of 17 years, 3 months, 23 days from the end of the fourth ISI interval (March 25, 2021) at which time all ASME Code, Section XI, Division 1 requirements were satisfied.

Technical Basis

A summary of the key aspects of the technical basis for this request is summarized below. The applicability of the technical basis to ANO-1 and 2 is shown in Attachments 1 and 2.

Applicability of the Degradation Mechanism Evaluation in References [9.1] and [9.2] to ANO-1 and 2

An evaluation of degradation mechanisms that could potentially impact the reliability of the SG welds and components was performed in References [9.1] and [9.2]. 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 known active degradation mechanisms identified that significantly affect the long-term structural integrity of the SG welds and components covered in this request. This observation was acknowledged by the NRC in Section 3.8, page 6, second paragraph of the Reference [9.16] Safety Evaluation (SE) for Vogtle Units 1 and 2 and Section 2.0, page 3, second paragraph of the Reference [9.18] SE for Millstone Unit 2. As shown in Attachments 1 and 2, the materials and operating conditions for the plants considered in this Request for Alternative are like those in the References [9.1] and [9.2] and therefore, the conclusions of these reports apply to the ANO-1 and 2. The fatigue-related mechanisms were considered in the PFM and DFM evaluations in References [9.1] and [9.2].

As part of the technical basis in References [9.1] and [9.2], a comprehensive industry survey involving 74 Pressurized Water Reactor (PWR) units was conducted to determine the degradation history of these components. The survey reviewed examination results from the start of plant operation. Most of these plants have operated for over 30 years and in some cases over 40 years. The results showed that no examinations identified any unknown degradation mechanisms (i.e., mechanisms other than those listed above). Based on this exhaustive industry survey, it is concluded that although the emergence of an unknown degradation mechanism cannot be completely ruled out, the possibility of the occurrence of such an unknown degradation mechanism is highly unlikely.

Applicability of the Stress Analysis in References [9.1] and [9.2] to ANO-1 and 2

Finite element analyses (FEA) were performed in References [9.1] and [9.2] to determine the stresses in the SG welds and components covered in this request. The finite element models used in References [9.1] and [9.2] are consistent with the configurations of ANO-1 and 2 and therefore no new FEA model is required for the stress analysis of these plants. The analysis in References [9.1] and [9.2] was performed using representative 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 ANO-1 and 2 is demonstrated in Attachments 1 and 2 and confirms that all plant-specific requirements are met. In particular, the key geometric parameters used in the Reference [9.1] and [9.2] stress analyses are compared to those of ANO-1 and 2 in Tables 1 and 2:

Table 1
SG Vessel Dimensions

Plant	Primary Lower Head Inside Diameter (ID) (in)	Primary Lower Head Thickness (in)	Primary Lower Head Ri/t	Secondary Upper Shell ID (in)	Secondary Upper Shell Thickness (in)	Secondary Upper Shell Ri/t
EPRI Report (Table 4-2 of [9.2])	155.33	6.94	11.2	230.87	4.91	23.5
ANO-1	118.6(2)	8.2(2)	7.23	139.95(1)	5.4(1)	13.0
ANO-2	157.42(5)	6.94(3)	11.34	230.87(4)	4.79(4)	24.1

Notes:
1. Reference [9.21]
2. Reference [9.22]
3. Reference [9.23]
4. Reference [9.24]
5. Reference [9.33]

Table 2
SG Nozzle Dimensions

Plant	FW Nozzle ID (in)	FW Nozzle Thickness (in)	FW Nozzle R_i/t	MS Nozzle ID (in)	MS Nozzle Thickness (in)	MS Nozzle R_i/t
EPRI Report (Figures 4-9 and 4-10 of [9.1])	16.5	6	1.38	22.25	4.53	2.46
ANO-1	N/A(1)	N/A(1)	N/A(1)	21.75(3)	5.31(4)	2.05
ANO-2	18.0(2)	7.25(2)	1.24	30.8	8.11	1.90

Notes:
1. Nozzle has no C2.21 or C2.22 components.
2. Reference [9.30]
3. Reference [9.34]
4. Reference [9.35]. This thickness value is for the shell which (per Reference [9.36]) is like the MS nozzle thickness.

As discussed in Sections 4.3.3 and 4.6 of Reference [9.1] and noted by the NRC in Section 3.8.3.1, page 9, third paragraph of the SE for Vogtle [9.16], the dominant stress is the pressure stress. Therefore, the variation in the R_i/t ratio determined in Tables 1 and 2 can be used to scale up the stresses of the Reference [9.1] and [9.2] reports to obtain the plant-specific stresses for each unit and component. From Tables 1 and 2, the stress ratio (R_i/t) of ANO-1 and ANO-2 relative to that used in the EPRI report) are as follows.

Primary lower head: ANO-1 (7.23/11.2) = 0.65 and ANO-2 (11.34/11.2) = 1.01 (applicable to primary side welds but conservatively assumed applicable to the rest of the SG welds)

Secondary upper shell: ANO-1 (13/23.5) = 0.55 and ANO-2 (24.1/23.5) = 1.03 (applicable to FW and MS nozzle-to-shell welds)

FW nozzle: ANO-2 (1.24/1.38) = 0.90 (applicable to FW inside radius sections) (Note: N/A for ANO-1 per Table 2)

MS nozzle: ANO-1 (2.05/2.46) = 0.83 and ANO-2 (1.90/2.46) = 0.92 (applicable to MS inside radius sections)

In the selection of the transients in Section 5 of References [9.1] and [9.2] and the subsequent stress analyses in Section 7, test conditions beyond a system leakage test were not considered since pressure tests at ANO-1 and 2 are performed at normal operating conditions. No hydrostatic testing had been performed at ANO-1 and 2 since the units went into operation.

In Reference [9.2], clad residual stress was not considered for the primary side welds. In a previous NRC Request for Additional Information (RAI) (Reference [9.19], RAI 3c), the NRC raised this issue. In response to the RAI (Reference [9.20], RAI Response 3.c), an evaluation was performed which showed that the clad residual stress has no significant impact on the conclusions of Reference [9.2] and this was found acceptable by the NRC in Section 5.3 of Reference [9.18].

Applicability of the Flaw Tolerance Evaluation in References [9.1] and [9.2] to ANO-1 and 2

Flaw tolerance evaluations were performed in References [9.1] and [9.2] consisting of PFM evaluations and confirmatory DFM evaluations. The results of the PFM analyses indicate that, after a PSI followed by subsequent ISI, the NRC's safety goal of 1.0E-06 failures per year is met.

The PFM analysis in Reference [9.1] was performed using the **Pr**obabilistic **O**pti**M**ization of **I**n**S**p**E**ction (**PROMISE**) Version 1.0 software, developed by Structural Integrity Associates. As part of the NRC's review of Southern Nuclear's alternative request, the NRC performed an audit of the **PROMISE** Version 1.0 software as discussed in the NRC's audit plan Reference [9.37]. The PFM analysis in Reference [9.2] was performed using the **PROMISE** Version 2.0 software which has not been audited by the NRC. The only technical difference between the two versions is that in **PROMISE** Version 1.0, the user-specified examination coverage is applied to all inspections, whereas in **PROMISE** Version 2.0, the examination coverage can be specified by the user uniquely for each inspection. In both Versions 1.0 and 2.0, the software assumes 100% coverage for the PSI examination.

In Section 8.2.2.2 of Reference [9.1] and Section 8.3.2.2 of Reference [9.2], 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 for Vogtle in Reference [9.14], 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 in Reference [9.2] 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 1.0E-06 per year. A comparison of the PSI/ISI scenarios used in the sensitivity studies performed in References [9.1] and [9.2] to those at ANO-1 and 2 is provided below. Note that the assumption below of a 30-year ISI deferral is conservative compared to the end of currently licensed operating life for each plant.

<u>ANO-1</u>

For the ANO-1 replacement SGs installed in 2005 (third period of the third inspection interval), PSI examinations have been performed followed by ISI examinations in the one completed 10-year interval (fourth interval) following SG replacement. The PSI/ISI scenario considered is therefore PSI plus one 10-year ISI examinations to be followed by two 30-year ISI deferrals (PSI+10+40+70).

ANO-2

For the ANO-2 replacement SGs installed in 2000 (first period of the third inspection interval), PSI examinations have been performed followed by ISI examinations in the two completed 10-year intervals (third and fourth intervals) following SG replacement. The PSI/ISI scenario considered is therefore PSI plus two 10-year ISI examinations to be followed by a 30-year ISI deferral (PSI+10+20+50).

Limiting PSI/ISI Scenario

From Reference [9.1], the limiting component for Item Nos. C2.21 and C2.22 is the FW nozzle. However, there are no Item No. C2.21 and C2.22 components for the ANO-1 Babcock & Wilcox (B&W) SG FW nozzle configuration; therefore, the MS nozzles will be considered at ANO-1. Five separate evaluations are performed with the following limiting PSI/ISI scenarios:

1. ANO-1 MS nozzle inside radius sections (PSI+10+40+70)
2. ANO-1 MS nozzle-to-shell welds PSI+10+40+70)
3. ANO-2 FW nozzle inside radius sections (PSI+10+20+50)
4. ANO-2 FW nozzle to-shell welds (PSI+10+20+50)
5. The remainder of the SG welds (applicable to both ANO-1 and ANO-2) (PSI+10+40+70)

The above limiting PSI/ISI scenarios for ANO-1 and 2 were not specifically considered in the Reference [9.1] and [9.2] PFM evaluations in combination with key variables, as evaluated by the NRC in Section 4.0 (page 6) of the Reference [9.16] SE. Therefore, the following additional plant-specific evaluations were performed with the limiting PSI/ISI scenarios shown above.

ANO-1 MS Nozzle Inside Radius Section

From Reference [9.1], the critical Case ID for the MS nozzle inside radius section is SGB-P1N. An evaluation like that shown in Table 8-28 of Reference [9.1] was performed for this location assuming a nozzle flaw density of 0.1, a fracture toughness of 200 kilo pound per square inch square root inch (ksi√in) and a standard deviation 5 ksi√in as described by the NRC in Reference [9.13]. A relatively high stress multiplier of 2.35 was applied to get close to the acceptance criteria. The results of the evaluation, using **PROMISE** Version 1.0, are summarized in Table 3 and show that after 80 years of plant operation the probabilities of rupture and leakage are well below the acceptance criterion of 1.0E-06.

Table 3
Sensitivity to Combined Effects of Fracture Toughness, Stress, and Nozzle Flaw Density for 80 Years for B&W MS Nozzle Inside Radius Section (Case ID SGB-P1N from Reference [9.1])

Time (yr)	Probability per Year for Combined Case K_{IC} = 200 ksi√in. SD = 5 ksi√in. Stress Multiplier = 2.35 Nozzle Flaw Density = 0.1 PSI+10+40+70	
	Rupture	Leak
10	1.00E-09	1.00E-09
20	5.00E-10	5.00E-10
30	2.83E-08	2.00E-09
40	3.26E-07	1.61E-07
50	2.61E-07	1.30E-07
60	2.20E-07	1.12E-07
70	1.92E-07	1.01E-07
80	1.68E-07	8.88E-08

ANO-1 MS Nozzle-to-Shell Weld

For the MS nozzle-to-shell weld, Table 8-15 of Reference [9.1] 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 [9.1]. A fracture toughness of 200 ksi√in and standard deviation of 5 ksi√in were also used. A relatively high stress multiplier of 1.95 was applied. The results of the evaluation, using **PROMISE** Version 1.0, are summarized in Table 4 and show that after 80 years of plant operation the probabilities of rupture and leakage are well below the acceptance criterion of 1.0E-06. The results indicate that a much higher stress multiplier than 1.95 could have been used and the acceptance criteria would still be met.

Table 4
Sensitivity to Combined Effects of Fracture Toughness, Stress, and Nozzle Flaw Density for 80 Years for the B&W MS Nozzle-to-Shell Weld Case ID SGB-P3A from Reference [9.1])

Time (year)	Probability per Year for Combined Case K_{IC} = 200 ksi√in. SD = 5 ksi√in. Stress Multiplier = 1.95 Nozzle Flaw Density = 1 PSI+10+40+70	
	Rupture	Leak
10	1.00E-08	1.00E-08
20	5.00E-09	5.00E-09
30	2.00E-08	3.33E-09
40	7.20E-07	2.50E-09
50	5.80E-07	2.00E-09
60	5.00E-07	1.67E-09
70	4.80E-07	1.43E-09
80	4.20E-07	1.25E-09

ANO-2 FW Nozzle Inside Radius Section

From Reference [9.1], the critical location for the inside radius section is FW nozzle Case ID FEW-P1N. An evaluation like that shown in Table 8-28 of Reference [9.1] was performed for this location assuming a nozzle flaw density of 0.1, a fracture toughness of 200 ksi√in and a standard deviation 5 ksi√in as recommended by the NRC in Reference [9.14]. A stress multiplier of 1.75 was applied. 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 5 and show that after 80 years of plant operation the probabilities of rupture and leakage are below the acceptance criterion of 1.0E-06.

Table 5
Sensitivity to Combined Effects of Fracture Toughness, Stress, and Nozzle Flaw Density for 80 Years for Westinghouse (bounds CE design) FW Nozzle Inside Radius Section (Case ID FEW-P1N from Reference [9.1])

Time (yr)	Probability per Year for Combined Case K_{IC} = 200 ksi√in. SD = 5 ksi√in. Stress Multiplier = 1.75 Nozzle Flaw Density = 0.1 PSI+10+20+50	
	Rupture	Leak
10	3.97E-07	1.58E-07
20	2.41E-07	9.80E-08
30	1.60E-07	6.57E-08
40	1.22E-07	4.98E-08
50	1.06E-07	4.26E-08
60	8.85E-08	3.55E-08
70	7.60E-08	3.04E-08
80	6.65E-08	2.66E-08

ANO-2 FW Nozzle-to-Shell Weld

For the FW nozzle-to-shell weld, Table 8-16 of Reference [9.1] indicates that the critical Case ID is FEW-P3A. For the evaluation, a nozzle flaw density of 1 flaw per nozzle was assumed. A fracture roughness of 200 ksi√in and standard deviation 5 ksi√in were also used. A stress multiplier of 1.45 was applied such that probability of rupture or probability of leakage are close to the acceptance criteria after 80 years. The results of the evaluation, using **PROMISE** Version 1.0, are summarized in Table 6 and show that after 80 years of plant operation the probabilities of rupture and leakage are below the acceptance criterion of 1.0E-06.

Table 6
Sensitivity to Combined Effects of Fracture Toughness, Stress, and Nozzle Flaw Density for 80 Years for Westinghouse (bounds CE design) FW Nozzle-to-Shell Weld (Case ID FEW-P3A from Reference [9.1])

Time (year)	Probability per Year for Combined Case K_{IC} = 200 ksi√in. SD = 5 ksi√in. Stress Multiplier = 1.45 Nozzle Flaw Density = 1 PSI+10+20+50	
	Rupture	Leak
10	1.00E-08	1.00E-08
20	5.00E-09	1.00E-08
30	3.33E-09	6.67E-09
40	2.50E-09	3.75E-08
50	2.00E-09	7.48E-07
60	1.67E-09	6.30E-07
70	1.43E-09	5.63E-07
80	1.25E-09	5.61E-07

Remainder of ANO-1 and 2 SG Welds

For the remaining SG welds, Table 8-32 of Reference [9.2] indicates that the critical Case ID is SGPTH-P4A. This case was evaluated for the limiting inspection scenario of PSI+10+40+70, a flaw density of 1.0 flaw per weld, a fracture toughness of 200 ksi√in and a standard deviation 5 ksi√in. A relatively high stress multiplier of 1.8 was applied. The results of the evaluation, using **PROMISE** Version 2.0, are summarized in Table 7 and show that after 80 years of plant operation, the probabilities of rupture and leakage are well below the acceptance criterion of 1.0E-06. The results indicate that a much higher stress multiplier than 2.1 could have been used and the acceptance criteria would still be met.

Table 7
Sensitivity to Combined Effects of Fracture Toughness, Stress, and Nozzle Flaw Density for 80 Years for the Remaining SG Welds (CE or B&W)
(Case ID SGPTH-P4A from Reference [9.2])

Time (year)	Probability per Year for Combined Case K_{IC} = 200 ksi√in. SD = 5 ksi√in. Stress Multiplier = 1.8 Nozzle Flaw Density = 1 PSI+10+40+70	
	Rupture	Leak
10	1.00E-08	1.00E-08
20	5.00E-09	5.00E-09
30	2.33E-08	3.33E-09
40	2.33E-07	2.50E-09
50	1.86E-07	2.00E-09
60	1.70E-07	1.67E-09
70	1.71E-07	1.43E-09
80	1.50E-07	1.25E-09

The plant-specific PFM evaluation presented above for ANO-1 and 2 indicates that with conservative inputs of the critical parameters, the probabilities of rupture and leakage are well below the acceptance criterion of 1.0xE-06 failures per year. The stress multipliers applied Tables 3 through 7 are greater than the plant specific stress ratios determined previously from the geometrical data in Tables 1 and 2 and therefore the stresses and fracture mechanics evaluations in the References [9.1] and [9.2] EPRI reports are conservative in application to ANO-1 and 2. It should also be noted that the evaluation incorporates conservative assumptions about the PSI/ISI scenarios. Furthermore, the evaluation was performed for 30 years, which is longer than the deferral being sought by Entergy in this Request for Alternative.

An evaluation was performed to show acceptability of the low K_{IC} values at the beginning and ending of the heat up/cooldown transient for the FW and MS nozzles to address Item No. 2.e.iii during the NRC audit of **PROMISE** [9.25]. The evaluation was performed using an RT_{NDT} value of 60°F, the maximum allowed by Branch Technical Position (BTP) 5-3 [9.26]. The RT_{NDT} value of 60°F is consistent with the limiting value assumed in Attachment 2 for the SG materials at ANO-2. The evaluation showed acceptable results for the limiting Case IDs from the Reference [9.1] EPRI report. This was found acceptable by the NRC [9.27]. A similar evaluation was performed for the remainder of the SG welds in Reference [9.28] using the limiting Case ID from the Reference [9.2] EPRI report to address NRC RAI-6 in Reference [9.29]. In this evaluation, the limiting RT_{NDT} value of 60°F was used and acceptable results were also obtained.

The PFM evaluations documented in References [9.1] and [9.2] and the plant-specific evaluations above 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. In the case of ANO-1 and 2, ASME Code, Section V procedures may be used for at least some Class 2 components (e.g., the ANO-1 MS nozzle inside radius section examinations and the ANO-2 FW nozzle inside radius section examinations). Based on the observations made by the NRC in Section 3.8.8.2, page 21 of the Vogtle SE [9.16], the use of the ASME Code, Section XI, Appendix VIII based POD curve for inspections based on ASME Code, Section V procedures would have minimal impact of the PFM results since the POD curve is not one of the parameters that significantly affect the PFM results.

The DFM evaluations in Table 8-31 of Reference [9.1] and Table 8-3 of Reference [9.2] provide verification of the above PFM results for ANO-1 and 2 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

As described in Section 8.2.4.1.1 of Reference [9.1] and Section 8.3.4.1 of Reference [9.2], PSI examination refers to the collective examinations required by ASME Code, Section III during fabrication and any ASME Code, Section XI examinations performed prior to service. The Section III fabrication examinations required for these components were robust and any Section XI preservice examinations further contributed to thorough initial examinations.

ANO-1

Inspection history for ANO-1 (including examinations performed to date, examination findings, inspection coverage, and relief requests) is presented in Attachment 1. As shown in the attachment, one weld/component has limited examination coverage of 73.75%. Examination coverage greater than 50% is acceptable per Section 3.8.7 of the Vogtle SE [9.16]. As shown in Attachment 1, no flaws that exceeded the ASME Code, Section XI acceptance standards were identified during any examinations.

ANO-2

Inspection history for ANO-2 (including examinations performed to date, examination findings, inspection coverage, and relief requests) is presented in Attachment 2. As shown in the attachment, one weld/component has limited exam coverage of 69.60%. Examination coverage greater than 50% is acceptable per Section 3.8.7 of the Vogtle SE [9.16]. 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

It is concluded that the SG pressure-retaining welds and full penetration welded nozzles are very flaw tolerant. PFM and DFM evaluations performed as part of the technical basis reports [9.1] and [9.2], supplemented by plant-specific evaluations performed as part of this Request for Alternative, demonstrate that using conservative PSI/ISI inspection scenarios for all plants, the NRC safety goal of 10^{-6} failures per reactor year is met with considerable margins. Plant-specific applicability of the technical basis to ANO-1 and 2 is demonstrated in Attachments 1 and 2. The requested ISI deferrals provide an acceptable level of quality and safety in lieu of the current ASME Code, 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. Attachments 1 and 2 show the examination history for the SG welds examined in the two most recent 10-year inspection intervals.

In addition to the required PSI examinations for these SG welds and components, ANO-1 and 2 have performed multiple ISI examinations through the current 10-year inspection interval at each plant.

No flaws that exceeded the ASME Code, Section XI acceptance standards were identified during any examinations, as shown in Attachments 1 and 2.

Finally, as discussed in Reference [9.3], 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 have been confirmed to be free of defects.

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

DURATION OF PROPOSED ALTERNATIVE:

ANO-1

The proposed alternative is requested for the remainder of the current fifth inspection interval and through the end of currently licensed operating life, which is currently scheduled to end on May 20, 2034.

ANO-2

The proposed alternative is requested for the remainder of the current fifth inspection interval and through the end of currently licensed operating life, which is currently scheduled to end on July 17, 2038.

PRECEDENTS:

The following previous submittal has been made by Southern Nuclear to provide relief from the ASME Code, Section XI Examination Category C-B (Item Nos. C2.21 and C2.22) surface and volumetric examinations based on the Reference [9.1] technical basis report:

- Letter from C. A. Gayheart (Southern Nuclear) to the NRC, 'Vogtle Electric Generating Plant, Units 1 & 2 Proposed Inservice Inspection Alternative VEGP-ISI-ALT-04-04 Version 2.0," ADAMS Accession No. ML20253A311. dated September 9, 2020, [9.15].

The NRC issued a safety evaluation of the Southern Nuclear request for alternative on January 11, 2021.

- Letter from Michael T. Markley (NRC) to Cheryl A. Gayheart (Southern Nuclear), "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)," ADAMS Accession No. ML20352A155, dated January 11, 2021, [9.16].

The following previous submittal has been made by Dominion Energy to provide relief from the ASME Section XI Examination Category B-B (Item No. B2.40) and Category C-A (Item Nos. C1.10, C1.20 and C1.30) surface and volumetric examinations based on the Reference [9.2] technical basis report:

- Letter from Mark D. Sartain (Dominion Energy) to the NRC, "Dominion Energy Nuclear Connecticut, Inc. Millstone Power Station Unit 2 Alternative Request RR-05-06 – Inspection Interval Extension for Steam Generator Pressure-Retaining Welds and Full-Penetration Welded Nozzles," ADAMS Accession No. ML20198M682, dated July 15, 2020 [9.17].

The NRC issued a safety evaluation of the Dominion Energy request for alternative on July 16, 2021.

- 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)," ADAMS Accession No. ML21167A355, dated July 16, 2021 [9.18].

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 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),” ADAMS Accession No. ML003730922, dated July 24, 2000.

- Letter from R. L. Emch (NRC) to J. B. Beasley, Jr. (Sothern Nuclear Operating Company), “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),” ADAMS Accession No. ML011640178, dated June 20, 2001.

- Letter from T. H. Boyce (NRC) to C. L. Burton (Carolina Power & Light), “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),” ADAMS Accession No. ML093561419, dated January 7, 2010.

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

- Letter from R. J. Pascarelli (NRC) to E. D. Halpin (Pacific Gas & Electric), “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),” ADAMS Accession No. ML15337A021, dated December 8, 2015.

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

- Based on studies presented in Reference [9.4], the NRC approved extending PWR reactor vessel nozzle-to-shell welds from 10 to 20 years in Reference [9.5].

- Based on work performed in Boiling Water Reactor Vessel and Internals Program (BWRVIP)-108 [9.6] and BWRVIP-241 [9.8], the NRC approved the reduction of Boiling Water Reactor (BWR) vessel FW 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 [9.7] and [9.9]. The work performed in BWRVIP-108 and BWRVIP-241 provided the technical basis for ASME Code Case N-702 [9.10], which has been conditionally approved by the NRC in Revision 19 of Regulatory Guide 1.147 [9.11].

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
PDI	Probability of detection
PFM	Probabilistic fracture mechanics
PSI	Preservice inspection
PWR	Pressurized Water Reactor
SCC	Stress corrosion cracking
SG	Steam Generator
WEC	Westinghouse Electric Company

REFERENCES:

9.1 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.

9.2 Technical Bases for Inspection Requirements for PWR Steam Generator Class 1 Nozzle-to-Vessel Welds and Class 1 and Class 2 Vessel Head, Shell, Tube sheet-to-Head and Tube sheet-to-Shell Welds. EPRI, Palo Alto, CA: 2019. 3002015906.

9.3 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.

9.4 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.

9.5 US 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," ADAMS Accession No. ML111600303, July 26, 2011.

9.6 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.

9.7 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)," ADAMS Accession No. ML073600374, dated December 19, 2007.

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9.8 BWRVIP-241: BWR Vessels and Internals Project, Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Shell Welds and Nozzle Blend Radii, EPRI, Palo Alto, CA 2010. 1021005.

9.9 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)," ADAMS Accession Nos. ML13071A240 and ML13071A233, dated April 19, 2013.

9.10 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.

9.11 NRC Regulatory Guide 1.147, Revision 18, "Inservice Inspection Code Case Acceptability, ASME Code Section XI, Division 1," dated March 2017.

9.12 N. Palm (EPRI), BWR Vessel & Internals Project (BWRVIP) Memo No. 2019-016, "White Paper on Suggested Content for PFM Submittals to the NRC," ADAMS Accession No. ML19241A545, February 27, 2019.

9.13 NRC Regulatory Guide 1.245, Revision 0, "Preparing Probabilistic Fracture Mechanics Submittals," January 2022.

9.14 NRC Report NUREG/CR-7278, "Technical Basis for the use of Probabilistic Fracture Mechanics in Regulatory Applications," January 2022.

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

9.16 Letter from Michael T. Markley (NRC) to Cheryl A. Gayheart (Southern Nuclear), "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)," ADAMS Accession No. ML20352A155, dated January 11, 2021.

9.17 Letter from Mark D. Sartain (Dominion Energy) to the NRC, "Dominion Energy Nuclear Connecticut, Inc. Millstone Power Station Unit 2 Alternative Request RR-05-06 – Inspection Interval Extension for Steam Generator Pressure-Retaining Welds and Full-Penetration Welded Nozzles," ADAMS Accession No. ML20198M682, dated July 15, 2020.

9.18 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)," ADAMS Accession No. ML21167A355, dated July 16, 2021.

9.19 Electronic mail Letter from R. Guzman (NRC) to S. Sinha (Dominion Energy Nuclear Connecticut, Inc.), "Millstone Unit 2 – Request for Additional Information – Alternative Request RR-05-06 Inspection Interval Extension for SG Pressure Retaining Welds and Full-Penetration Welded Nozzles (EPID: L-2020-LLR-0097)," ADAMS Accession No. ML21034A576, dated February 3, 2021.

9.20 Letter from G. T. Bischof (Dominion Energy Nuclear Connecticut, Inc.) 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," ADAMS Accession No. ML21081A136, dated March 19, 2021.

9.21 UT Vessel Examination Report No. 1-ISI-UT-11-007 (file *1R23 03-104 UT.pdf*), 10/27/11.

9.22 UT Vessel Examination Report No. 1-ISI-UT-11-014 (file *1R23 03-103 UT.pdf*), 10/27/11.

9.23 Drawing No. M-2001-C6-293, "Arkansas Nuclear One Unit 2 Δ 109 Replacement Steam Generator Primary Side Channel Head Assy," Sheet 3, Revision 0.

9.24 Drawing No. M-2001-C6-204, "Arkansas Nuclear One Unit 2 Δ 109 Replacement Steam Generator Upper Shell Assembly," Sheet 2, Revision 0.

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

9.26 NUREG-0800 - Chapter 5, Branch Technical Position (BTP) 5-3, Revision 2, Fracture Toughness Requirements.

9.27 Letter from J. G. Lamb (NRC) to C. A. Gayheart (Southern Nuclear), "Vogtle Electric Generating Plant, Units 1 and 2 – Audit Report for the PROMISE Version 1.0 Probabilistic Fracture Mechanics Software Code Used in Relief Request VEGP-ISI-ALT-04-04 (EPID L-2020-LLR-0109), ADAMS Accession No. ML20258A002, dated December 10, 2020.

9.28 Letter RS-22-084 from D. T, Gudger (Constellation Energy Generation, LLC) to NRC, “Response to Request for Additional Information - Proposed Alternative for Examinations of Examination Categories B-B, B-D, and C-A Steam Generator Pressure Retaining Welds and Full Penetration Welded Nozzles, ADAMS Accession No. ML22168A005, dated June 17, 2022.

9.29 Electronic mail Letter from J. Wiebe (NRC) to T. Loomis (Constellation Energy Generation, LLC), “Draft RAIs for Requests for Alternatives I4R-17, I4R-23, ISI-05-018, I6R-10 (EPID Nos.: L-2021-LLR-091, L-2021-LLR-092, L-2021-LLR-093, L-2021-LLR-094),” dated May 6, 2022.

9.30 Drawing No. M-2001-C6-245, “Feedwater Nozzle/Thermal Sleeve and Feedwater Ring Assembly,” Sheet 2, Revision N.

9.31 UT Vessel Examination Report No. 1-ISI-UT-11-016 (Summary No. 1-03-117), 10/30/11.

9.32 Drawing No. M-2001-C6-204, “Arkansas Nuclear One Unit 2 Δ 109 Replacement Steam Generator Upper Shell Assembly,” Sheet 2, Revision 0.

9.33 Drawing No. M-2001-C6-200, “Arkansas Nuclear One Unit 2 Δ 109 Replacement Steam Generator Outline,” Sheet 2, Revision 0.

9.34 Drawing No. 5022694E (M1D-212), “ANO-1 EOTSG Steam Outlet Nozzle Elbows,” Revision 3.

9.35 UT Vessel Examination Report No. ISI-UT-08-126 (Summary No. 1-03-118), 11/12/08.

9.36 Entergy Drawing No. MID-295, Sheet 1, “Replacement Steam Generator for Arkansas Plant – Unit 1, Principal Side Weld Map Drawing, Repere Des Soudures oe L’enceinte Principale,” Revision 1.

9.37 Letter from John G. Lamb (NRC) to Cheryl A. Gayfield (Southern Nuclear Operating Co., Inc.), " Vogtle Electric Generating Plant, Units 1 and 2 - Audit Report for the PROMISE Version 1.0 Probabilistic Fracture Mechanics Software used in Relief Request VEGP-ISI-ALT-04-04 (EPID L-2019-LLR-0109)," ADAMS Accession No. ML20128J311, dated December 10, 2020.

ENCLOSURE, ATTACHMENT 1

0CAN062402

PLANT-SPECIFIC APPLICABILITY
ANO-1

Plant-Specific Applicability – ANO-1

Section 9 of References [1-1] and [1-2] provide requirements that must be demonstrated to apply the representative stress and flaw tolerance analyses to a specific plant. Plant-specific evaluation of these requirements for ANO-1 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 ANO-1.

Table 1-1

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

Items No. B2.31 and B2.40 (SG Primary Side Shell Welds)

Category	Requirement from Reference [1-1]	Applicability to ANO-1
General Requirements	The Loss of Power transient (involving auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of a 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.	The ANO-1 Enhanced Once-Through Steam Generators (EOTSG) have not experienced a loss of power transient resulting in unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following a blackout, resulting in thermal shock of any portion of the vessel. (Notes: Attachment 8.1 of Reference [1-6] records 3 occurrences of Transient 12 (Hot, Dry, Depressurized SG Refill) in the ANO-1 EOTSG (on 12/20/08, 2/5/09 and 2/7/09). However, per Attachment 8.2 of Reference [1-6], the actual scenario for each of these 3 occurrences was "After manual RX trip, MFW to OTSG temperature dropped below design limit of 135F. CR-ANO-1-2009-00735 conservatively determined this was a Transient 12 cycle." Therefore, the actual transients experienced do not resemble the "introducing unheated AFW into a hot, dry SG" transient described above. The updated transient counts record no additional occurrences of Transient 12.) (Reference [1-8]

	The materials of the SG vessel head, and tube sheet must be low alloy ferritic steels which conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.	The ANO-1 SG vessel heads and tube sheets are fabricated from SA-508 Class 3A material (per Table 3-3 of Reference [1-3]). The maximum Reference Temperature for Nil Ductility Transition (RT_{NDT}) value for the material is 0°F [1-4] and is therefore bounded by the value used in the EPRI report. SA-508, Class 3A material is a low alloy ferritic steel which conforms to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.
Specific Requirements	The weld configurations must conform to those shown in Figures 1-1 and Figure 1-2 of Reference [1-1].	The ANO-1 tube sheet-to-head weld configuration is shown in Figure 1-2 and shows conformance with Figure 1-2 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 9.4.3 of Reference [1-1].	Per Table 1 in the Enclosure, the ANO-1 SG vessel dimensions are as follows: • SG Lower Head Outside Diameter (OD) = 135.00 inches • SG Upper Shell OD = 150.75 inches The dimensions are within 10% of those specified in Table 9-2 in Section 9.4.3 of Reference [1-1] for Babcock & Wilcox (B&W) plants.
	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.	As shown in Table 1-2, the ANO-1 number of cycles projected to occur over a 60-year operating life are significantly lower than those shown in Table 5-7 of Reference [1-1].

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

Category	Requirement from Reference [1-1]	Applicability to ANO-1
General Requirements	The Loss of Power transient (involving auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of a 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.	The ANO-1 EOTSGs have not experienced a loss of power transient resulting in unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following a blackout, resulting in thermal shock of any portion of the vessel. (Notes: Attachment 8.1 of Reference [1-6] records 3 occurrences of Transient 12 (Hot, Dry, Depressurized SG Refill) in the ANO-1 EOTSG (on 12/20/08, 2/5/09 and 2/7/09). However, per Attachment 8.2 of Reference [1-6], the actual scenario for each of these 3 occurrences was "After manual RX trip, MFW to OTSG temperature dropped below design limit of 135F. CR-ANO-1-2009-00735 conservatively determined this was a Transient 12 cycle." Therefore, the actual transients experienced do not resemble the "introducing unheated AFW into a hot, dry SG" transient described above. The updated transient counts record no additional occurrences of Transient 12.) (Reference [1-8]
	The materials of the SG vessel shell and tube sheet must be low alloy ferritic steels which conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.	The ANO-1 SG vessel shell and tube sheet are fabricated from SA-508 Class 3A material (per Table 3-3 of Reference [1-3]). The maximum RT_{NDT} value for the material is 0°F [1-4] and is therefore bounded by the value used in the EPRI report. SA-508, Class 3A material is a low alloy ferritic steel which conforms to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.
Specific Requirements	The weld configurations must conform to those shown in Figure 1-7 and Figure 1-8 of Reference [1-1].	The ANO-1 weld configuration is shown in Figure 1-3 and conforms to Figure 1-8 of Reference [1-1].

Category	Requirement from Reference [1-1]	Applicability to ANO-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].	Per Table 1 of the Enclosure, the ANO-1 SG vessel dimensions are as follows: • SG Lower Head OD = 135.00 inches • SG Upper Shell OD = 150.75 inches The dimensions are within 10% of those specified in Table 9-3 in Section 9.4.4 of Reference [1-1] for B&W plants.
	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, the ANO-1 number of cycles projected to occur over a 60-year operating life are significantly lower than those shown in Table 5-9 of Reference [1-1].

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

Category	Requirement from Reference [1-2]	Applicability to ANO-1
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].	The ANO-1 MS nozzle-to-shell weld is shown in Figure 1-4 and is representative of the configuration shown in Figure 1-2 of Reference [1-2]. Per Section 4.3.1.3, Item 3 of Reference [1-2], B&W plants (like ANO-1) do not have FW nozzles welded into the SG shells (the nozzle is 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 C2.21 or C2.22 components for the FW nozzle.
	The materials of the SG shell, FW nozzles, and MS nozzles must be low alloy ferritic steels which conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G 2110.	The ANO-1 SG vessel shell and MS nozzles are fabricated from SA-508 Class 3A material (per Table 3-3 of Reference [1-3]). The maximum RT_{NDT} value for the materials is 0°F [1-4] and is therefore bounded by the value used in the EPRI report. SA-508, Class 3A material is a low alloy ferritic steel which 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 [1-2] over a 60-year operating life.	As shown in Table 1-4, the ANO-1 SGs are not projected to experience more than the number of transients shown in Table 5-5 of Reference [1-2] over a 60-year operating life.
SG Feedwater Nozzle	The piping attached to the FW nozzle must be 14-inch to 18-inch Nominal Pipe Size (NPS).	There are no C2.21 or C2.22 components for the FW nozzle.
	The FW nozzle design must have an integrally attached thermal sleeve.	There are no 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 ANO-1 (B&W design).

Category	Requirement from Reference [1-2]	Applicability to ANO-1
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 ANO-1 (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 ANO-1 MS nozzle is 24 inch Inside Diameter (ID) (per Table 4-4 of Reference [1-3]).
	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.	The ANO-1 (B&W) MS nozzles are shown in Figure 1-1 and exit the side of the SG.
	The main steam nozzle shall not significantly protrude into the SG (e.g., see Figure 4-7 of Reference [1-2]) or have a unique nozzle weld configuration (e.g., see Figure 4-6 of Reference [1-2]).	The ANO-1 (B&W) MS nozzles are shown in Figure 1-1 and do not significantly protrude or have a unique nozzle weld configuration.

Table 1-2

ANO-1 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(1)	Number of Cycles for 60 Years from Table 5-7 of Reference [1-1]	ANO-1 60-Year Projection
Heat up / Cooldown	300	192 / 187(2)
Plant Loading / Unloading	5,000	673 / 673(3)
Reactor Trip	360	187(2)

Notes:

1. Table 5-7 of Reference [1-1] also includes allowable transient temperatures and pressures. From previous experience with B&W plants, these values are typically within 2% of the values used in the EPRI report. This is acceptable based on the large margins in the evaluation as demonstrated by the low plant specific stress ratio compared to the maximum allowed stress ratio.

2. Calculated based on values in Attachment 8.1, PDF pages 5 and 6 (of 45) of Reference [1-6]. It has been confirmed that these calculated values bound those obtained by using Reference [1-8].

3. Table 6 of Reference [1-7].

Table 1-3

ANO-1 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(1)	Number of Cycles for 60 Years from Table 5-9 of Reference [1-1]	ANO-1 60-Year Projection
Heat up / Cooldown	300	192 / 187(2)
Plant Loading / Unloading	5000	673 / 673(3)
Reactor Trip	360	187(2)

Notes:

1. Table 5-9 of Reference [1-1] also includes allowable transient temperatures and pressures. From previous experience with B&W plants, these values are typically within 2% of the values used in the EPRI report. This is acceptable based on the large margins in the evaluation as demonstrated by the low plant specific stress ratio compared to the maximum allowed stress ratio.

2. Calculated based on values in Attachment 8.1, PDF pages 5 and 6 (of 45) of Reference [1-6]. It has been confirmed that these calculated values bound those obtained by using Reference [1-8].

3. Table 6 of Reference [1-7].

Table 1-4

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

Transient	Number of Cycles for 60 Years from Table 5-5 of Reference [1-2]	ANO-1 60-Year Projection
Heat up / Cooldown	300	192 / 187(1)
Plant Loading	5000	673(3)
Plant Unloading	5000	673(3)
Loss of Load	360	187(1)(2)
Loss of Power	60	7(4)

Notes:

1. Calculated based on values in Attachment 8.1, PDF pages 5 and 6 (of 45) of Reference [1-6]. It has been confirmed that these calculated values bound those obtained by using Reference [1-8].
2. Loss of Load = Reactor Trip.
3. Table 6 of Reference [1-7].
4. Based on a review of the NRC Licensee Event Report (LER) database for Loss of Offsite Power events at ANO-1. The tally does not factor in the SG replacement and is therefore conservative.

Table 1-5

ANO-1 Inspection History

Item No.	Component ID	Exam Date	Interval/Period/Outage	Exam Results	Coverage	Relief Request
B2.40	03-103	10/23/2011	4^{th} / 2^{nd} / 1R23	No Recordable Indications (NRI)	100%	N/A
B2.40	03-103	5/7/2024	5^{th} / 2^{nd} / 1R31	NRI	100%	N/A
B2.40	03-109	4/8/2010	4^{th} / 1^{st} / 1R22	NRI	>90%	N/A
B2.40	03-109	10/14/2019	5^{th} / 1^{st} / 1R28	NRI	>90%	N/A
B2.31	03-102	10/22/2011	4^{th} / 2^{nd} / 1R23	NRI	73.75%	N/A
B2.31	03-102	5/7/2024	5^{th} / 2^{nd} / 1R31	NRI	76%	N/A
B2.31	03-110	10/29/2011	4^{th} / 2^{nd} / 1R23	NRI	100%	N/A
B2.31	03-110	5/3/2024	5^{th} / 2^{nd} / 1R31	NRI	100%	N/A
C1.30	03-104	10/24/2011	4^{th} / 2^{nd} / 1R23	NRI	96.17%	N/A
C1.30	03-108	4/6/2010	4^{th} / 1^{st} / 1R22	NRI	>90%	N/A
C1.30	03-108	4/12/2018	5^{th} / 1^{st} / 1R27	NRI	99.78%	N/A
C2.21	03-117	10/25/2011	4^{th} / 2^{nd} / 1R23	NRI	92.10%	N/A
C2.21	03-118	11/9/2008	4^{th} / 1^{st} / 1R21	NRI	92.10%	N/A
C2.22	03-115IR	10/26/2011	4^{th} / 2^{nd} / 1R23	NRI	96.76%	N/A
C2.22	03-116IR	4/11/2018	5^{th} / 1^{st} / 1R27	NRI	96.76%	N/A

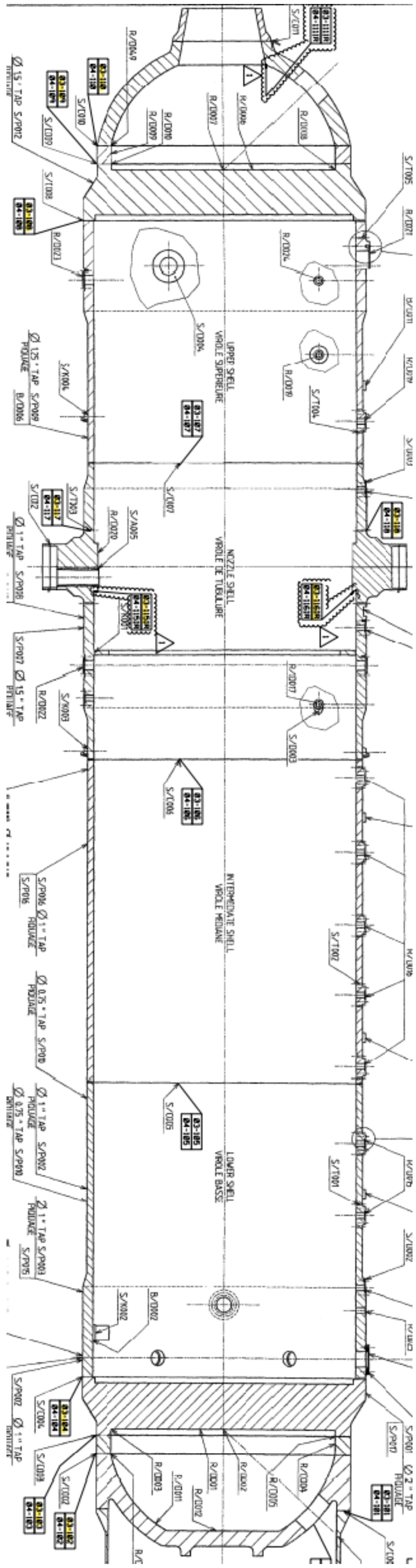

Figure 1-1. ANO-1 Steam Generator Layout [1-5]

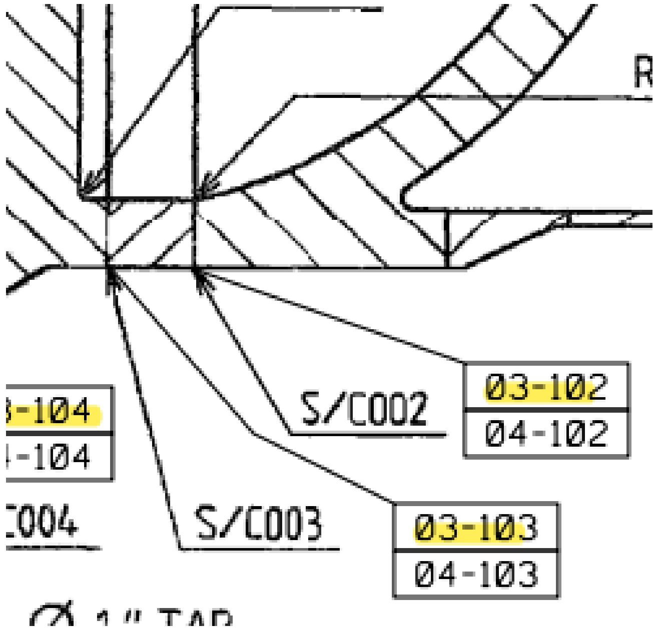

Figure 1-2. ANO-1 Item No. B2.40 Weld Configuration [1-5]

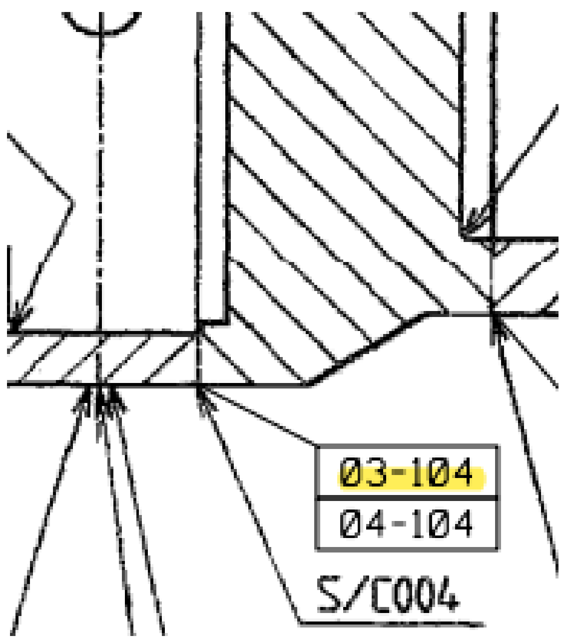

Figure 1-3. ANO-1 Item No. C1.30 Weld Configuration [1-5]

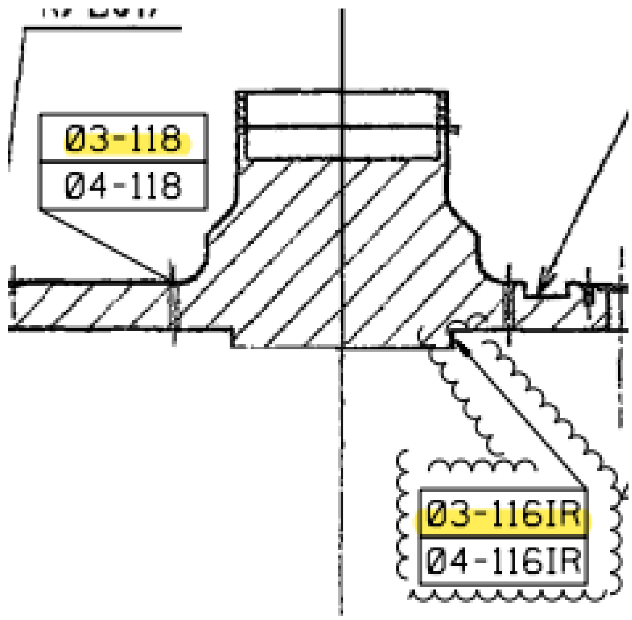

Figure 1-4. ANO-1 Main Steam Nozzle Configuration [1-5]

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, Tube sheet-to-Head and Tube sheet-to-Shell Welds*. EPRI, Palo Alto, CA: 2019. 3002015906.

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.

1-3. Engineering Report No. ER-ANO-2002-1381-000, "ANO-1 Steam Generator Replacement," Revision 0.

1-4. E-mail from Andy Nettles (Entergy) to Scott Chesworth (SI), "Subject: RE: ANO Project Inputs," dated May 11, 2023.

1-5. Drawing No. M1D-295, "Replacement Steam Generator for Arkansas Plant – Unit 1, Principal Side Weld Map Drawing, Repere Des Soudures oe L'Enceinte Principale," Sheet 1, Revision 1.

1-6. Entergy Engineering Report No. CALC-ANO1-SE-17-00001, "ANO Unit 1 Transient Cycle Report for 2016," Revision 00.

1-7. SI Calculation No. 2200654.301, "ANO Units 1 & 2 Plant Loading/Unloading and Pressurizer Insurge/Outsurge Transients," Revision 0.

1-8. Entergy Engineering Report No. CALC-ANO1-SE-18-00002, "ANO Unit 1 Transient Cycle Report for 2017 Through 2022," Revision 00.

ENCLOSURE, ATTACHMENT 2

0CAN062402

PLANT-SPECIFIC APPLICABILITY
ANO-2

Plant-Specific Applicability – ANO-2

Section 9 of References [2-1] and [2-2] provide requirements that must be demonstrated to apply the representative stress and flaw tolerance analyses to a specific plant. Plant-specific evaluation of these requirements for ANO-2 is provided in Table 2-1 and indicates that all plant-specific requirements are met. Therefore, the results and conclusions of the EPRI reports are applicable to ANO-2.

Table 2-1

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

<u>Items No. B2.40 (Steam Generator (SG) Primary Side Shell Welds)</u>

Category	Requirement from Reference [2-1]	Applicability to ANO-2
General Requirements	The Loss of Power transient (involving auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of a 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.	The ANO-2 Replacement Steam Generators (RSGs) have not experienced a loss of power transient resulting in unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following a blackout, resulting in thermal shock of any portion of the vessel.
	The materials of the SG vessel head and tube sheet must be low alloy ferritic steels which conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.	The ANO-2 RSG vessel heads and tube sheet are fabricated from SA-508 Class 3A material (per Table 5.2-3 of Reference [2-3]). The Reference Temperature for Nil Ductility Transition (RT_{NDT}) value for the material is assumed to be 60°F, the maximum value allowed by Branch Technical Position (BTP) 5-3 [2-4] and therefore consistent with that used in the EPRI report. SA-508, Class 3A material is a low alloy ferritic steel which conforms to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.
Specific Requirements	The weld configurations must conform to those shown in Figures 1-1 and Figure 1-2 of Reference [2-1].	The ANO-2 tube sheet-to-head weld configuration is shown in Figure 2-2 and shows conformance with Figure 1-2 of Reference [2-1].

	The SG vessel dimensions must be within 10% of the upper and lower bounds of the values provided in the table in Section 9.4.3 of Reference [2-1].	Using the Inside Diameter (ID) and thickness values from Table 1 of the Enclosure, the ANO-2 SG Outside Diameter (OD) vessel dimensions are as follows: • SG Lower Head OD = 171.30 inches • SG Upper Shell OD = 240.45 inches The dimensions are within 10% of those specified in Table 9-2 in Section 9.4.3 of Reference [2-1] for Combustion Engineering (CE) plants. (Note: The values given in Table 5.5-2 of Reference [2-3] are 171.07 inches and 240.69 inches respectively; these values are extremely close to the values above from Table 1.)
	The component must experience transients and cycles bounded by those shown in Table 5-7 of Reference [2-1] over a 60-year operating life.	As shown in Table 2-2, the ANO-2 number of cycles projected to occur over a 60-year operating life are significantly lower than those shown in Table 5-7 of Reference [2-1].

Item No. B3.130 (SG Primary Inlet/Outlet Nozzles)

Category	Requirement from Reference [2-1]	Applicability to ANO-2
General Requirements	The Loss of Power transient (involving unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of portion of the vessel) is not considered in this evaluation due to its rarity. If such a significant thermal event occurs at a plant, its impact on the K_{IC} value may require more frequent examinations and other plant actions outside the scope of this report's guidance.	The ANO-2 RSGs have not experienced a loss of power transient resulting in unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following a blackout, resulting in thermal shock of any portion of the vessel.
	The materials of the SG vessel head and tube sheet must be low alloy ferritic steels which conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.	The ANO-2 RSG vessel heads and tube sheet are fabricated from SA-508 Class 3A material (per Table 5.2-3 of Reference [2-3]). The RT_{NDT} value for the material is assumed to be 60°F, the maximum value allowed by BTP 5-3 [2-4] and therefore consistent with that used in the EPRI report. SA-508, Class 3A material is a low alloy ferritic steel which conforms to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.
Specific Requirements	The weld configurations must conform to those shown in Figures 1-3 through 1-5 of Reference [2-1].	The ANO-2 weld configurations are shown in Figure 2-3 and conform to those shown in Figures 1-3 through 1-5 of Reference [2-1].
	The piping attached to the primary inlet and outlet nozzles (RCS piping) for the various designs must be within 10% of the values provided in the table in Section 9.4.2 of Reference [2-1].	The piping attached to the ANO-2 primary inlet and outlet nozzles (RCS piping) is 42 inches ID and 30 inches ID, respectively (per Table 5.5-2 of [2-3]). These values are within 10% of the values provided in the table in Section 9.4.2 of Reference [2-1].
	The component must experience transients and cycles bounded by those shown in Table 5-8 of Reference [2-1] over a 60-year operating life.	As shown in Table 2-3, the ANO-2 number of cycles projected to occur over a 60-year operating life are significantly lower than those shown in Table 5-8 of Reference [2-1].

Items No. C1.20 and C1.30 (SG Secondary Side Shell Welds)

Category	Requirement from Reference [2-1]	Applicability to ANO-2
General Requirements	The Loss of Power transient (involving auxiliary feedwater being introduced into a hot SG that has been boiled dry following blackout, resulting in thermal shock of a 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.	ANO-2 has not experienced a loss of power transient resulting in unheated auxiliary feedwater being introduced into a hot SG that has been boiled dry following a blackout, resulting in thermal shock of any portion of the vessel.
	The materials of the SG vessel shell and tube sheet must be low alloy ferritic steels which conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.	The ANO-2 RSG vessel heads and tube sheet are fabricated from SA-508 Class 3A material (per Table 5.2-3 of Reference [2-3]). The RT_{NDT} value for the material is assumed to be 60°F, the maximum value allowed by BTP 5-3 [2-4] and therefore consistent with that used in the EPRI report. SA-508, Class 3A material is a low alloy ferritic steel which conforms to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.

Specific Requirements	The weld configurations must conform to those shown in Figure 1-7 and Figure 1-8 of Reference [2-1].	The ANO-2 weld configurations are shown in Figures 2-4 and 2-5 and conform to Figures 1-7 and 1-8 of Reference [2-1].
	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 [2-1].	Using the ID and thickness values from Table 1 in the Enclosure, the ANO-2 SG OD vessel dimensions are as follows: • SG Lower Head OD = 171.30 inches • SG Upper Shell OD = 240.45 inches The dimensions are within 10% of those specified in Table 9-3 in Section 9.4.4 of Reference [2-1] for CE plants. (Note: The values given in Table 5.5-2 of Reference [2-3] are 171.07 inches and 240.69 inches respectively; these values are extremely close to the values above from Table 1.)
	The component must experience transients and cycles bounded by those shown in Table 5-9 of Reference [2-1] over a 60-year operating life.	As shown in Table 2-4, the ANO-2 number of cycles projected to occur over a 60-year operating life are significantly lower than those shown in Table 5-9 of Reference [2-1].

Items Nos. C2.21 and C2.22 (Main Steam (MS) and Feedwater (FW) Nozzle to Shell Welds and Inside Radius Sections)

Category	Requirement from Reference [2-2]	Applicability to ANO-2
General Requirements	The nozzle-to-shell weld shall be one of the configurations shown in Figure 1-1 or Figure 1-2 of Reference [2-2].	The ANO-2 FW nozzle-to-shell weld is shown in Figure 2-6 and is representative of the configuration shown in Figure 1-2 of Reference [2-2]. The ANO-2 MS nozzle-to-shell weld is shown in Figure 2-7 and is representative of the configuration shown in Figure 1-2 of Reference [2-2]
	The materials of the SG shell, FW nozzles, and MS nozzles must be low alloy ferritic steels which conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.	The ANO-2 RSG shell, FW nozzles, and MS nozzles are fabricated from SA-508 Class 3A material (per Table 5.2-3 of Reference [2-3]). The RT_{NDT} value for the material is assumed to be 60°F, the maximum value allowed by BTP 5-3 [2-4] and therefore consistent with that used in the EPRI report. SA-508, Class 3A material is a low alloy ferritic steel which 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 [2-2] over a 60-year operating life.	As shown in Table 2-5, the ANO-2 SGs are not projected to experience more than the number of transients shown in Table 5-5 of Reference [2-2] over a 60-year operating life.
SG Feedwater Nozzle	The piping attached to the FW nozzle must be 14-inch to 18-inch Nominal Pipe Size (NPS).	The piping attached to the ANO-2 FW nozzle is 18 inches nominal size (per Table 5.5-2 of [2-3]).
	The FW nozzle design must have an integrally attached thermal sleeve.	The ANO-2 FW nozzle design has an integrally attached thermal sleeve per Reference [2-8].
	Auxiliary feedwater nozzles connected directly to the SG are not covered in this evaluation.	N/A for ANO-2 (CE Design).
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 Nominal Pipe Size (NPS).	The piping attached to the ANO-2 (CE) main steam nozzle is 34" ID (per Table 5.5-2 of [2-3]).

Category	Requirement from Reference [2-2]	Applicability to ANO-2
	For Babcock and Wilcox (B&W) SGs, the piping attached to the main steam nozzle must be 22-inch to 26-inch NPS.	N/A for ANO-2 (CE design).
	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.	The ANO-2 (CE) MS nozzle is shown in Figure 2-1 and exits the top dome of the SG.
	The main steam nozzle shall not significantly protrude into the SG (e.g., see Figure 4-7 of Reference [2-2]) or have a unique nozzle weld configuration (e.g., see Figure 4-6 of Reference [2-2]).	The ANO-2 MS nozzle is shown in Figure 2-7 and does not significantly protrude or have a unique nozzle configuration.

Table 2-2

ANO-2 Data for Thermal Transients for Stress Analysis of the PWR SG Primary-Side Head Welds (Comparison to Table 5-7 of Reference [2-1])

Transient(1)	Number of Cycles for 60 Years from Table 5-7 of Reference [2-1]	ANO-2 60-Year Projection
Heat up / Cooldown	300	237 / 237(2)
Plant Loading / Unloading	5000	493 / 493(3)
Reactor Trip	360	156(2)

Notes:

1. Table 5-7 of Reference [2-1] also includes allowable transient temperatures and pressures. From previous experience with Westinghouse plants, these values are with 2% of the values used in the EPRI report. This is acceptable based on the large margins in the evaluation as demonstrated by the low plant specific stress ratio compared to the maximum allowed stress ratio.

2. Calculated based on values on Page 8 (of 46) of Reference [2-11].

3. Table 7 of Reference [2-12].

Table 2-3

ANO-2 Data for Thermal Transients for Stress Analysis of the PWR SG Inlet Nozzle-to-Vessel Welds (Item No. B3.130) (Comparison to Table 5-8 of Reference [2-1])

Transient[(1)]	Number of Cycles for 60 Years from Table 5-7 of Reference [2-1]	ANO-2 60-Year Projection
Heat up / Cooldown	300	237 / 237[(2)]
Plant Loading / Unloading	5000	493 / 493[(3)]
Reactor Trip	360	156[(2)]

Notes:

1. Table 5-7 of Reference [2-1] also includes allowable transient temperatures and pressures. From previous experience with Westinghouse plants, these values are with 2% of the values used in the EPRI report. This is acceptable based on the large margins in the evaluation as demonstrated by the low plant specific stress ratio compared to the maximum allowed stress ratio.

2. Calculated based on values on Page 8 (of 46) of Reference [2-11].

3. Table 7 of Reference [2-12].

Table 2-4

ANO-2 Data for Thermal Transients for Stress Analysis of the PWR SG Secondary-Side Vessel Welds (Comparison to Table 5-9 of Reference [2-1])

Transient(1)	Number of Cycles for 60 Years from Table 5-9 of Reference [2-1]	ANO-2 60-Year Projection
Heat up / Cooldown	300	237 / 237(2)
Plant Loading / Unloading	5000	493 / 493(3)
Reactor Trip	360	156(2)

Notes:

1. Table 5-9 of Reference [2-1] also includes allowable transient temperatures and pressures. From previous experience with Westinghouse plants, these values are with 2% of the values used in the EPRI report. This is acceptable based on the large margins in the evaluation as demonstrated by the low plant specific stress ratio compared to the maximum allowed stress ratio.

2. Calculated based on values on Page 8 (of 46) of Reference [2-11].

3. Table 7 of Reference [2-12].

Table 2-5

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

Transient	Number of Cycles for 60 Years from Table 5-5 Reference [2-2]	ANO-2 60-Year Projection
Heat up / Cooldown	300	237 / 237(1)
Plant Loading	5000	493(3)
Plant Unloading	5000	493(3)
Loss of Load	360	156(1)(2)
Loss of Power	60	9(4)

Notes:

1. Calculated based on values on Page 8 (of 46) of Reference [2-11].
2. Loss of Load = Reactor Trip.
3. Table 7 of Reference [2-12].
4. Based on a review of the NRC Licensee Event Report (LER) database for Loss of Offsite Power events at ANO-2.

Table 2-6

ANO-2 Inspection History

Item No.	Component ID	Exam Date	Interval/Period/Outage	Exam Results	Coverage	Relief Request
B2.40	04-004	10/10/2006	3rd / 2nd / 2R18	No Recordable Indications (NRI)	>90%	N/A
B2.40	04-004	10/17/2015	4th / 2nd / 2R24	NRI	100%	N/A
B3.130	03-005	5/19/2014	4th / 2nd / 2R23	NRI	87%	N/A
B3.130	04-005	10/8/2006	3rd / 2nd / 2R18	NRI	<90%	N/A
B3.130	04-005	10/3/2015	4th / 2nd / 2R24	NRI	88%	N/A
B3.130	03-006	5/19/2014	4th / 2nd / 2R23	NRI	87%	N/A
B3.130	04-006	10/8/2006	3rd / 2nd / 2R18	NRI	<90%	N/A
B3.130	04-006	10/3/2015	4th / 2nd / 2R24	NRI	88%	N/A
B3.130	03-007	5/30/2014	4th / 2nd / 2R23	NRI	94%	N/A
B3.130	04-007	10/10/2006	3rd / 2nd / 2R18	NRI	<90%	N/A
B3.130	04-007	10/10/2015	4th / 2nd / 2R24	NRI	87%	N/A
C1.20	04-001	10/8/2015	4th / 2nd / 2R24	NRI	100%	N/A
C1.30	04-003	10/16/2006	3rd / 2nd / 2R18	NRI	>90%	N/A
C1.30	04-003	10/12/2018	4th / 3rd / 2R26	NRI	98.90%	N/A
C2.21	04-002	10/12/2006	3rd / 2nd / 2R18	NRI	<90%	N/A
C2.21	04-002	10/7/2015	4th / 2nd / 2R24	NRI	69.60%	N/A
C2.22	04-002IR	10/7/2015	4th / 2nd / 2R24	NRI	100%	N/A

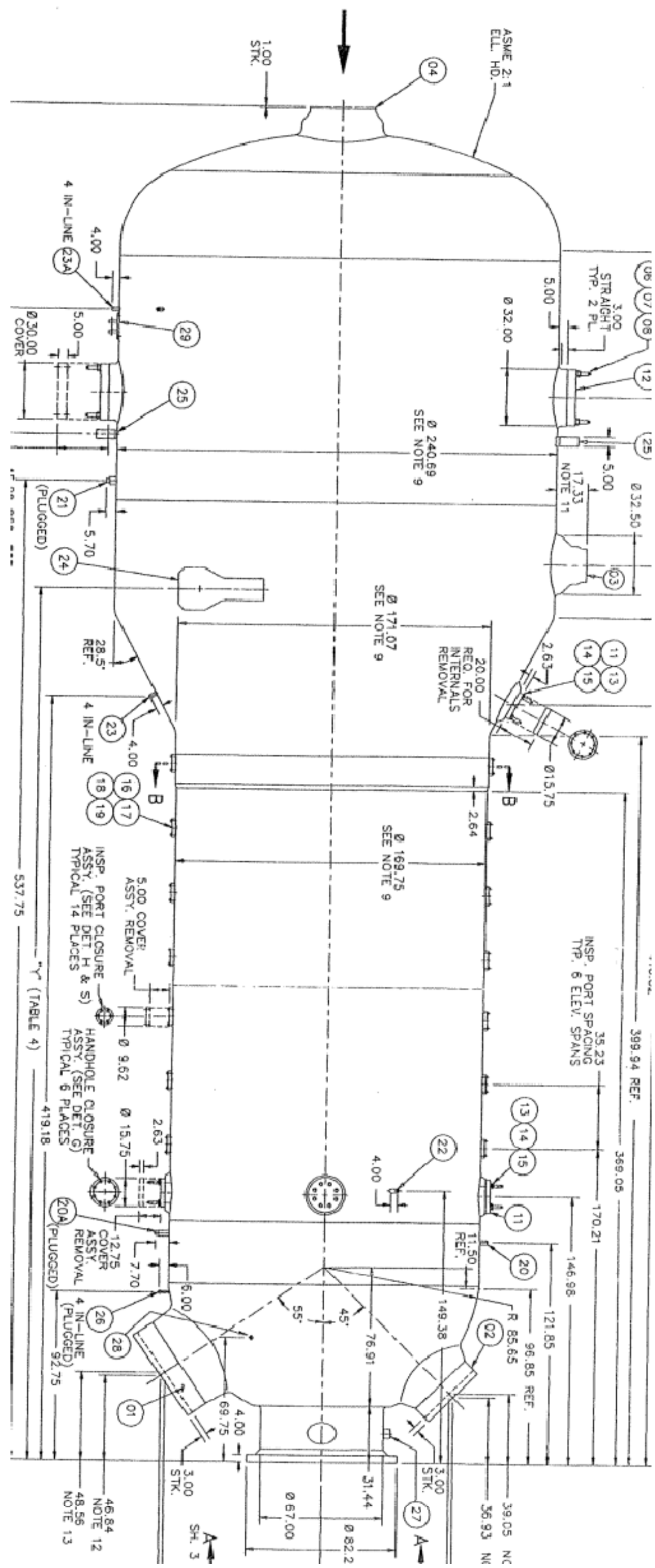

Figure 2-1. ANO-2 Steam Generator Layout [2-5]

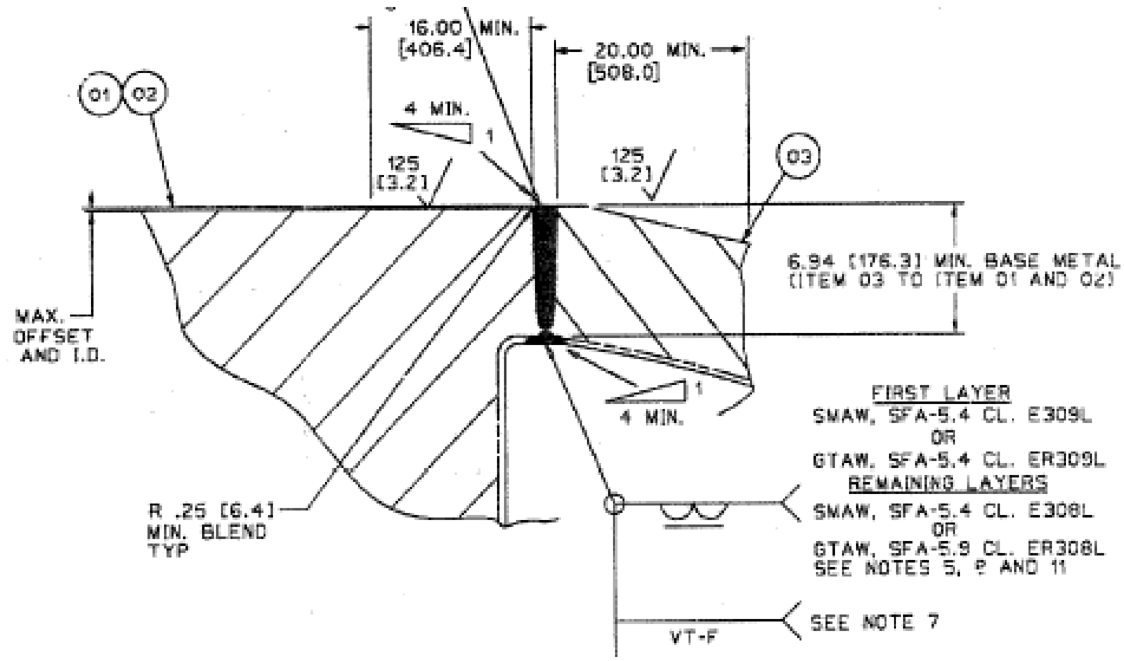

Figure 2-2. ANO-2 Item No. B2.40 Weld Configuration [2-6]

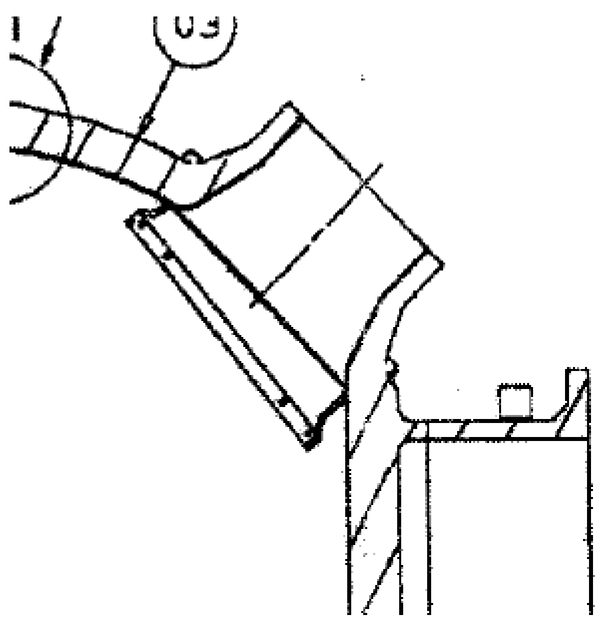

Figure 2-3. ANO-2 Item No. B3.130 Weld Configuration [2-6]

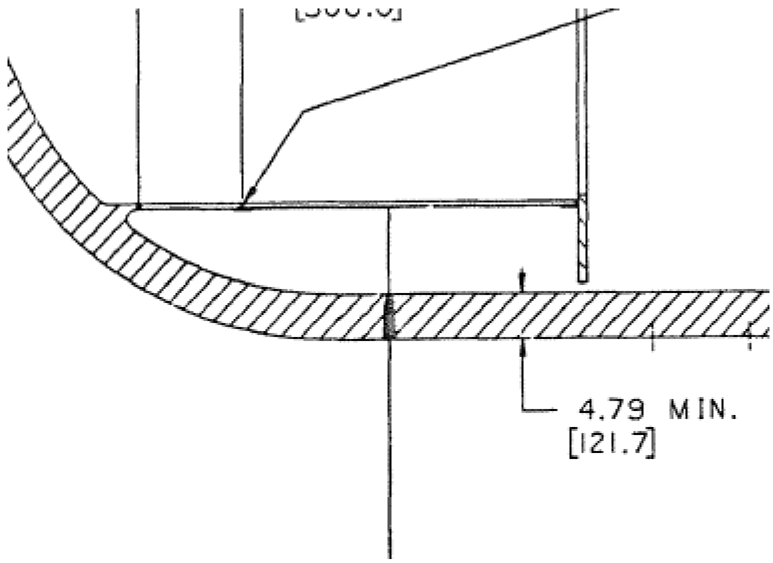

Figure 2-4. ANO-2 Item No. C1.20 Weld Configuration [2-10]

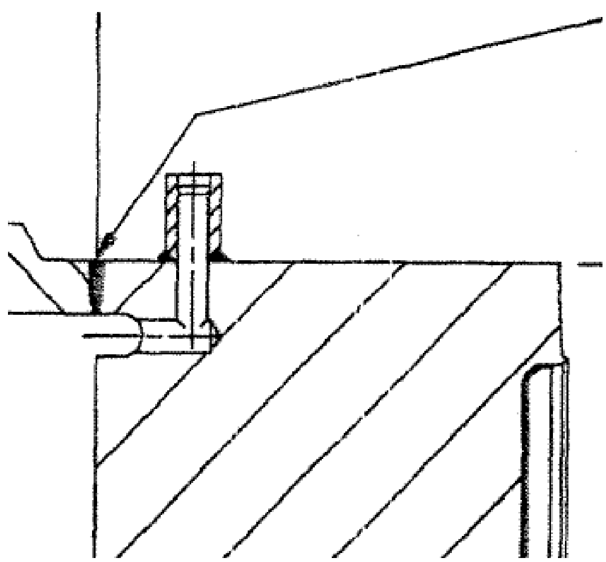

Figure 2-5. ANO-2 Item No. C1.30 Weld Configuration [2-7]

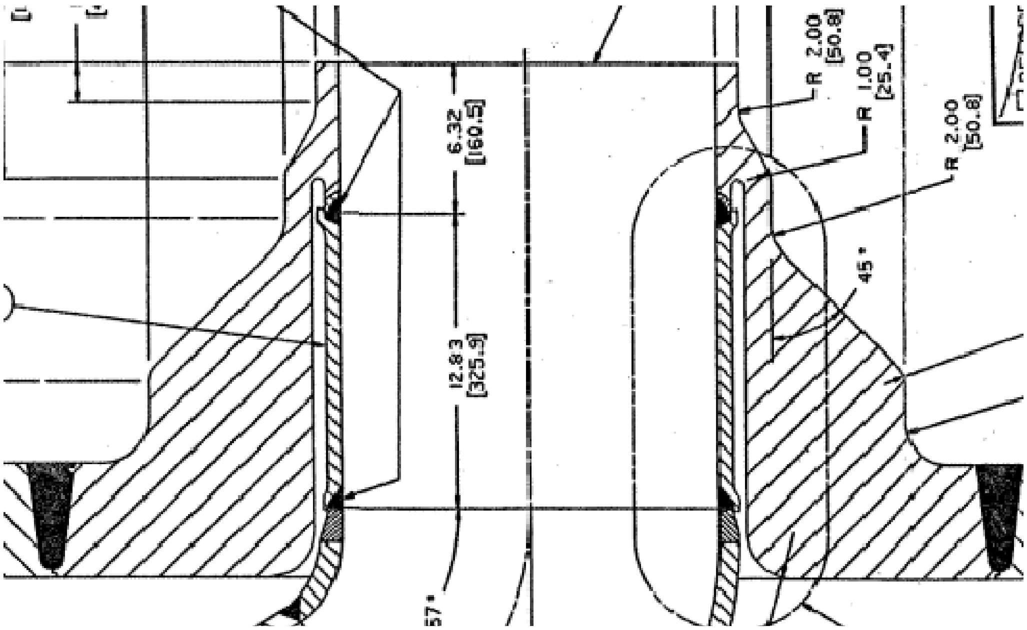

Figure 2-6. ANO-2 Feedwater Nozzle Configuration [2-8]

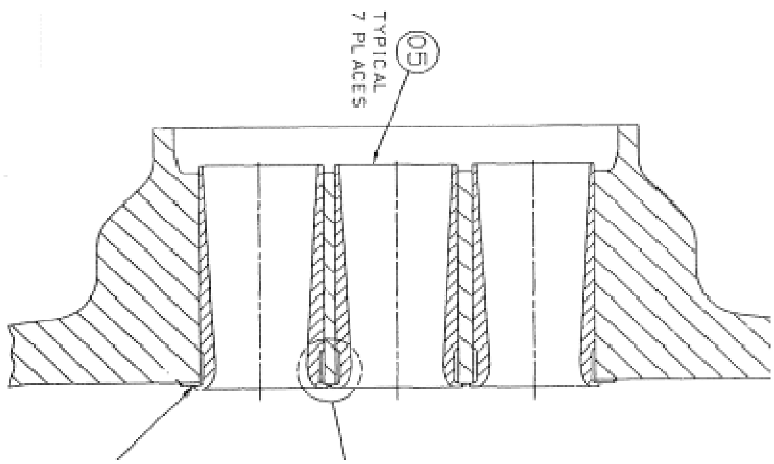

Figure 2-7. ANO-2 Main Steam Nozzle Configuration [2-9]

Attachment 2 References

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

2-2. *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.

2-3. Arkansas Nuclear One – Unit 2, SAR Amendment 22, Facility Operating License Number NPF-6, Docket Number 50-368.

2-4. NUREG-0800 - Chapter 5, Branch Technical Position (BTP) 5-3, Revision 2, "Fracture Toughness Requirements".

2-5. Drawing No. M-2001-C6-200, "Arkansas Nuclear One Unit 2 Δ 109 Replacement Steam Generator Outline," Sheet 2, Revision 0.

2-6. Drawing No. M-2001-06-293, "Arkansas Nuclear One Unit 2 Δ 109 Replacement Steam Generator, Primary Side Channel Head Assy," Sheet 2, Revision 0.

2-7. Drawing No. M-2001-C6-213, "Steam Generator Tube Plate and Lower Barrel Weld Assembly and Machining," Sheet 1, Revision 0.

2-8. Drawing No. M-2001-C6-245, "Feedwater Nozzle/Thermal Sleeve and Feedwater Ring Assembly," Sheet 2, Revision N.

2-9. Drawing No. M-2001-C6-204, "Arkansas Nuclear One Unit 2 Δ 109 Replacement Steam Generator, Upper Shell Assembly," Sheet 2, Revision 0.

2-10. Drawing No. M-2001-C6-204, "Arkansas Nuclear One Unit 2 Δ 109 Replacement Steam Generator, Upper Shell Assembly," Sheet 1, Revision 0.

2-11. Entergy Engineering Report No. CALC-ANO2-SE-18-00002, "ANO Unit 2 Transient Cycle Report for 2018," Revision 0.

2-12. SI Calculation No. 2200654.301, "ANO Units 1 & 2 Plant Loading/Unloading and Pressurizer Insurge/Outsurge Transients," Revision 0.

ENCLOSURE, ATTACHMENT 3

0CAN062402

RESULTS OF INDUSTRY SURVEY

Results of Industry Survey

<u>Overall Industry Inspection Summary for Code Items B2.31, B2.32, B2.40, B3.130, C1.10, C1.20, and C1.30</u>

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 Reference [3-1]. Table 3-1 provides a summary of the combined survey results for Item Nos. B2.31, B2.32 (see Table Note 3), B.240, B3.130, C1.10, C1.20, and C1.30. The results of the industry survey identified numerous SG examinations being performed with no service-induced flaws being detected. Performing these examinations adversely impact outage activities including worker exposure, personnel safety, and radwaste. A total of 74 domestic and international boiling water reactor (BWR) and pressurized water reactor (PWR) units responded to the survey and provided information representing all PWR plant designs currently in operation in the United States. This included 2-loop, 3-loop, and 4-loop PWR designs from each of the PWR nuclear steam supply system (NSSS) vendors (i.e., Babcock and Wilcox (B&W), Combustion Engineering (CE), and Westinghouse). A total of 1374 examinations for the components of the affected Item Nos. were conducted, with 1148 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 American Society of Mechanical Engineers (ASME) Code Section XI; however, these were determined to be subsurface-embedded fabrication flaws and non-service-induced (see Table 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 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 per 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	163	195	0	0	0
Totals	**226**	**1148**	**1374**	**0**	**Notes 1 and 2**	**Notes 1 and 2**

Notes:

1. Two PWR Westinghouse (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. C2.22, C2.21, and C2.32 (see Table Note 1). The results identify that SG MS and FW Nozzle-to-Shell Welds and Nozzle Inside Radius Section examinations being performed with no service-induced flaws being detected. Performing these examinations adversely impact outage activities including worker exposure, personnel safety, and radwaste. A total of 74 domestic and international 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 (i.e., B&W, CE, and Westinghouse). 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 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, Tube sheet-to-Head and Tube sheet-to-Shell Welds*. EPRI, Palo Alto, CA: 2019. 3002015906.

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.

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

3-4. Letter from J. M. Sorensen (NMC) 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," ADAMS Accession Number ML011550346, dated May 29, 2001.

3-5. Letter from J. P. Solymossy (NMC) 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)," ADAMS Accession Number ML031040553, dated April 4, 2003.