From: Billy Gleaves
Sent: Thursday, October 20, 2022 8:28 AM
To: Vogtle PEmails
Cc: Cayetano Santos
Subject: FW: Public Copy - ASME Code Alternative Request Drafts VEGP 3&4-ISI1-ALT-17 (Non-Proprietary) and VEGP 3&4-ISI1-ALT-18 (Non-Proprietary)
Attachments: VEGP 3&4-ISI1-ALT-17 (FINAL DRAFT) - Non-Proprietary - NRC.pdf; VEGP 3&4-ISI1-ALT-18 (FINAL DRAFT) - Non-Proprietary - NRC.pdf; ALT-17 ALT-18 Presubmittal Presentation.pdf

The attached files are to be used and presented at a public meetin□ currently scheduled for October 2□□2022.

From: Dorsey, Keith A. <kadorsey@southernco.com>
Sent: Wednesday, October 19, 2022 2:46 PM
To: Billy Gleaves <Bill.Gleaves@nrc.gov>
Cc: Cayetano Santos <Cayetano.Santos@nrc.gov>; Chapman, Nathan B. <NBCHAPMA@SOUTHERNCO.COM>; Garrett, William <WGARRETT@SOUTHERNCO.COM>; Leighty, Steven <sleighty@southernco.COM>; Chamberlain, Amy Christine <ACCHAMBE@southernco.com>; Coleman, Jamie Marquess <JAMIEMCO@SOUTHERNCO.COM>
Subject: [External_Sender] Public Copy - ASME Code Alternative Request Drafts VEGP 3&4-ISI1-ALT-17 (Non-Proprietary) and VEGP 3&4-ISI1-ALT-18 (Non-Proprietary)

Mr. Gleaves:

Attached you will find the Pre-Submittal Presentation, a Non-Proprietary draft copy of ASME Code Alternative Request Draft VEGP 3&4-ISI1-ALT-17, and a draft copy of VEGP 3&4-ISI1-ALT-18 (also Non-Proprietary) for NRC review prior to the pre-submittal meeting scheduled for next Thursday (10/27).

These copies may be made available to the public. The Proprietary / Non-public copy of VEGP 3&4-ISI1-ALT-17 will be provided in a separate email.

Thanks,

Keith Dorsey P.E.
Lead Engineer
Nuclear Development
Regulatory Affairs

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Subject: FW□Public Copy - □SME Code □lternati□e Re□uest Drafts VE□P 3□□-ISI1-□LT-1□ □□on-Proprietary□and VE□P 3□□-ISI1-□LT-1□ □□on-Proprietary□
Sent Date: 10□20□2022 □2□□□□ □M
Received Date: 10□20□2022 □2□□0□ □M
From: Billy □lea□es

Created By: Bill.□lea□es□ nrc.□o□

Recipients:
"Cayetano Santos" □Cayetano.Santos□ nrc.□o□□
Trac□in□ Status□□one
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Options
Priority: □ormal
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Sensitivity: □ormal
Expiration Date:

10CFR50.55a Alternative VEGP3&4-ISI1-ALT-17
Proposed Alternative in Accordance with 10CFR50.55a(z)(1)
--Alternative Provides Acceptable Level of Quality and Safety--
Revision 0

1. ASME Code Component(s) Affected

Code Class:	1
Reference:	IWB-2500, Table IWB-2500-1 10CFR50.55a(b)(2)(xlii)
Examination Category:	B-F
Item Number:	B5.71
Description:	Alternative Inspection Requirements for Steam Generator Nozzle to Reactor Coolant Pump Casing Welds

Component Number:

Unit 3	Unit 4
SV3-SGA-Nozzle A-201-96A	SV4-SGA-Nozzle A-201-96A
SV3-SGA-Nozzle B-201-96B	SV4-SGA-Nozzle B-201-96B
SV3-SGB-Nozzle A-201-96A	SV4-SGB-Nozzle A-201-96A
SV3-SGB-Nozzle B-201-96B	SV4-SGB-Nozzle B-201-96B

Drawing Number: Figures 1 and 2

2. Applicable Code Edition

The First Interval of the Vogtle Electric Generating Plant (VEGP), Units 3 and 4 Inservice Inspection (ISI) Program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section XI, 2017 Edition.

3. Applicable Code Requirement

Subarticle IWB-2500 requires components specified in Table IWB-2500-1 to be examined. Table IWB-2500-1 requires a volumetric and surface examination of all NPS 4 (DN 100) or larger nozzle-to-component butt welds each inspection interval (Examination Category B-F, Item Number B5.71). The applicable examination volume is shown in Figure IWB-2500-8(f).

In accordance with the provisions of 10CFR50.55a(b)(2)(xlii), Section XI condition: Steam Generator Nozzle-to-Component welds and Reactor Vessel Nozzle-to-Component welds:

> Licensees applying the provisions of Table IWB–2500–1, Examination Category B–F, Pressure Retaining Dissimilar Metal Welds in Vessel Nozzles… Item B5.71 (NPS 4 or Larger Nozzle-to-Component Butt Welds) of the 2011a Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of 10CFR50.55a must also meet the following conditions:
>
> (A) Ultrasonic examination procedures, equipment, and personnel shall be qualified by performance demonstration in accordance with Mandatory Appendix VIII.

(B) When applying the examination requirements of Figure IWB–2500–8, the volumetric examination volume shall be extended to include 100 percent of the weld volume, except as provided in paragraph (b)(2)(xlii)(B)(1) of 10CFR50.55a :

(1) If the examination volume that can be obtained by performance demonstration qualified procedures is less than 100 percent of the weld volume, the licensee may ultrasonically examine the qualified volume and perform a flaw evaluation of the largest hypothetical crack that could exist in the volume not qualified for ultrasonic examination, subject to prior NRC authorization in accordance with paragraph (z) of 10CFR50.55a .

4. Reason for Request

In accordance with 10CFR50.55a(z)(1), relief is requested on the basis that the proposed alternative will provide an acceptable level of quality and safety.

Southern Nuclear Operating Company, Inc. (SNC) is requesting an alternative from the conditions listed in 10CFR50.55a(b)(2)(xlii)(B) to extend the examination volume to include 100 percent of the weld volume, and to seek NRC approval to use exception 10CFR50.55a(b)(2)(xlii)(B)(1) to ultrasonically examine the qualified volume shown in the 2017 Edition of Section XI, and to perform a flaw evaluation of the largest hypothetical crack that could exist in the volume not qualified for ultrasonic examination.

5. Proposed Alternative and Basis for Use

The AP1000® design is unique in that the reactor coolant pump inlet nozzle is welded directly to the steam generator cold leg outlet nozzle (two per steam generator). The dissimilar metal circumferential butt weld joining the low alloy steel with austenitic stainless steel cladding steam generator nozzle to the cast austenitic stainless steel reactor coolant pump casing is classified as an ASME Section XI Class 1 weld (see Figure 1). The AP1000 design has two steam generators and four reactor coolant pumps in each unit at VEGP.

SNC proposes to perform an inservice inspection encoded volumetric examination of the required 2017 Edition of ASME Section XI inspection volume, not the entire weld volume. The volumetric examination will be performed from the inner diameter (ID) surface. SNC will also perform the required surface examination on the outer diameter (OD) surface. The ultrasonic testing techniques will be qualified in accordance with the Performance Demonstration Initiative (PDI) Program which satisfies the requirements of ASME Section XI, Appendix VIII, Supplement 10, including 10CFR50.55a.

In addition to the ASME Section XI examinations, SNC proposes to perform an eddy current examination from the ID surface. Although not an ASME mandatory examination, the eddy current examination utilized on the ID surface will be qualified in accordance with ASME Section V, Article 14 (2017 Edition).

This approach minimizes the impact of sound beam re-direction and scattering. The capability of detecting and length sizing of ID-initiated flaws in the weld and in the cast austenitic stainless steel (CASS) material was demonstrated on a representative blind test specimen. Therefore, full volume of the inner third of the weld, as required by the 2017 Edition of ASME Section XI, will be achieved. For clarification, SNC's proposed volumetric coverage for the First ISI Interval is depicted in Figure IWB-2500-8(f) and applied to the VEGP steam generator nozzle to reactor coolant pump casing butt welds in Figure 2.

The figures in the ASME Section XI, 2017 Edition (Figure IWB-2500-8 (c) – (e)) are illustrative with respect to the weld joint configuration. These figures are intended only to define the examination volume and examination surface extent for similar and dissimilar metal welds in components, nozzles, and piping. It is noted that the examination volume and examination surface extent is defined with respect to the weld (or weld end buttering) edges at the widest part of the weld (and weld end buttering) regardless of whether it is located on the inside or outside surface. For the examination volume, these weld (or weld end buttering) edges are extended to the inside surface and the ¼-inch of adjacent base material is added to both edges to obtain the width extent of the examination volume. The 1/3t examination volume depth is taken from the inside surface. This approach ensures that the entire weld and weld end buttering width is captured in the examination volume regardless of the weld preparation configuration.

Figure 2 shows the proposed exam volume and that the widest part of the weld (and weld end buttering) is the same on the inside and outside surfaces. The weld is defined by the edges of the weld end buttering on the Reactor Coolant Pump Casing and the Steam Generator Nozzle. The ¼-inch of adjacent pump casing and nozzle base material is taken from these weld end buttering points and the 1/3t depth is taken from the inside surface. Figure 2 is to scale and the proposed exam volume captures all of the innermost weld. As noted in Figure 2, the entire weld and weld end buttering width is captured in the examination volume.

The examination procedure to be utilized has been qualified on a mock-up representative of the thickness and configuration of the steam generator outlet nozzle to reactor coolant pump casing weld and contains ID-initiated planar flaws in the weld, buttering and in the cast stainless steel material. Detection and length sizing qualification was extended to the full thickness. Because the examinations are performed from the ID surface, coverage of the examination volume is not limited. It is important to note, the examination volume is the inner 1/3 of the thickness of the weld and includes the weld and 0.25-inch of adjacent base metal on both sides of the weld and buttering. The weld and buttering on the steam generator outlet nozzle are composed of Alloy 690 weld metals. Alloy 690 weld metals used in the AP1000 design are much more resistant to developing flaws and have significantly improved flaw growth tolerance as compared to Alloy 600 weld metals. Examination of the outer 2/3 is not required unless performing depth sizing of a flaw indication.

The ultrasonic testing and eddy current examination are the same as those applied to Reactor Pressure Vessel (RPV) nozzle to safe end dissimilar metal welds from the ID surface by the inspection vendor, except for the addition of larger and deeper focused ultrasonic testing transducers. These added transducers allow for through-wall coverage through the full thickness of the weld in the event flaw indications are detected within the inner 1/3 of the thickness of the weld and adjacent base material or the defined examination volume. The AP1000 steam generator nozzle to pump casing dissimilar metal butt weld is thicker than the RPV nozzle to safe end dissimilar metal welds found in other pressurized water reactors.

To extend the PDI qualification to this greater thickness and to account for the specific weld configuration, an AP1000 steam generator to pump casing weld specimen was designed and fabricated by the Electric Power Research Institute (EPRI) in accordance with the EPRI/PDI Program. This specimen serves as a blind test specimen necessary to qualify the ultrasonic testing procedure and the ultrasonic testing personnel. The ultrasonic testing procedure and personnel qualifications are conducted by PDI under the PDI ASME Section XI, Appendix VIII program.

The eddy current examination techniques are qualified internally by the inspection vendor in accordance with ASME Section V, Article 14, intermediate rigor, using test data obtained from an additional AP1000 steam generator to pump casing butt weld specimen, containing ID surface breaking planar flaws.

This combination of the ID surface applied ultrasonic testing and eddy current examination, that have been qualified or demonstrated, will allow detection of primary water stress corrosion cracking, the failure mechanism identified for dissimilar metal welds in operational pressurized water reactors. It is also noted that the ultrasonic testing techniques are capable of detecting, and length sizing, embedded planar flaws throughout the 2017 Edition of ASME Section XI examination volume as demonstrated on an open AP1000 steam generator to pump casing butt weld specimen.

The preservice inspection (PSI) Interval examinations requested in Alternative VEGP 3&4-PSI-ALT-05 were complimented by the required ASME BPV Code Section III radiography examinations and the design organization's required ASME Code Section V ultrasonic testing imposed during component fabrication. The ultrasonic testing included in-progress inspections of the buttering material on both the steam generator and reactor coolant pump materials from the end face, and post-weld inspections of the full volume of the weld using multiple angles, four directional angle beam techniques, from both the

ID and OD surfaces. The post-weld ultrasonic testing results were evaluated against the ASME Code Section III and Section XI standards for acceptance. No relevant indications were identified during the fabrication process or PSI examinations.

Enclosures 1 and 2 provide additional information and analysis that concludes a postulated defect in the outer 2/3 of the weld would not exceed the allowable flaw size over the licensed lifetime of the plant based upon the ASME BPV Code Section XI flaw tolerance evaluation and the ASME BPV Section III design evaluation.

Enclosures 3 and 4 provide justification as to why the constant flaw aspect ratios of 2 and 6 for axial and circumferential flaws was used in Enclosures 1 and 2. Enclosures 3 and 4 also provide justification as to why only flaw depth was evaluated in Enclosures 1 and 2.

Enclosure 5 provides an affidavit from SNC supporting withholding the Proprietary information. Enclosures 1 and 3 contain the revised Non-Proprietary response. Enclosures 2 and 4 contain the revised Proprietary response subject to withholding under 10CFR2.390.

Enclosure 6 and 7 are Westinghouse's Proprietary Information Notices, Copyright Notices, and CAW-16-4509 and CAW-17-4534, Application for Withholding Proprietary Information from Public Disclosure and Affidavit. The affidavits set forth the basis upon which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information that is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations. Enclosures 1 through 7 were submitted as part of the approval process of the PSI VEGP3&4-PSI-ALT-05 Alternative, under ADAMS Accession Numbers ML16355A222 and ML17032A524, and the justification of this alternative has been proven for the life of the plants.

The proposed examinations are in accordance with the 2017 Edition of ASME Section XI, as described above. Therefore, SNC concludes that the proposed examinations will provide an acceptable level of quality and safety.

6. Duration of Proposed Alternative

The proposed alternative is requested for the First ISI Interval for Vogtle Electric Generating Plant (VEGP), Units 3 and 4.

7. Precedents

- Vogtle Electric Generating Plant, Units 3 and 4 Preservice Inspection Proposed Alternative VEGP3&4-PSI-ALT-05 was authorized by an NRC SE dated April 17, 2017 (i.e., NRC Accession Nos. ML17097A337 and ML17097A450).

8. References

1. ASME Boiler and Pressure Vessel Code, Section XI, Division 1, "Inservice Inspection of Nuclear Power Plant Components," 2017 Edition.
2. ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Rules for Construction of Nuclear Facility Components," 1998 Edition through the 2000 Addenda.
3. ASME Boiler and Pressure Vessel Code, Section V, Division 1, "Nondestructive Examination," 2017 Edition.
4. Vogtle Units 3 and 4, Updated Final Safety Analysis Report (UFSAR), Subsection 5.1.3.3.

9. Enclosures

Enclosure 1: Westinghouse LTR-PAFM-16-59-NP, NRC RAI Response Regarding Inspection of AP1000 Vogtle Units 3 & 4 and V.C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Suction Nozzle Weld (Non-Proprietary) Rev. 1

Enclosure 2: Westinghouse LTR-PAFM-16-59-P, NRC RAI Response Regarding Inspection of AP1000 Vogtle Units 3 & 4 and V.C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Suction Nozzle Weld **(Proprietary)** Rev. 1

Enclosure 3: Westinghouse LTR-PAFM-17-6, Rev. 0, NP-Enclosure 1 (Non-Proprietary) - Additional Information Regarding AP1000 Vogtle Units 3 & 4 and V. C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Suction Nozzle Weld Flaw Evaluation

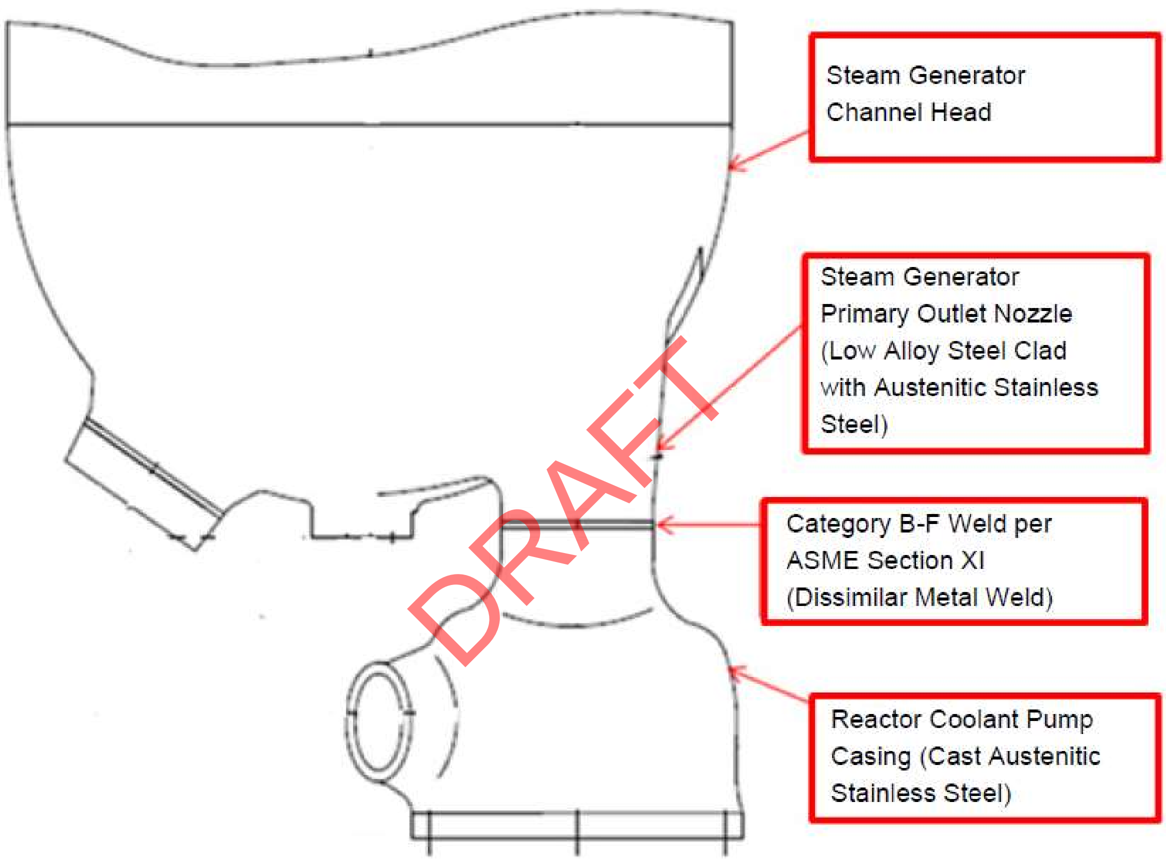
Enclosure 4: Westinghouse LTR-PAFM-17-6, Rev. 0, P-Enclosure 2 **(Proprietary)** - Additional Information Regarding AP1000 Vogtle Units 3 & 4 and V. C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Suction Nozzle Weld Flaw Evaluation

Enclosure 5: Vogtle Electric Generating Plant (VEGP) Units 3 and 4 – Affidavit from Southern Nuclear Operating Company for Withholding Under 10 CFR 2.390

Enclosure 6: Vogtle Electric Generating Plant (VEGP) Units 3 and 4 – Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-16-4509, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice

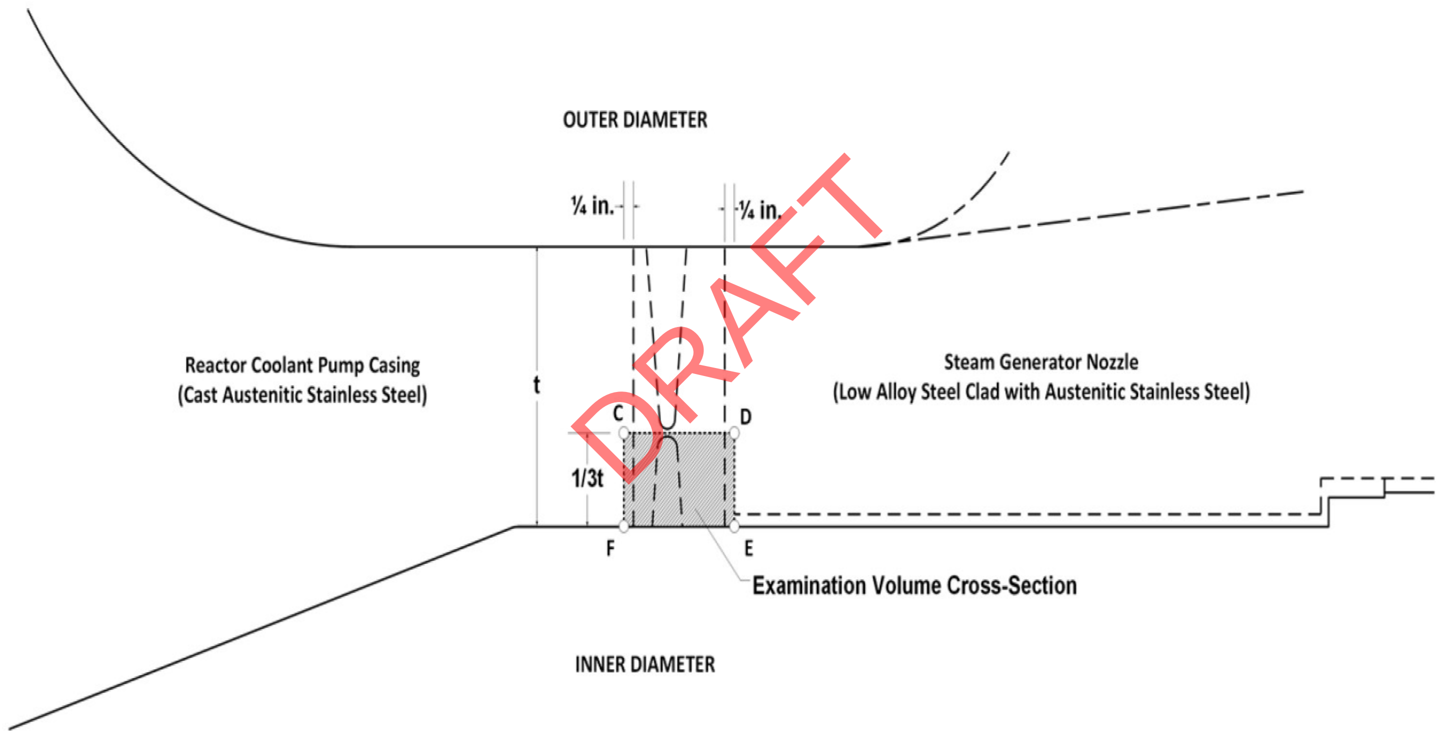
Enclosure 7: Vogtle Electric Generating Plant (VEGP) Units 3 and 4- Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-17-4534, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice

Figure 1



RCP CASING ATTACHMENT

Figure 2


OUTER DIAMETER
¼ in.
¼ in.
Reactor Coolant Pump Casing
(Cast Austenitic Stainless Steel)
t
C
D
1/3t
F
E
Steam Generator Nozzle
(Low Alloy Steel Clad with Austenitic Stainless Steel)
Examination Volume Cross-Section
INNER DIAMETER

10CFR50.55a Alternative VEGP3&4-ISI1-ALT-17
Proposed Alternative in Accordance with 10CFR50.55a(z)(1)
--Alternative Provides Acceptable Level of Quality and Safety--
Revision 0

Enclosure 1

Westinghouse LTR-PAFM-16-59-NP, NRC RAI Response Regarding Inspection of AP1000 Vogtle Units 3 & 4 and V.C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Suction Nozzle Weld (Non-Proprietary) Rev. 1

LTR-PAFM-16-59-NP
Revision 1

NRC RAI Response Regarding Inspection of *AP1000* Vogtle Units 3 & 4 and V. C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Suction Nozzle Weld

November 2016

DRAFT

Author: Alexandria Carolan*, Piping Analysis and Fracture Mechanics
Rick Rishel*, Wesdyne

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Stephan Sabo*, Wesdyne

Approved: Benjamin Leber*, Manager, Piping Analysis and Fracture Mechanics

**Electronically approved records are authenticated in the electronic document management system.*

Westinghouse

Record of Revisions

Rev.	Date	Revision Description
0	September 2016	Original Version
1	November 2016	Incorporate fabrication-related inspection data on flaw sizes from post-weld Radiographic Testing (RT) and Ultrasonic Testing (UT) examinations from Vogtle Units 3 and 4 and V.C. Summer Units 2 and 3. Major changes in Revision 1 are identified by vertical bars in the left-hand margins of the document.

Trademark Note:

FOREWORD

This document contains Westinghouse Electric Company LLC proprietary information and data which has been identified by brackets. Coding $^{(a,c,e)}$ associated with the brackets sets forth the basis on which the information is considered proprietary. These codes are listed with their meanings in WCAP-7211 Revision 8 (September 2015), "Proprietary Information and Intellectual Property Management Policies and Procedures."

The proprietary information and data contained in this report were obtained at considerable Westinghouse expense and its release could seriously affect our competitive position. This information is to be withheld from public disclosure in accordance with the Rules of Practice 10CFR2.390 and the information presented herein is to be safeguarded in accordance with 10CFR2.903. Withholding of this information does not adversely affect the public interest.

This information has been provided for your internal use only and should not be released to persons or organizations outside the Directorate of Regulation and the ACRS without the express written approval of Westinghouse Electric Company LLC. Should it become necessary to release this information to such persons as part of the review procedure, please contact Westinghouse Electric Company LLC, which will make the necessary arrangements required to protect the Company's proprietary interests.

The proprietary information in the brackets has been deleted in this report, the deleted information is provided in the proprietary version of this report (LTR-PAFM-16-59-P Revision 1).

1.0 Introduction

The objective of this letter report is to provide responses to the NRC Request for Additional Information (RAI) [1] regarding the *AP1000*® Steam Generator (SG) to Reactor Coolant Pump (RCP) suction nozzle dissimilar metal (DM) weld inspection coverage. The NRC RAI requests additional information or analyses to justify why an ultrasonic examination of the inner 1/3 of the weld thickness and a surface examination of the inner and outer weld surfaces is an acceptable alternative to examining the full weld volume.

The responses to the NRC RAI will be based on two separate assessments that have been performed for the region of interest. The first assessment is based on an ASME Section XI flaw tolerance analysis, and the second assessment is based on the ASME Section III design evaluation.

1.1 ASME Section XI Flaw Tolerance Evaluations for Flaws on the Outside Surface and Embedded Within the Weld Examination Volume

The first evaluation is based on an ASME Section XI flaw tolerance analysis for the DM weld, with the consideration of a surface postulated flaw size in the outer 2/3 of the wall thickness. This flaw evaluation considers a crack growth calculation for 60 years (design life) and the maximum end-of-evaluation flaw size determinations based on limit load analysis, typical of an ASME Section XI flaw evaluation using the rules of Appendix C. The intent is to show that a large postulated outside surface flaw will not grow to the maximum allowable end-of-evaluation period flaw size (i.e., allowable flaw size) after 60 years of growth. The maximum allowable end-of-evaluation period flaw size is calculated based on the ASME IWB-3640 guidelines [2]. The postulated outside surface flaws used in this crack growth analysis are an axial flaw with Aspect Ratio (AR), flaw length/flaw depth, AR = 2, and a circumferential flaw with AR = 6. The aspect ratio of 2 is reasonable because the axial flaw growth is limited to the width of the DM weld configuration, and an aspect ratio of 6 for postulated circumferential flaw is typical for fracture mechanics analyses. The primary crack growth mechanism for flaws within the nozzle welds is Fatigue Crack Growth (FCG). The fatigue crack growth rates as well as the stress intensity factor equations required to complete a FCG analysis are further discussed in Section 2 of this letter report. Crack growth due to Primary Water Stress Corrosion Crack (PWSCC) growth does not need to be investigated since the base metals around the DM weld, the stainless steel buttering, and the Alloy 152/52 DM weld material have a low susceptibility to stress corrosion cracking. Furthermore, the evaluation considered in this letter is for postulated outside surface flaws (which conservatively bound postulated embedded fabrication flaws) which are not exposed to the primary coolant, and thus the susceptibility to PWSCC is not of concern. Any potential indications in the inner 1/3 of the dissimilar metal weld wall thickness will be detected by volumetric inspection during the in-service inspections.

While the fracture mechanics evaluation assumed postulated outside surface flaws, the eight (8) steam generator to RCP casing DM welds for Vogtle Units 3 and 4, and six (6) of the eight (8) steam generator to RCP casing DM welds for V. C. Summer Units 2 and 3 (Note: V.C. Summer Unit 3 Steam Generator 'A' is still in the fabrication facility) have been examined in accordance with the requirements of ASME Section III, NB-5000. These examinations included a liquid penetrant (PT) examination of the outside surface. No relevant indications were observed. Relevant indications are those having major dimensions greater than 1/16-inch (0.0625-inch). For the remaining two (2) DM welds for V.C. Summer Unit 3, there can be no cracks or linear indications greater than

1/16-inch (0.0625-inch) long or rounded indications greater than 3/16-inch (0.188-inch) long on the outside surface.

This same evaluation of postulated outside surface flaws conservatively covers evaluations for embedded fabrication flaws which may be present at the beginning of service. The stress intensity factors in the fracture mechanics analysis for surface flaws are more limiting than embedded fabrication flaws.

The eight (8) steam generator to RCP casing DM welds for Vogtle Units 3 and 4, and six (6) of the eight (8) steam generator to RCP casing DM welds for V.C. Summer Units 2 and 3 have been examined volumetrically in accordance with the requirements of ASME Section III, NB-5000 and Westinghouse design requirements. These volumetric examinations included radiographic (RT) and ultrasonic (UT) testing of the buttering materials on both the steam generator and RCP materials as in-process examinations, and RT and UT of the weld, buttering materials and 0.25-inch of adjacent base material on both sides of the weld for the full thickness after completion of the welds. The post-weld RT results were evaluated against the acceptance standards in ASME Code Section III, NB-5320. The post-weld UT of the full volume of the weld used multiple angle, four directional angle beam techniques, from both the ID (inside diameter) and OD (outside diameter) surfaces. The post-weld UT results were evaluated against the acceptance standards of ASME Code Section III, NB-5331 and Section XI, IWB-3514 (for preservice examination).

The UT from the OD surface was performed with a 0° probe and 45°, 60°, and 70° 1.0 MHz transmit-receive, longitudinal wave probes focused at various depths below the OD surface; the angle beam examinations were focused on the outer half of the examination volume. The UT from the ID surface was performed with 37°, 45°, and 70°, 1.0 – 2.0 MHz transmit-receive longitudinal wave probes focused at various depths below the ID surface; these examinations were focused on the inner half of the examination volume.

[

]^{a,c,e}

The remaining two (2) steam generator to RCP casing DM welds in the V.C. Summer Unit 3 SG 'A' that have not been examined may contain volumetric RT flaw indications of less than 0.75-inch long. If there are any UT flaw indications these flaw indications must satisfy the allowable flaw standards in ASME Code Section XI 1998 Edition with the 2000 Addenda, Table IWB-3514-2 for preservice examination with a nominal wall thickness of 3.0-inches. For a subsurface flaw indication, the allowable a/t ranges from 7.6% to 8.9% depending on the aspect ratio (a/l) of the flaw indication. This converts to a through-wall extent (2a/t) of 15.2% to 17.8%, or for a [

]^{a,c,e} Therefore, the allowable standards in Table IWB-3514-2 are sufficiently large to show acceptance for any flaws detected during fabrication and pre-service at Vogtle and V. C. Summer. It should be noted that the ASME Section XI 2007 Edition, 2008 Addenda have even larger allowable standards than those in the 1998 Edition with 2000 Addenda.

1.2 ASME Section III Design Evaluations

In addition to the ASME Section XI flaw tolerance analysis for postulated axial and circumferential flaw on the outside surface (and postulated embedded fabrication flaws) of the SG to RCP suction nozzle DM weld, a Section III design evaluation [4] assessment was already completed for the *AP1000* Steam Generator and the adjacent DM weld. The primary goal of the Section III evaluation is to demonstrate acceptable margins to avoid cracks initiating as a result of fatigue cycling. The ASME Section III discussion and results are provided here also to demonstrate that the region of interest (i.e., SG outlet nozzle and DM weld) meet the structural design requirements of the ASME Code. The select results that are provided in Section 3 of this letter aim to demonstrate that the primary and secondary stress analysis, fatigue usage, and non-ductile failure (fracture mechanics) assessment per the ASME Section III code are all satisfied. Therefore, meeting the requirements of ASME Section III further demonstrates the justification that the DM weld location is structurally qualified for the design life of the plant.

2.0 ASME Section XI Flaw Tolerance Analysis for Postulated Outside Surface (and Embedded) Flaws

This section provides a brief discussion for the fracture mechanics analysis of outside surface postulated flaws per the ASME Section XI guidelines. The evaluation first calculates the maximum allowable end-of-evaluation period flaw sizes for the two different flaw orientations (axial and circumferential flaws) based on ASME Section XI at the Steam Generator to RCP DM weld location. Next, fatigue crack growth calculations at the dissimilar metal weld are performed for 60 years for large postulated outside surface flaws. The initial postulated outside surface flaw sizes are sufficiently larger than any existing fabrication-related outside surface flaws detected during the fabrication examinations of the welds.

It is also noted that since the stress intensity factor is lower for embedded flaws than that of surface flaws for a given through-wall stress distribution, the initial postulated outside surface flaw sizes are bounding for existing fabrication-related embedded flaws.

2.1 Maximum Allowable End-of-Evaluation Period Flaw Sizes

The calculation of the maximum allowable end-of-evaluation period flaw sizes for austenitic steel and nickel base alloys is based on limit load analysis. The procedures and acceptance criteria for the limit load analysis in austenitic components and weld metals are contained in paragraph IWB-3640 of ASME Section XI [2]. These criteria were used to determine the maximum allowable end-of-evaluation period flaw size for axial (AR = 2) and circumferential (AR = 6) flaw configurations. The aspect ratio of 2 is reasonable because the axial flaw growth is limited to the width of the DM weld configuration, and an aspect ratio of 6 for postulated circumferential flaw is typical for fracture mechanics analyses. The procedure to evaluate the allowable flaw sizes is based on IWB-3640 and subsequently Appendix C of Section XI of the code.

The maximum end-of-evaluation period flaw sizes determined for both axial and circumferential flaws have incorporated the relevant material properties, nozzle loadings and geometry. Loadings under normal, upset, emergency, and faulted conditions were considered in conjunction with the applicable safety factors for the corresponding service conditions required in the ASME Code Section XI. For circumferential flaws, axial stress due to the [

]^{a,c,e} were considered in the evaluation. As for the axial flaws, hoop stress resulting from pressure loading was used, since none of the other loadings have an impact on such flaws.

Per ASME Section XI, the thermal expansion loads do not need to be considered in the maximum end-of-evaluation period flaw size determination since the nozzle welds are GTAW (Gas Tungsten Arc Weld) and are non-flux welds.

The *AP1000* SG to RCP suction nozzle DM weld dimensions and operating parameters are shown in Table 1. A design temperature of [ ]^{a,c,e} was conservatively used in determining the end-of-evaluation period flaw size and for the fatigue crack growth analysis. The nozzle loads at the SG to RCP suction nozzle weld are based on conservatively bounding both the SG and RCP design specification allowable loads (Table 2). The loads given in Table 2 are in the local coordinate system, where the x-axis is axial along the component centerline, y-axis and zaxis by right-hand-rule. Furthermore, all loads are conservatively applied as absolute values. The design mechanical loads cover normal pump vibration loadings.

Table 1: *AP1000* SG to RCP Suction Nozzle Weld Geometry and Operating Parameters

a,c,e

Table 2: *AP1000* SG to RCP Suction Nozzle Weld Allowable Loads

a,c,e

The maximum end-of-evaluation period allowable flaw sizes are determined based on the weaker of the base metal and weld metal material properties flow strength value (average of the yield and ultimate strengths) at the SG to RCP suction nozzle weld for a maximum temperature of [ ]a,c,e. The ASME code limiting material properties at the DM weld location are based on the [

]a,c,e

The maximum allowable end-of-evaluation period flaw sizes for the SG to RCP suction nozzle DM weld are shown in Table 3. It should be noted that the maximum end-of-evaluation period allowable flaw sizes are limited to only 75% of the wall thickness in accordance with the requirements of ASME Section XI paragraph IWB-3640 [2]. Next, the fatigue crack growth analyses are performed to determine the largest postulated allowable initial flaw size for 60 years of plant operation such that the final crack growth flaw sizes will not reach the maximum-end-of-evaluation period flaw sizes shown in Table 3.

Table 3: Maximum Allowable End-of-Evaluation Period Flaw Size

Flaw Configuration	Aspect Ratio (flaw length/flaw depth)	Maximum End-of-Evaluation Period Flaw Size (a/t)
Axial Flaw	2	0.75
Circumferential Flaw	6	0.47

The wall thickness, denoted as 't', and the flaw depth and flaw length, denoted as 'a' and 'ℓ' respectively, are shown in Figure 1 for an axial flaw on the outside diameter of the SG to RCP suction nozzle DM weld. A circumferential flaw on the outside diameter has the same denotation for thickness and flaw configuration variables.

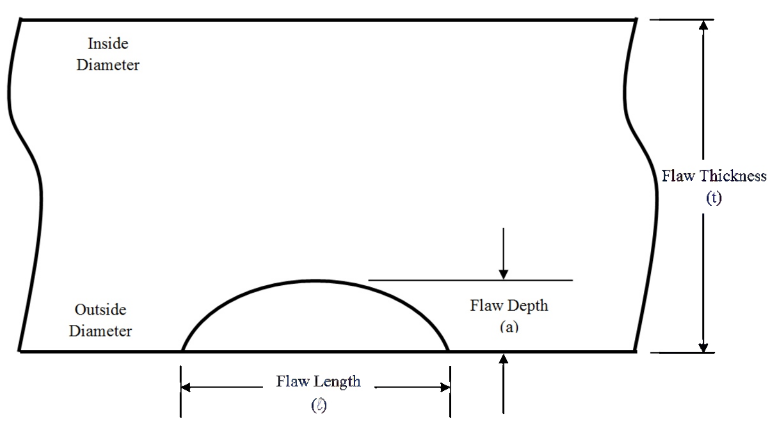

Figure 1: Illustration of SG to RCP Suction Nozzle DM Weld Outside Diameter Axial Flaw

2.2 Crack Growth Evaluations

As discussed in Section 1 of this letter report, the primary crack growth mechanism for flaws within the nozzle welds is fatigue crack growth. Crack growth due to Primary Water Stress Corrosion Crack growth is not applicable since the Alloy 152/52 DM weld material has very low susceptibility to stress corrosion cracking, especially for outside surface flaws and embedded flaws which are not exposed to the primary coolant. The fatigue crack growth rate as well as the stress intensity factor equations required to complete a FCG analysis are discussed in this section. The inputs that are used in the crack growth evaluations are the applicable design transients and cycles that contribute significantly to the FCG (see Section 2.2.1), along with the consideration of welding residual stresses to the crack growth analysis. The through-wall transient stresses, along with the residual stresses are used to calculate the stress intensity factor, which will then be used with the cycles to determine the 60 year FCG results for the axial and circumferential flaw.

2.2.1 Transient Stresses

The transient hoop and axial stress data used in the FCG analysis is representative of the stresses at the SG to RCP suction nozzle DM weld. The transients and cycles are listed in Table 4, and these particular transients are considered limiting for use in the FCG analysis. The remaining *AP1000* design transients do not have high fatigue usage factor contributions to the ASME Section III RCP nozzle fatigue evaluation. It is assumed that the fatigue crack growth is also negligible for the other remaining design transients, since the fatigue usage factor contribution for those transients is negligible. It should be noted that the mechanical loadings are included in the transient stresses. A high-cycle pump vibration transient has also been considered, and determined to have stress intensity factor that are below the threshold needed for fatigue crack growth as described later in this report (Section 2.2.5).

Table 4: Limiting Transients Used for the FCG Analysis

a,c,e

2.2.2 Residual Stresses

For the FCG analysis, the welding residual stresses at the SG to RCP suction nozzle DM weld are also considered along with the transient stresses. The inclusion of residual stresses will not change the range of stress intensity factor for the fatigue crack growth calculations; however, it will affect the Load Ratio (R) in the FCG equation (see Section 2.2.5). [

]a,c,e

2.2.3 Fatigue Crack Growth Analysis

In order to determine the growth of a postulated flaw after 60 years, a fatigue crack growth analysis is completed. Fatigue crack growth is the only credible mechanism for crack growth in the material between the SG and RCP since both the weld and the base metals have very low susceptibility to PWSCC, especially since the outside postulated surface flaw is not exposed to the reactor coolant loop environment. The fatigue crack growth analysis procedure involves postulating an initial flaw at the weld region and predicting the growth of that flaw due to an imposed series of loading transients. The input required for a fatigue crack growth analysis is essentially the information necessary to calculate the range of crack tip stress intensity factor, which depends on the crack size and shape, geometry of the structural component where the crack is postulated, and the applied cyclic stresses. Provided below is the methodology used to calculate the stress intensity factor for the axial and circumferential surface flaws.

2.2.4 Generation of Crack Tip Stress Intensity Factors

The FCG analysis in this letter involves calculating growth for a flaw on the outside surface of the SG to RCP suction nozzle DM weld, for an axial (AR = 2) and circumferential (AR = 6) flaw. The aspect ratio of 2 is reasonable because the axial flaw growth is limited to the width of the DM weld configuration, and an aspect ratio of 6 for postulated circumferential flaw is typical for fracture mechanics analyses. The postulated flaws are subjected to cyclic loads due to the transients and residual stresses described previously. The inputs required for the fatigue crack growth analysis is the range in stress intensity factor, ΔK, and the R ratio, K_{min}/K_{max}.

The stress intensity factors expression for surface flaws utilizes a representation of the actual stress profile rather than a linearization between data points. The stress distribution profiles are represented by a cubic polynomial:

$$\sigma (x) = A_0 + A_1 x + A_2 x^2 + A_3 x^3$$

where:

A_0, A_1, A_2, and A_3 are the stress profile curve fitting coefficients,

x is the distance from the wall surface where the crack initiates, and

σ is the stress perpendicular to the plane of the crack.

The stress intensity factor expression for semi-elliptical flaw shapes was used. The methodology for calculating the crack tip stress intensity factors is documented in an ASME publication [5] for axial flaws. The stress intensity factor from [5] can also be used conservatively for circumferentially oriented flaws. When evaluating axial and circumferential flaws, semi-elliptical surface flaws with aspect ratios (flaw length/flaw depth) of 2 for axial flaws and 6 for circumferential flaws are considered. Stress intensity factors can be expressed in the general form as follows:

$$K_I = \left(\frac{\pi a}{Q}\right)^{0.5} \sum_{j=0}^{3} G_j\,(a/c, a/t, t/R, \phi)\, A_j\, a^j$$

where:

a:	crack depth
c:	half of the crack length along the surface
t:	wall thickness
R:	inside radius of the component
A_j:	coefficients A_0, A_1, A_2, and A_3 for the stress profile cubic fit
ϕ:	angular position of a point of the crack front (ϕ= 0° at the deepest point; 90° at the surface point)
G_j:	G_0, G_1, G_2, G_3 are boundary correction factors provided in [5] for axial and used conservatively for circumferential flaws
Q:	shape factor of an elliptical crack. Q is approximated by: $Q = 1 + 1.464(a/c)^{1.65}$ for $a/c \leq 1$, or $Q = 1 + 1.464(c/a)^{1.65}$ for $a/c > 1$

2.2.5 Fatigue Crack Growth Rate

Once R (load ratio = K_{min}/K_{max}) and ΔK are calculated, the crack growth due to any given stress cycle can be calculated for each transient. This increment of crack growth is then added to the original crack size, and the analysis proceeds to the next transient.

Fatigue crack growth for each transient for a given time interval and number of cycles (N) can be computed using the following equation:

New Crack Depth = Initial Crack Depth + Incremental Crack Depth

with the incremental crack depth, Δa, given by:

$$\Delta a = C\,(\Delta K)^n\,N$$

The procedure is continued in this manner until all the transients known to occur in the period of evaluation have been analyzed. The design transient load cycles used in the analysis for the *AP1000* SG to RCP suction nozzle weld are listed in Table 4. The above equation is the most fundamental form of fatigue crack growth law, where C and n are material constants.

The general fatigue crack growth rate for materials in air environments is given by the equation of the type:

$$\frac{da}{dN} = F_{weld}\, C(T)\, S(R)\, (\Delta K)^n$$

where:

$C(T)$	=	Scaling Factor for Temperature Effects
$S(R)$	=	Scaling Factor for Load Ratio Effects
F_{weld}	=	Factor for Weld Material
ΔK	=	Stress Intensity Factor Range = K_{max} - K_{min}
R	=	Load Ratio K_{min} / K_{max}
K_{max}	=	Maximum Stress Intensity Factor
K_{min}	=	Minimum Stress Intensity Factor
da/dN	=	Crack Growth Rate in Environment
n	=	Crack Growth Law Exponent

The fatigue crack growth is performed for the Alloy 152/52 weld material between the SG and RCP suction nozzle. Note that the buttering on the steam generator outlet nozzle is Alloy 152/52, and the weld is Alloy 52. The FCG reference curves for Alloy 152/52 have not been developed in Section XI of the ASME Code; therefore, information available in NUREG/CR-6907 [6] is used. According to [6], in an air environment the Alloy 52 and Alloy 182 weld is approximately 2 times the Alloy 600 FCG rate in air. Due to limited number of test data for Alloy 152 in air environment, Reference [6] concludes that a factor of 2 on the Alloy 600 in air can be used to approximate the Ni-alloy welds, such as Alloy 152/52, FCG rate in air. It should be noted that the buttering on the RCP pump suction nozzle is stainless steel; however, the crack growth results for the Ni-alloy in air are more limiting than the stainless steel material in air (FCG curves for stainless steel based on ASME Section XI Appendix C).

Thus, the crack growth evaluation used herein are based on the FCG rate expression for Alloy 600 in air in SI units with a factor of 2 to represent the Alloy 152/52 weld in air environment [6]:

$$\frac{da}{dN} = F_{weld}\, C(T)\, S(R)\, (\Delta K)^n$$

$$C(T) = 4.835 \text{ X } 10^{-14} + (1.622 \text{ x } 10^{-16})T - (1.490 \text{ x } 10^{-18})T^2 + (4.355 \text{ x } 10^{-21})T^3$$

$$S(R) = (1 - 0.82R)^{-2.2}$$

$$F_{weld} = 2$$

where:

T	= Temperature (°C)
ΔK	= Stress Intensity Factor Range, K_{max} - K_{min}, MPa$\sqrt{m}$
K_{max}	= Maximum Stress Intensity Factor, MPa$\sqrt{m}$
K_{min}	= Minimum Stress Intensity Factor, MPa$\sqrt{m}$
n	= Crack Growth Law Exponent (= 4.1)
R	= Load Ratio, K_{min} / K_{max}
$\frac{da}{dN}$	= Crack Growth Rate in Environment, m/Cycle

[

]^{a,c,e} As such, the stress profile and stress range through the DM weld thickness due to RCP vibrations will be small. The stresses that are produced by the vibration are below the endurance limit of the Alloy 152/52 DM weld. Furthermore, the range in stress intensity factors for pump vibrations are less than the $\Delta K_{threshold}$. Therefore, the contribution of RCP vibrations to the FCG analysis would be negligible.

2.2.6 Fatigue Crack Growth Charts

The fatigue crack growth charts (Figures 2 and 3) are constructed by plotting the fatigue crack growth results over a period of 60 years. The flaw depth to through-wall thickness ratio (a/t) is plotted as the ordinate, and time is plotted as the abscissa. The charts are generated for the SG to RCP suction nozzle DM weld for an outside surface axial flaw (AR = 2) and circumferential flaw (AR = 6) as shown in Figures 2 and 3 respectively. The fatigue crack growth results are compared to the maximum allowable end-of-evaluation flaw size. The maximum allowable flaw size is tabulated in Table 3 for the axial and circumferential flaws, and also plotted in Figures 2 and 3. The initial flaw size is a sufficiently large postulated flaw which would not reach the maximum allowable flaw size in 60 years.

As shown in Figure 2 for an axial flaw, even a 60 percent through the wall thickness flaw would not reach the maximum allowable flaw size in 60 years. Figure 3 for circumferential flaws shows that a postulated flaw as large as 30 percent through the wall thickness flaw would not reach the maximum allowable flaw size in 60 years. Any initial axial and circumferential flaw sizes less than the 60 and 30 percent of the wall thickness, respectively, are encompassed by these curves and will not grow to the maximum allowable flaw size in 60 years. The large axial and circumferential surface flaw sizes described above do not exist in the eight (8) DM welds of Vogtle Units 3 and 4, and six (6) of the eight (8) DM welds of V.C. Summer Units 2 and 3 as evidenced by the ASME Section III fabrication PT examination results. The remaining two (2) DM welds in V.C. Summer Unit 3 will not contain surface flaws as described above given the allowable standards of ASME Code Section III, NB-5000. Surface flaws will be detected by the regular ISI surface examinations of the outside surface of the SG to RCP suction nozzle DM weld.

The stress intensity factor correlations for an embedded flaw are lower than that for surface flaws. Therefore, for a given through-wall stress distribution, it can be concluded that the fatigue crack growth for outside surface flaws is more limiting than the embedded flaws due to higher stress intensity factor. [

]^{a,c,e} Thus, all the UT detected indications are below the ASME Section XI IWB-3500 Allowable Standards. Furthermore, the initial surface flaw sizes used in the crack growth analysis in Figures 2 and 3 bound not only all the UT detected indications in the existing SG to RCP welds that have been inspected, but also bound the ASME Section XI allowable flaw sizes. Therefore, for the remaining two V. C. Summer Unit 3 SG A DM welds that have not been inspected, it can be conservatively assumed to have initial embedded flaw sizes (a/t) no larger than the ASME Section XI allowable flaw sizes in the (2a/t) range of 15.2% (0.845-inch) to 17.8% (0.99-inch) of the wall thickness per ASME Section XI Table IWB-3514-2 (1998 edition 2000 addenda). Thus, the initial surface flaw sizes used in Figures 2 and 3 will also bound the remaining welds that are not yet inspected.

In conclusion, all detected indications and any other potential fabrication indications are within the ASME Section XI allowable standards (1998 edition 2000 addenda), and moreover bounded by the initial flaw sizes considered in the crack growth analysis performed in this report for the design life of the plant (60 years).

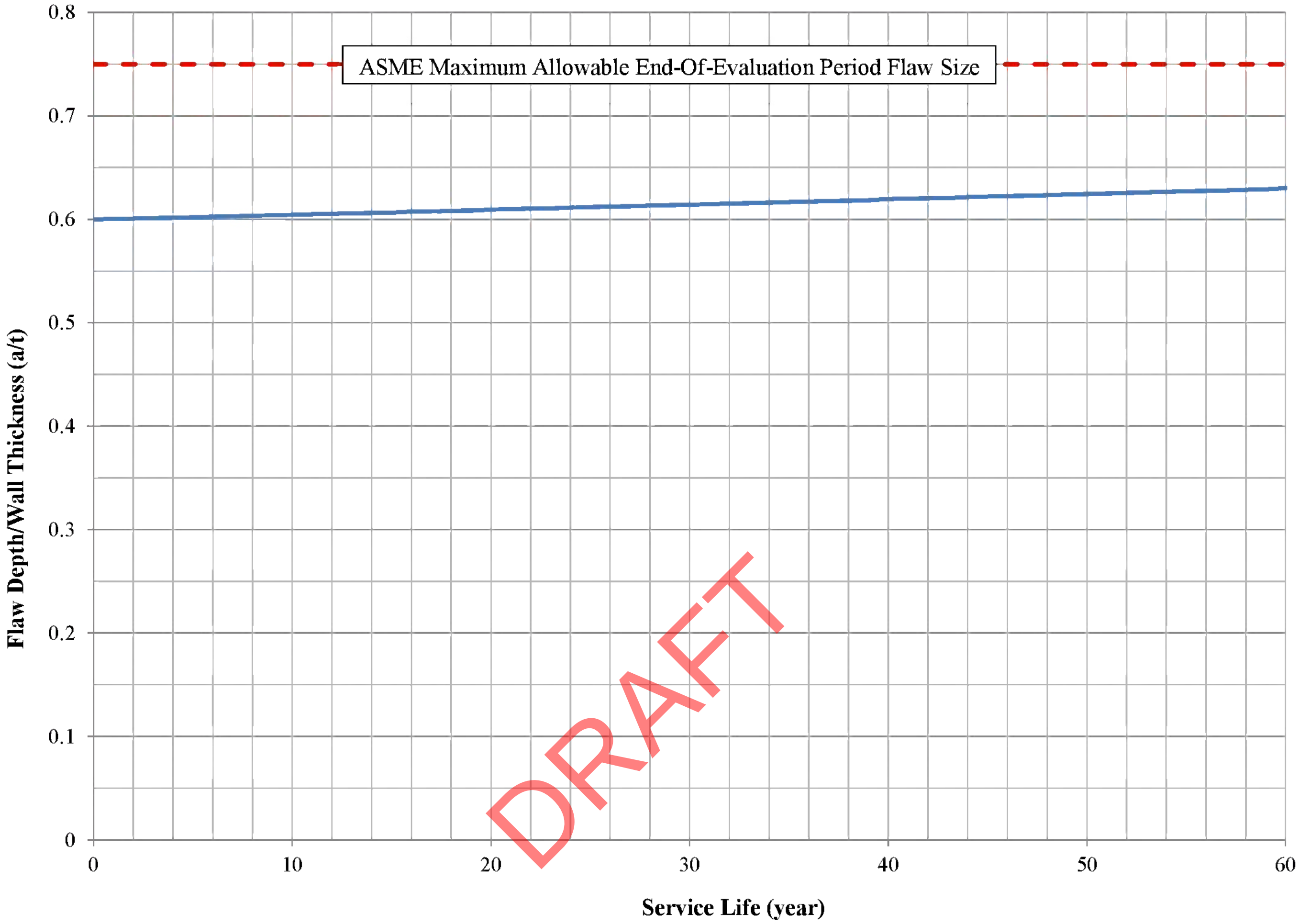

Figure 2: Crack Growth Chart for the *AP1000* Steam Generator to Reactor Coolant Pump Suction Nozzle Dissimilar Metal Weld, Outside Surface Axial Flaw with Aspect Ratio = 2

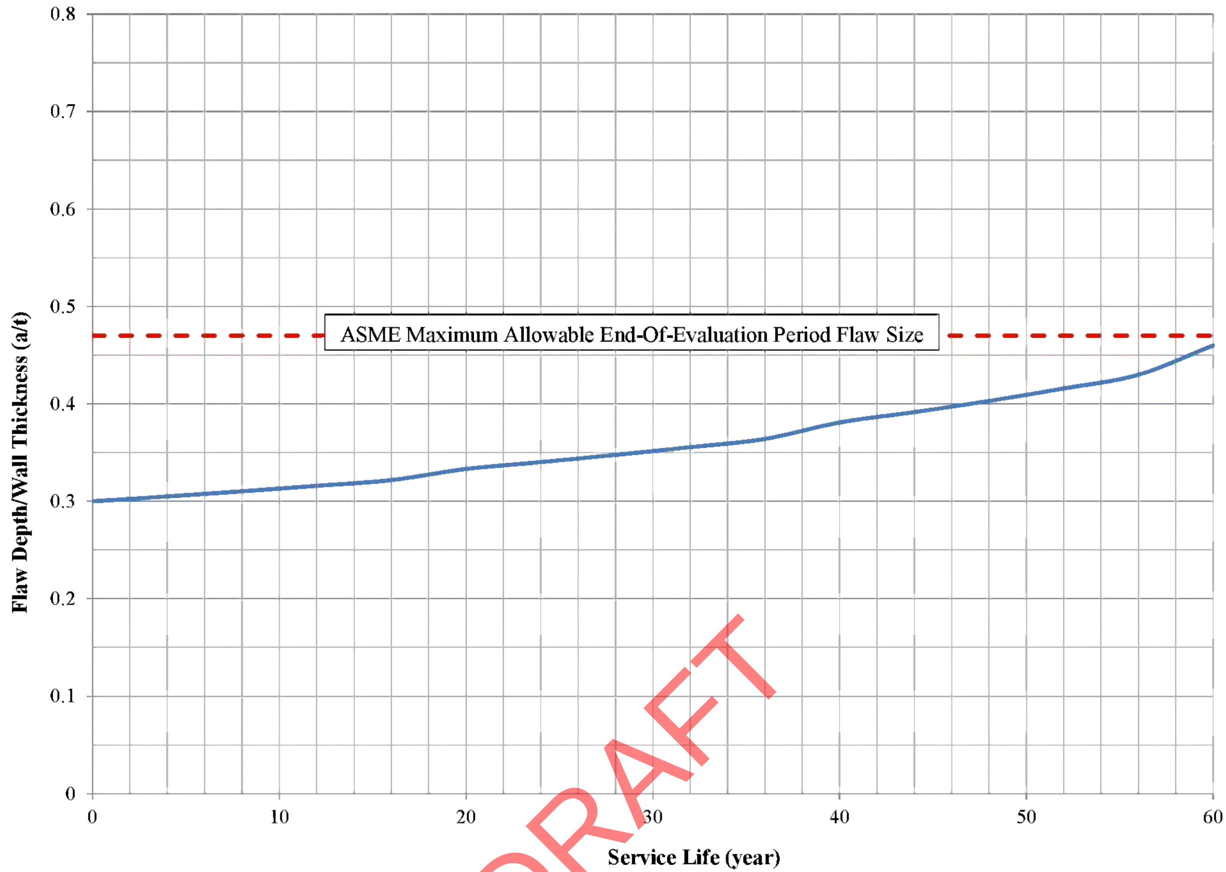

Figure 3: Crack Growth Chart for the *AP1000* Steam Generator to Reactor Coolant Pump Suction Nozzle Dissimilar Metal Weld, Outside Surface Circumferential Flaw with Aspect Ratio = 6

3.0 Section III Design Evaluation for Steam Generator to Pump DM Weld

The goal of the discussion herein on ASME Section III design evaluation [4] is to supplement the primary assessment provided in Section 2 based on ASME Section XI flaw tolerance analysis. The aim here is to provide a brief summary of the primary and secondary stress analyses, including the fatigue usage and ASME Section III Appendix G fracture mechanics (low alloy ferritic steel region) results.

The ASME Section III evaluations for the SG primary outlet nozzle and the SG to RCP suction nozzle DM weld are based on the pressure loads, thermal loads, and external mechanical loads obtained using finite element analysis and also based on strength of materials equations. It should be noted that the loads due to pump fluctuations and vibrations were included in the evaluation for all conditions to determine the fatigue usage factors. Furthermore, the high cycle loading due to pump vibrations was also evaluated separately for an infinite number of cycles to determine the maximum alternating stress. All alternating stress intensities for this high cycle pump loading are below one-half the endurance limits.

Provided in Table 5 below are select results of the ASME Section III allowable stress limits and fatigue usage for the DM weld. Note that all ASME Section III design criteria are satisfied for this region. Furthermore, the low fatigue usage shown in Table 5 demonstrates low susceptibility for any fatigue crack initiations at either inside or outside surfaces.

Table 5: ASME Section III Select Results for DM Weld Location

a,c,e

[ ]

A non-ductile fracture mechanics evaluation was also performed per ASME Section III Appendix G for the SG nozzle ferritic material adjacent to the DM weld. The non-ductile brittle fracture evaluation per ASME Section III Appendix G can be used as a conservative fracture mechanics assessment of the more ductile Alloy 152/52 weld. The Appendix G results for the ferritic location in the SG next to the DM weld are shown in Table 6 below.

Results from Table 6 can be used to demonstrate the structural stability of the region at and around the DM weld based on a large postulated flaw of 25 percent of the wall thickness with a length-to-depth (aspect ratio) of 6, as required by ASME Section III Appendix G design analysis. Normal, upset, emergency, test, and faulted condition transients were all evaluated. The limiting transients within these service conditions were chosen to be those transients that result in low metal temperatures and high stresses, which would give the worst brittle fracture assessment. The most limiting hoop and axial stresses from either the inside or outside surface were used in the Appendix G evaluation; as a result, the evaluation cover postulated flaws on the inside or outside, and axial or circumferential flaw configurations. The lower bound fracture toughness values were used based on the limiting temperature and material reference nil-ductility temperature (RT_{NDT}) based on the design specification. Table 6 provides the ASME Section III results for the postulated 1/4T flaw for the ferritic steel location adjacent to the DM weld. Based on the results in Table 6, it is demonstrated that the ferritic steel and the adjacent ductile DM weld would be flaw tolerant for large postulated flaws based on the ASME Section III Appendix G non-ductile evaluations.

Table 6: ASME Section III Appendix G Results[1] for SG Nozzle Ferritic Steel Location Adjacent to the DM weld

a,c,e

4.0 Conclusions

The objective of this letter report is to provide responses to the NRC RAI (Reference 1) to support justification of a volumetric inspection of the inner 1/3 of the weld, with a surface examination of the inner and outer weld surfaces by ET and PT, respectively, and to demonstrate that this is an acceptable alternative to examining the full weld volume during the ISI. The responses to the NRC RAI were based on two separate assessments that have been performed for this particular region of interest. The first assessment was based on ASME Section XI flaw tolerance analysis, and the second assessment is based on the ASME Section III design evaluation.

The ASME Section XI flaw tolerance evaluation is provided in Section 2 of this report. Postulated outside surface axial and circumferential flaws with aspect ratios of 2 and 6, respectively, were evaluated at the SG to RCP suction nozzle DM weld locations. AP1000 specific geometry, loadings, and material properties were considered in the maximum end-of-evaluation period flaws and the fatigue crack growth analysis. [

]^{a,c,e} Thus, all detected indications are well within the ASME Section XI allowable standards and the fracture mechanics calculations performed in this report.

Section 3 of this report provides the ASME Section III design evaluation that was performed for the DM weld and the surrounding low alloy steel region. Based on the design criteria, all requirements of the ASME Section III code were met for this region based on the primary and secondary stress analyses, and fatigue usage calculations. The fatigue usage at the DM weld region is very low at both the inside and outside surface, and this region has acceptably low susceptibility to crack initiations for the design life of the plant. The non-ductile ASME Section III Appendix G fracture mechanics evaluation was also performed and shown to be acceptable for the low alloy ferritic steel of the steam generator. The ferritic steel material is considered in the Appendix G evaluation since it is susceptible to brittle fracture, whereas the DM weld is more ductile than the SG base metal, and therefore is not required to be considered in the Appendix G evaluation. Thus, per the design ASME Section III fracture mechanics, it is also demonstrated that the DM weld region is flaw tolerant for large flaws of size 25% of the wall thickness with an aspect ratio of 6.

In conclusion, based on the ASME Section XI and III discussions provided in this report, it is demonstrated that with the volumetric and surface examinations performed during fabrication, and the outer surface examinations and inner 1/3 of the wall thickness surface and volumetric examinations to be performed every ISI, a volumetric inspection of the entire DM weld that includes the outer 2/3 of the wall thickness is not necessary. This conclusion is based on a fracture mechanics evaluation per ASME Section XI and ASME Section III Appendix G assessments. It was demonstrated that the outer 2/3 of the wall thickness is flaw tolerant for the design life of the plant, and that the initiation of any active degradation of the weld would not be expected to occur over the licensed lifetime of the plant.

5.0 References

1) NRC email from Steven Downey to Chandu Patel, "Requests for Additional Information related to Vogtle Alternative Request VEGP3&4-PSI-ALT-05," dated: August 5, 2016, (NRC ADAMS: ML16218A439).

2) ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2007 Edition including 2008 Addenda.

3) Request for Alternative Requirement for Preservice Inspection at Vogtle Units 3 & 4 and V.C. Summer Units 2 and 3.

   a. VEGP 3&4-PSI-ALT-05, "Southern Nuclear Operating Company, Vogtle Electric Generating Plant Units 3 and 4 Request for Alternative: Preservice Inspection Requirements for Steam Generator Nozzle to Reactor Coolant Pump Casing Welds," June 24, 2016, (NRC ADAMS: ML16176A312).

   b. NND-16-0246, "Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3, Docket Numbers 52-027 and 52-028, Request for Alternative: Preservice Inspection Requirements for Steam Generator Nozzle to Reactor Coolant Pump Casing Welds," July 7, 2016, (NRC ADAMS: ML16189A312).

4) ASME Code Section III, "Rules for Construction of Nuclear Power Plant Components," 1998 Edition including 2000 Addenda.

5) Raju, I.S. and Newman, J.C., "Stress Intensity Factor Influence Coefficients for Internal and External Surface Cracks in Cylindrical Vessels," ASME Publication PVP, Volume 58, 1982, pp. 37-48.

6) NUREG/CR-6907, ANL-04/3 "Crack Growth Rates of Nickel Alloy Welds in a PWR Environment," U.S. Nuclear Regulatory Commission (Argonne National Laboratory), May 2006.

7) NUREG/CR-6383, ANL-95/37, "Corrosion Fatigue of Alloys 600 and 690 in Simulated LWR Environments," April 1996.

8) Nomura, Y., Yamamoto, K., Hojo, K., ASME PVP2014-28098, "Fatigue Crack Growth Rates for Nickel Base Alloys in Air," Proceedings of ASME 2014 Pressure Vessels & Piping Conference, Anaheim, California, USA, July 20-24, 2014.

9) U.S. Nuclear Regulatory Commission Letter Dated May 19, 2000, License Renewal Issue No. 98-0030, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components."

10) Doosan Radiographic Test Reports for Vogtle Unit 3 SG A (R150209-017-001 and R150209-020-001) and SG B (R141201-043-001 and R141201-044-001) and for Vogtle Unit 4 SG A (R150430-005-001 and R150430-006-001) and SG B (R150709-015-001 and R150709-016-001).

11) Doosan Ultrasonic Test Reports for Vogtle Unit 3 SG A (U150212-042-001) and SG B (U141201-029-001) and Vogtle Unit 4 SG A (U150417-005-001) and SG B (U150709-036-001).

12) Doosan Radiographic Test Reports for V.C. Summer Unit 2 SG A (R140805-021-001 and R140805-022-001) and SG B (R140902-014-001 and R140901-020-001) and for V.C. Summer Unit 3 SG B (R151116-014-001 and R151116-015-001).

13) Doosan Ultrasonic Test Reports for V.C. Summer Unit 2 SG A (U140730-037-001) and SG B (U140829-002-001) and for V.C. Summer Unit 3 SG B (U151119-012-001).

10CFR50.55a Alternative VEGP3&4-ISI1-ALT-17
Proposed Alternative in Accordance with 10CFR50.55a(z)(1)
--Alternative Provides Acceptable Level of Quality and Safety--
Revision 0

Enclosure 2

Westinghouse LTR-PAFM-16-59-P, NRC RAI Response Regarding Inspection of AP1000 Vogtle Units 3 & 4 and V.C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Suction Nozzle Weld (Proprietary) Rev. 1

This Enclosure includes proprietary information, and is withheld from the public in accordance with 10 CFR 2.390.

10CFR50.55a Alternative VEGP3&4-ISI1-ALT-17
Proposed Alternative in Accordance with 10CFR50.55a(z)(1)
--Alternative Provides Acceptable Level of Quality and Safety--
Revision 0

Enclosure 3

Westinghouse LTR-PAFM-17-6, Rev. 0, NP-Enclosure 1 (Non-Proprietary) - Additional Information Regarding AP1000 Vogtle Units 3 & 4 and V. C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Suction Nozzle Weld Flaw Evaluation

Attachment B

NP-Attachment (Non-Proprietary)

Additional Information Regarding AP1000 Vogtle Units 3 & 4 and V. C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Suction Nozzle Weld Flaw Evaluation

This document contains Westinghouse Electric Company LLC proprietary information and data which has been identified by brackets. Coding (a,c,e) associated with the brackets sets forth the basis on which the information is considered proprietary. These codes are listed with their meanings in WCAP-7211 Revision 8 (September 2015), "Proprietary Information and Intellectual Property Management Policies and Procedures."

The proprietary information and data contained in this report were obtained at considerable Westinghouse expense and its release could seriously affect our competitive position. This information is to be withheld from public disclosure in accordance with the Rules of Practice 10CFR2.390 and the information presented herein is to be safeguarded in accordance with 10CFR2.903. Withholding of this information does not adversely affect the public interest.

This information has been provided for your internal use only and should not be released to persons or organizations outside the Directorate of Regulation and the ACRS without the express written approval of Westinghouse Electric Company LLC. Should it become necessary to release this information to such persons as part of the review procedure, please contact Westinghouse Electric Company LLC, which will make the necessary arrangements required to protect the Company's proprietary interests.

The proprietary information in the brackets has been deleted in this report. The deleted information is provided in the proprietary version of this report.

Additional Information Regarding *AP1000*® Vogtle Units 3 & 4 and V. C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Suction Nozzle Weld Flaw Evaluation

1. Additional information regarding justification of the constant flaw aspect ratios (AR) of 2 and 6 for axial and circumferential flaws in Reference 1 is provided below.

   Axial Flaw

   For the postulated axial flaw, the analysis in Reference 1 considers an AR = 2 (flaw length/flaw depth). This flaw shape is based on the understanding that in the DM (dissimilar metal) weld, the axial flaw will follow the characteristic shape of the DM weld width and thickness. The DM weld inspection volume consists of the width of the dissimilar metal weld and the Heat Affected Zone (HAZ) – see Figure 1. The width of the dissimilar metal weld is approximately [ ]a,c,e based on the AP1000 steam generator and pump drawings. The inspection volume includes the ¼" examination zones adjacent to the weld on either side to account for the HAZ. Therefore, the total width of the DM weld inspection region is approximately [ ]a,c,e. The weld thickness is [ ]a,c,e (Reference 1). Therefore the shape or aspect ratio of the weld is 0.6 [ ]a,c,e, thus an aspect ratio of 2 is sufficiently large to account for any existing and hypothetical axial flaws.

   Also, based on the fabrication ultrasonic testing (UT) results, the flaw aspect ratios that are observed are bounded by the analyzed aspect ratio of 2 for axial flaws. For example, based on the available axial flaw UT results for the AP1000 Vogtle and V. C. Summer units, [

   ]a,c,e This particular detected aspect ratio is bounded by the axial flaw aspect ratio of 2 analyzed in Reference 1.

Trademark Note:

Circumferential Flaw

For the postulated circumferential flaw, the analysis in Reference 1 considers AR = 6 (flaw length/flaw depth). This flaw shape is a typical aspect ratio for various applications in fracture mechanics. For instance, the Pressure Temperature (P-T) limits evaluation in ASME Section XI Appendix G also considers postulated flaw shapes to have an aspect ratio of 6:1. Industry experiences have also shown that the flaws found in-service are typically below AR = 6 (on the order of AR = 2 to 4 or even less). It should be noted that the AP1000 steam generator to pump DM weld region does not experience any high thermal stratification, as evident by the minimal fatigue usage discussed in Section 3 of Reference 1; therefore, there is low susceptibility for any fatigue crack initiations or propagation of existing fabrication indications. Therefore, the aspect ratio of 6 is sufficient to account for any existing and hypothetical circumferential flaws.

Also, based on the fabrication ultrasonic testing results, the circumferential flaw aspect ratios that are observed are bounded by the analyzed aspect ratio of 6 for circumferential flaws. For example, based on the available circumferential flaw UT results for the AP1000 Vogtle and V. C. Summer units, [

]^{a,c,e} This particular detected aspect ratio is bounded by the circumferential aspect ratio of 6 in the analysis (Reference 1).

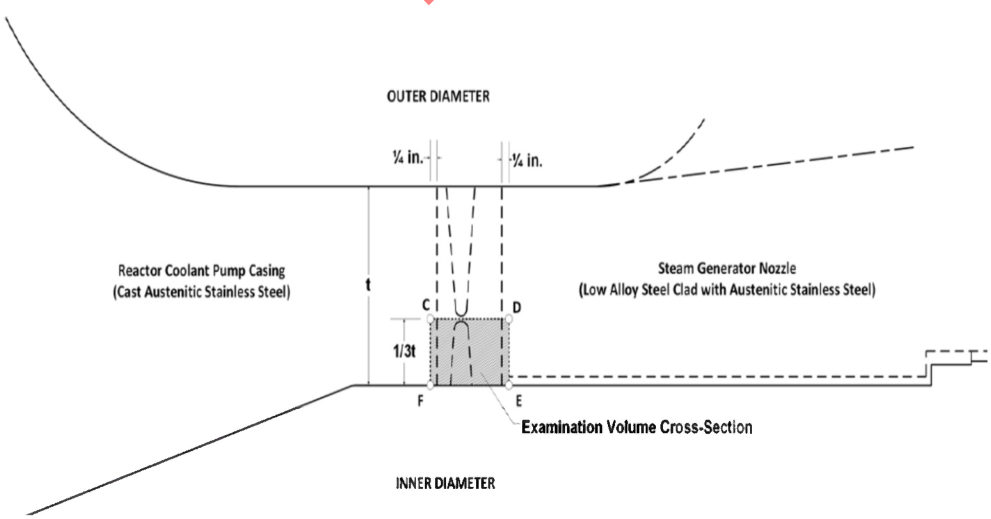

Figure 1: Schematic of AP1000 Steam Generator and Reactor Coolant Pump Inspection Region

2. Appendix C, Paragraph C-5300, requires that the allowable flaw depth and length be evaluated to determine acceptability. However, only flaw depth was evaluated. Additional information justifying this approach is provided below.

For the Appendix C-5000 evaluation, the evaluation is per fully-plastic fracture mechanics using limit load. The limiting allowable flaw parameter for failure of this type is the flaw depth which was reported in the analysis (Reference 1). The allowable flaw length was not reported as it is not the limiting flaw parameter. However, based on the maximum end-of-evaluation allowable flaw sizes that were calculated (see Table 1), the allowable axial and circumferential flaw lengths can be calculated by multiplying the allowable flaw depths by the aspect ratios (see Table 1).

Table 1: Maximum End-of-Evaluation Allowable Flaw Size, Depth, and Length

Flaw Orientation	AR (flaw length /flaw depth)	Maximum End of Evaluation Allowable Flaw size (a/t)	Thickness (in.)	Allowable Flaw Depth (in.)	Allowable Flaw Length (in.)
Axial	2	0.75 (Reference 1, Table 3)	[ ]a,c,e	[ ]a,c,e	[ ]a,c,e
Circumferential	6	0.47 (Reference 1, Table 3)	[ ]a,c,e	[ ]a,c,e	[ ]a,c,e

a = flaw depth, t = wall thickness

The axial and circumferential maximum end-of-evaluation allowable flaw lengths are [

]a,c,e Therefore, the detected flaw lengths are below the calculated maximum end-of-evaluation allowable flaw lengths.

If fatigue crack growth is considered, then the maximum allowable initial flaw depths and lengths for 60 years of growth are shown in Table 2.

Table 2: Maximum Allowable Initial Flaw Size, Depth, and Length Accounting for 60 Years of Fatigue Crack Growth

Flaw Orientation	AR	Maximum Allowable Initial Flaw Size for 60 Years (a/t)	Thickness (in.)	Allowable Flaw Depth-60 years (in.)	Allowable Flaw Length-60 years (in.)
Axial	2	0.60 (Reference 1, Fig 2)	[ ]a,c,e	[ ]a,c,e	[ ]a,c,e
Circumferential	6	0.30 (Reference 1, Fig 3)	[ ]a,c,e	[ ]a,c,e	[ ]a,c,e

a = flaw depth, t = wall thickness

The maximum allowable initial flaw lengths for 60 years are [

]a,c,e

Reference:

1. LTR-PAFM-16-59-P, Revision 1, "NRC RAI Response Regarding Inspection of AP1000 Vogtle Units 3 & 4 and V. C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Suction Nozzle Weld," November 2016.

10CFR50.55a Alternative VEGP3&4-ISI1-ALT-17
Proposed Alternative in Accordance with 10CFR50.55a(z)(1)
--Alternative Provides Acceptable Level of Quality and Safety--
Revision 0

Enclosure 4

Westinghouse LTR-PAFM-17-6, Rev. 0, P-Enclosure 2 (Proprietary) - Additional Information Regarding AP1000 Vogtle Units 3 & 4 and V. C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Suction Nozzle Weld Flaw Evaluation

This Enclosure includes proprietary information, and is withheld from the public in accordance with 10 CFR 2.390.

Enclosure 5

Affidavit from Southern Nuclear Operating Company for Withholding Under 10 CFR 2.390

Affidavit of Jamie M. Coleman

1. My name is Jamie M. Coleman. I am the Regulatory Affairs Director, Vogtle Electric Generating Plant (VEGP) Units 3 & 4, for Southern Nuclear Operating Company (SNC). I have been delegated the function of reviewing proprietary information sought to be withheld from public disclosure and am authorized to apply for its withholding on behalf of SNC.
2. I am making this affidavit on personal knowledge, in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations, and in conjunction with Westinghouse Electric Company documents LTR-PAFM-16-59-P, Revision 1, which is the Westinghouse Electric Company document titled, "NRC RAI Response Regarding Inspection of AP1000 Vogtle Units 3 & 4 and V. C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Suction Nozzle Weld", and LTR-PAFM-17-6, P-Attachment, Revision 0, which is the Westinghouse Electric Company Document titled, "Additional Information Regarding AP1000 Vogtle Units 3 & 4 and V. C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Section Nozzle Weld Flaw Evaluation." I have personal knowledge of the criteria and procedures used by SNC to designate information as a trade secret, privileged, or as confidential commercial or financial information.
3. Based on the reason(s) at 10 CFR 2.390(a)(4), this affidavit seeks to withhold from public disclosure Westinghouse Electric Company documents LTR-PAFM-16-59-P, Revision 1, and LTR-PAFM-17-6, P-Attachment, Revision 0, which are included in Enclosure 2 to SNC letter ND-22-0817 for VEGP Units 3 and 4, Request for Alternatives: Alternative Inspection Requirements for Steam Generator Nozzle to Reactor Coolant Pump Casing Welds (VEGP 3&4-ISI1-ALT-17), and Alternative for Use of Code Case N-648-2 for Inservice Inspection of the Reactor Vessel Nozzle Inside Radius Sections (VEGP 3&4-ISI1-ALT-18).

4. The following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.
    a. The information sought to be withheld from public disclosure has been held in confidence by SNC and Westinghouse Electric Company.
    b. The information is of a type customarily held in confidence by SNC and Westinghouse Electric Company and not customarily disclosed to the public.
    c. The release of the information might result in the loss of an existing or potential competitive advantage to SNC and/or Westinghouse Electric Company.
    d. Other reasons are identified in Enclosure 2 to SNC letter ND-22-0817 for VEGP Units 3 and 4, which includes Westinghouse Applications for Withholding Proprietary Information from Public Disclosure CAW-16-4509 and CAW-17-4534, accompanying Affidavits, Proprietary Information Notices, and Copyright Notices, and those reasons are incorporated here by reference.
5. Additionally, release of the information may harm SNC because SNC has a contractual relationship with the Westinghouse Electric Company regarding proprietary information. SNC is contractually obligated to seek confidential and proprietary treatment of the information.
6. The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.
7. To the best of my knowledge and belief, the information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method.

I declare under penalty of perjury that the foregoing is true and correct.

Jamie M. Coleman
Jamie M. Coleman

Executed on 10/19/2022
Date

10CFR50.55a Alternative VEGP3&4-ISI1-ALT-17
Proposed Alternative in Accordance with 10CFR50.55a(z)(1)
--Alternative Provides Acceptable Level of Quality and Safety--
Revision 0

Enclosure 6

Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-16-4509, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice

Westinghouse

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Direct tel: (412) 374-4643
Direct fax: (724) 940-8560
e-mail: greshaja@westinghouse.com

CAW-16-4509

November 18, 2016

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-PAFM-16-59-P, Revision 1, "NRC RAI Response Regarding Inspection of *AP1000* Vogtle Units 3 & 4 and V.C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Suction Nozzle Weld" (Proprietary)

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC ("Westinghouse"), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-16-4509 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Southern Nuclear Operating Company and South Carolina Electric and Gas Company.

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-16-4509, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

Very truly yours,

James A. Gresham, Manager

Regulatory Compliance

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, James A. Gresham, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse"), and that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

James A. Gresham, Manager
Regulatory Compliance

Date: 11/18/16

(1) I am Manager, Regulatory Compliance, Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-PAFM-16-59-P, Revision 1, "NRC RAI Response Regarding Inspection of *AP1000* Vogtle Units 3 & 4 and V.C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Suction Nozzle Weld" (Proprietary), for submittal to the Commission, being transmitted by Southern Nuclear Operating Company and South Carolina Electric & Gas Company letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with technical justification to support inspection coverage for *AP1000*® plant steam generator to reactor coolant pump dissimilar metal weld at Vogtle Units 3 and 4 and V.C. Summer Units 2 and 3, and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to:

(i) Provide technical justification to support inspection coverage for *AP1000* plant steam generator to reactor coolant pump dissimilar metal weld at Vogtle Units 3 and 4 and V.C. Summer Units 2 and 3.

(b) Further this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of providing technical justification to support extended volumetric examination interval for reactor vessel nozzle to safe end dissimilar metal welds.

(ii) Westinghouse can sell support and defense of industry guidelines and acceptance criteria for plant-specific applications.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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The reports transmitted herewith each bear a Westinghouse copyright notice. The NRC is permitted to make the number of copies of the information contained in these reports which are necessary for its internal use in connection with generic and plant-specific reviews and approvals as well as the issuance, denial, amendment, transfer, renewal, modification, suspension, revocation, or violation of a license, permit, order, or regulation subject to the requirements of 10 CFR 2.390 regarding restrictions on public disclosure to the extent such information has been identified as proprietary by Westinghouse, copyright protection notwithstanding. With respect to the non-proprietary versions of these reports, the NRC is permitted to make the number of copies beyond those necessary for its internal use which are necessary in order to have one copy available for public viewing in the appropriate docket files in the public document room in Washington, DC and in local public document rooms as may be required by NRC regulations if the number of copies submitted is insufficient for this purpose. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

Southern Nuclear Operating Company and South Carolina Electric & Gas Company

Letter for Transmittal to the NRC

The following paragraphs should be included in your letter to the NRC Document Control Desk:

Enclosed are:

1. LTR-PAFM-16-59-P, Revision 1, "NRC RAI Response Regarding Inspection of *AP1000* Vogtle Units 3 & 4 and V.C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Suction Nozzle Weld" (Proprietary)

2. LTR-PAFM-16-59-NP, Revision 1, "NRC RAI Response Regarding Inspection of *AP1000* Vogtle Units 3 & 4 and V.C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Suction Nozzle Weld" (Non-Proprietary)

Also enclosed is the Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-16-4509, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice.

As Item 1 contains information proprietary to Westinghouse Electric Company LLC, it is supported by an Affidavit signed by Westinghouse, the owner of the information. The Affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-16-4509 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

10CFR50.55a Alternative VEGP3&4-ISI1-ALT-17
Proposed Alternative in Accordance with 10CFR50.55a(z)(1)
--Alternative Provides Acceptable Level of Quality and Safety--
Revision 0

Enclosure 7

Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-17-4534, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice

Westinghouse

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Direct tel: (412) 374-4643
Direct fax: (724) 940-8560
e-mail: greshaja@westinghouse.com

CAW-17-4534

January 19, 2017

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-PAFM-17-6 P-Attachment, "Additional Information Regarding **AP1000** Vogtle Units 3 & 4 and V.C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Suction Nozzle Weld Flaw Evaluation" (Proprietary)

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC ("Westinghouse"), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-17-4534 signed by the owner of the proprietary information, Westinghouse. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Southern Nuclear Operating Company and South Carolina Electric and Gas Company.

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-17-4534, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

J.A. Gresham

James A. Gresham, Manager
Regulatory Compliance

CAW-17-4534

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, James A. Gresham, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse") and declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

Executed on: 1/19/17

JAGresham

James A. Gresham, Manager
Regulatory Compliance

(1) I am Manager, Regulatory Compliance, Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-PAFM-17-6, P-Attachment, "Additional Information Regarding **AP1000** Vogtle Units 3 & 4 and V.C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Suction Nozzle Weld Flaw Evaluation" (Proprietary), for submittal to the Commission, being transmitted by Southern Nuclear Operating Company and South Carolina Electric & Gas Company letter. The proprietary information as submitted by Westinghouse is that associated with technical justification to support inspection coverage for **AP1000**® [1]plant steam generator to reactor coolant pump dissimilar metal weld at Vogtle Units 3 and 4 and V.C. Summer Units 2 and 3, and may be used only for that purpose.

[1] **AP1000** is a trademark or registered trademark of Westinghouse Electric Company LLC, its affiliates and/or its subsidiaries in the United States of America and may be registered in other countries throughout the world. All rights reserved. Unauthorized use is strictly prohibited. Other names may be trademarks of their respective owners.

(a) This information is part of that which will enable Westinghouse to provide technical justification to support inspection coverage for **AP1000** plant steam generator to reactor coolant pump dissimilar metal weld at Vogtle Units 3 and 4 and V.C. Summer Units 2 and 3.

(b) Further, this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of providing technical justification to support extended volumetric examination interval for reactor vessel nozzle to safe end dissimilar metal welds.

(ii) Westinghouse can sell support and defense of industry guidelines and acceptance criteria for plant-specific applications.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and non-proprietary versions of a document, furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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Southern Nuclear Operating Company and South Carolina Electric & Gas Company

Letter for Transmittal to the NRC

The following paragraphs should be included in your letter to the NRC Document Control Desk:

Enclosed are:

1. LTR-PAFM-17-6 P-Attachment, "Additional Information Regarding **AP1000** Vogtle Units 3 & 4 and V.C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Suction Nozzle Weld Flaw Evaluation" (Proprietary)

2. LTR-PAFM-17-6 NP-Attachment, "Additional Information Regarding **AP1000** Vogtle Units 3 & 4 and V.C. Summer Units 2 & 3 Steam Generator to Reactor Coolant Pump Suction Nozzle Weld Flaw Evaluation" (Non-Proprietary)

Also enclosed are the Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-17-4534, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice.

As Item 1 contains information proprietary to Westinghouse Electric Company LLC ("Westinghouse"), it is supported by an Affidavit signed by Westinghouse, the owner of the information. The Affidavit sets forth the basis on which the information may be withheld from public disclosure by the Nuclear Regulatory Commission ("Commission") and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-17-4534 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

1. ASME Code Component(s) Affected

Code Class: 1

Reference: IWB-2500, Table IWB-2500-1
Regulatory Guide 1.147, Revision 20

Examination Category: B-D

Item Number: B3.100

Description: Alternative for Use of Code Case N-648-2 for Inservice Inspection of the Reactor Vessel Nozzle Inside Radius Sections

Component Number:

Unit 3	Unit 4
SV3-RPV-24A-101-IRS	SV4-RPV-24A-101-IRS
SV3-RPV-24B-101-IRS	SV4-RPV-24B-101-IRS
SV3-RPV-24C-101-IRS	SV4-RPV-24C-101-IRS
SV3-RPV-24D-101-IRS	SV4-RPV-24D-101-IRS
SV3-RPV-25A-102-IRS	SV4-RPV-25A-102-IRS
SV3-RPV-25B-102-IRS	SV4-RPV-25B-102-IRS
SV3-RPV-26A-103-IRS	SV4-RPV-26A-103-IRS
SV3-RPV-26B-103-IRS	SV4-RPV-26B-103-IRS

2. Applicable Code Edition

The First Interval of the Vogtle Electric Generating Plant (VEGP), Units 3 and 4 Inservice Inspection (ISI) Program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section XI, 2017 Edition.

3. Applicable Code Requirement

Subarticle IWB-2500 requires components specified in Table IWB-2500-1 to be examined. Table IWB-2500-1 requires a volumetric examination of all Reactor Vessel nozzle inside radius sections (IRS) each inspection interval (Examination Category B-D, Item Number B3.100). The examination volume is shown in Figure IWB-2500-7.

ASME Code Case N-648-2 (N-648-2), "Alternative Requirements for Inner Radius Examinations of Class 1 Reactor Vessel Nozzles, Section XI, Division 1," (conditionally approved for use under Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 20) provides an alternative to the ASME Section XI requirements stated above by allowing a VT-1 visual examination in lieu of the required volumetric examination.

Regulatory Guide 1.147, Revision 20, approved N-648-2, with the following condition:

> "This Code Case shall not be used to eliminate the preservice or inservice volumetric examination of plants with a combined operating license under 10CFR Part 52, or a plant that receives its operating license after October 22, 2015."

4. Reason for Request

In accordance with 10CFR50.55a(z)(1), relief is requested on the basis that the proposed alternative will provide an acceptable level of quality and safety.

Regulatory Guide 1.147, Revision 20, which conditionally approves Code Case N-648-2, does not allow VEGP to utilize the code case in order to perform a VT-1 visual examination in lieu of a UT examination as currently required in Table IWB-2500-1, Examination Category B-D, Item B3.100 during the Inservice Interval. It is the intention of Southern Nuclear Operating Company, Inc. (SNC) to adopt this Code Case for the current ISI Interval. The proposed alternative is requested to align the First ISI Interval with the performed Preservice Inspections and with the subsequent planned Inservice Inspections. This code case is used extensively in operating units.

5. Proposed Alternative and Basis for Use

SNC proposes to perform a VT-1 visual examination of the reactor vessel nozzle inside radius sections for the two outlet nozzles, the four inlet nozzles and the two direct vessel injection (DVI) nozzles using a remote underwater visual examination process that will be comparable to the historical preservice inspections. This visual examination will be conducted in accordance with the ASME BPV Code, Section XI, 2017 Edition. All of the requirements defined in Section 2, "Inservice Examinations," of Code Case N-648-2 will be applied.

The PSI Alternative VEGP3&4-PSI-ALT-07 was approved using Code Case N-648-1. Code Case N-648-2 has since superseded Code Case N-648-1 and is conditionally approved for use under Regulatory Guide 1.147. Code Case N-648-2 was revised to include use of the code case for preservice examinations. The PSI alternative addressed the NRC condition with N-648-1 by utilizing the ASME Section XI Table IWB-3512-1 acceptance criteria, and VEGP went beyond what would currently be required of Code Case N-648-2, in that a manual UT was performed as part of the preservice examination. Another concern with Code Case N-648-2, was that the staff requested a plant specific flaw tolerance be performed for the AP1000 nozzle at the inside radius corner. The PSI and this current ISI alternative satisfies that request by providing the flaw tolerance evaluation for the AP1000 nozzles. Therefore, Code Case N-648-2 is applicable for use and PSI examinations have been performed over and above what is currently requested under N-648-2, which provides more assurance to the integrity of the nozzles.

The proposed remote VT-1 visual examination will be performed in accordance with ASME Code, Section XI, which requires that a visual examination performed instead of an ultrasonic examination has a magnification that has a resolution sensitivity to resolve 0.044 inch (1.1 mm) lower case characters without an ascender or descender (e.g., a, e, n, v).

This technical basis addresses a VT-1 visual examination approach that includes a deterministic fracture assessment similar to that performed as a basis for Code Case N-648-2. The code alternative provides an acceptable level of quality and safety in accordance with 10CFR50.55a(z)(1) because the VT-1 visual examinations are sufficient to detect service-induced flaw mechanisms (fatigue) occurring at the inner diameter (ID) surface well before the nozzle experiences degradation of its structural integrity. The proposed remote VT-1 visual examination will cover essentially 100 percent of the ASME Code, Section XI, Figure IWB-2500-7(b) section M-N surface area for the inlet, outlet, and DVI nozzles.

<u>Fracture Assessment</u>

A fracture assessment was performed to determine the maximum initial flaw size that will not grow beyond the allowable end of evaluation period flaw size for the life of the plant (60 years) considering Level A/B/Test conditions which were limiting in comparison to the Level C/D/Test. The allowable end of evaluation period flaw sizes (depths) for the AP1000® inlet, outlet, and DVI nozzles were determined using both linear elastic fracture mechanics (LEFM) and elastic plastic fracture mechanics (EPFM) methods.

For the most limiting case, the DVI nozzle using the LEFM analysis, the acceptance criteria for the VT-1 visual examinations using the requirements of ASME Section XI are much more limiting than the governing initial flaw depth for each nozzle and would not allow a flaw length that results in an unacceptable flaw depth during the examination period without performance of repair/replacement and regulatory review in accordance with ASME Section XI, IWB-3113. For example, for the limiting DVI Nozzle Case, the initial limiting flaw size that would grow to the limiting flaw depth of 0.358″ over a 10-year period is 0.351″. Using a 0.5 flaw depth to length ratio, this corresponds to a 1.14″ flaw length on the surface of the cladding ((0.351″ + 0.22″) / 0.5), given a 0.22″ cladding thickness. The proposed ASME Section XI Table IWB-3512-1 acceptance criteria of 0.144″ for the maximum allowable flaw length detected during the VT-1 visual examination is much more stringent than supported by the fracture mechanics analysis of 1.14″ in length.

The VT-1 visual examination acceptance criteria are conservative for the limiting LEFM case; however, it is important to note that more realistic EPFM results show that a flaw over 3″ in depth for each reactor vessel nozzle can be tolerated for a 60-year plant life.

Examinations performed during fabrication of the nozzle forgings include magnetic particle examination in accordance with ASME Section III and ultrasonic examination in accordance with ASME Section V, Article 7 and Article 5. These examinations ensure that examination surfaces have been appropriately prepared for the application of future volumetric examinations. Following deposition of the cladding, a PT examination was performed using the acceptance standards of ASME Code Section III, NB-5350. Following intermediate heat treatment, a UT in accordance with the examination procedure requirements of ASME Section V, using the acceptance standards of Section XI, IWB-3512 was performed with no recordable indications. Following the hydrostatic test, a PT examination was performed using the acceptance standards of ASME Code Section III, NB-5350, with no relevant indications (i.e., an indication greater than 1/16″ long). These examinations, in addition to the proposed visual examination, provide assurance that existing flaws are limited in length and flaws do not exist on the cladding inner diameter surface prior to implementing a VT-1 visual examination. These examinations provide the bases for using the postulated flaw size of 0.16″ (this depth correlates to a 0.32″ flaw length) in the ASME Code Section III, Appendix G analysis. Each reactor vessel satisfies ASME Section III, Appendix G requirements.

<u>Comparison to Operating Fleet</u>

The purpose of the examination of nozzle inner radii is to detect fatigue cracking due to operation and service conditions of the component. The absence of fatigue cracking in nozzle inside radius sections during the operating history of Pressurized Water Reactor (PWR) commercial nuclear power plants, which have been inspected either ultrasonically or visually, indicates that these areas are not readily susceptible to fatigue cracking. The ability to visually detect fatigue cracks has been demonstrated successfully with probability of detection of 80% or greater.

This data is supported most recently via the joint round robin conducted by the industry Electric Power Research Institute (EPRI) and the research arm of the NRC, Pacific Northwest National Laboratory (PNNL), in a 3-phase joint project. The types of cracking the round robin was attempting to detect were much more challenging than fatigue cracking (Intergranular Stress Corrosion Cracking (IGSCC) & Irradiation Assisted Stress Corrosion Cracking (IASCC)) due to the inherent morphology of fatigue cracks versus IGSCC or IASCC. The round robin and previous operating experience clearly demonstrate that detection of cracking is primarily dependent upon the crack opening which would be greater for fatigue cracks.

Visual examination for critical reactor vessel components is routinely conducted industry wide via the Boiling Water Reactor Vessel Internals Program (BWRVIP) for BWRs and Materials Reliability Program (MRP) for PWRs. The VT-1 visual examination method as well as EVT-1 enhanced visual examination and VT-3 visual examination are applied for these critical reactor components. Based on the above, VT-1 visual examinations of the nozzle inner radii are adequate for detection of fatigue cracks should they be present.

In addition to the above discussion, it is noted that the nozzles to which the alternative applies are fabricated from nozzle forgings, in the same manner in which the nozzles now in service in operating plants were fabricated. The material properties (yield strength, ultimate strength, and fracture toughness) for the AP1000 nozzles are the same or better as compared to the operating fleet. The geometries are also similar, in that there are no welds in the region of the nozzle corner. Furthermore, the stresses at the nozzle corner region for the AP1000 are similar to the operating fleet; nevertheless, a plant specific stress analysis evaluation was performed for the AP1000 nozzle corner regions based on finite element analysis. The stresses were then used to perform a plant specific flaw tolerance evaluation based on ASME Section XI (as described in the alternative request) and design basis ASME Section III Appendix G evaluation to demonstrate the structural integrity of the nozzle corner with presence of a large postulated flaw.

The water chemistry and PWR environment for the AP1000 are also similar to the operating fleet. The AP1000 chemistry requirements follow the latest provided EPRI water chemistry requirements, and over time the requirements have become stricter due to the advances in instrumentation and their sensitivities. Nevertheless, the AP1000 water chemistry ranges (such as pH, boron concentration, conductivity, dissolved hydrogen and oxygen) are similar to that of the operating fleet. In general, lack of oxygen in the water chemistry precludes general corrosion and wastage in the carbon steel during normal operating conditions (where primary water chemistry is controlled and which generally represents about 90% of the plant lifetime). During shutdown conditions, any potential for corrosion is prevented with the presence of stainless-steel cladding which is layered over the carbon steel base material.

During PSI, in addition to these VT-1 visual examinations, a liquid penetrant (PT) surface examination was performed at the plant site on the nozzle inside radius sections of the two outlet nozzles, the four inlet nozzles and the two DVI nozzles. The PT surface examination results were evaluated in accordance with ASME Code Section III, NB-5350. The PT examinations were performed prior to the VT-1 visual examinations.

The technical basis for this proposed alternative is included in Enclosure 1. Enclosure 1 was submitted as part of the approval process of the PSI VEGP3&4-PSI-ALT-07 Alternative, under ADAMS Accession Number ML17192A125, and the justification of this alternative has been proven for the life of the plants. This proposed alternative provides an acceptable level of quality and safety in accordance with 10 CFR 50.55a(z)(1).

6. Duration of Proposed Alternative

The proposed alternative is requested for the First ISI Interval for Vogtle Electric Generating Plant (VEGP), Units 3 and 4.

7. Precedents

- Vogtle Electric Generating Plant, Units 3 and 4 Preservice Inspection Interval Proposed Alternative VEGP3&4-PSI-ALT-07 was authorized by an NRC SE dated September 25, 2018 (i.e., NRC Accession Nos. ML18263A215 and ML18263A219).

8. References

1. ASME Boiler and Pressure Vessel Code, Code Case N-648-2, "Alternative Requirements for Inner Radius Examinations of Class 1 Reactor Vessel Nozzles, Section XI, Division 1," dated September 4, 2014.
2. NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 20, dated December 2021.
3. ASME Boiler and Pressure Vessel Code, Section XI, Division 1, "Inservice Inspection of Nuclear Power Plant Components," 2017 Edition.
4. ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Rules for Construction of Nuclear Facility Components," 1998 Edition through the 2000 Addenda.

9. Enclosure

Enclosure 1: ND-17-1121, Technical Basis for the Alternative Request on Inservice Inspection Requirements for Reactor Vessel Nozzle Inner Radius Sections (Originally Enclosure 2 in VEGP 3&4-PSI-ALT-07)

10CFR50.55a Alternative VEGP3&4-ISI1-ALT-18
Proposed Alternative in Accordance with 10CFR50.55a(z)(1)
--Alternative Provides Acceptable Level of Quality and Safety--
Revision 0

Enclosure 1

ND-17-1121, Technical Basis for the Alternative Request on Inservice Inspection Requirements for Reactor Vessel Nozzle Inner Radius Sections
(Originally Enclosure 2 in VEGP 3&4-PSI-ALT-07)

TECHNICAL BASIS FOR THE ALTERNATIVE REQUEST ON PRESERVICE INSPECTION REQUIREMENTS FOR REACTOR VESSEL NOZZLE INNER RADIUS SECTIONS

1.0 Introduction

ASME Code Section XI Code Case (CC) N-648-1 **[1]** allows for the use of a VT-1 visual examination in lieu of the volumetric examination requirement defined in ASME Code Section XI, Table IWB-2500-1, Examination Category B-D, Item No. B3.100 **[2]**. This code case is conditionally accepted by the NRC in Regulatory Guide 1.147 **[3]**. The condition is that the allowable flaw length criteria of ASME Code Section XI, Table IWB-3512-1 must be used with limiting assumptions on the flaw aspect ratio. CC N-648-1 applies only to inservice inspection (ISI).

The technical basis for CC N-648-1 is documented in a paper prepared for and presented at the ASME 2001 Pressure Vessels and Piping Conference **[4]**. The key arguments to justify elimination of the volumetric ISI requirements are good inspection history, a large flaw tolerance, and a risk argument concluding that there is negligible change in core damage frequency with the elimination of the inspection . The logic for the VT-1 visual examination is that service-induced flaw mechanisms (fatigue) will be associated with the inner diameter (ID) surface of the cladding and that the VT-1 examinations are sufficient to detect such mechanisms occurring at the ID surface well before the nozzle suffers degradation of its structural integrity.

It is proposed to extend the application of VT-1 visual examination to the preservice inspection (PSI) subject to the following requirements:

- The surface M-N shown in Figure IWB-2500-7 sketches (a) through (d) is examined using a surface examination method and shall meet the Section III fabrication acceptance standards at least once after the Construction Code hydrostatic test. The surface examination is performed prior to the preservice VT-1 visual examination.

- The volume O-P-Q-R shown in Figure IWB-2500-7 sketches (a) through (d) is examined using a manual volumetric examination method and shall meet the Section XI acceptance standards at least once after the Construction Code hydrostatic test.

- The appropriate surface is prepared in accordance with IWA-2200(b) for application of a future volumetric examination in accordance with Table IWB-2500-1, Examination Category B-D.

- An evaluation that includes the following is performed:
    - Review of the fabrication examination history for the nozzle inner radius region
    - Verification that the nozzle of interest meets the requirements of Section III, Nonmandatory Appendix G.

This technical basis addresses a VT-1 visual examination approach that includes a deterministic fracture assessment similar to that performed as a basis for CC N-648-1, provides a description and justification of a preservice inspection process that addresses the requirements provided above, and will provide an acceptable level of quality and safety in accordance with 10 CFR 50.55a(z)(1).

2.0 PWR Nozzle Inner Radius Section Inspection History in Industry

The ASME Code Section XI 1971 Edition through the 2015 Edition requires volumetric examination of the reactor vessel inner radius section. The original requirement for an examination of this region was developed as a result of cracking in a non-nuclear vessel that occurred around the time when the ASME Code Section XI inspection requirements were being established **[6]**.

Up until the implementation of ASME Code Section XI Code Case N-648-1 after 2001[1], volumetric examinations of PWR reactor vessel nozzle inner radius sections were conducted as required by ASME Section XI using the ultrasonic test method. No recordable flaw indications were detected **[6]**. Subsequently, enhanced VT-1 visual examinations with a resolution capability of distinguishing a 1-mil wire or crack have been applied to PWR reactor vessel nozzle inner radius sections. Again, no recordable flaw indications have been detected.

3.0 Reactor Vessel Nozzle Inner Radius Section Design and Fabrication Inspection History

The reactor vessel and the reactor vessel nozzles are designed in accordance with the ASME Code Section III, Subsection NB **[7]**. The reactor vessel nozzles are fabricated of

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[1] Code Case N-648-1 was conditionally approved by the NRC in Regulatory Guide 1.147, Revision 13 issued in June 2003.

SA-508, Grade 3, Class 1 **[27]** ferritic steel forgings clad on the inner diameter surface with multiple layers of stainless steel cladding (Type 309L first layer and Type 308L subsequent layers). The **AP1000**® reactor vessel has two outlet nozzles, four inlet nozzles, and two direct vessel injection (DVI) nozzles. Figure 1 through Figure 3 show elevation view cross-sections of the three nozzle types, respectively.

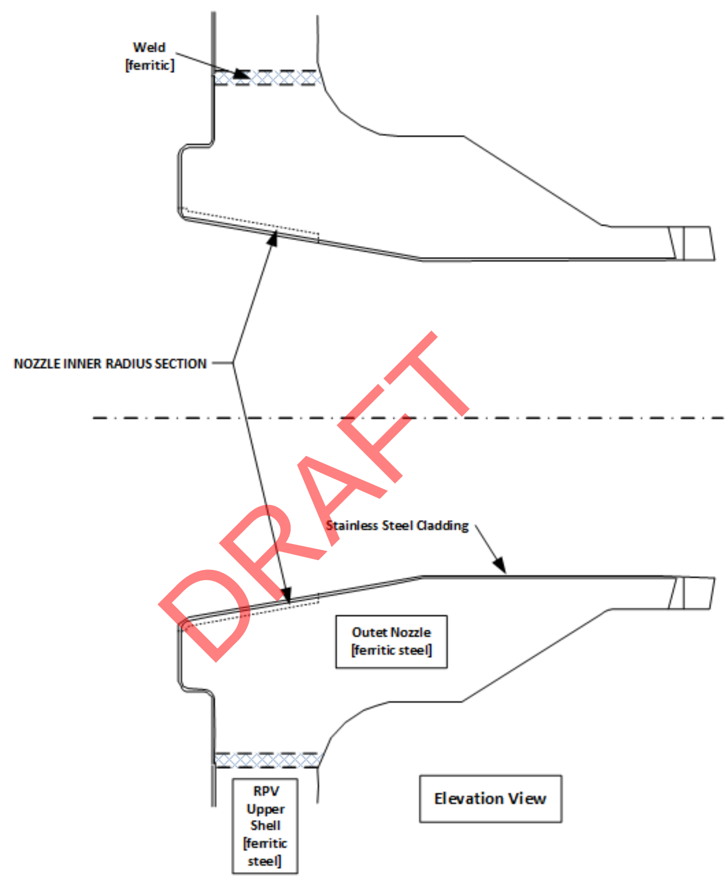

Figure 1: AP1000® Reactor Vessel Outlet Nozzle – Elevation View Cross-Section

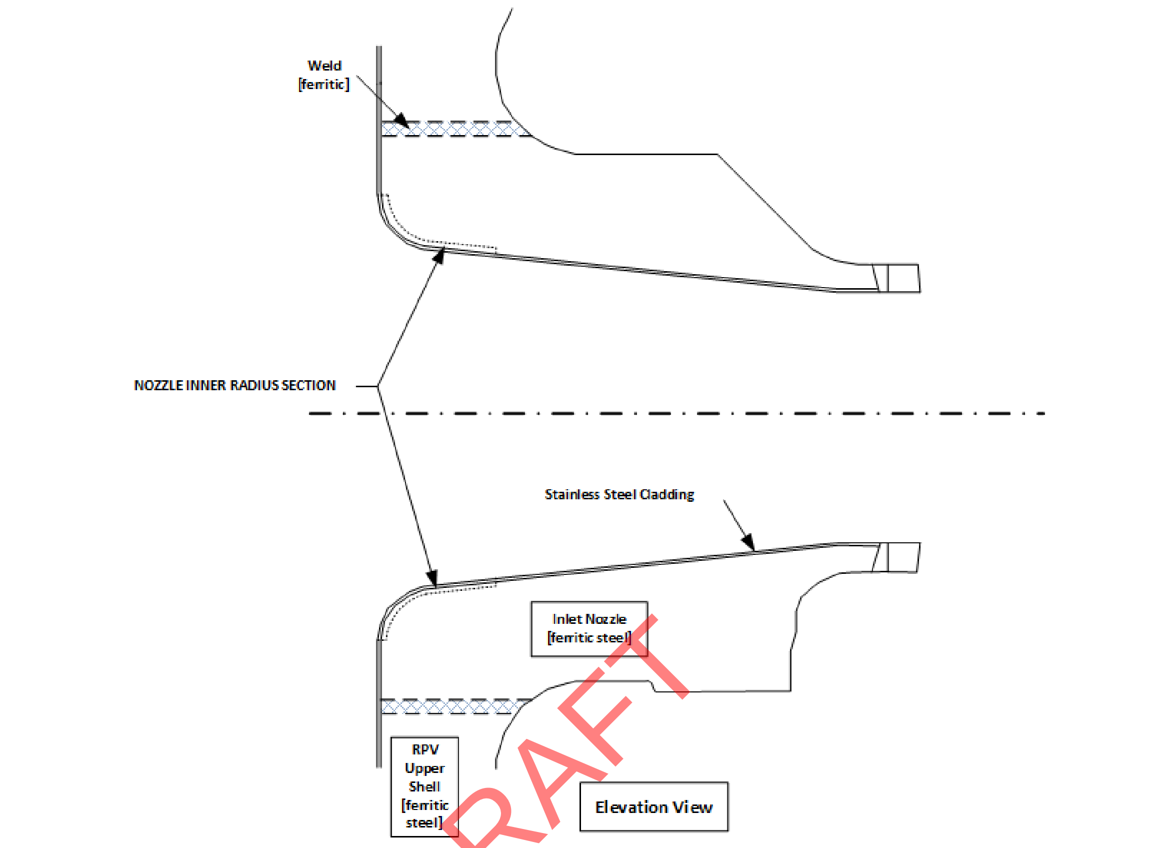

Figure 2: AP1000® Reactor Vessel Inlet Nozzle – Elevation View Cross-Section

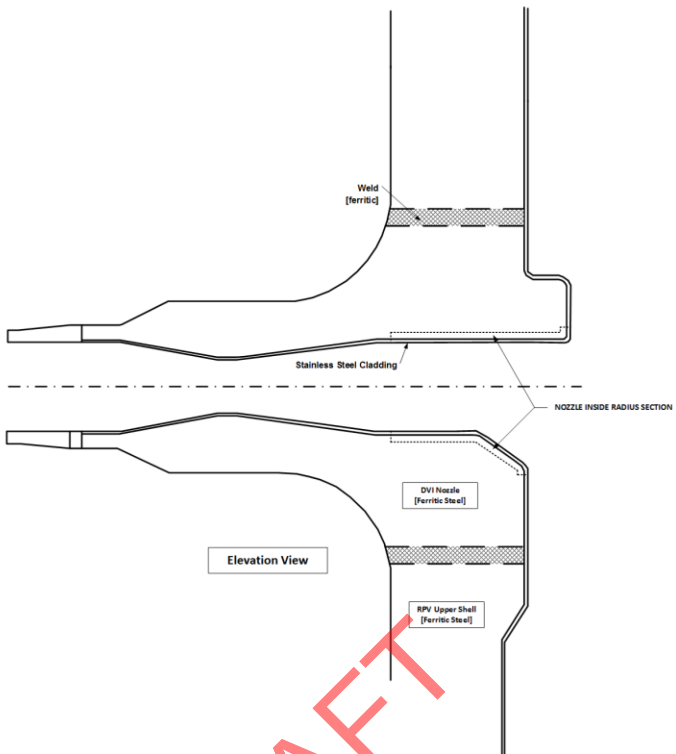

Figure 3: AP1000® Reactor Vessel Direct Vessel Injection (DVI) Nozzle – Elevation View Cross-Section

In accordance with ASME Code Section III, NB-2540, the nozzle forgings are subject to magnetic particle examination over all external surfaces and accessible internal surfaces, and ultrasonic examination of the nozzle volume in accordance with the ASME Code Section V, Article 7 and Article 5, respectively **[8]**. The ultrasonic test requirements are enhanced by the Westinghouse material specification for SA-508 forging materials. Such enhancements include the implementation of Supplementary Requirement S2 of SA-508 **[9]** that requires the use of a higher sensitivity straight beam examination calibrated on 1/4-inch diameter flat-bottomed holes rather than the forging back surface, recording and investigating of angle beam indications equal to or exceeding 20% of the reference level rather than equal to or exceeding 50% of the reference level, and specifically identifying all recordable angle beam indications located near a surface (within 15% of the wall thickness) and/or all indications that display crack-like characteristics for separate disposition.

In accordance with ASME Code Section III, NB-5120 (d), the nozzle base metal surface is examined by the magnetic particle method prior to the deposition of the stainless steel cladding using the acceptance standards of ASME Code Section III, NB-5340. After the cladding is deposited, the cladding surface is examined by liquid penetrant method using the acceptance standards of ASME Code Section III, NB-5350. The cladding is also subject to an ultrasonic examination for lack of bond as defined in the Westinghouse fabrication specification.

After completion of welding and the intermediate heat treatment but before the post-weld heat treatment, the ultrasonic test method is applied for the examination of the inlet, outlet and DVI nozzle inner radius section volumes as defined by ASME Code Section XI, Figure IWB-2500-7(b). The nozzle inner radius section volumes are shown in Figures 1 through 3. These examinations are conducted from the inside and outside diameter surfaces in accordance with the examination procedure requirements of ASME Code Section V, Article 4 **[8]** and using the acceptance standards of ASME Code Section XI, IWB-3512. These are mandatory supplemental requirements defined in the Westinghouse fabrication specification.

After the vessel hydrostatic test, the Westinghouse fabrication specification requires a liquid penetrant examination of all internal vessel surfaces including the stainless steel cladding in the nozzle inner radius sections using the acceptance standards of ASME Code Section III, NB-5350. This specification also requires a repeat of the ultrasonic test method on the nozzle inner radius section volumes applied before the post-weld heat treatment including the examinations from the inner and outer diameter surfaces.

The examinations described above were applicable to the reactor vessel nozzles of Vogtle Units 3 and 4 and V.C. Summer Units 2 and 3.

For these four units, the post-hydrostatic test liquid penetrant examinations detected no relevant flaw indication of cracking or linear indication **[10, 11, 12, 13]**. A relevant indication is defined as being greater than 1/16-inch long.

For these four units, the post-hydrostatic test nozzle inner radius section ultrasonic examinations of the two outlet nozzles, four inlet nozzles and two DVI nozzles of each unit detected no recordable indications **[14, 15, 16, 17]**. The inner and outer diameter surface applied ultrasonic examinations consisted of the techniques defined in Table 1.

Table 1: Ultrasonic Test Techniques Applied in the Shop on the Inlet, Outlet and DVI Nozzle Inner Radius Sections Prior to the Vessel Post-Weld Heat Treatment and After the Hydrostatic Test					
Applied Surface	**Test Angle(s)**	**Test Mode**	**Test Frequency**	**Reference Sensitivity**	**Nozzle Inner Surface Radius Section**
Outside [1],[3]	27°, 30°, 45°	Shear Wave	2.25 MHz	ID notch (2%T)	Inlet and Outlet Nozzles
Outside [1],[3]	13°, 22°	Shear Wave	2.25 MHz	ID notch (2%T)	DVI Nozzle
Inside [4]	70°	Transmit-receive longitudinal wave	2 MHz	ID notch (2.5% a/t)[2]	Inlet, Outlet and DVI Nozzles

Note 1: 0° transducer applied for the detection of laminar flaw indications that would limit or affect the interpretation of the angle beam examination results.
Note 2: Notch depth consistent with ASME Code Section XI, Table IWB-3512-1 for inside corner region.
Note 3: Examinations in two circumferential directions around nozzles.
Note 4: Examinations in four directions, two axial and two circumferential directions.

4.0 Section III, Appendix G Verification

The **AP1000®** reactor vessel was evaluated for its ability to protect against non-ductile failure in accordance with ASME Code Section III, Appendix G **[18]** requirements for postulated defects. The inlet nozzle, outlet nozzle and DVI nozzle regions were part of this linear elastic fracture mechanics (LEFM) evaluation. The fracture mechanics evaluation considered the Level A/B service condition, Level C/D service condition and Test Condition (at 70°F and at 110°F) design transients and mechanical loads.

The results demonstrate that the maximum K_I values, resulting from the design transients and mechanical loads, meet the requirements of ASME Code Section III, Appendix G for the postulated flaw sizes. The **AP1000®** reactor vessel is in compliance with ASME Code Section III, Appendix G. To meet these requirements, flaw sizes smaller than one-quarter of the section thickness were assumed. For the reactor vessel nozzle inner radius regions, the smallest postulated flaw size was 0.16-inch at a hydrostatic test temperature of 70°F. Such a small postulated flaw was justified based on the manufacturing inspections described in Section 3.0 and the ultrasonic and visual examinations to be performed prior to service as described in Section 6.0.

5.0 Fracture Assessment

Reference **[4]** provides a basis to eliminate inservice volumetric examinations at the inner radius of reactor vessel nozzles for the operating reactor vessels in the US. The American Society of Mechanical Engineers (ASME) approved Code Case N-648-1 **[1]** based on the results documented in **[4]**. At the time of publication of **[1]**, the Code Case was only applicable to operating plants in the US. The fracture assessment results documented in this section support the technical basis for the **AP1000®** plant design. This includes calculation of the end of evaluation period flaw sizes for the **AP1000®** inlet, outlet and direct vessel injection (DVI) nozzles as well as fatigue crack growth analyses using the rules of ASME Section XI **[2]**.

The allowable end of evaluation period flaw sizes (depths) for the **AP1000®** inlet, outlet, and DVI nozzles were determined using both linear elastic fracture mechanics (LEFM) and elastic plastic fracture mechanics (EPFM) methods. The LEFM flaw tolerance calculations were performed per ASME Section XI IWB-3600 and Appendix A, and the EPFM method followed the guidelines of Code Case N-749 **[5]**. Fatigue crack growth (FCG) analyses were also performed in order to determine the maximum initial flaw size that will not grow beyond the allowable end of evaluation period flaw size within the life of the plant (60 years) considering Level A/B/Test conditions. In addition, FCG analyses were also performed to determine the maximum initial flaw size for a 10 year period using LEFM only. In all cases the crack growth law for ferritic steels not susceptible to environmentally assisted cracking (EAC) given in Code Case N-643-2 **[26]** was used for the FCG calculations.

Table 2 shows the fracture assessment results for Level A/B/Test conditions for all three nozzle types using both LEFM and EPFM methods. The LEFM method is very conservative because it does not take into account the ductile behavior of the nozzle material, due to the lack of constraint present in this geometry. The EPFM results listed in Table 2 were produced using Code Case N-749 **[5]** and provide a more realistic fracture assessment considering the resistance to crack extension of the ductile nozzle material.

For the LEFM results, the DVI nozzle design produced the smallest end of evaluation period flaw size (0.358 inch), as well as the most limiting FCG result (0.326 inch) for a 60 year operating life. The results for ten years of operation show tolerance for slightly larger flaws and demonstrate the flaw sizes that might be of concern between inspection intervals based on conservative LEFM evaluations.

The EPFM evaluations demonstrate tolerance for much larger flaws, as shown in Table 2. The most limiting end of evaluation period flaw size is 4.5 inches for the DVI nozzle. However, the most limiting FCG result occurs for the outlet nozzle with a flaw depth of 3.088 inches. In all cases, tolerance for flaws over three inches in depth is demonstrated for 60 years of operation. Because the 60 year results demonstrate tolerance for such large flaws, it was not necessary to evaluate a 10 year period as was done for the LEFM cases.

The end of evaluation period flaw sizes for Level C/D conditions were also determined for each nozzle type using the LEFM method. As can be seen in Table 3, the Level C/D flaw evaluation results are not limiting in comparison to the Level A/B/Test LEFM results reported in Table 2. For all cases listed in Table 3, the limiting flaw sizes are over 3 inches.

These initial flaw size results for 60 years are considered to be acceptable based on Section III flaw acceptance criteria prior to the components being placed into service. The largest permissible flaw length for magnetic partial examination per NB-2545 for forgings is 3/16 inch. The analyses were performed using the nozzle corner, quarter-circular stress intensity factor solution from API 579-1 **[25]** with the built-in assumed length-to-depth ratio of 2. Thus, the depth corresponding to a 3/16 (0.1875) inch flaw length would be 3/32 (0.094) inch. Additionally, the in-process and post-hydrostatic test UT examinations of the nozzle inner radius sections from the ID and OD surfaces (described in Section 3.0) detected no indications that may have appeared after cladding of the ID surface. Therefore, any flaw that would have been placed into service would have a depth less than the limiting flaw size reported in Table 2, even for the conservative LEFM cases.

Table 2: End of Evaluation Period Flaw Size and Fatigue Crack Growth Results for Inlet, Outlet and DVI Nozzles (Level A/B/Test Conditions)				
LEFM/EPFM	**Component\Location**	**End of Evaluation Period Flaw Size (in)**	**FCG Results Period = 10 Years (in)**	**FCG Results Period = 60 Years (in)**
LEFM	Inlet Nozzle\Cut 5	1.034	0.946	0.663
	Inlet Nozzle\Cut 6	0.988	0.944	0.786
	Outlet Nozzle\Cut 5	1.151	1.066	0.793
	Outlet Nozzle\Cut 6	0.922	0.884	0.766
	DVI Nozzle\Cut 8	0.362	0.356	0.335
	DVI Nozzle\Cut 9	0.358	0.351	0.326
EPFM	Inlet Nozzle\Cut 5	7.0	N/A	4.542
	Inlet Nozzle\Cut 6	5.0	N/A	4.195
	Outlet Nozzle\Cut 5	6.0	N/A	3.595
	Outlet Nozzle\Cut 6	4.0	N/A	3.088
	DVI Nozzle\Cut 8	5.0	N/A	4.652
	DVI Nozzle\Cut 9	4.5	N/A	4.132

Table 3: End of Evaluation Period Flaw Size for Level C/D Conditions Using LEFM Method	
Nozzle/Cut	**End of Evaluation Period Flaw Size (in.)**
Inlet/5	7.200
Inlet/6	7.200
Outlet/5	9.480
Outlet/6	9.480
DVI/8	5.260
DVI/9	3.477

6.0 Preservice Inspection Process for Reactor Vessel Nozzle Inner Radius Sections

The ASME Code Section XI preservice inspection (PSI) of the nozzle inner radius sections for the two outlet nozzles, the four inlet nozzles and the two DVI nozzles will be done using a VT-1 visual examination method using an underwater camera system attached to a submersible. This process will be comparable to the subsequent inservice inspections. This visual examination will be conducted in accordance with the ASME

Code Section XI, 2007 Edition with the 2008 Addenda. The allowable flaw length criteria of ASME Code Section XI, Table IWB-3512-1 with a flaw aspect ratio (*a/l*) of 0.5 will be applied for any detected flaw indication. This exception is consistent with the condition defined in NRC Regulatory Guide 1.147 for Code Case N-648-1. Table 4 provides the acceptance standards for the VT-1 visual examination specific to the **AP1000®** reactor vessel nozzles.

However, prior to the VT-1 visual examination PSI, liquid penetrant (PT) surface examinations will be performed at the plant site on the nozzle inner radius sections of the two outlet nozzles, the four inlet nozzles and the two DVI nozzles. The liquid penetrant examinations will be conducted in accordance with ASME Code Section XI, IWA-2222 using the ASME Code Section III, NB-5350 acceptance standards. This is a repeat of the surface examinations performed in the manufacturer's shop as described in Section 3.0 after the Construction Code hydrostatic test. These repeat surface examinations are applied to ensure that no relevant surface-breaking flaws are present on the cladding surfaces prior to service. The PT examination report is to be included in the preservice inspection (PSI) documentation package.

After the PT examinations, manual ultrasonic examinations (UT) will be conducted at the plant site. These UT examinations will be applied from the inner diameter surface using two opposing circumferential beam directions around the nozzle inner radius sections of the two outlet nozzles, the four inlet nozzles and the two DVI nozzles. Dual focused 70-degree transmit-receive longitudinal wave transducers with acoustic focusing at or near the clad/base metal interface will be used to interrogate the nozzle inner radius section examination volume as defined in ASME Code Section XI, Figure IWB-2500-7(b) for radial-axial flaws (see Figures 1 through 3). The ultrasonic examination procedure requirements will be in accordance with ASME Code Section XI, Appendix III as supplemented by ASME Code Section XI, Appendix I Supplements 1 – 8, 10 and 11. These supplements are:

- Supplement 1 – Calibration Block Material and Thickness
- Supplement 2 – Calibration Blocks for Clad Welds or Components
- Supplement 3 – Calibration Blocks for Examination of Parts with Curved Surfaces
- Supplement 4 – Alternative Weld Calibration Block Design
- Supplement 5 – Electronic Simulators
- Supplement 6 – Pulse Repetition Rate
- Supplement 7 – Instrument Calibration
- Supplement 8 – Scan Overlap and Search Unit Oscillation
- Supplement 10 – Recording Criteria
- Supplement 11 – Geometric Indications

The reference sensitivity will be established on a radial-axial notch at the inside corner region with a depth equal to an '*a/t*' of 2.5% consistent with the ASME Code Section XI, Table IWB-3512-1 allowable planar flaw size.

It is noted that dual focused 70-degree transmit-receive longitudinal wave transducers have proven to be effective at detecting near surface flaws initiating at the cladding surface or at the clad/base metal interface **[19 – 24]**.

These ultrasonic examinations ensure that the appropriate surfaces have been prepared for application of future volumetric examinations in accordance with ASME Code Section XI, Table IWB-2500-1, and they provide a baseline volumetric examination of the nozzle inner radius section volumes. The UT examination report is to be included in the preservice inspection (PSI) report.

Table 4: AP1000® Reactor Vessel Nozzle Inside Corner Region VT-1 Visual Examination Acceptance Standards in Accordance with ASME Code Section XI, Table IWB-3512-1

Nozzle Description	Component Thickness[1] [inch]	Max. '*a/l*' Allowed[2]	Max. '*a/t*' Allowed[3] [%]	Max. '*a*' Allowed[4] [inch]	Max. '*l*' Allowed[5] [inch]
Outlet	t_{n1} = 12.03	0.5	2.5	0.196	0.392
	t_{n2} = 7.85				
	t_s = 10.15				
Inlet	t_{n1} = 11.91	0.5	2.5	0.192	0.384
	t_{n2} = 7.68				
	t_s = 10.15				
DVI	t_{n1} = 2.87	0.5	2.5	0.072	0.144
	t_{n2} = 2.87				
	t_s = 10.15				

Note 1: Thickness is the smallest of the three thicknesses shown in ASME Code Section XI, Figures IWB-2500-7(a) and -7(b) as defined by Table IWB-3512-2. The smallest component thickness is shown in bold print.
Note 2: Based on Reg. Guide 1.147, Rev. 17 condition to Code Case N-648-1 and approach defined in this technical basis; '*a/l*' is the flaw depth / flaw length ratio.
Note 3: Based on ASME Code Section XI, Table IWB-3512-1 for Inside Corner Region; applicable for nominal wall thickness ranging from 2.5-inches or less to 12-inches; '*a/t*' is the flaw depth / component thickness ratio. This table was defined in the Reg, Guide 1.147 condition on CC N-648-1 and is the same approach defined in this technical basis.
Note 4: Calculated as 2.5% of the smallest component thickness or 0.025 times the smallest component thickness.
Note 5: Calculated as '*a*'/0.5.

7.0 Conclusions

The following conclusions can be made:

1. The combination of the post-hydrostatic test surface (PT) examinations performed at the manufacturer's facility and at the plant site ensures that fabrication flaws, particularly cracks and linear flaws, do not exist on the cladding inner diameter

surface prior to implementing the VT-1 visual examination method. The ASME Code Section III, NB-5340 acceptance standards consider all cracks and linear flaws greater than 1/16-inch long to be unacceptable.

2. The combination of the post-hydrostatic test volumetric (UT) examinations performed at the manufacturer's facility and at the plant site ensures that the appropriate surfaces have been prepared in accordance with ASME Code Section XI, IWA-2200(b) for application of future volumetric examinations. This is consistent with the Owner's responsibility to provide adequate design and access provisions for periodic inservice inspection in compliance with ASME Code Section III, NCA-3220(r), ASME Code Section XI, IWA-1400(b) and IWA-1500, and 10 CFR 50.55a(g)(3)(i). The UT examinations performed at the plant site also provide for a baseline volumetric examination of the nozzle inner radius section volumes.

3. The fabrication examination history for the nozzle inner radius sections have been reviewed and documented. This examination history includes the required ASME Code Section III, NB-2500 and NB-5000 examinations of the forging material and cladding as supplemented by Westinghouse requirements for UT of the cladding for lack of bond, in-process and post-hydrostatic test UT examinations applied from the ID and OD surfaces for flaws in the base metal, and post-hydrostatic test PT examinations of the cladding surfaces for flaws on the ID surface. Such examinations support the maximum postulated flaw sizes used in the ASME Code Section III, Appendix G evaluation.

4. The nozzles meet the requirements of ASME Code Section III, Appendix G.

5. A deterministic fracture mechanics assessment has shown that the governing initial flaw size for the **AP1000®** inlet, outlet and DVI nozzles over a 10 year period consistent with the 10-year inspection interval is 0.351-inch deep within the underlying nozzle base metal, using very conservative LEFM methods. The limiting flaw size for a 60 year period is 0.326 inch, also using LEFM. More realistic EPFM results show that a flaw over 3 inches in depth can be tolerated for a 60 year plant life. The flaw depth to length aspect ratio consistent with the condition on Code Case N-648-1 in Regulatory Guide 1.147 is $a/l = 0.5$. Thus for a flaw depth (a) of 0.326-inch, the flaw length (l) is 0.652-inch for the base metal.

6. The governing initial flaw depth of 0.326-inch in the base metal (calculated for the DVI nozzle as indicated on Table 2) corresponds to a flaw length of 1.09-inches on the surface of the cladding given the nominal clad thickness of 0.22-inch, the total flaw depth (a) of 0.546-inch (0.326 + 0.22) and a 0.5 flaw depth to flaw length (a/l) aspect ratio. Table 4 indicates that for the VT-1 visual examination of the DVI nozzle

the allowable flaw length is 0.144-inch which is smaller than the governing initial flaw length of 1.09-inch given a 0.5 flaw depth to flaw length aspect ratio. A VT-1 visual examination finding exceeding this allowable acceptance standard would result in repair/replacement and reexamination, and regulatory review in accordance with ASME Code Section XI, IWB-3113 and IWB-3114 to ensure fitness for service.

8.0 References

1. ASME Boiler and Pressure Vessel Code, Case N-648-1: Alternative Requirements for Inner Radius Examinations of Class 1 Reactor Vessel Nozzles, Section XI, Division 1, approved September 7, 2001.

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3. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.147: Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1, Revision 17.

4. W. H. Bamford et al, 'Technical Basis for Elimination of Reactor Vessel Nozzle Inner Radius Inspections,' Proceedings of ASME 2001 Pressure Vessels and Piping Conference, Atlanta, GA.

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11. Doosan Heavy Industries & Construction, P140103-031-001: PT Report for the Examination of All Inside Surfaces of the Reactor Vessel (After Hydrostatic Test) for Vogtle Unit 4, January 2014.

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14. Doosan Heavy Industries & Construction, U120706-021-001: UT Report for the Examination of Vessel Girth Welds, Nozzle to Shell Welds and Nozzle Inner Radius Regions (After Hydrostatic Test) for Vogtle Unit 3, July 2012.

15. Doosan Heavy Industries & Construction, U140107-017-002: UT Report for the Examination of Nozzle Inner Radius Regions (After Hydrostatic Test) for Vogtle Unit 4, January 2014.

16. Doosan Heavy Industries & Construction, U120907-022-002: UT Report for the Examination of Nozzle Inner Radius Regions (After Hydrostatic Test) for V.C Summer Unit 2, September 2012.

17. Doosan Heavy Industries & Construction, U150330-031-001: UT Report for the Examination of Nozzle Inner Radius Regions (After Hydrostatic Test) for V.C Summer Unit 3, April 2015.

18. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Appendix G: Protection Against Nonductile Failure, 1998 Edition through 2000 Addenda.

19. S.Crutzen, C. Vinche, Ph. Doubret and N. Haines, “The Evaluation of the PISC II Round Robin Tests,” 8th International Conference on NDE in the Nuclear Industry, Kissimmee, Florida, November 17 – 20, 1986.

20. D.L. Lock, K.J. Cowburn and B. Watkins, "The Results Obtained in the UKAEA Defect Detection Trials on Test Pieces 3 and 4," Nuclear Energy: Journal of the British Nuclear Energy Society, Volume 22, Number 5, October 1983.

21. D.B. Langston and R. Wilson, "An Assessment of the Defect Detection Capability of the Ultrasonic Inspection Techniques Used by the CEGB in the UKAEA Defect Detection Trials," International Journal – Pressure Vessel & Piping, Volume 23, 1986.

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SNC Pre-Submittal Meeting for Proposed Alternatives VEGP 3&4-ISI1-ALT-17 and VEGP 3&4-ISI1-ALT-18

October 27, 2022

Southern Nuclear

Meeting Purpose and Agenda

- Purpose: Discuss proposed alternatives with NRC
  - VEGP 3&4-ISI1-ALT-17 (Based on VEGP 3&4-PSI-ALT-05)
  - VEGP 3&4-ISI1-ALT-18 (Based on VEGP 3&4-PSI-ALT-07)
  - Applicable to first Inservice Inspection Interval for VEGP 3&4

- This presentation will cover the following topics:
  - Schedule
  - Background
  - Proposed Alternative
  - Basis

Schedule

- Submit Alternatives – October 2022
- Request Approval – February 2023

VEGP 3&4-ISI1-ALT-17

Alternative Inspection Requirements for Steam Generator Nozzle to Reactor Coolant Pump Casing Welds

Background – VEGP 3&4-ISI1-ALT-17

- 10 CFR 50.55a(b)(2)(xlii)(B) requires volumetric examination volume to be extended to include 100% of the weld volume, when applying requirements of Figure IWB-2500-8
- 10 CFR 50.55a(b)(2)(xlii)(B)(1) allows for ultrasonic examination of the qualified volume and largest hypothetical crack flaw evaluation to be performed when the exam volume that can be obtained is less than 100%, subject to prior NRC authorization

Proposed Alternative – VEGP 3&4-ISI1-ALT-17

- SNC proposes to perform an inservice inspection encoded volumetric examination of the required ASME Section XI, 2017 Edition inspection volume, not the entire weld volume.

- Required surface examinations will also be performed on the outer diameter surface

- Ultrasonic testing techniques will be qualified in accordance with the Performance Demonstration Initiative (PDI) Program, which satisfies the requirements of ASME Section XI, Appendix VIII, Supplement 10, including 10 CFR 50.55a.

- SNC proposes to perform an eddy current examination from the inner diameter surface

Basis for Request – VEGP 3&4-ISI1-ALT-17

- PDI qualification was extended to account for the greater thickness of the AP1000 Steam Generator (SG) to Reactor Coolant Pump (RCP) casing weld
  - EPRI designed and fabricated a blind test specimen in accordance with EPRI/PDI Program
  - Qualifies ultrasonic examination procedures, equipment, and personnel under PDI ASME Section XI, Appendix VIII Program

- Eddy current examination techniques are qualified in accordance with ASME Section V, Article 14, 2017 Edition

- Combination of UT and eddy current examination allows detection of primary water stress corrosion cracking

- UT techniques are capable of detecting embedded planar flaws throughout ASME Section XI, 2017 Edition examination volume

Questions?

VEGP 3&4-ISI1-ALT-18

Alternative for Use of Code Case N-648-2 for Inservice Inspection of the Reactor Vessel Nozzle Inside Radius Sections

Background – VEGP 3&4-ISI1-ALT-18

- IWB-2500-1 requires volumetric examination of all Reactor Vessel nozzle inside radius sections each inspection interval (Examination Category B-D, Item No. B3.100)

- Code Case N-648-2, “Alternative Requirements for Inner Radius Examinations of Class 1 Reactor Vessel Nozzles, Section XI, Division 1”

  – Provides alternative to ASME Section XI Requirements

  – Allows a VT-1 visual examination in lieu of the required volumetric examination

  – Conditionally approved for use under Reg Guide 1.147

Proposed Alternative – VEGP 3&4-ISI1-ALT-18

- SNC proposes to perform a VT-1 visual examination of the Reactor Vessel nozzle inside radius sections for the two outlet nozzles, the four inlet nozzles, and two direct vessel injection nozzles
  - Comparable to historical preservice inspection examinations
  - Conducted in accordance with ASME Section XI, 2017 Edition
  - Requirements in Code Case N-648-2 will be applied

Basis for Request – VEGP 3&4-ISI1-ALT-18

- PSI Alternative VEGP 3&4-PSI-ALT-07 was approved using Code Case N-648-1

- N-648-2 has since superseded N-648-1, to include use of the code case for preservice examinations

- PSI examinations were performed in accordance with PSI-ALT-07
  - Utilized ASME Section XI Table IWB-3512-1 acceptance criteria
  - Manual UT examinations were performed

- Flaw Tolerance Evaluation demonstrated requirements in ASME Section III, Appendix G were met for postulated flaw sizes
  - Addressed in enclosure to VEGP 3&4-ISI1-ALT-18 (ND-17-1121)

Basis for Request – VEGP 3&4-ISI1-ALT-18

- Fracture Assessment determined maximum initial flaw size that will not grow beyond allowable end of evaluation period flaw size for life of plant
  - Addressed in enclosure to VEGP 3&4-ISI1-ALT-18 (ND-17-1121)

- VT-1 visual examinations are sufficient to detect service-induced flaw mechanisms occurring at the inner diameter surface before structural degradation

Questions?