

September 9, 2020

Docket Nos.: 50-424
50-425

NL-20-1011

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

Vogtle Electric Generating Plant, Units 1 & 2
Proposed Inservice Inspection Alternative VEGP-ISI-ALT-04-04 Version 2.0

Ladies and Gentlemen:

By letters dated December 11, 2019 (Agencywide Documents Access and Management System Accession No. ML19347B105), Southern Nuclear Operating Company (SNC) submitted a request for relief for the Vogtle Electric Generating Plant (VEGP), Units 1 and 2. SNC requests to increase the inspection interval for ASME Section XI, Table IWC-2500-1, exam Category C-B, item number C2.21 and C2.22, exams from 10 years to 30 years through for the remainder of the 6th ISI Interval.

SNC has discovered two minor errors in the Alternative after the Nuclear Regulatory Commission (NRC) staff's audit, and is resubmitting the Alternative to provide an editorial correction within the Inspection History section and a correction in the Section 6.0 conclusions regarding pressure test frequency. SNC is also submitting supplemental information requested by the NRC during the audit. Enclosure 1 resubmits the VEGP Alternative and supersedes Version 1.0 of proposed ISI Alternative VEGP-ISI-ALT-04-04.

Enclosure 2 provides SI Report No. 1900064.406.R0, Evaluations to Address Limited Examination Coverage of Vogtle Electric Generating Plant Units 1 and 2 Steam Generator Main Steam and Feedwater Nozzle-to-Shell Welds and Nozzle Inside Radius Sections.

Enclosure 3 contains SI Report No. 1900064.407.R2, Evaluations to Address Benchmarking of the PROMISE Software to Include the Effects of Inspections.

This letter contains no NRC commitments. If you have any questions, please contact Jamie Coleman at 205.992.6611.

Respectfully submitted,

Cheryl A. Gayheart
Regulatory Affairs Director

CAG/DSP/sm

Enclosure 1: Proposed Alternative VEGP-ISI-ALT-04-04, Version 2.0,
in Accordance with 10 CFR 50.55a(z)(1)

Enclosure 2: SI Report No. 1900064.406.R0

Enclosure 3: SI Report No. 1900064.407.R2

cc: Regional Administrator
NRR Project Manager – Vogtle 1 & 2
Senior Resident Inspector – Vogtle 1 & 2
RType: CVC7000

**Vogtle Electric Generating Plant, Units 1 & 2
Proposed Inservice Inspection Alternative VEGP-ISI-ALT-04-04**

Enclosure 1

**Proposed Alternative VEGP-ISI-ALT-04-04, Version 2.0,
in Accordance with 10 CFR 50.55a(z)(1)**

1.0 ASME CODE COMPONENTS AFFECTED:

Code Class: Class 2
Description: Nozzle-to-shell welds and inside radius sections
Examination Category: C-B (Pressure Retaining Nozzle Welds in Pressure Vessels, Section XI, Division 1)
Item Numbers: C2.21 - Nozzle-to-shell (nozzle-to-head or nozzle-to-nozzle) welds
C2.22 - Nozzle inside radius sections

Component IDs:

11201-B6-001-W18	32" STEAM OUTLET NOZZLE TO UPPER HEAD WELD
11201-B6-001-W19	16" MAIN FEEDWATER NOZZLE TO SHELL WELD
11201-B6-002-W18	32" STEAM OUTLET NOZZLE TO UPPER HEAD WELD
11201-B6-002-W19	16" MAIN FEEDWATER NOZZLE TO SHELL WELD
11201-B6-003-W18	32" STEAM OUTLET NOZZLE TO UPPER HEAD WELD
11201-B6-003-W19	16" MAIN FEEDWATER NOZZLE TO SHELL WELD
11201-B6-004-W18	32" STEAM OUTLET NOZZLE TO UPPER HEAD WELD
11201-B6-004-W19	16" MAIN FEEDWATER NOZZLE TO SHELL WELD
11201-B6-001-IR04	MAIN FEEDWATER NOZZLE INNER RADIUS
11201-B6-002-IR04	MAIN FEEDWATER NOZZLE INNER RADIUS
11201-B6-003-IR04	MAIN FEEDWATER NOZZLE INNER RADIUS
11201-B6-004-IR04	MAIN FEEDWATER NOZZLE INNER RADIUS
21201-B6-001-W18	32" STEAM OUTLET NOZZLE TO UPPER HEAD WELD
21201-B6-001-W19	16" MAIN FEEDWATER NOZZLE TO SHELL WELD
21201-B6-002-W18	32" STEAM OUTLET NOZZLE TO UPPER HEAD WELD
21201-B6-002-W19	16" MAIN FEEDWATER NOZZLE TO SHELL WELD
21201-B6-003-W18	32" STEAM OUTLET NOZZLE TO UPPER HEAD WELD
21201-B6-003-W19	16" MAIN FEEDWATER NOZZLE TO SHELL WELD
21201-B6-004-W18	32" STEAM OUTLET NOZZLE TO UPPER HEAD WELD
21201-B6-004-W19	16" MAIN FEEDWATER NOZZLE TO SHELL WELD
21201-B6-001-IR04	MAIN FEEDWATER NOZZLE INNER RADIUS
21201-B6-002-IR04	MAIN FEEDWATER NOZZLE INNER RADIUS
21201-B6-003-IR04	MAIN FEEDWATER NOZZLE INNER RADIUS
21201-B6-004-IR04	MAIN FEEDWATER NOZZLE INNER RADIUS

2.0 REQUESTED APPROVAL DATE:

Approval is requested by December 31, 2020.

3.0 APPLICABLE CODE EDITION AND ADDENDA:

The Fourth Inservice Inspection (ISI) Interval Code of record for Vogtle Units 1 & 2 is the 2007 Edition with 2008 Addenda of ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

4.0 APPLICABLE CODE REQUIREMENT:

ASME Section XI IWC-2500(a), Table IWC-2500-1, Examination Category C-B, Item No. C2.21 requires surface and volumetric examination of all representative steam generator nozzles at terminal ends of piping runs once during each Section XI inspection interval. ASME Section XI IWC-2500(a), Table IWC-2500-1, Examination Category C-B, Item No. C2.22 requires volumetric examination of representative steam generator all nozzle at terminal ends of piping runs once during each Section XI inspection interval. The examination areas for Item Nos. C2.21 and C2.22 are shown in Figures IWC-2500-4(a), (b), and (d).

5.0 REASON FOR REQUEST:

The Electric Power Research Institute (EPRI) performed an assessment [1] of the basis for the ASME Section XI examination requirements specified for Examination Category C-B of ASME Section XI, Division 1 for Steam Generator (SG) Main Steam (MS) and Feedwater (FW) Nozzle-to-Shell Welds and Nozzle Inside Radius Sections. The assessment includes a survey of inspection results from 74 units as well as flaw tolerance evaluations using probabilistic fracture mechanics (PFM) and deterministic fracture mechanics (DFM). The Reference [1] report concluded that the current ASME Code Section XI inspection interval of ten years can be increased significantly with no impact to plant safety. It is upon the basis of this conclusion that an alternate inspection interval is being requested. The Reference [1] report was developed consistent with the recommendations provided in EPRI's White Paper on PFM [14].

6.0 PROPOSED ALTERNATIVE AND BASIS FOR USE:

Southern Nuclear Company (SNC) is requesting an inspection alternative to the examination requirements of ASME Section XI, Table IWC-2500-1, Examination Category C-B, Item Nos. C2.21 and C2.22. The proposed alternative is to increase the inspection interval for these examination items to 30 years (from the current ASME Code Section XI 10-year requirement) for the remainder of the 6th Inservice Inspection (ISI) Interval. Although the EPRI report [1] supports a longer inspection period, 30 years was selected as a prudent alternative to ensure that one more examination was conducted prior to the end of the current license period for Vogtle Units 1 & 2. A summary of the key aspects of the technical basis for this request are summarized below. The applicability of the technical basis to Vogtle Units 1 & 2 is shown in Appendix A.

Degradation Mechanism Evaluation

An evaluation of degradation mechanisms that could potentially impact the reliability of the SG MS and FW Nozzle-to-Shell Welds and Nozzle Inside Radius Sections was performed in Reference [1]. Evaluated mechanisms included stress corrosion cracking (SCC), environmental assisted fatigue (EAF), microbiologically influenced corrosion (MIC), pitting, crevice corrosion, erosion-cavitation, erosion, flow accelerated corrosion (FAC), general corrosion, galvanic corrosion, and mechanical/thermal fatigue. Other than the potential for EAF and mechanical/thermal fatigue, there were no active degradation mechanisms identified that significantly affect the long-term structural integrity of the SG MS and FW nozzles.

Stress Analysis

Finite element analysis (FEA) was performed in Reference [1] to determine the stresses in the SG MS and FW Nozzle-to-Shell Welds and Nozzle Inside Radius Sections. The analysis was performed using representative pressurized water reactor (PWR) geometries, bounding transients, and typical material properties. The results of the stress analyses were used in a flaw tolerance evaluation. The applicability of the FEA analysis to Vogtle Units 1 & 2 is shown in Appendix A and confirms that all plant-specific requirements are met. Therefore, the evaluation results and conclusions of Reference [1] are applicable to Vogtle Units 1 & 2.

Flaw Tolerance Evaluation

Flaw tolerance evaluations were performed in Reference [1] consisting of PFM evaluations and confirmatory DFM evaluations. The results of the PFM analyses indicate that, after a preservice inspection (PSI), no other inspections are required for up to 60 years of plant operation to meet the U.S. Nuclear Regulatory Commission's (NRC's) safety goal of 10^{-6} failures per year. For the specific case of Vogtle Units 1 and 2 where PSI followed by three 10-year interval inspections have been performed, Table 8-10 of Reference [1] indicates that if the inspection interval is increased to 30 years after these previous inspections, the NRC safety goal is met (with considerable margin) for up to 80 years of plant operation. The DFM evaluations provide verification of the PFM results by demonstrating that it takes approximately 80 years for a postulated flaw with an initial depth equal to the ASME Code Section XI acceptance standards to grow to a depth where the maximum stress intensity factor (K) exceeds the ASME Code Section XI allowable fracture toughness.

Inspection History

Plant Vogtle Unit 1 and 2 operating experience (including examinations performed to date, examination findings, inspection coverage, and Relief Requests) is presented in Appendix B. As shown in this Appendix, Item No. C2.21 (FW nozzle and MS nozzle) examinations have had limited coverage. Also, as shown in Appendix B, no flaws that exceeded the ASME Code, Section XI acceptance standards were identified during any examinations.

Industry inspection history for these components (as obtained from an industry survey [1]) is presented in Appendix C. The results of the survey [1] indicate that these components are very flaw tolerant.

Conclusion

It is concluded that the SG MS and FW Nozzle-to-Shell Welds and Nozzle Inside Radius Sections are very flaw tolerant. PFM and DFM evaluations performed as part of the technical basis [1] demonstrate that, after PSI, no other inspection is required until 60 years to meet the NRC safety goal of 10^{-6} failures per reactor year. Plant-specific applicability of the technical basis to Vogtle Units 1 & 2 is demonstrated in Appendix A. An inspection interval of 30 years provides an acceptable level of quality and safety in

lieu of the ASME Examination Category C-B, Item Nos. C2.21 and C2.22 surface and volumetric examination 10-year inspection frequency.

Operating and examination experience demonstrates that these components have performed with very high reliability, mainly due to their robust design. As shown in Appendix B, to date, SNC has performed 20 inspections of SG MS and FW Nozzle-to-Shell Welds and Nozzle Inside Radius Sections at Vogtle Units 1 & 2. No flaws that exceeded the ASME Code, Section XI acceptance standards were identified during any examinations, as shown in Appendix B. Some of the inspections listed in Appendix B involved limited coverage ranging from 50% to 80%. Section 8.2.5 of Reference [1] discusses limited coverage and determines that the conclusions of the report are applicable to components with limited coverage. In addition, it is important to note all other inspection activities, including the system leakage test (Examination Category C-H) conducted each inservice inspection period (approximately every other refueling outage), will continue to be performed, providing further assurance of safety.

Finally, as discussed in Reference [2], for situations where no active degradation mechanism is present, it was concluded that subsequent inservice inspections do not provide additional value after PSI has been performed and the inspection volumes have been confirmed to have no flaws that exceeded the ASME Code, Section XI acceptance standards. The Vogtle Units 1 & 2 SG MS and FW nozzles have received the required PSI examinations and 20 follow-on inservice inspections with no flaws that exceeded the ASME Code, Section XI acceptance standards.

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

7.0 DURATION OF PROPOSED ALTERNATIVE:

The proposed Alternative is requested for the remainder of the 4th Inservice Inspection through 6th Inspection (ISI) Interval for Vogtle Units 1 & 2, currently scheduled to end on 5/30/47.

8.0 PRECEDENT:

No previous submittals have been made requesting relief from the ASME Examination Category C-B, Item Nos. C2.21 and C2.22 surface and volumetric examinations on the basis of the Reference [1] technical basis. However, the following is a list of approved Relief Requests related to inspections of SG MS and FW nozzles:

- Letter from J. W. Clifford (NRC) to S. E. Scace (Northeast Nuclear Energy Company), "Safety Evaluation of the Relief Request Associated with the First and Second 10-Year Interval of the Inservice Inspection (ISI) Plan, Millstone Nuclear Power Station, Unit 3 (TAC No. MA 5446)," dated July 24, 2000, ADAMS Accession No. ML003730922.
- Letter from R. L. Emch (NRC) to J. B. Beasley, Jr. (SNOC), "Second 10-Year Interval Inservice Inspection Program Plan Requests for Relief 13, 14, 15, 21 and 33 for Vogtle Electric Generating Plant, Units 1 and 2 (TAC No. MB0603 and MB0604)," dated June 20, 2001, ADAMS Accession No. ML011640178.

- Letter from T. H. Boyce (NRC) to C. L. Burton (CP&L), “Shearon Harris Nuclear Power Plant Unit 1 – Request for Relief 2R1-019, 2R1-020, 2R1-021, 2R1-022, 2R2-009, 2R2-010, 2R2-011 for the Second Ten-Year Interval Inservice Inspection Program Plan (TAC Nos. ME0609, ME0610, ME0611, ME0612, ME0613, ME0614 and ME0615),” dated January 7, 2010, ADAMS Accession No. ML093561419.
- Letter from M. Khanna (NRC) to D. A. Heacock (Dominion Nuclear Connecticut Inc.), Millstone Power Plant Unit No. 2 – Issuance of Relief Requests RR-89-69 Through RR-89-78 Regarding Third 10-Year Interval Inservice Inspection plan (TAC Nos. ME5998 Through ME6006),” dated March 12, 2012, ADAMS Accession No. ML120541062.
- Letter from R. J. Pascarelli (NRC) to E. D. Halpin (PG&E), “Diablo Canyon Plant, Units 1 and 2 – Relief Request; NDE SG-MS-IR, Main Steam Nozzle Inner Radius Examination Impracticality, Third 10-Year Interval, American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Inservice Inspection Program (CAC Nos. MF6646 and MF6647),” dated December 8, 2015, ADAMS Accession No. ML15337A021.

In addition, there are precedents related to similar requests for relief for Class 1 nozzles:

- Based on studies presented in Reference [3], the NRC approved extending PWR reactor vessel nozzle-to-shell welds from 10 to 20 years in Reference [4].
- Based on work performed in BWRVIP-108 [5] and BWRVIP-241 [7], the NRC approved the reduction of BWR vessel feedwater nozzle-to-shell weld examinations (Item No. B3.90 for BWRs from 100% to a 25% sample of each nozzle type every 10 years) in References [6] and [8]. The work performed in BWRVIP-108 and BWRVIP-241 provided the technical basis for ASME Code Case N-702 [9], which has been conditionally approved by the NRC in Revision 18 of Regulatory Guide 1.147 [10].

Finally, there are precedents that used generic industry guidance in a similar approach to the approach requested in this submittal:

- Based on EPRI generic analysis, the Vogtle and Farley plants requested an alternative to the Reactor Pressure Vessel Threads in Flange examination requirements of ASME Section XI in References [11] and [12].
- NRC relief was granted for the Vogtle and Farley requests for alternatives to the Reactor Pressure Vessel Threads in Flange examination requirements in the reference [13] Safety Evaluation.

9.0 ACRONYMS:

ASME	American Society of Mechanical Engineers
B&W	Babcock and Wilcox
BWR	Boiling Water Reactor
BWRVIP	Boiling Water Reactor Vessel and Internals Program
CE	Combustion Engineering
CFR	Code of Federal Regulations
DFM	Deterministic fracture mechanics
EAF	Environmentally assisted fatigue
EPRI	Electric Power Research Institute
FAC	Flow accelerated corrosion
FEA	Finite element analysis
FW	Feedwater
ISI	Inservice Inspection
MIC	Microbiologically influenced corrosion
MS	Main Steam
NPS	Nominal pipe size
NRC	Nuclear Regulatory Commission
NSSS	Nuclear steam supply system
PFM	Probabilistic fracture mechanics
PWR	Pressurized Water Reactor
SCC	Stress corrosion cracking
SG	Steam Generator
SNC	Southern Nuclear Company

10.0 **REFERENCES:**

1. Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Inside Radius Sections. EPRI, Palo Alto, CA: 2019. 3002014590.
2. American Society of Mechanical Engineers, Risk-Based Inspection: Development of Guidelines, Volume 2-Part 1 and Volume 2-Part 2, Light Water Reactor (LWR) Nuclear Power Plant Components. CRTD-Vols. 20-2 and 20-4, ASME Research Task Force on Risk-Based Inspection Guidelines, Washington, D.C., 1992 and 1998.
3. B. A. Bishop, C. Boggess, N. Palm, "Risk-Informed extension of the Reactor Vessel In-Service Inspection Interval," WCAP-16168-NP-A, Rev. 3, October 2011.
4. US NRC, "Revised Safety Evaluation by the Office of Nuclear Reactor Regulation; Topical Report WCAP-16168-NP-A, Revision 2, 'Risk-Informed Extension of the Reactor Vessel In-service Inspection Interval,' Pressurized Water Reactor Owners Group, Project No. 694," July 26, 2011, ADAMS Accession No. ML111600303.
5. BWRVIP-108: BWR Vessels and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Shell Welds and Nozzle Blend Radii, EPRI, Palo Alto, CA 2002. 1003557.
6. US NRC, Safety Evaluation of Proprietary EPRI Report, "BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Inner Radius (BWRVIP-108)," December 19, 2007, ADAMS Accession No. ML073600374.
7. BWRVIP-241: BWR Vessels and Internals Project, Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Shell Welds and Nozzle Blend Radii, EPRI, Palo Alto, CA 2010. 1021005.
8. US NRC, Safety Evaluation of Proprietary EPRI Report, "BWR Vessel and Internals Project, Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Shell Welds and Nozzle Blend Radii (BWRVIP-241)," April 19, 2013, ADAMS Accession Nos. ML13071A240 and ML13071A233.
9. Code Case N-702, "Alternate Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds," ASME Code Section XI, Division 1, Approval Date: February 20, 2004.
10. U. S. NRC Regulatory Guide 1.147, Revision 18, "Inservice Inspection Code Case Acceptability, ASME Code Section XI, Division 1," dated March 2017.
11. Southern Nuclear Company, NL-16-0724, "Vogtle Electric Generating Plant, Units 1 & 2, Proposed Inservice Inspection Alternative VEGP-ISI-ALT-11, Version 1.0," June 28, 2016, ADAMS Accession No. ML16180A046.
12. Southern Nuclear Company, NL-16-0723, "Joseph M. Farley Nuclear Plant, Unit 1, Proposed Inservice Inspection Alternative FNP-ISI-ALT-19, Version 1.0," June 30, 2016, ADAMS Accession No. ML16182A475.
13. Michael T. Markley (NRC) to Charles R. Pierce (Southern Nuclear), "Vogtle Electric Generating Plant, Units 1 and 2, and Joseph M. Farley Nuclear Plant, Unit 1 – Alternative to Inservice Inspection Regarding Reactor Pressure Vessel Threads

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- Inflange Inspection (CAC Nos. MF8061, MF8062, MF8070),” January 26, 2017,
ADAMS Accession No. ML ML17006A109.
14. N. Palm (EPRI), BWR Vessel & Internals Project (BWRVIP) Memo No. 2019-016,
“White Paper on Suggested Content for PFM Submittals to the NRC,” February 27,
2019, ADAMS Accession No. ML19241A545.
 15. Structural Integrity Associates, Inc. Calculation No, FP-VOG-323, Revision 0,
“FatiguePro Analysis of Plant Data for Vogtle Units 1 and 2 through 2018”.

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APPENDIX A

VOGTLE UNIT 1 AND UNIT 2 APPLICABILITY

Plant-Specific Applicability

Section 9 of Reference [1] provides requirements that must be demonstrated in order to apply the representative stress and flaw tolerance analyses to a specific plant. Plant-specific evaluation of these requirements for Vogtle Units 1 & 2 is provided in Table A1.

Table A1 indicates that all plant-specific requirements are met for Vogtle Units 1 & 2. Therefore, the results and conclusions of the EPRI report are applicable to Vogtle Units 1 & 2.

Table A1. Applicability of Reference [1] Representative Analyses to Vogtle Units 1 & 2

Category	Requirement from Reference [1]	Applicability to Vogtle Units 1 & 2
General Requirements	The nozzle-to-shell weld shall be one of the configurations shown in Figure 1-1 or Figure 1-2 of Reference [1].	The Vogtle Units 1 & 2 MS and FW nozzle configurations are shown in Figures A1 and A2, and are representative of the configuration shown in Figure 1-1 of Reference [1].
	The materials of the SG shell, FW nozzles, and MS nozzles must be low alloy ferritic steels which conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.	The Vogtle Units 1 & 2 nozzles are fabricated of SA-508, Class 2A material, and the SG vessel heads/shells are fabricated from SA-533, Gr. A, Cl. 2 material. Both of these materials conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.
	The number of transients shown in Table 5-5 of Reference [1] are bounding for application over a 60-year operating life.	The transient cycles in Table 5-5 of Reference [1] meet or exceed the 60-year projected cycles for Vogtle Units 1 and 2 as shown in Table A2 [15].
SG Feedwater Nozzle	The piping attached to the FW nozzle must be 14-inch to 18-inch NPS.	The Vogtle Units 1 & 2 FW piping lines are both 16-inch NPS.

Category	Requirement from Reference [1]	Applicability to Vogtle Units 1 & 2
	The FW nozzle design must have an integrally attached thermal sleeve	The Vogtle Units 1 & 2 FW nozzle configuration is shown in Figure A1 and has an integrally attached thermal sleeve.
SG Main Steam Nozzle	For Westinghouse and CE plants, the piping attached to the SG MS nozzle must be 28-inch to 36-inch NPS.	Vogtle Units 1 & 2 are Westinghouse 4-loop PWRs. The Vogtle Units 1 & 2 MS nozzles have 32" to 26" reducers. The pipe size of the attached reducer to the nozzle end is 32" NPS which satisfies the intent of this requirement.
	For B&W SGs, the piping attached to the main steam nozzle must be 22-inch to 26-inch NPS	This requirement is not applicable for Vogtle Units 1 & 2 because they are both Westinghouse 4-loop units.
	The SG must have one main steam nozzle that exits the top dome of the SG.	As shown in Figure A3, Vogtle Units 1 & 2 both have one MS nozzle per SG that exits the top dome of each SG.
	The main steam nozzle shall not significantly protrude into the SG (e.g., see Figure 4-7 of Reference [1]) or have a unique nozzle weld configuration (e.g., see Figure 4-6 of Reference [1]).	The Vogtle Units 1 & 2 MS nozzle configuration is shown in Figures A2 and A3, and does not protrude significantly into the SG. The Vogtle Units 1 & 2 MS nozzles are NOT unique. They are similar to the configuration selected for analysis (Figure 4-8 of Reference [1]).

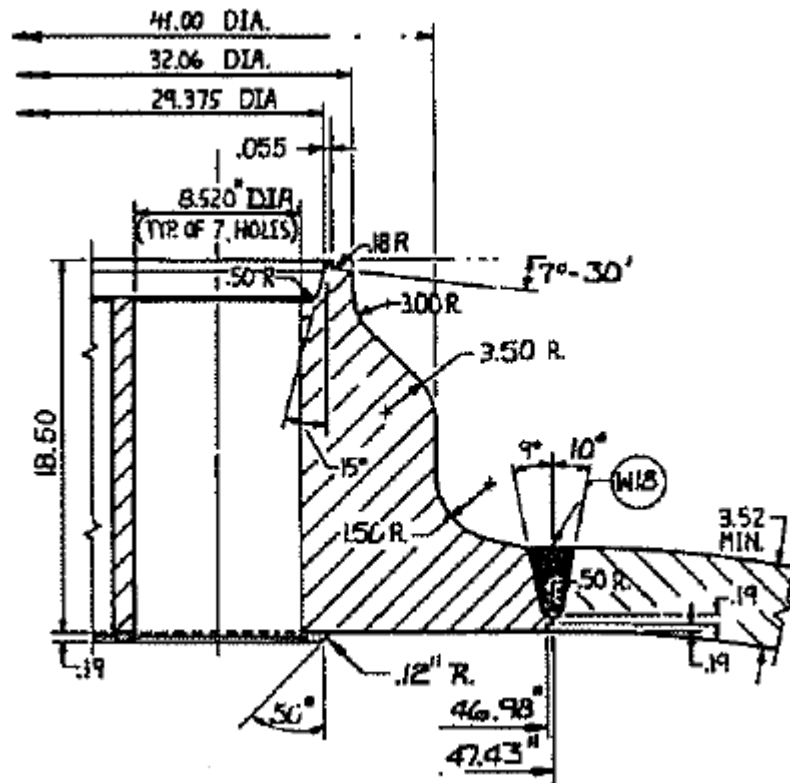


Figure A2 Vogtle Units 1 & 2 SG Main Steam Nozzle Configuration

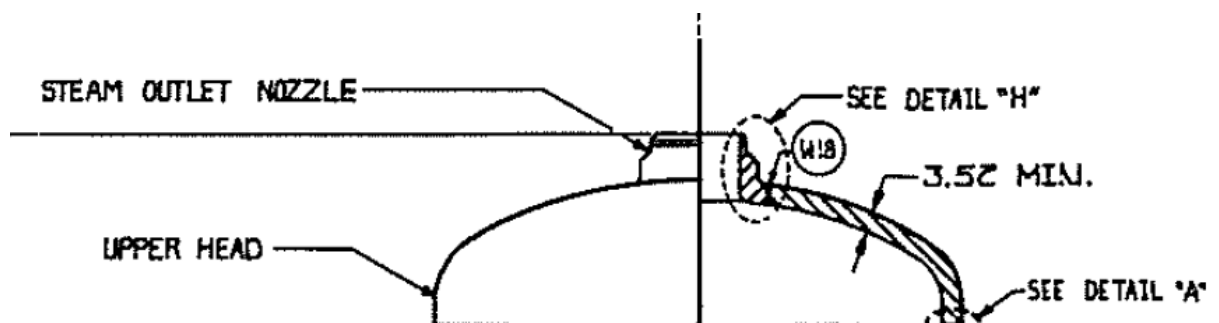


Figure A3 Steam Generator Upper Head

Table A2

Transient Cycles for Vogtle Units 1 and 2 in Comparison to the Requirements in Reference [1]

Transient	Cycles From Table 5-5 of EPRI Report 3002014590 [1]	Unit 1 60-Year Projected Cycles From Table 3 of [15]	Unit 2 60-Year Projected Cycles From Table 4 of [15]	Allowable Cycles From Tables 3 and 4 of [15]
Heatup/Cooldown	300	69	76/75 ⁽⁵⁾	200
Plant Loading ⁽¹⁾	5000	164	141	500
Plant Unloading ⁽²⁾	5000	66	36	500
Loss of Load ⁽³⁾	360	119	89	760
Loss of Power ⁽⁴⁾	60	3	3	40

Notes:

- (1) Transient listed as Plant Loading 0-15% Power in Tables 3 and 4 of [15].
- (2) Transient listed as Plant Unloading 0 – 15% Power in Tables 3 and 4 of [15].
- (3) Loss of Load transient is a bundled to conservatively envelope a combination of several transients listed in Tables 3 and 4 of [15]:
 - Loss of Load w/o Rx Trip
 - Loss of RC Flow 1 Loop @ Power
 - Large Step Load Decrease
 - Reactor Trip (CD and SI)
 - Reactor Trip (CD no SI)
 - Reactor Trip (No Cooldown)
- (4) Transient listed as Loss of Offsite Power in Tables 3 and 4 of [15].
- (5) Cycles for Heatup and Cooldown, respectively.

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APPENDIX B

VOGTLE UNITS 1 & 2 INSPECTION HISTORY

VOGTLE UNITS 1 & 2 INSPECTION HISTORY

Currently, the MS and FW nozzle components for VEGP Units 1 & 2 satisfy all of the inspection requirements of ASME Code, Section XI, 2007 Edition including the 2008 Addenda.

(Note: The first digit in each Component ID depicts the unit for each component.)

MS Nozzle

	Date	Interval/Period	Components ID	Exam Results	Coverage⁽³⁾
Item No. C2.21	10/26/88	1 st /1 st	11201-B6-001-W18	NRI	50%
	10/2/97	2 nd /1 st	11201-B6-001-W18	NRI	50%
	4/1/08	3 rd /1 st	11201-B6-001-W18	NRI	50%
	3/29/14	3 rd /3 rd	11201-B6-001-W18	NRI	50%
	10/9/90	1 st /1 st	21201-B6-001-W18	RI ⁽¹⁾	50%
	10/19/99	2 nd /1 st	21201-B6-001-W18	NRI	50%
	10/02/08	3 rd /1 st	21201-B6-001-W18	NRI	50%
	9/24/14	3 rd /3 rd	21201-B6-001-W18	NRI	50%

(1) Subsurface Planer flaw acceptable per IWC-3510-1.

(3) The following relief requests address <90% inspection coverage for the 1st, 2nd, and 3rd Intervals: RR-29, RR-14, and VEGP-ISI-RR-05.

FW Nozzle

	Date	Interval/Period	Components ID	Exam Results	Coverage⁽³⁾
Item No. C2.21	9/30/94	1 st / 3 rd	11201-B6-002-W19	NRI	50%
	3/29/05	2 nd /3 rd	11201-B6-002-W19	NRI	50%
	10/7/09	3 rd /1 st	11201-B6-002-W19	NRI	80%
	10/10/96	1 st /3 rd	21201-B6-002-W19	RI ⁽²⁾	50%
	10/1/05	2 nd /3 rd	21201-B6-002-W19	RI ⁽²⁾	50%
	3/19/10	3 rd /3 rd	21201-B6-002-W19	RI ⁽²⁾	80%
Item C2.22	10/6/94	1 st /3 rd	11201-B6-002-IR04	NRI	100%
	3/25/05	2 nd /3 rd	11201-B6-002-IR04	NRI	100%
	10/7/09	3 rd /1 st	11201-B6-002-IR04	NRI	100%
	9/27/96	1 st /3 rd	21201-B6-002-IR04	NRI	100%
	10/1/05	2 nd /3 rd	21201-B6-002-IR04	NRI	100%
	3/18/10	3 rd /2 nd	21201-B6-002-IR04	NRI	100%

(2) Subsurface Planer flaw acceptable per IWC-3510-1.

(3) The following relief requests address <90% inspection coverage for the 1st, 2nd, and 3rd Intervals: RR-29, RR-14, and VEGP-ISI-RR-05.

Enclosure 1 to NL-20-1011
Proposed Alternative VEGP-ISI-ALT-04-04, Version 2.0,
in Accordance with 10 CFR 50.55a(z)(1)

APPENDIX C
RESULTS OF INDUSTRY SURVEY

Overall Industry Inspection Summary

The results of an industry survey of past inspections of SG MS and FW nozzles are summarized in Section 3 of Reference [1]. Table C1 provides a summary of the combined survey results for Item Nos. C2.22, C2.21, and C2.32⁽¹⁾. The results identify that SG MS and FW Nozzle-to-Shell Welds and Nozzle Inside Radius Section examinations adversely impact outage activities including worker exposure, personnel safety, and radwaste. A total of 74 domestic and international BWR and PWR units responded to the survey and provided information representing all PWR plant designs currently in operation in the U.S. This included 2-loop, 3-loop, and 4-loop PWR designs from each of the PWR nuclear steam supply system (NSSS) vendors (i.e., Babcock and Wilcox (B&W), Combustion Engineering (CE), and Westinghouse). A total of 727 examinations for Item Nos. C2.21, C2.22, and C2.32⁽¹⁾ components were conducted, with 563 of these specifically for PWR components. The majority of the PWR examinations were performed on SG MS and FW nozzles. Only one PWR examination identified two (2) flaws that exceeded ASME Code Section XI acceptance criteria. The flaws were linear indications of 0.3” and 0.5” in length and were detected in a MS nozzle-to-shell weld using magnetic particle examination techniques. The indications were dispositioned by light grinding (ADAMS Accession No. ML13217A093).

Table C1 – Summary of Survey Results

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

¹ Item No. C2.32 is similar to Item No. C2.21 and was evaluated in the Reference [1] technical basis and included in the industry survey. Vogtle Units 1 & 2 have not performed any examinations on Item No. C2.32 components.

**Vogtle Electric Generating Plant, Units 1 & 2
Proposed Inservice Inspection Alternative VEGP-ISI-ALT-04-04 Version 2.0**

Enclosure 2

SI Report No. 1900064.406.R0



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July 16, 2020
SI Report No. 1900064.406.R0

Mr. Robert Grizzi
Program Manager, NDE PD Operations and Issue Program Support
Nuclear Sector
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1300 West WT Harris Blvd.
Charlotte, NC 28262

Subject: Evaluations to Address Limited Examination Coverage of Vogtle Electric
Generating Plant Units 1 and 2 Steam Generator Main Steam and Feedwater
Nozzle-to-Shell Welds and Nozzle Inside Radius Sections

Dear Bob:

Per your request, Structural Integrity Associates, Inc. (SI) performed an evaluation to determine the failure probabilities (rupture and leakage) considering the plant specific examination coverage for the Vogtle Electric Generating Plant (VEGP) steam generator main steam and feedwater nozzle-to-shell welds and nozzle inside radius sections using the PROMISE software. The evaluation methodology and results are presented in Attachment A to this letter report.

We appreciate the opportunity to provide you with this service. Please do not hesitate to let me know if you have any questions.

Very truly yours,

A handwritten signature in black ink, appearing to read 'Scott Chesworth', is written over a light blue horizontal line.

Scott Chesworth
Senior Consultant

cc: G. Stevens (EPRI)

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ATTACHMENT A

**EVALUATIONS TO DETERMINE FAILURE PROBABILITIES CONSIDERING 50%
EXAMINATION COVERAGE FOR VOGTLE ELECTRIC GENERATING PLANT UNITS 1 AND 2
STEAM GENERATOR MAIN STEAM AND FEEDWATER NOZZLE-TO-SHELL WELDS AND
NOZZLE INSIDE RADIUS SECTIONS**

BACKGROUND

In Section 8.2.5 of EPRI Report 3002014590 (Reference [1]), the impacts of reduced examination coverage for steam generator (SG) main steam (MS) and feedwater (FW) nozzle-to-shell welds and nozzle inside radius sections were qualitatively evaluated. It was concluded that inservice inspection (ISI) examination coverage of any extent after the preservice inspection (PSI) examination (which has an assumed 100% coverage) is acceptable, since the probabilities of rupture and leakage with only a PSI examination (and no other follow-on ISI examinations for 80 years) are three orders of magnitude below the acceptance criteria for all but a single case, as shown in Table 1 (reproduction of Table 8-9 of Reference [1]).

The sole exception is the probability of leakage (but not rupture) for Case ID FEW-P3A, **highlighted** in Table 1. The probability of leakage after PSI and after 60 and 80 years of operation for this case exceeds the acceptance criteria by one order of magnitude (2.44×10^{-6} and 1.19×10^{-5} vs. an allowed value of 1×10^{-6}). As explained in Sections 8.2.5 and 8.2.4.1.1 of Reference [1], this result was considered acceptable because (1) the increased likelihood of pressure boundary leakage is detectable by plant operators, (2) plant procedures allow for safe plant shutdown once any leakage is detected, and (3) the probability of rupture values are maintained three orders of magnitude below the acceptance criterion, thus eliminating the possibility of failure of the pressure boundary.

Southern Nuclear Operating Company (SNOC) is requesting an examination alternative [2] to that mandated in ASME Code, Section XI for Examination Category C-B, Item No. C.2.21 and C.2.22 components associated with the SG MS and FW nozzles for Vogtle Electric Generating Plant (VEGP) Units 1 and 2. The proposed alternative is to extend the ISI interval for these examinations to 30 years (from the current ASME Code, Section XI 10-year requirement). The examination history for VEGP Units 1 and 2 is provided in Appendix B of Reference [2] and is reproduced in Tables 2 and 3. As indicated in Tables 2 and 3, in addition to the PSI examination, four 10-year ISI examinations have been performed for VEGP Units 1 and 2 MS nozzle-to-shell welds, and three 10-year ISI examinations have been performed for VEGP Units 1 and 2 FW nozzle-to-shell welds, subsequent to the PSI examinations. All examinations had limited coverage ranging from 50% to 80%.

In the present study, additional probabilistic fracture mechanics (PFM) evaluations were performed to determine the probabilities of rupture and leakage based on the actual examination coverage obtained at VEGP Units 1 and 2. This evaluation only addresses limited coverage for the Item No. C.2.21 nozzle-to-shell welds, since the Item No. C.2.22 MS nozzle inner radii are exempt from ISI examinations and the FW nozzle inner radii had 100% coverage for all past ISI examinations (see Table 3).

Table 1 Probability of Rupture (per Year) and Probability of Leakage (per Year) for PSI Only (Table 8-9 from Ref. [1])

Component	Case Identification	P(Rupture) at 80 yrs.	P(Leakage) at			
			20 yrs.	40 yrs.	60 yrs.	80 yrs.
Westinghouse Main Steam Nozzle (SGW)	SGW-P1N	1.25E-12	5.00E-12	2.50E-12	1.67E-12	1.25E-12
	SGW-P2C	1.25E-09	5.00E-09	2.50E-09	1.67E-09	1.25E-09
	SGW-P2A	1.25E-09	5.00E-09	2.50E-09	1.67E-09	1.25E-09
B&W Main Steam Nozzle (SGB)	SGB-P1N	1.25E-12	5.00E-12	2.50E-12	1.67E-12	2.50E-12
	SGB-P2N	1.25E-12	5.00E-12	2.50E-12	1.67E-12	1.25E-12
	SGB-P3C	1.25E-09	5.00E-09	2.50E-09	1.67E-09	1.25E-09
	SGB-P3A	1.25E-09	5.00E-09	2.50E-09	1.67E-09	1.25E-09
	SGB-P4A	1.25E-09	5.00E-09	2.50E-09	1.67E-09	1.25E-09
	SGB-P4C	1.25E-09	5.00E-09	2.50E-09	1.67E-09	1.25E-09
Westinghouse Feedwater Nozzle (FEW)	FEW-P1N	1.25E-12	4.50E-11	9.88E-09	9.68E-08	3.20E-07
	FEW-P2N	1.25E-12	5.00E-12	2.50E-12	1.67E-12	1.25E-12
	FEW-P3C	1.25E-09	5.00E-09	2.50E-09	1.67E-09	1.25E-09
	FEW-P3A	1.25E-09	5.00E-09	2.08E-07	2.44E-06	1.19E-05
	FEW-P4A	1.25E-09	5.00E-09	2.50E-09	1.67E-09	1.25E-09
	FEW-P4C	1.25E-09	5.00E-09	2.50E-09	1.67E-09	1.25E-09

Note: The limiting case is displayed in **bold red text highlighted in yellow.**

Table 2 VEGP Units 1 and 2 Main Steam Nozzle Examination History Summary [2]

	Date	Interval/Period	Components ID	Exam Results	Coverage⁽³⁾
Item No. C2.21	10/26/88	1 st /1 st	11201-B6-001-W18	NRI	50%
	10/2/97	2 nd /1 st	11201-B6-001-W18	NRI	50%
	4/1/08	3 rd /1 st	11201-B6-001-W18	NRI	50%
	3/29/14	3 rd /3 rd	11201-B6-001-W18	NRI	50%
	10/9/90	1 st /1 st	21201-B6-001-W18	RI ⁽¹⁾	50%
	10/19/99	2 nd /1 st	21201-B6-001-W18	NRI	50%
	10/02/08	3 rd /1 st	21201-B6-001-W18	NRI	50%
	9/24/14	3 rd /3 rd	21201-B6-001-W18	NRI	50%

(1) Subsurface Planer flaw acceptable per IWC-3510-1.

(3) The following relief requests address <90% inspection coverage for the 1st, 2nd, and 3rd intervals: RR-29, RR-14, and VEGP-ISI-RR-05.

Table 3 VEGP Units 1 and 2 Feedwater Nozzle Examination History Summary [2]

	Date	Interval/Period	Components ID	Exam Results	Coverage⁽³⁾
Item No. C2.21	9/30/94	1 st / 3 rd	11201-B6-002-W19	NRI	50%
	3/29/05	2 nd /3 rd	11201-B6-002-W19	NRI	50%
	10/7/09	3 rd /1 st	11201-B6-002-W19	NRI	80%
	10/10/96	1 st /3 rd	21201-B6-002-W19	RI ⁽²⁾	50%
	10/1/05	2 nd /3 rd	21201-B6-002-W19	RI ⁽²⁾	50%
	3/19/10	3 rd /3 rd	21201-B6-002-W19	RI ⁽²⁾	80%
Item C2.22	10/6/94	1 st /3 rd	11201-B6-002-IR04	NRI	100%
	3/25/05	2 nd /3 rd	11201-B6-002-IR04	NRI	100%
	10/7/09	3 rd /1 st	11201-B6-002-IR04	NRI	100%
	9/27/96	1 st /3 rd	21201-B6-002-IR04	NRI	100%
	10/1/05	2 nd /3 rd	21201-B6-002-IR04	NRI	100%
	3/18/10	3 rd /2 nd	21201-B6-002-IR04	NRI	100%

(2) Subsurface Planer flaw acceptable per IWC-3510-1.

(3) The following relief requests address <90% inspection coverage for the 1st, 2nd, and 3rd intervals: RR-29, RR-14, and VEGP-ISI-RR-05.

EVALUATION

As shown in Tables 2 and 3, examination coverage for Item No. C2.21 for both the VEGP Units 1 and 2 MS and FW nozzle-to-shell welds ranged from 50% to 80%. Item No. C2.21 was therefore evaluated using the minimum coverage of 50% achieved during these exams. Per Table 1, Case ID FEW-P3A has the highest probability of leakage value of 1.19×10^{-5} , so the evaluation was performed for this limiting case.

Two ISI scenarios were considered:

1. The current ASME Code, Section XI examination requirement, which involves 10-year interval examinations after the PSI examination. For this case, the evaluation was performed assuming VEGP Units 1 and 2 will continue with the current 10-year inspection interval through 70 years of operation (i.e., ISI at 10, 20, 30, 40, 50, 60, and 70 years).
2. The alternative [2] requested for VEGP Units 1 and 2, where only the first three 10-year ISI examinations are performed after the PSI examination, followed by one examination on a 30-year interval (i.e., ISI at 10, 20, 30, and 60 years).

Plant records reflect that ASME Code, Section III PSI examinations involving radiographic testing (RT) were performed and found acceptable for the affected welds of the VEGP Units 1 and 2 MS and FW nozzles. The acceptability of the ASME Section III RT examinations indicates that 100% coverage was achieved during these examinations. PSI ultrasonic testing (UT) examinations were not performed on the vessel side of the MS or FW welds due to difficulties associated with the nozzle configurations. However, because of the success with the RT examinations, this is considered acceptable since as discussed in Section 8.2.4.1.1 of the Reference report [1], PSI refers to the collective initial ASME Code, Section III and Section XI examinations.

A comprehensive study performed in References [4] and [5] concluded that detection and sizing of flaws utilizing RT is as effective as UT. Figure 3.6 of Reference [4] provides theoretical probability of detection (POD) curves for RT examinations. The most conservative of these POD curves is compared to that from UT examinations used in the Reference [1] evaluations and is presented in Figure 1. As shown in this figure, except for extremely shallow flaws (less than 0.04 inches), the POD curve used in Reference [1] can be conservatively applied to RT examinations. It should be noted that the minimum flaw depth in all the simulations is 0.075 inches which is greater than the flaw depths at which RT governs.

The **PROMISE** software [3], which was used to perform the PFM evaluations in Reference [1], was used to perform the present evaluation which covers 80 years of plant operation. The probabilities of rupture and leakage were determined for the two ISI scenarios discussed above (Section XI and the requested alternative) for the limiting case (Case ID FEW-P3A).

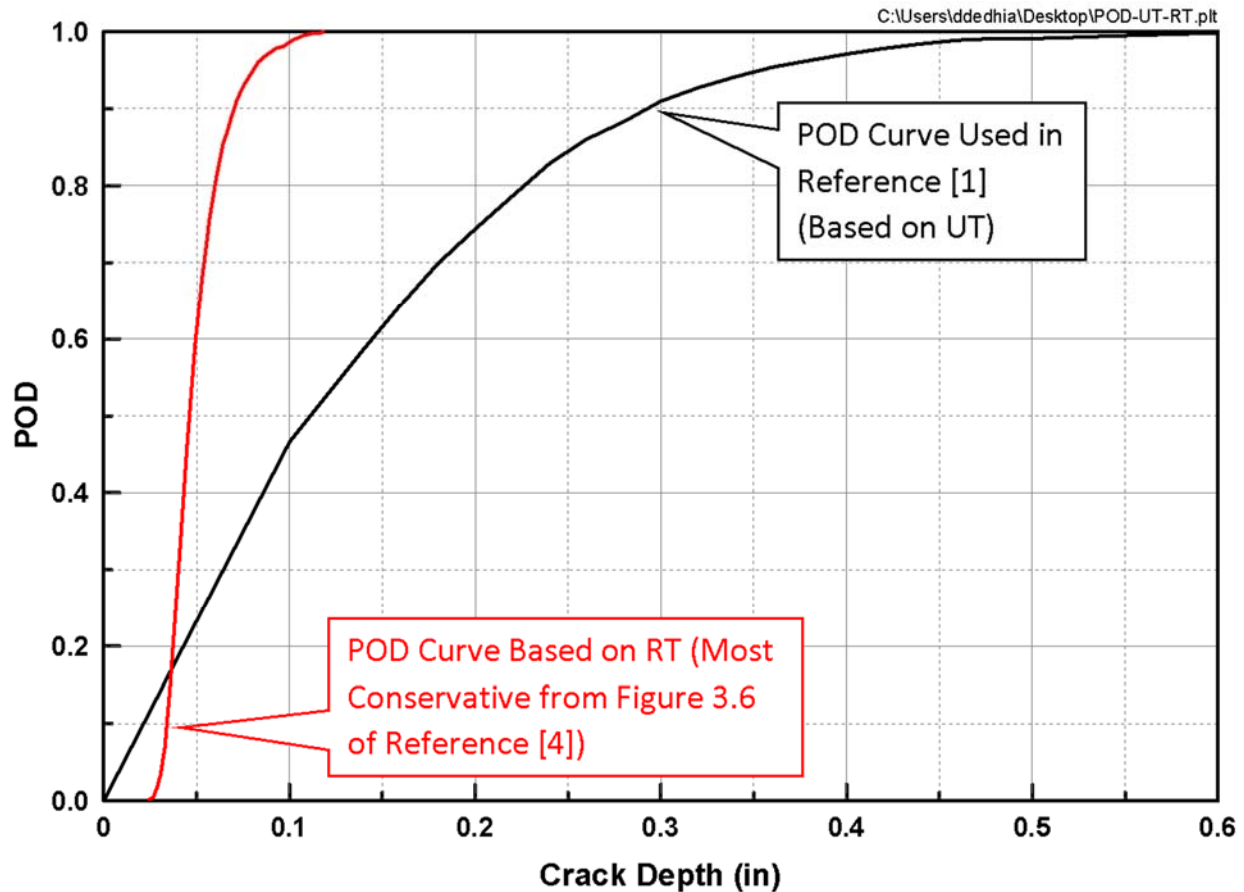


Figure 1 Comparison of POD Curve Used in Reference [1] (Based on UT) to that Based on RT [4]

RESULTS

The results of the evaluation are presented in Table 4. As shown in this table, the probabilities of rupture for 100% ISI coverage and 50% ISI coverage are identical and remain unchanged from the results shown in Table 1 for PSI examination only. The values of 1.25×10^{-9} for both scenarios are approximately three orders of magnitude less than the acceptance criteria of 1×10^{-6} specified in Reference [1].

The probabilities of leakage for the ASME Section XI (PSI+10+20+30+40+50+60+70) and alternative ISI (PSI+10+20+30+60) scenarios assuming 50% ISI coverage are both nearly equal at 5.9×10^{-6} , which is above the acceptance criterion. However, these values are lower than the PSI only (100% coverage) value of 1.19×10^{-5} reported in Table 1 (Table 8-9 of Reference [1]). Therefore, for the same reasons as for the rupture probabilities, the probability of leakage is decreased by performing ISI, regardless of coverage.

As indicated by comparison of the two scenarios in Table 4, the probability of leakage for 80 years for the alternative examination scenario (Scenario #2) is essentially identical to Scenario #1 where the ASME Code, Section XI 10-year examinations are performed through 70 years of

operation. A further comparison of the cumulative probabilities of leakage vs. time for the two ISI scenarios is shown in Figure 2. The results presented in this figure show that the probability of leakage for the alternative ISI scenario is almost identical to the scenario where the ASME Code, Section XI 10-year ISI examinations are continued through 70 years of operation. Regardless of which of the two ISI scenarios is considered, the probability of leakage remains the same after 80 years of operation. Therefore, changing the ISI schedule to the proposed alternative [2] does not alter the probability of leakage compared to the current ASME Code, Section XI schedule of repeated 10-year examinations.

Table 4 Sensitivity Study for ISI Coverage - Limiting Case FEW-P3A

Case ID	Scenario	PSI/ISI Schedule	Probability of Rupture (per Year) at 80 years		Probability of Leakage (per Year) at 80 years	
			100% ISI Coverage	50% ISI Coverage	100% ISI Coverage	50% ISI Coverage
FEW-P3A	1	PSI+10+20+30+40+50+60+70	1.25E-09	1.25E-09	1.25E-09	5.93E-06
	2	PSI+10+20+30+60	1.25E-09	1.25E-09	2.50E-09	5.95E-06

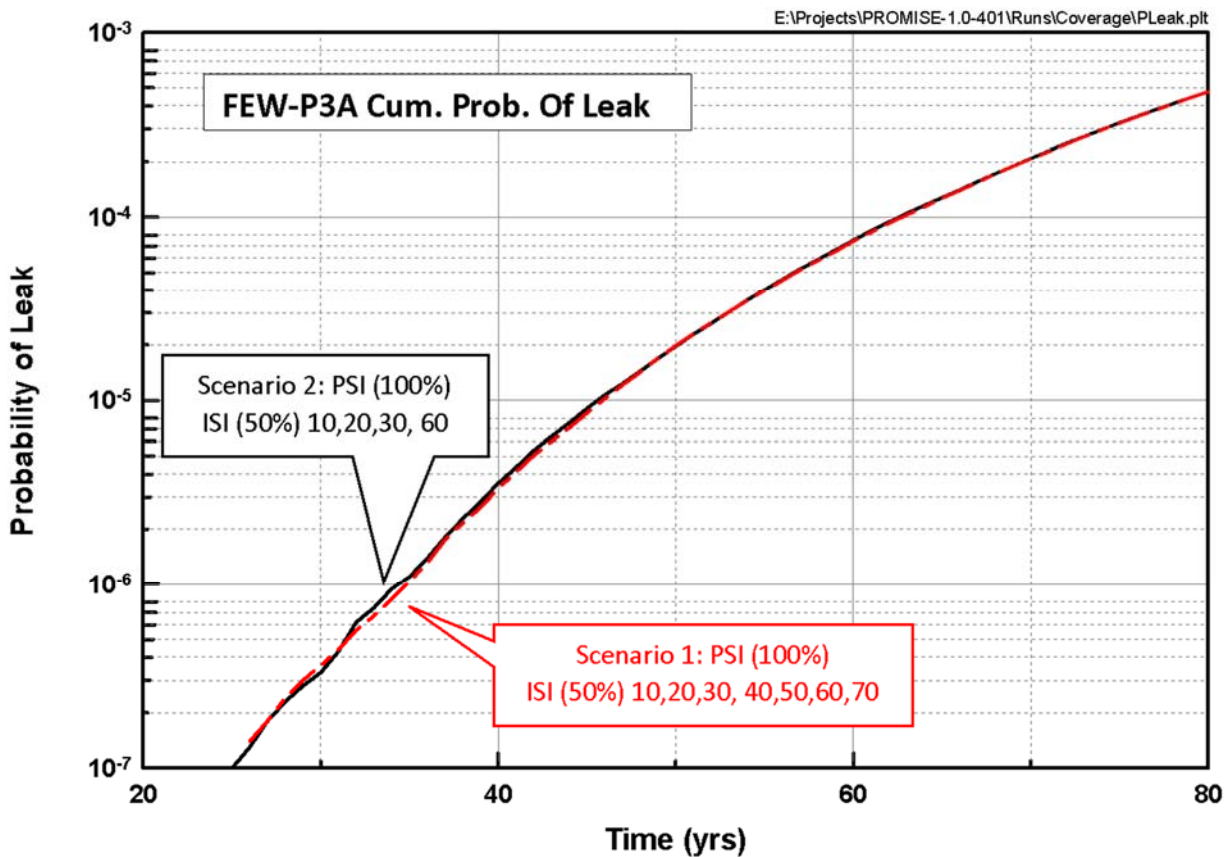


Figure 2 Comparison of Probability of Leakage for Two ISI Scenarios (Scenario #1 = PSI+10+20+30+40+50+60+70 = Current ASME Code, Section XI Requirement) and (Scenario #2 = PSI+10+20+30+60 = VEGP's Alternative Request) for Limiting Case ID FEW-P3A

SUMMARY AND CONCLUSIONS

PFM evaluations for the minimum 50% coverage achieved during examinations at VEGP Units 1 and 2 with various ISI scenarios were performed to determine the impact of reduced ISI coverage for the SG MS and FW nozzle-to-shell welds (ASME Code, Section XI, Examination Category C-B, Item No. C2.21). Because of the successful ASME Code, Section III RT examinations performed after fabrication of the welds, 100% coverage was used for the PSI examinations. The POD curve for UT used for ISI examinations in Reference [1] was conservatively applied to the PSI examinations.

It is concluded that limited coverage of as low as 50% for the VEGP Units 1 and 2 SG MS and FW nozzle-to-shell welds is acceptable for continued operation for the alternative requested by SNOC in Reference [2]. The probabilities of rupture for 50% coverage for the limiting case are three orders of magnitude below the acceptance criteria for 80 years of operation. The probability of leakage for the limiting case using the alternative ISI scenario (PSI+10+20+30+60) is slightly above the acceptance criterion (5.95×10^{-6} vs. 1×10^{-6}); however, this probability of leakage is almost identical to the scenario where the ASME Code, Section XI 10-year ISI examinations are continued through 70 years of operation (5.93×10^{-6}). Therefore, the examination interval associated with the VEGP Request for Alternative [2] does not increase the probabilities of rupture and does not significantly increase the probability of leakage from the currently required ASME Code, Section XI ISI examination interval.

As discussed in Sections 8.2.4.1.1 and 8.2.5 of Reference [1], the fact that the probability of leakage at location FEW-P3A slightly exceeds the acceptance criterion does not compromise plant safety. This is because pressure boundary leakage is detectable by plant operators, plant procedures allow for safe plant shutdown once any leakage is detected, and the probability of rupture values are maintained well below the acceptance criterion for 80 years of operation even under the scenario of PSI examination only.

REFERENCES

1. *Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Nozzle Inside Radius Sections*. EPRI, Palo Alto, CA: 2019. 3002014590.
2. Letter No. NL-19-0832 from Cheryl A. Gayheart (Southern Nuclear) to U. S. Nuclear Regulatory Commission, *Vogtle Electric Generating Plant, Units 1 & 2, Proposed Inservice Inspection Alternate VEGP-ISI-ALT-04-04*, December 11, 2019.
3. Structural Integrity Associates Report DEV1806.402, *PROMISE 1.0 Theory and User's Manual*, Revision 0.
4. T. L. Moran, P. Ramuhalli, A. F. Pardini, M. T. Anderson and S. R. Doctor, *Replacement of Radiography with Ultrasonics for Nondestructive Inspection of Welds - Evaluation of Technical Gaps - An Interim Report*, Pacific Northwest National Laboratory Report PNNL-19086, April 2010, ADAMS Accession No. ML101031254.
5. T. L. Moran, M. Prowant, C. A. Nove, A. F. Pardini, S. L. Crawford, A. D. Cinson and M. T. Anderson, *Applying Ultrasonic Testing in Lieu of Radiography for Volumetric Examination of Carbon Steel Piping*, NUREG/CR-7204 (PNNL-24232), September 2015.

Prepared by:



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Date

**Vogtle Electric Generating Plant, Units 1 & 2
Proposed Inservice Inspection Alternative VEGP-ISI-ALT-04-04 Version 2.0**

Enclosure 3

SI Report No. 1900064.407.R2



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August 5, 2020
SI Report No. 1900064.407.R2

Mr. Robert Grizzi
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Nuclear Sector
Electric Power Research Institute
1300 West WT Harris Blvd.
Charlotte, NC 28262

Subject: Evaluations to Address Benchmarking of the **PROMISE** Software to Include the Effects of Inspections

Dear Bob:

Per your request, Structural Integrity Associates, Inc. (SI) has performed evaluations to benchmark the **PROMISE** software against the **VIPER-NOZ** software to include the effect of inspections. The evaluation methodology and results are presented in Attachment A to this letter report. The updated results include an additional case with a different PSI/ISI combination. Furthermore, probabilities of rupture are provided in addition to the probabilities of leakage, as requested by the U.S. NRC during the July 27th, 2020 **PROMISE** software audit.

We appreciate the opportunity to provide you with this service. Please do not hesitate to let me know if you have any questions.

Very truly yours,

A handwritten signature in black ink, appearing to read 'Scott Chesworth', is written over a light blue horizontal line.

Scott Chesworth
Senior Consultant

cc: G. Stevens (EPRI)

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ATTACHMENT A
EVALUATIONS TO BENCHMARK THE PROMISE SOFTWARE TO INCLUDE THE EFFECT
OF PRESERVICE AND INSERVICE INSPECTION

BACKGROUND

In Section 8.2.3.2.2 of EPRI Report 3002014590 (Reference [1]), the **PROMISE** probabilistic fracture mechanics (PFM) software [2] was benchmarked against the **VIPER-NOZ** software code [3]. The **VIPER-NOZ** software, used for performing PFM analyses of the BWR vessel nozzle-to-shell welds and the nozzle inner radii in BWRVIP-108-A [4], was chosen for the benchmarking because it was reviewed extensively by the NRC as a part of their Safety Evaluation (SE) approving BWRVIP-108-A, and subsequently BWRVIP-241-A [5]. A summary of the key inputs to the benchmarking exercise is provided in Table 8-4 of Reference [1], which is reproduced in Table 1. The results of the benchmarking are shown Table 8-5 of Reference [1], which is reproduced in Table 2. As shown in Table 2, the probabilities of leakage are very similar.

Table 1
Benchmarking Inputs (Table 8-4 of Reference [1])

Input	Value
No. of cracks per inner radius section	1, constant
Crack depth distribution	PVRUF
Fracture toughness (ksi√in)	Normal (200,5)
PSI	None
ISI	None
POD Curve	Not applicable
Fatigue crack growth law and threshold	BWRVIP-108-A
Uncertainties on transients	None
Residual stresses (ksi)	None

Table 2
Comparison of Cumulative Probability of Leakage Between PROMISE and VIPER-NOZ for Benchmarking (Table 8-5 of Reference [1])

Cyclic Stress (ksi)	Cycles/year	PROMISE	VIPER-NOZ
25	500	2.8E-2	3.1E-2
15	500	1.7E-4	3.0E-4

As shown in Table 1, neither preservice inspection (PSI) nor inservice inspection (ISI) were considered in the benchmarking. Since one of the key features of the **PROMISE** software is its ability to evaluate the impacts of ISI on failure probabilities, the NRC requested during their July 1, 2020 audit of the **PROMISE** software that the benchmarking should include consideration of inspections to provide a benchmarking demonstration of the ISI capabilities of **PROMISE**. Furthermore, during the July 27, 2020 follow-on audit, the NRC requested that in addition to the probability of leakage, a comparison of the probability of rupture be included in the benchmarking.

TECHNICAL APPROACH

Two scenarios using different PSI/ISI scenarios were considered in the benchmarking. The input parameters for the two scenarios are provided in Tables 3 and 4. The changes to the inputs from the previous benchmarking exercise included in Reference [1] are shown in *red italic text* in Tables 3 and 4. As shown in these tables, a combination of PSI/ISI cases were considered in each of the two scenarios to determine the trending associated with ongoing inspections with both software codes. The two scenarios are defined as follows:

- Scenario No. 1 considered four cases of PSI alone and PSI followed by 20-year ISI examinations up to 60 years as shown in Table 3.
- Scenario No. 2 considered eight cases of PSI alone and PSI followed by 10-year ISI examinations up to 70 years as shown in Table 4.

The POD curve used in Reference [1] was also employed in this updated benchmarking exercise (i.e., Figure 8-2 of Reference [1] = the POD curve from BWRVIP-108-A and BWRIP-241-A).

Similar to the benchmarking exercise performed in Reference [1], the nozzle corner crack model was used to determine the stress intensity factors since the model is common to both software codes. A conservative combination of stress and fracture toughness was used to increase the likelihood of failure. For simplicity, a constant through-wall stress of 30 ksi was applied. The mean fracture toughness was lowered to 100 ksi $\sqrt{\text{in}}$ (from the original benchmarking value of 200 ksi $\sqrt{\text{in}}$) with a standard deviation of 20 ksi $\sqrt{\text{in}}$ (compared to the original benchmarking value of 5 ksi $\sqrt{\text{in}}$).

The BWRVIP-108-A fatigue crack growth (FCG) equation was used, along with a Weibull distribution for the coefficient of the FCG equation, and the FCG threshold was assumed to be zero. In addition, the PVRUF crack depth distribution was used for the initial crack size. All of these crack growth and crack depth distribution inputs remain identical to what was used in the initial benchmarking exercise in Section 8.2.3.2.2 of Reference [1]. In both software codes, the inputs for the time increment for updating the crack growth was set to one-tenth of a year.

Table 3
Updated Benchmarking Inputs for First PSI/ISI Combination (Scenario No. 1)

Input	Value
No. of cracks per inner radius section	1, constant
Crack depth distribution	PVRUF
Fracture toughness (ksi√in)	<i>Normal (100, 20)</i>
PSI	<i>Yes (at 0 years)</i>
ISI	<i>Yes</i> <i>4 cases:</i> 1) <i>No ISI</i> 2) <i>ISI at 20 yrs</i> 3) <i>ISI at 20, 40 yrs</i> 4) <i>ISI at 20, 40, 60 yrs</i>
POD Curve	<i>Figure 8-2 of Reference [1]</i>
Fatigue crack growth law and threshold	BWRVIP-108-A
Applied Stress	<i>30 ksi through thickness</i>
Transient stresses and uncertainties	None
Residual stresses (ksi)	None
Time increment for updating crack growth calculation	<i>One tenth of a year</i>
Cycles	<i>500 per Year</i>

Table 4
Updated Benchmarking Inputs for Second PSI/ISI Combination (Scenario No. 2)

Input	Value
No. of cracks per inner radius section	1, constant
Crack depth distribution	PVRUF
Fracture toughness (ksi√in)	<i>Normal (100, 20)</i>
PSI	<i>Yes (at 0 years)</i>
ISI	<i>Yes</i> <i>8 cases:</i> 1) <i>No ISI</i> 2) <i>ISI at 10 yrs</i> 3) <i>ISI at 10, 20 yrs</i> 4) <i>ISI at 10, 20, 30 yrs</i> 5) <i>ISI at 10, 20, 30, 40 yrs</i> 6) <i>ISI at 10, 20, 30, 40, 50 yrs</i> 7) <i>ISI at 10, 20, 30, 40, 50, 60 yrs</i> 8) <i>ISI at 10, 20, 30, 40, 50, 60, 70 yrs</i>
POD Curve	<i>Figure 8-2 of Reference [1]</i>
Fatigue crack growth law and threshold	BWRVIP-108
Applied Stress	<i>30 ksi through thickness</i>
Transient stresses and uncertainties	None
Residual stresses (ksi)	None
Time increment for updating crack growth calculation	<i>One tenth of a year</i>
Cycles	<i>1,000 per Year</i>

EVALUATION

The probabilities of leakage and rupture were determined for multiple cases for each of the two PSI/ISI scenarios shown in Tables 3 and 4 using both the **PROMISE** and **VIPER-NOZ** software codes. The constant stress of 30 ksi applied to the nozzle corner was cycled from the constant value to zero with 500 cycles per year for Scenario No. 1 and 1,000 cycles per year for Scenario No. 2. The added number of cycles for Scenario No. 2 is to increase the likelihood of failure since it involves many more inspections.

RESULTS

The probabilities of leakage and rupture as a function of the cases for each PSI/ISI scenario are presented in Figures 1 and 2 for Scenario No. 1 and Figures 3 and 4 for Scenario No. 2. As shown in these figures, there is very good agreement between the **PROMISE** and **VIPER-NOZ** software results for all cases for both scenarios. As expected, both software codes indicate that the probabilities of leakage and rupture decrease as more inspections are performed.

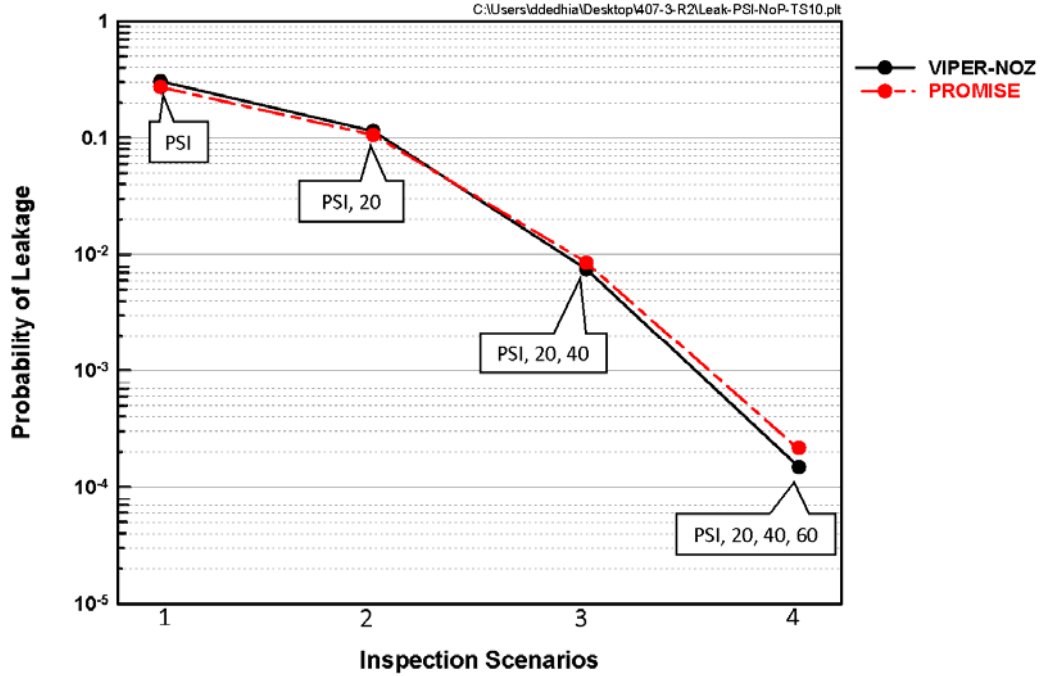


Figure 1
 Comparison of Cumulative Probabilities of Leakage Between the PROMISE and the VIPER-NOZ Software Codes for Scenario No. 1

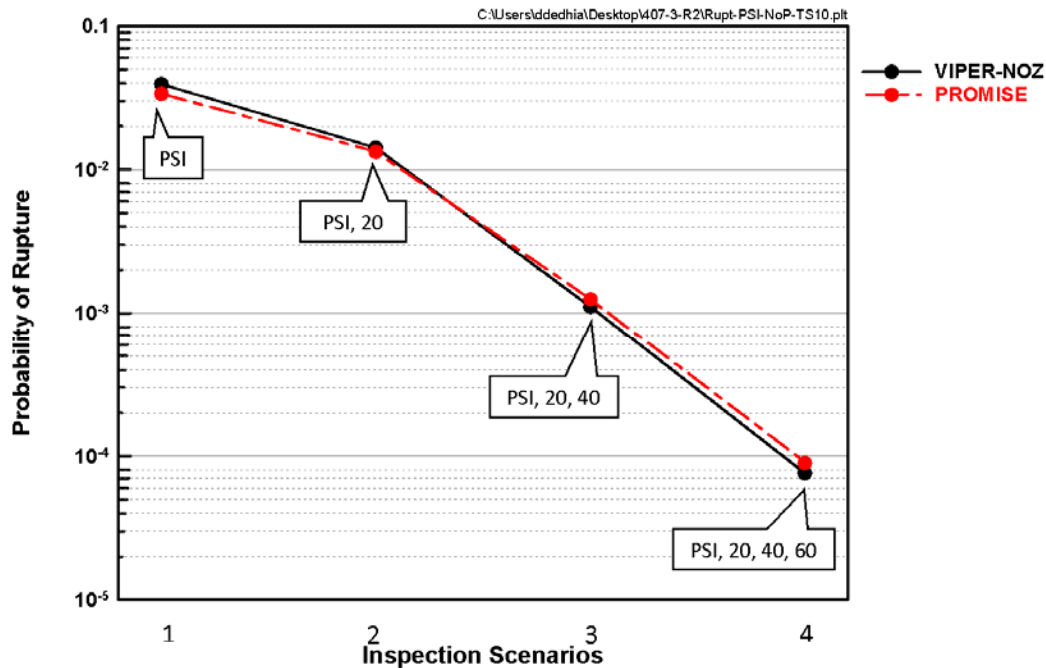


Figure 2
 Comparison of Cumulative Probabilities of Rupture Between the PROMISE and the VIPER-NOZ Software Codes for Scenario No. 1

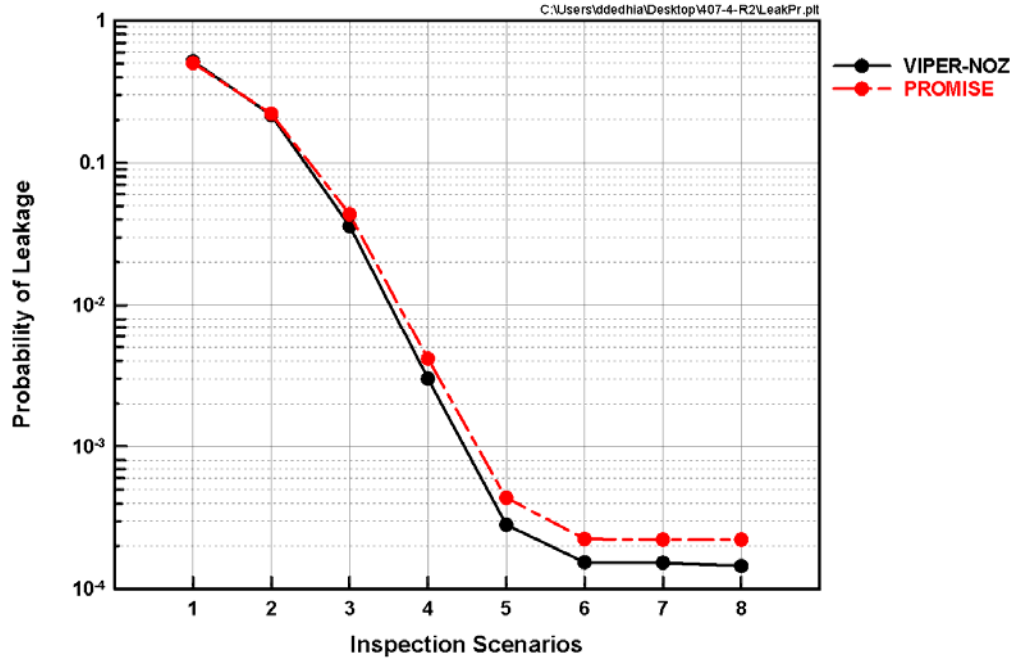


Figure 3
 Comparison of Cumulative Probabilities of Leakage Between the PROMISE and the VIPER-NOZ Software Codes for Scenario No. 2

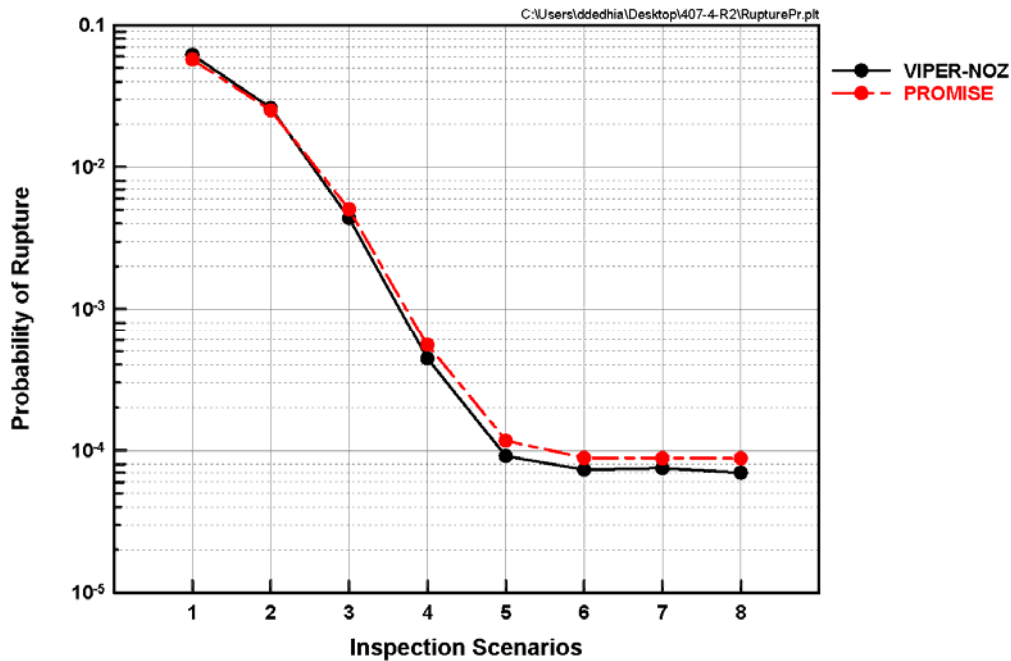


Figure 4
 Comparison of Cumulative Probabilities of Rupture Between the PROMISE and the VIPER-NOZ Software Codes for Scenario No. 2

SUMMARY AND CONCLUSION

PFM evaluations were performed to benchmark the **PROMISE** software against the **VIPER-NOZ** software including the effect of inspections (both PSI and ISI) using multiple cases of different PSI/ISI combinations for two scenarios to determine the effect of ISI on failure probabilities for both leakage and rupture. Scenario No. 1 considered four cases of PSI alone and PSI followed by 20-year ISI examinations up to 60 years. Scenario No. 2 considered eight cases of PSI alone and PSI followed by 10-year ISI examinations up to 70 years. The results from the two software codes are in very good agreement for all cases for both scenarios, indicating that both software codes produce consistent probabilities of leakage and rupture using identical inputs.

REFERENCES

1. *Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Nozzle Inside Radius Sections*. EPRI, Palo Alto, CA: 2019. 3002014590.
2. Structural Integrity Associates Report DEV1806.402, **PROMISE 1.0 Theory and User's Manual**, Revision 0.
3. Structural Integrity Associates, **VIPER-NOZ Version 1.1**.
4. *BWRVIP-108-A: BWR Vessels and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Shell Welds and Nozzle Blend Radii*, EPRI, Palo Alto, CA 2018. 3002013092.
5. *BWRVIP-241-A: BWR Vessel and Internals Project, Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii*. EPRI, Palo Alto, CA: 2018. 3002013093.

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