Southern Nuclear

Cheryl A. Gayheart
Regulatory Affairs Director

3535 Colonnade Parkway
Birmingham, AL 35243
205 992 5316

cagayhea@southernco.com

November 23, 2020

Docket Nos.: 50-424
50-425

NL-20-1312

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

Vogtle Electric Generating Plant, Units 1 & 2
Response to Request for Additional Information Related to
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 (ADAMS) Accession No. ML19347B105), and September 9, 2020 (ADAMS Accession No. ML20253A311), 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 CB, item number C2.21 and C2.22, exams from 10 years to 30 years through for the remainder of the 6th ISI Interval.

By email dated October 22, 2020, the US NRC notified SNC that additional information is needed for the staff to perform their review.

Enclosure 1 to this letter provides the SNC response to the NRC request for additional information (RAI). Attachment 1 provides supplemental information regarding Item 2.c.i of the NRC Audit of the PROMISE Version 1.0 Software held on July 27, 2020.

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/cbg

Enclosure 1: SNC Response to Request for Additional Information (RAI)

Attachment 1: SNC Response to Item 2.c.i During the Continuation of the NRC Audit of the PROMISE Version 1.0 Software on July 27, 2020

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

Vogtle Electric Generating Plant, Units 1 & 2
Response to Request for Additional Information Related to
Proposed Inservice Inspection Alternative VEGP-ISI-ALT-04-04 Version 2.0

Enclosure 1

SNC Response to Request for Additional Information (RAI)

RAI 1

Issue

Section 5.2 of the EPRI report states that it did not consider test conditions beyond a system leakage test in the analyses and that, since any pressure tests will be performed at operating pressure, no separate test conditions need to be included in the analyses because the test conditions are captured in the other transients included in the analyses. Even though the test conditions are not included in the analyses, the NRC staff determined that the appropriate temperature conditions for an upper shelf fracture toughness (KIC) value of 200 ksi√in assumed in the EPRI report must exist during the secondary side system leakage and secondary side hydrostatic tests. The NRC staff noted in Sections 3.9.N.1.1.1.15 and 3.9.N.1.1.5.2 of the Vogtle, Units 1 and 2, Updated Final Safety Analysis Report (UFSAR, ADAMS Accession No. ML19296C722) the minimum temperature of 120°F specified for the secondary side system leakage and secondary side hydrostatic tests. The NRC staff noted this minimum temperature is too low for an upper shelf KIC value of 200 ksi√in assumed in the EPRI report. The NRC staff further noted that the value of the parameter "T – RTNDT" used in calculating the ASME Code KIC value must be at least 105°F for the material to be on the upper shelf and a KIC value of 200 ksi√in to be appropriate. Section 8.2.2.7 of the EPRI report states that it assumed an RTNDT value of 60°F for the subjection SG components. Therefore, the temperature "T" in T – RTNDT must be at least 105°F + 60°F = 165°F in order for a KIC value of 200 ksi√in to be appropriate.

Request

Confirm that, when the secondary side system leakage and secondary side hydrostatic tests at Vogtle, Units 1 and 2, are performed at the maximum pressures specified for the tests, the temperature is least 165°F.

SNC Response: With respect to secondary side system leakage tests, leakage tests (ASME Code, Section XI, Examination Category C-H) are conducted during each inservice inspection period (approximately every other refueling outage). The secondary side system leakage tests are performed in parallel with the Class 1 leakage test which is performed at nominal operating temperature and pressure; as a result, the secondary side leakage test for the steam generators (SGs) is performed at nominal operating temperature and pressure. Per UFSAR Table 10.1-1, the Vogtle Units 1 and 2 SG normal operating inlet feedwater and outlet main steam temperatures are 449°F and 543°F, respectively. Therefore, the secondary side system leakage tests are performed at SG temperatures well above the temperature of 105°F applicable for the assumed K_{IC} value of 200 ksi$\sqrt{\text{inch}}$ in the EPRI study.

With respect to secondary side system hydrostatic tests, the hydrostatic events referred to in the RAI from the Vogtle UFSAR refer to tests that are performed during construction prior to plant startup. For any major SG repairs that may occur in the future, Section XI, Paragraph IWA-4540(a) requires an operating system leakage test or a secondary side system hydrostatic test. SNC performs operating system leakage tests rather than secondary side system hydrostatic tests on the Vogtle SGs following repair and replacement activities, so secondary side system hydrostatic tests are not applicable and were therefore not considered in the supporting EPRI study.

Nevertheless, using the bounding RT_{NDT} value from the EPRI report of 60°F, the K_{IC} value applicable for a 120°F lower bound transient temperature is:

$$K_{IC} = 33.2 + 20.734 \exp[0.02 (T - RT_{NDT})]$$

$$= 33.2 + 20.734 \exp[0.02\,(120 - 60)]$$

$$= 102.0 \text{ ksi}\sqrt{\text{inch}}$$

Section 8.2.4.3.1 of the EPRI report presents the results of sensitivity studies for fracture toughness. The results of those sensitivity studies are presented in Tables 8-13 and 8-14 of the EPRI report for probabilities of rupture and leakage, respectively. One of the sensitivity runs that was made used a K_{IC} value of 100 ksi$\sqrt{\text{inch}}$ (with a standard deviation of 5 ksi$\sqrt{\text{inch}}$). The results of that sensitivity study run indicate that the probabilities of rupture and leakage for the two limiting cases (FEW-P1N and FEW-P3A) are at least three orders of magnitude below the acceptance criteria of $1\text{x}10^{-6}$ per year, as summarized in the table below.

Summary of Results of the Fracture Toughness Sensitivity Study for a K_{IC} Value of 100 ksi$\sqrt{\text{inch}}$ for the Limiting Cases from the EPRI Report

Component	Case Identification	Probability of Rupture (from Table 8-13 of the EPRI Report)	Probability of Leakage (from Table 8-14 of the EPRI Report)
Westinghouse Feedwater Nozzle (FEW)	FEW-P1N	4.61E-09	1.25E-12
	FEW-P3A	3.75E-09	1.25E-09

Therefore, the secondary side system leakage tests, as defined in the Vogtle Units 1 and 2 UFSAR, are addressed by the sensitivity studies included in the EPRI report, which produce acceptable results.

RAI 2

Issue

Note 6 in Table 5-5 of the EPRI report states that the Loss of Power transient affects only the feedwater nozzle, and therefore, was applied only to the feedwater nozzle analysis. Table A2 in Enclosure 1 to the September 9, 2020 supplement compares the three cycles of the Loss of Power transient of Vogtle, Units 1 and 2, to the cycles analyzed in the EPRI report. The NRC staff conducted an audit of the PFM software that was used for the PFM analyses in the EPRI report. The NRC staff expects to issue the audit report shortly (ADAMS Accession No. ML20258A002). During the audit, the NRC staff noted that two of the output files for the limiting feedwater nozzle case contain all the transients listed in Table 5-5 of the EPRI report except for the Loss of Power transient (Items 2.e.i and 2.e.ii of the audit report). Table 5-5 indicates that during the Loss of Power transient the pressure is 1,120 psig, and the through-wall stress distribution plots in Figures 7-32 through 7-35 of the EPRI report for the feedwater nozzle clearly show thermal transient stress distributions for the Loss of Power transient at 619 seconds. Pressure and thermal stresses due to the Loss of Power transient could lead to an applied stress intensity factor that exceeds KIC. Also, Table 5-4 of the EPRI report suggests that Loss of Power could have large pressure and temperature fluctuations; large pressure and temperature fluctuations could have a large impact on fatigue crack growth. The description of the Loss of Power transient on page 5-10 of the EPRI report does not provide sufficient details on the transient.

The NRC staff also noted in Items 2.e.i and 2.e.ii of the audit report what appears to be a low temperature overpressure (LTOP) event in two of the input files for the limiting feedwater nozzle case. The NRC staff noted that the EPRI report did not contain information on LTOP; additionally, the NRC staff noted that the design bases in Section 5.2.2.1 of the Vogtle, Units 1 and 2 UFSAR (ADAMS Accession No. ML19296C741) states that the overpressure protection for the steam system is provided by the SG safety valves, which suggests there might be LTOP events that could affect the subject SG components of Vogtle, Units 1 and 2. Therefore, the NRC staff requests the following regarding the Loss of Power transient and LTOP.

Request

a) Explain if the Loss of Power transient was included in the feedwater nozzle stress analyses used as input to the PFM analyses. If it was included, explain why it is not in the output files for the feedwater nozzle case. If not included, explain why in terms of the pressure specified for the Loss of Power transient in Table 5-5 of the EPRI report, the thermal transient stress distributions in Figures 7-32 through 7-35 of the EPRI report, the total of which can lead to a potential exceedance of KIC, and possible pressure/temperature fluctuations during this transient that warrant exclusion from fatigue crack growth calculations.

b) Clarify if LTOP was included in the PFM analyses in the EPRI report and explain whether LTOP is an event that could affect the subject SG components of Vogtle, Units 1 and 2.

SNC Response:

a) The Loss of Power (LOP) transient is described on page 5-10 of the EPRI report. This transient initiates at 100% power and causes the initiation of cold (minimum of 32°F) auxiliary feedwater flow into the hot (560°F) SG via the feedwater nozzle at a constant pressure of 1,120 psig. This event was assumed to occur once per year for 60 years of operation, or a total of 60 cycles. The LOP transient was explicitly included in the deterministic fracture mechanics (DFM) analysis for the feedwater nozzle, the results of which are presented in Table 8-31 of the EPRI report. It was also implicitly included in the probabilistic fracture mechanics (PFM) analyses in the EPRI report as discussed at the top of page 5-10 of the EPRI report where it is stated that the number of LOL cycles was increased from 100 to 360 cycles for 60 years of operation to account for other transients including the LOP transient.

   Nevertheless, for the purposes of demonstrating the insignificance of the LOP event, an additional PFM run was made for the Base Case (the results of which are shown in Table 8-8 of the EPRI report). The Base Case was rerun for the two critical Case IDs for the feedwater nozzle (FEW-P1N and FEW-P3A) with the inclusion of the LOP transient. The comparison of the results with and without the LOP event is shown in the table below:

Case ID	Without LOP Transient (1)		With LOP Transient	
	P(Rupture) at 80 years	P(Leakage) at 80 Years	P(Rupture) at 80 years	P(Leakage) at 80 years
FEW-P1N	1.25E-12	1.88E-11	1.25E-12	1.88E-11
FEW-P3A	1.25E-09	2.5E-09	1.25E-09	3.75E-09

Note: 1. These results are also reported in Table 8-8 of the EPRI report.

   As shown in the table, there is no change in the probabilities of rupture or leakage for Case ID FEW-P1N. Even though there is a change in the probability of leakage for Case ID FEW-P3A, the probability of leakage values with and without the LOP transient are the same order of magnitude, and they are three orders of magnitude below the acceptance criteria of

$1x10^{-6}$ failures/year. Note also that the number of LOL events was not decreased for this run, so the number of cycles used for the LOL event still conservatively includes the LOP cycles. Therefore, the LOP event has an insignificant impact on the PFM results.

For reference, the relevant portions of the PROMISE output files for these two runs are included below for Case ID FEW-P1N. The ***italicized and boldfaced highlighting*** for the second output file indicates the inputs for the LOP event.

Case ID FEW-P1N (Without LOP Transient)

```
FEA Case # 7
    Transient Case ID:UNLOAD5
    FEA File: STR_UNLD_MAP_P1.CSV
        Column# for X = 1
        Column# for Axial Stress = 3
        Column# for Hoop Stress = 4
        Multiplier to Convert to ksi = 0.001
        Uncertainty Type: Aleatory
        Distribution Type: Normal
         Mean =       1.0000
         Standard Deviation =       0.0000
         Deterministic Value =       1.0000
         Minimum Cut-off =       0.0000
         Maximum Cut-off =       2.0000

    No. of Load Cases created = 14

    Probability of Rupture (Kop > KIC)
           Time              Quantiles
                 Mean       0.5000

     0.000E+00 0.000E+00 0.000E+00
     1.000E+00 0.000E+00 0.000E+00
     7.900E+01 0.000E+00 0.000E+00
     8.000E+01 0.000E+00 0.000E+00

    Probability of Through-Wall (a > t)
           Time              Quantiles
                 Mean       0.5000

     0.000E+00 0.000E+00 0.000E+00
     1.000E+00 0.000E+00 0.000E+00
     7.900E+01 1.400E-06 1.400E-06
     8.000E+01 1.400E-06 1.400E-06
```

FEW-P1N (With LOP Transient)

```
FEA Case # 7
Transient Case ID:UNLOAD5
FEA File: STR_UNLD_MAP_P1.CSV
    Column# for X = 1
    Column# for Axial Stress = 3
    Column# for Hoop Stress = 4
    Multiplier to Convert to ksi = 0.001
    Uncertainty Type: Aleatory
    Distribution Type: Normal
     Mean =       1.0000
     Standard Deviation =       0.0000
```

```
     Deterministic Value =       1.0000
     Minimum Cut-off =       0.0000
     Maximum Cut-off =       2.0000
```

```
FEA Case # 8
Transient Case ID:LOP
FEA File: STR_LOP_MAP_P1.CSV
    Column# for X = 1
    Column# for Axial Stress = 3
    Column# for Hoop Stress = 4
    Multiplier to Convert to ksi = 0.001
    Uncertainty Type: Aleatory
    Distribution Type: Normal
     Mean =       1.0000
     Standard Deviation =       0.0000
     Deterministic Value =       1.0000
     Minimum Cut-off =       0.0000
     Maximum Cut-off =       2.0000

No. of Load Cases created = 16

Probability of Rupture (Kop > KIC)
       Time              Quantiles
             Mean       0.5000

 0.000E+00 0.000E+00 0.000E+00
 1.000E+00 0.000E+00 0.000E+00
 7.900E+01 0.000E+00 0.000E+00
 8.000E+01 0.000E+00 0.000E+00

Probability of Through-Wall (a > t)
       Time              Quantiles
             Mean       0.5000

 0.000E+00 0.000E+00 0.000E+00
 1.000E+00 0.000E+00 0.000E+00
 7.900E+01 1.400E-06 1.400E-06
 8.000E+01 1.400E-06 1.400E-06
```

b) The PROMISE PFM software has the ability to evaluate LTOP events for those components where LTOP is applicable. LTOP events are applicable to the primary side (reactor coolant system including the reactor pressure vessel). NRC guidance for overpressure protection is provided in Section 5.2.2 of NUREG-0800, "*Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition*." Paragraph 4 under SRP Acceptance Criteria in Section 5.2.2 of NUREG-0800 identifies that PWRs operating at low temperatures should be designed in accordance with Branch Technical Position (BTP) 5-2, "*Overpressure Protection of Pressurized-Water Reactors While Operating at Low Temperatures*." Paragraph I.1.C of Section 5.2.2 of NUREG-0800 identifies that the applicable areas for LTOP protection are the pressurizer, safety relief valves (SRVs), and the piping from these valves to a quench tank or to containment atmosphere on the primary side, as well as SG SRVs on the secondary side. Therefore, the SG itself is protected from LTOP events from the primary side by the SG SRVs. As a result, the LTOP event was not considered in the EPRI report and the supporting evaluations for the SG welds included in SNC's Request for Alternative.

RAI 3

Issue

Section 8.2.2.3 of the EPRI report states that the probability of detection (POD) curve used in the analyses is from the PFM analyses performed in proprietary report BWRVIP-108, "BWR [Boiling Water Reactor] Vessel and Internals Project; Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii, October 2002." The NRC staff noted that the NVWs and NIR sections analyzed in BWRVIP-108 were associated with the reactor pressure vessel and that the POD curve was, therefore, developed based on the ultrasonic testing (UT) requirements in ASME Code, Section XI, Appendix VIII. The NVWs and NIR sections in the EPRI report are associated with the SG vessel for which the UT requirements of ASME Code, Section V apply. The NRC staff noted that in practice the POD curve based on the UT requirements of ASME Code, Section V, could be lower than the POD curve based on the UT requirements of ASME Code, Section XI, Appendix VIII. Since ASME Code, Section V is used for the UT examination of SG vessel components, applying the ASME Code, Section XI, Appendix VIII-based POD curve to the PFM analyses of SG vessel components could be nonconservative.

Request

Explain how the ASME Code, Section XI, Appendix VIII-based POD curve is sufficient for the PFM analyses of the subject SG vessel components in the EPRI report.

SNC Response

The EPRI report used a Section XI, Appendix VIII-based POD curve in the PFM evaluation because most ISI examinations of major plant Class 1 and Class 2 components are performed using Appendix VIII procedures. However, for Class 2 components, the use of Appendix VIII procedures is plant-specific. Many plants adopt and the use their Appendix VIII procedures for major Class 2 components (such as SGs) for consistency across all their examinations. In the case of Vogtle Units 1 and 2, SNC does not use Appendix VIII procedures for all of the examination categories included in the SNC submittal, so use of the Appendix VIII POD curve may not be appropriate for all of the items. However, as shown below, the EPRI report demonstrates that the 30-year interval is supported for these welds, regardless of the POD curve used.

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

Feedwater Nozzle Inner Radius (C2.22)

The Vogtle SG feedwater nozzle inner radius sections materials, sizes and shapes included in SNC's Request for Alternative are within the scope of qualified Appendix VIII procedures, and those procedures are used to examine the applicable SG feedwater nozzle inner radii sections.

Therefore, the Appendix VIII POD curves are applicable to the Vogtle feedwater nozzle inner radii examinations, and all results in the EPRI report for these components are applicable to the Vogtle SG feedwater nozzle inner radius sections.

<u>Main Steam Nozzle Inner Radius (C2.22)</u>

The Vogtle SG main steam nozzle does not have an inner radius section, so the inner radius examination is not applicable for this component.

<u>Feedwater and Main Steam Nozzle-to-Shell Welds (C2.21)</u>

The Vogtle Units 1 and 2 feedwater and main steam nozzle-to-shell weld examinations are performed in accordance with Section V, Article 4 requirements, so application of the Appendix VIII POD curves may not be appropriate nor conservative for these welds. Section V examinations do not have established POD curves. However, the EPRI report demonstrates that the 30-year interval is supported for these welds because the results of the Base Case, where only preservice inspection (PSI) was assumed without the benefit of any follow-on ISI examinations, are acceptable. Any NDE examinations performed after PSI examinations, including those performed in accordance with Section V, only serve to strengthen those results.

As explained in the response to Item 2.c.i during the continuation of the NRC audit of the PROMISE Version 1.0 software on July 27, 2020 (included as Attachment 1), PSI examinations were performed on the feedwater and main steam nozzle-to-shell welds using radiographic testing (RT). The Appendix VIII POD curve for ultrasonic testing (UT) examinations, used in the PROMISE analyses for both PSI (only) and ISI examination cases, is considered more conservative than the RT POD curve. As discussed on page 8-18 of the EPRI report, by performing PSI examination only, the acceptance criteria is met for 80 years. Therefore, any additional NDE examinations performed after the PSI examination, regardless of the POD curve used, will reduce the probability of failure numbers reported for the feedwater and main steam nozzle-to-shell welds analyzed for the PSI (only) case.

Table 8-9 of the EPRI report provides the probabilities of rupture and leakage for PSI examinations only (with no follow-on ISI examinations). The following observations are repeated from page 8-18 of the EPRI report based on the review of the results in Table 8-9:

- *The probability of rupture is below the acceptance criteria at all locations for all NSSS design types by about three orders of magnitude after 80 years of operation.*
- *The probability of leakage is below the acceptance criteria for all B&W and Westinghouse MS nozzles after 80 years.*
- *For the Westinghouse FW nozzles, five out of six locations are below the acceptance criteria by about three orders of magnitude for 80 years. The remaining location (FEW-P3A) is below the acceptance criteria for almost 60 years.*
- *The acceptance criteria exceedance for leakage at the one FW nozzle location (FEW-P3A) increases the likelihood of leakage but does not compromise plant safety as leakage of the pressure boundary is detectable by plant operators and plant procedures allow for safe plant shutdown under leaking conditions.*

Based on the above observations, the PFM technical assessments conclude that performing only the PSI examination without any other follow-on ISI examinations is acceptable for up to 80 years of operation while still maintaining plant safety.

The PFM analyses are also supported by the results of the deterministic fracture mechanics (DFM) evaluations shown in Table 8-31 of the EPRI report. The DFM results demonstrate that it would take a very long operating period (more than 100 years) for a postulated initial flaw with a depth equal to the ASME Code, Section XI acceptance standards to propagate through-wall and cause leakage without performing any subsequent inspections. After 80 years and without any subsequent inspections, the maximum K obtained from the analysis is below or close to the ASME Code, Section XI allowable fracture toughness, including the Section XI Appendix G structural factor for primary stress, which indicates that ASME Code, Section XI structural margins have been satisfied.

Therefore, the feedwater and main steam nozzle-to-shell welds are acceptable for the 30-year inspection interval requested in SNC's submittal based on the PSI (only) examination case PFM analyses. When additional NDE examinations are performed and applied to the analyses, no matter the method, procedure, or technique, they further improve the analytical results.

RAI 4

Issue

Section 8.2.2.4.1 of the EPRI report states that transient stresses are normally distributed. The NRC staff noted that this is consistent with Table 8-3 of the EPRI report, which indicates a normal distribution for transient stresses. However, in the list of inputs for the PFM base cases in Table 8-7, EPRI stated that the uncertainties on transients are "None," which implies that there is no statistical distribution on transient stresses. During the audit of the software used in the PFM analyses in the EPRI report, the NRC staff reviewed one of the input files, which seemed to indicate that the transient stresses have a statistical distribution.

Request

Clarify whether transient stresses (pressure and thermal) were random or not. If random, provide and justify the mean and standard deviation values.

SNC Response

As identified in Table 8-3 of the EPRI report and discussed during the NRC's audit of the PROMISE software, transient stresses are input to the PROMISE software as random variables. However, the transient stresses were treated as constant for the PFM assessments in the EPRI report. As an example, the relevant portion of the PROMISE software input file "FEW-3PA.vnz3" for the Loss of Load transient is reproduced below:

```
LOL
STR_LOL_MAP_P3.CSV
1 4 3 8 0.001
Aleatory
Normal
1,0
1,0,2
```

The following table describes each of the line items in the above PROMISE input file for the Loss of Load transient:

Line No.	Line Content	Description [1]
1	LOL	Record 12c: Load Case ID
2	STR_LOL_MAP_P3.CSV	Record 12d: Filename for the Load Case This file contains through-wall stress profiles at various time steps. The main fields in the file are time, x-coordinate, stresses (S_x, S_y, S_z) and temperature.
3	1 4 3 8 0.001	Record 12e: Structure of the FEA file First Number – Column number of the x-coordinate (always 1). Second Number – Column number for the axial stress. Third Number – Column number for the hoop stress. Fourth Number – Column number for the temperature. Fifth Number – Stress multiplier (psi to ksi).
Record 12f: Distribution for the random multiplier		
4	Aleatory	Page A-11: UncType = Uncertainty Type
5	Normal	Page A-11: aepDT = Distribution Type The distribution for the stress multiplier.
6	1, 0	Page A-11: aep1, aep2 = the mean and standard deviation for the multiplier distribution.
7	1, 0, 2	Page A-11: detValue, minValue, maxValue detValue = Multiplier for the deterministic analysis. minValue = Lower cutoff value for the sampled value of the multiplier. maxValue = Upper cutoff value for the sampled value of the multiplier.

Note: 1. The Record and Page numbers included in this column refer to the PROMISE Version 1.0 User's Manual.

As indicated by the descriptions in the above table, the inputs for Record 12f identify that the Loss of Load transient was input as a normally distributed random variable, with a mean value of 1.0 and a standard deviation of 0. This is equivalent to a constant input.

Attachment 1
SNC Response to Item 2.c.i During the Continuation of the NRC Audit of the PROMISE Version 1.0 Software on July 27, 2020

Audit Item 2.c.i (1 of 9)

2.c.i: Technical aspects of PROMISE Version 1.0 - Probability of Detection (POD) and Inservice Inspection (ISI) - the EPRI Technical Basis Report does not discuss the interaction between the POD curve and ISI. Show how POD and ISI are implemented into the PROMISE Version 1.0 code by showing the software requirements for this feature and running some example problems. Refer to Sections 8.2.2.3 and 8.2.2.8 of the EPRI Technical Basis Report

Open Items:

A. Provide the rationale for the use of the same POD curve for both ISI and PSI.
B. Provide the results of a PROMISE run using a limiting case with an assumption of 50% ISI coverage.

Response to Open Audit Item 2.c.i.A (2 of 9)

Regarding the Assumption of 100% Coverage for PSI:

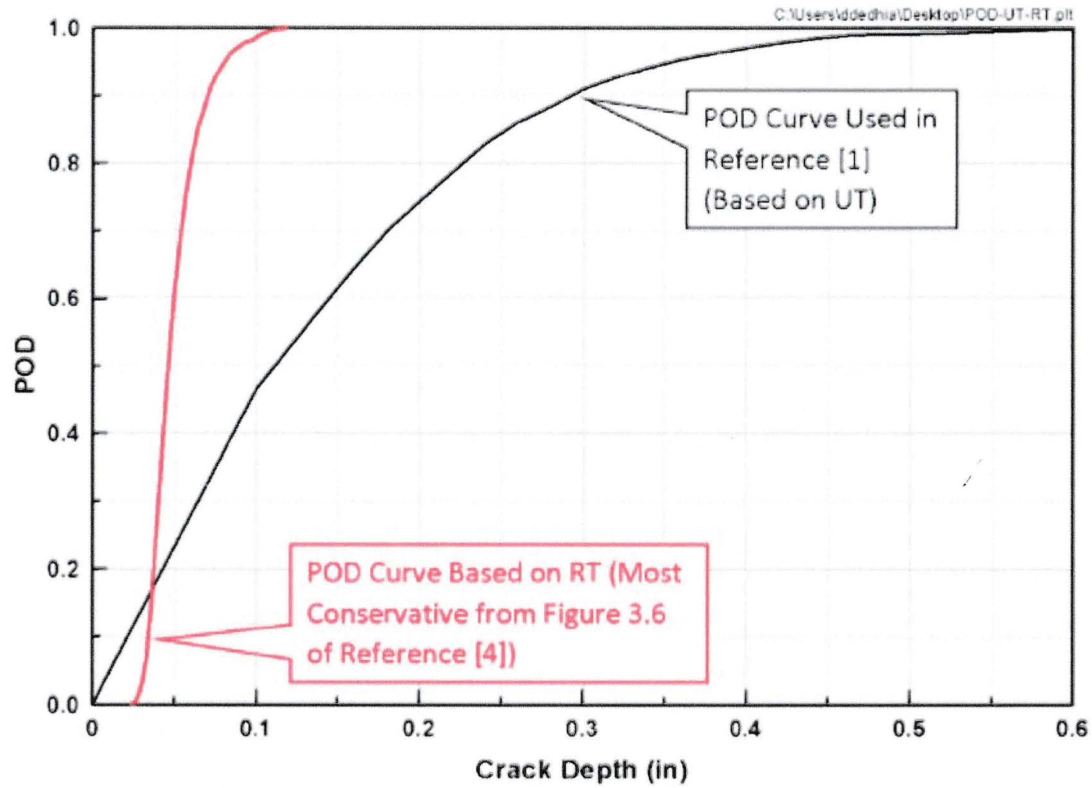
- In Section 8.2.4.1.1 of the EPRI report, PSI refers to the collective initial ASME Code, Section III and Section XI examinations
- ASME Code Section III requires 100% coverage for RT exams
- ASME Code, Section III PSI examinations involving radiographic testing (RT) were performed and found acceptable for the affected welds of the VEGP Units 1 and 2 MS and FW nozzles
  - The acceptability of the ASME Code, Section III RT examinations indicates that 100% coverage was achieved during these examinations
- In addition, Section XI PSI (UT) exams were performed, some with limited coverage
- Collectively, both examinations covered 100% of the weld

Response to Open Audit Item 2.c.i.A (3 of 9)

Regarding the Assumption of Applying ISI POD to PSI:

- From PNNL-19086 (ML101031254) and NUREG/CR-7204
  - Detection and sizing of flaws utilizing RT examination is as effective as UT examination
  - Figure 3.6 of PNNL-19086 provides theoretical probability of detection (POD) curves for RT examinations
  - The most conservative of the POD curves for RT from PNNL-19086 is compared to the POD curve for UT used in the EPRI Technical Basis Report (Figure 8-2 of EPRI Report = BWRVIP-108-A POD curve)
    - The POD comparison is shown on the next slide
- Except for extremely shallow flaws (less than 0.04 inches), the POD curve used in in the technical basis can be conservatively applied to RT examinations
  - The minimum flaw depth in all the PROMISE simulations is 0.075 inches, which is greater than the flaw depths at which RT governs

Response to Open Audit Item 2.c.i.A (4 of 9)



Note: Reference [1] = EPRI report, and Reference [4] = PNNL-19086.

Response to Open Audit Item 2.c.i.B (5 of 9)

Regarding the Inspection Coverage Issue:

- From SNOC Alternative VEGP-ISI-ALT-04-04
  - For VEGP Units 1 and 2, in addition to PSI examinations, four ISI examinations have been performed for the MS nozzle-to-shell welds, and three ISI examinations have been performed for the FW nozzle-to-shell welds
  - All ISI examinations for Item C2.21 for the MS and FW nozzles had limited coverage ranging from 50% to 80%.
  - Item No. C2.22 MS nozzle inner radii are exempt from ISI
  - Item No. C2.22 FW nozzle inner radii had 100% ISI coverage for all past ISI examinations
- Therefore an ISI coverage evaluation was performed for C2.21 to address 50% ISI coverage

Response to Open Audit Item 2.c.i.B (6 of 9)

Evaluation

- Per Table 8-9 if the EPRI Report, Case ID FEW-P3A, is the limiting location for the base case (PSI only)
- Evaluation was performed for Case ID FEW-P3A (Item No. C2.21) using the PROMISE Version 1.0 software
- PSI with 100% coverage and subsequent ISI with 50% coverage was assumed
- Two ISI scenarios were considered:
  1. The current ASME Code, Section XI examination requirement, which involves 10-year interval examinations after the PSI examination
     - ISI at 10, 20, 30, 40, 50, 60, and 70 years
  2. The alternative 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
     - ISI at 10, 20, 30, and 60 years

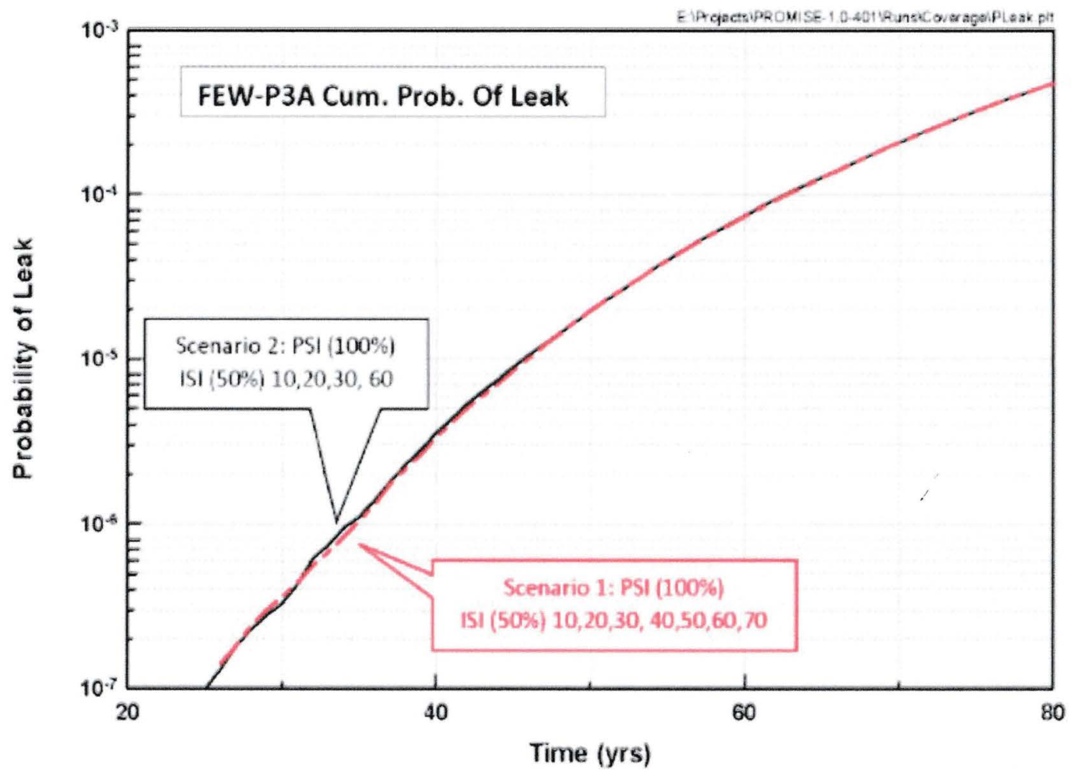
Response to Open Audit Item 2.c.i.B (7 of 9)

Evaluation Results

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

- The probabilities of rupture for 100% ISI coverage and 50% ISI coverage are identical and remain unchanged from the results for PSI examination only
- The probabilities of leakage for the ASME Code, Section XI ISI (PSI+10+20+30+40+50+60+70) and the alternative ISI (PSI+10+20+30+60) scenarios assuming 50% ISI coverage are essentially equal at 5.9×10^{-6}, which is slightly above the acceptance criterion
  - These values are lower than the PSI only (100% coverage) value of 1.19×10^{-5} for 80 years
  - As expected, ISI inspections with limited coverage decrease the probability of leakage compared to PSI alone
- The probability of leakage vs. time for the two investigated ISI scenarios is nearly identical over time (see next slide)

Response to Open Audit Item 2.c.i.B (8 of 9)



Response to Open Audit Item 2.c.i (9 of 9)

Conclusions

- 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
  - 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})
  - The slight exceedance of the probability of leakage does not compromise plant safety because pressure boundary leakage is detectable by plant operators and plant procedures allow for safe plant shutdown once any leakage is detected, as described in Section 8.2.4.1.1 of the EPRI report