Dominion Energy Nuclear Connecticut, Inc.
5000 Dominion Boulevard, Glen Allen, VA 23060
DominionEnergy.com

Dominion Energy

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U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

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Docket No.	50-336
License No.	DPR-65

DOMINION ENERGY NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION FOR ALTERNATIVE REQUEST RR-05-06 - INSPECTION INTERVAL EXTENSION FOR STEAM GENERATOR PRESSURE-RETAINING WELDS AND FULL-PENETRATION WELDED NOZZLES

By letter dated July 15, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20198M682), Dominion Energy Nuclear Connecticut, Inc. (DENC) submitted to the U.S. Nuclear Regulatory Commission (NRC) a proposed alternative to the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. The proposed alternative was associated with steam generator (SG) nozzle-to-shell welds, SG head welds, SG tubesheet-to-head welds, SG shell welds, and SG nozzle inside radius sections of Millstone Power Station, Unit No. 2 (MPS2). Specifically, pursuant to Title 10 of the Code of Federal Regulations (10 CFR), Part 50, Paragraph 50.55a(z)(1), DENC proposed to increase the ISI interval for the subject components to 30 years, from the current ASME Code Section, Section XI requirement of 10 years.

In an email dated January 22, 2021, the NRC issued a draft request for additional information (RAI) related to the proposed alternative request. On February 2, 2021, the NRC staff conducted a conference call with DENC staff and Electric Power Research Institute staff to clarify the request. In an email dated February 3, 2021, the NRC transmitted the final version of the RAI (ADAMS Accession No. ML21034A576). DENC agreed to respond to the RAI within 45 days of issuance, or no later than March 22, 2021.

Attachment 1 provides DENC's response to the RAI. The Stress Analysis, and Probabilistic and Deterministic Fracture Mechanics for the MPS2 SG Design are provided in Attachments 2 and 3, respectively, which support the response to Request RAI-2(b).

If you have any questions or require additional information, please contact Shayan Sinha at (804) 273-4687.

Sincerely,

Gerald T. Bischof
Senior Vice President – Nuclear Operations & Fleet Performance

Attachments:

1. Response to Request for Additional Information for Alternative Request RR-05-06 – Inspection Interval Extension for Steam Generator Pressure-Retaining Welds and Full Penetration Welded Nozzles

2. Supporting Millstone Power Station Unit 2 Steam Generator Design Stress Analysis for RAI-2(b)

3. Supporting Millstone Power Station Unit 2 Steam Generator Design Deterministic and Probabilistic Fracture Mechanics Evaluations for RAI-2(b)

Commitments made in this letter: None

cc: U.S. Nuclear Regulatory Commission
Region I
2100 Renaissance Blvd, Suite 100
King of Prussia, PA 19406-2713

R. V. Guzman
Senior Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North, Mail Stop 08-C 2
11555 Rockville Pike
Rockville, MD 20852-2738

NRC Senior Resident Inspector
Millstone Power Station

ATTACHMENT 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION FOR

ALTERNATIVE REQUEST RR-05-06

INSPECTION INTERVAL EXTENSION FOR STEAM GENERATOR PRESSURE–RETAINING WELDS AND FULL PENETRATION WELDED NOZZLES

MILLSTONE POWER STATION UNIT 2
DOMINION ENERGY NUCLEAR CONNECTICUT, INC.

By letter dated July 15, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20198M682), Dominion Energy Nuclear Connecticut, Inc. (DENC) submitted to the U.S. Nuclear Regulatory Commission (NRC) a proposed alternative to the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code for steam generator (SG) nozzle-to-shell welds (NSWs), SG head welds (HWs), SG tubesheet-to-head welds (THWs), SG shell welds (SWs), and SG nozzle inside radius (NIR) sections of Millstone Power Station, Unit No. 2 (MPS2). Specifically, pursuant to Title 10 of the Code of Federal Regulations (10 CFR), Part 50, Paragraph 50.55a(z)(1), DENC proposed to increase the ISI interval for the subject components to 30 years, from the current ASME Code Section, Section XI requirement of 10 years.

In an email dated January 22, 2021, the NRC issued a draft request for additional information (RAI) related to the proposed alternative request. On February 2, 2021, the NRC staff conducted a conference call with DENC staff and Electric Power Research Institute (EPRI) staff to clarify the request. In an email dated February 3, 2021, the NRC transmitted the final version of the RAI (ADAMS Accession No. ML21034A576). DENC agreed to respond to the RAI within 45 days of issuance, or no later than March 22, 2021.

This attachment provides DENC's response to the RAI.

Background

In 10 CFR 50.55a(z)(1), DENC is required to demonstrate that the proposed alternative provides an acceptable level of quality and safety. DENC referred to the analyses in non-proprietary EPRI Report No. 3002015906 "Technical Bases for Inspection Requirements for PWR [Pressurized Water Reactor] Steam Generator Class 1 Nozzle-to-Vessel Welds and Class 1 and Class 2 Vessel Head, Shell, Tubesheet-to-Head and Tubesheet-to-Shell Welds," 2019 (ADAMS Accession No. ML20225A141) and non-proprietary EPRI Report No. 3002014590, "Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Nozzle Inside Radius Sections", April 2019 (ADAMS Accession No. ML19347B107) to support the proposed alternative in the submittal. DENC also included an applicability evaluation of the EPRI reports to MPS2. The NRC staff has determined that additional information as requested below is needed to complete its review of DENC's proposed alternative.

RAI-1

Regulatory Basis

The NRC has established requirements in 10 CFR Part 50 to protect the structural integrity of structures and components in nuclear power plants. Among these requirements are the ISI requirements of Section XI of the ASME Code incorporated by reference in 10 CFR Part 50.55a to ensure that adequate structural integrity of the SG

vessel is maintained through the service life of the vessel. Therefore, the regulatory basis for the following RAI has to do with demonstrating that the proposed alternative ISI requirements would ensure adequate structural integrity of the SG vessel welds of MPS2 for which EPRI report 3002015906 is referenced, and thereby providing an acceptable level of quality and safety per 10 CFR 50.55a(z)(1) for these components

Issue
The NRC staff noted that the probabilistic fracture mechanics (PFM) software used in EPRI report 3002015906 is PROMISE Version 2.0, while in EPRI report 3002014590, the PFM software used is PROMISE Version 1.0. The NRC staff audited PROMISE Version 1.0 and issued the audit report by letter dated December 10, 2020 (ADAMS Accession No. ML20258A002). One of the objectives of the NRC staff's audit was to ensure that PROMISE Version 1.0 received adequate verification and validation (V&V). The NRC staff has not audited PROMISE Version 2.0, and therefore, has determined that an understanding of the difference between PROMISE Version 2.0 and PROMISE Version 1.0 and the V&V performed for its difference is needed to support its review of the licensee's submittal.

Request
To ensure that PROMISE Version 2.0 received adequate V&V for use in the referenced analyses for the SG vessel welds of MPS2, describe the difference between PROMISE Version 2.0 and PROMISE Version 1.0 and the V&V performed for its difference to ensure that (1) cases intended for the difference between PROMISE Version 2.0 and PROMISE Version 1.0 reflected the change; and (2) that cases not intended for the difference were not affected.

DENC Response to RAI-1

The usage of two different versions of the PROMISE software is due to the timing of the publication of the two EPRI reports and improvements made to the software between the publication dates. EPRI Report 3002014590 (for SG Class 2 pressure-retaining welds) was published in April 2019 and EPRI Report 3002015906 (for other SG Class 1 and 2 welds) was published in October 2019. PROMISE Version 2.0 was developed in May 2019. The NRC performed their audit of Version 1.0 of the PROMISE software because that version of the software was used in another plant-specific submittal that was based only on the earlier EPRI Report 3002014590.

The main difference between the two versions of the PFM software is that PROMISE Version 1.0 applies a single, user-specified examination coverage to all inspections assumed over the component evaluation time period, whereas PROMISE Version 2.0 applies unique, user-specified examination coverages to each inspection assumed over the component evaluation period. In both PROMISE Versions 1.0 and 2.0, the software assumes 100% coverage for the pre-service inspection (PSI) examination.

Structural Integrity Associates performed a V&V and documented the results in a V&V calculation package. With respect to Item (1) of the NRC's request, test cases were run

that were specific to testing the portions of the software that were modified. Table RAI-1-1, which has been reproduced from the V&V package, contains the results for four scenarios where the coverage for each inspection during the component lifetime were not all identical. Besides inspection coverage, all other PFM inputs were identical for all four V&V scenarios. For V&V Scenario A, which was considered to be the reference case, the coverage was assumed to be 100% for the pre-service inspection (PSI) at 0 years, followed by inspection coverage values of 50% for all remaining examinations at 20, 40, and 60 years of operation. The other three V&V scenarios assumed different inspection coverage values for 20, 40, and 60 years.

As shown by the results in Table RAI-1-1, there is a notable difference in the cumulative probability of leakage at 80 years for Scenario B compared to Scenario A. These results were verified to be correct in the V&V. The results for Scenarios C and D are the same as Scenario A because early inspections during the life of the component have the most impact in reducing the probability of leakage, while PFM sensitivity studies have shown that later inspections do not have a significant impact on the probability of leakage. This is demonstrated by the sensitivity study results presented in Table 8-30 of EPRI Report 3002015906 in which an inspection scenario of (0, 10, 20 and 30 years) resulted in a probability of failure of 1.25E-09, while an inspection scenario of (0, 10, 20, 30, 50 and 70 years) also resulted in a probability of failure of 1.25E-09 using either the BWRVIP-108 or Appendix L probability of detection (POD) curves. Similar results were obtained for EPRI Report 3002014590 as documented in the response to RAI 3 for the Vogtle Electric Generating Plant, Units 1 & 2 (ADAMS Accession No. ML20329A302 dated November 23, 2020, Response to RAI 3 and Attachment 1, “Response to Open Audit Item 2.c.i.B”). These results were verified to be correct in the V&V.

Finally, it is noted there were no failures due to ruptures in all of the V&V scenarios, so only the probability of leakage results are meaningful for these scenarios.

Based on the foregoing evaluation, these V&V test cases verified that the user-input inspection coverage was implemented correctly in PROMISE Version 2.0.

Table RAI-1-1: 80-Year Probability of Leakage Values for Various Assumed Inspection Coverage Scenarios

V&V Scenario	Coverage Values for Inspections at 0, 20, 40, 60 Years (%)	Probability of Leakage at 80 Years
A (reference case)	100, 50, 50, 50	0.149
B	100, 0, 50, 50	0.1686
C	100, 50, 0, 50	0.149
D	100, 50, 50, 0	0.149

With respect to Item (2) of the NRC’s request, the V&V for PROMISE Version 2.0 performed some of the same test runs that were performed in the V&V for PROMISE Version 1.0. Table RAI-1-2, which was reproduced from the V&V package, contains the

results of tests where the output from PROMISE Version 2.0 is compared with the output from PROMISE Version 1.0 for four cases where assumed inspection coverage is identical for all inspections during the component lifetime. Besides inspection coverage, other PFM inputs were identical for all four test runs. As indicated in this table, the cumulative probability of leakage results are the same for both versions of the software. There were no failures due to ruptures in all cases, so only the probability of leakage results are reported.

Furthermore, all the test cases for PROMISE Version 1.0 that investigated stress intensity factors, crack growth laws, crack size distributions and random sampling were re-run in PROMISE Version 2.0. The results of those cases were identical, and no discrepancies were observed. This assures that no differences other than those intended were introduced in PROMISE Version 2.0.

Table RAI-1-2: 80-Year Probability of Leakage Values for PROMISE Versions 1.0 and 2.0 for Various Assumed Inspection Coverage Scenarios

Inspection Coverage Assumed at 20, 40, and 60 Years (%)	PROMISE Version 1.0 Results for One Crack	PROMISE Version 2.0 Results for One Crack
0	0.296	0.296
25	0.223	0.223
50	0.149	0.149
100	8.52E-4	8.52E-4

Therefore, the V&V testing performed for PROMISE Version 2.0 demonstrated that (1) the results for the cases intended to test the differences between PROMISE Version 2.0 and PROMISE Version 1.0 correctly reflected the software changes; and (2) the results for the cases re-run from the PROMISE Version 1.0 testing were not affected.

RAI-2

Regulatory Basis

Similar to RAI-1, the regulatory basis for the following requests has to do with demonstrating that the proposed alternative ISI requirements would ensure adequate structural integrity of the "Stay Cylinder Base to Hemisphere (Head)" (SG-1-BHC-1-A and SG-2-BHC-1-A) and "Stay Cylinder to Tube Sheet" welds (SG-1-TSS-3-A and SG-2-TSS-3-A) identified as Examination Category B-B, Item No. B2.31 in Section 1.0 of Attachment 1 to the submittal, and thereby providing an acceptable level of quality and safety per 10 CFR 50.55a(z)(1) for these welds.

<u>Issue</u>

(a) Figure A2 in Attachment 2 to the licensee's submittal shows the configuration of the "Stay Tube to Head Weld" and "Stay Tube to Tubesheet Weld." It is not clear to the NRC staff whether "Stay Tube to Head Weld" is the same as the "Stay Cylinder Base to Hemisphere (Head)" (SG-1-BHC-1-A and SG-2-BHC-1-A) and whether "Stay Tube to Tubesheet Weld" is the same as "Stay Cylinder to Tube Sheet" (SG-1-TSS-3-A and SG-2-TSS-3-A). Also, "Stay Tube to Head Weld" and "Stay Tube to Tubesheet Weld" in Figure A2 in Attachment 2 to the licensee's submittal do not seem to be represented by any of the paths in Figure 7-24 of EPRI report 3002015906.

(b) In addition, the NRC staff noted that the configuration of the "Stay Tube to Tubesheet Weld" in Figure A2 appears to include a tube attached to the tubesheet, rather than a plate attached to the tubesheet as shown in Figure 7-24 of EPRI report 3002015906.

<u>Request – RAI-2(a)</u>

Clarify if the "Stay Tube to Head Weld" in Figure A2 in Attachment 2 of the submittal is the same as the "Stay Cylinder Base to Hemisphere (Head)" (SG-1-BHC-1-A and SG-2-BHC-1-A) and whether "Stay Tube to Tubesheet Weld" in Figure A2 is the same as "Stay Cylinder to Tube Sheet" (SG-1-TSS-3-A and SG-2-TSS-3-A).

<u>DENC Response to RAI-2(a)</u>

DENC has reviewed Figure A2 in Attachment 2 of the submittal, and has confirmed that the "Stay Tube to Head Weld" in Figure A2 is the same as the "Stay Cylinder Base to Hemisphere (Head)" (SG-1-BHC-1-A and SG-2-BHC-1-A) and that "Stay Tube to Tubesheet Weld" in Figure A2 is the same as "Stay Cylinder to Tube Sheet" (SG-1-TSS-3-A and SG-2-TSS-3-A).

<u>Request – RAI-2(b)</u>

Discuss which paths in Figure 7-24 of EPRI report 3002015906 represent the "Stay Cylinder Base to Hemisphere (Head)" (SG-1-BHC-1-A and SG-2-BHC-1-A) welds of MPS2 and "Stay Cylinder to Tube Sheet" (SG-1-TSS-3-A and SG-2-TSS-3-A) welds of MPS2, and justify that the stresses in the selected paths are appropriate for these welds.

<u>DENC Response to RAI-2(b)</u>

The evaluation performed in EPRI Report 3002015906 considered a steam generator bottom head with a welded divider plate. The configuration at MPS2 has a stay cylinder

with a divider plate that is mechanically restrained without welds. Therefore, a new finite element model representative of the MPS2 configuration was developed and the stress analyses and fracture mechanics analyses in EPRI Report 3002015906 were reperformed with the MPS2 model. Both the deterministic fracture mechanics (DFM) and the probabilistic fracture mechanics (PFM) evaluations were performed. The stress analyses are provided in Attachment 2 of this RAI response letter and the fracture mechanics analyses are provided in Attachment 3 of this RAI response letter.

The stress analyses presented in Attachment 2 of this RAI response letter were performed consistent with the approach used in Section 7.0 of EPRI Report 3002015906. The stress analyses results are provided in Section 6.0 of Attachment 2 of this RAI response letter. The location of the stress paths for the stay cylinder base-to-hemispherical head welds (Paths 15, 16 and 17) and the stay cylinder-to-tubesheet welds (Paths 18, 19 and 20) are provided in Figure 6 of Attachment 2 (which aligns with Figure 7-24 of EPRI Report 3002015906). Typical transient stresses for these stress paths are provided in Figures 7 through 12 of Attachment 2 of this RAI response letter (which align with Figures 7-25 through 7-28 of EPRI Report 3002015906).

The technical approach used in the DFM evaluation for the MPS2 SG design is consistent with Section 8.2 in EPRI Report 3002015906. An initial flaw size of 5.2% of the wall thickness was assumed equivalent to the most conservative ASME Code, Section XI acceptance standard for these components. The ASME Code, Section XI, Appendix A, Paragraph A-4300 fatigue crack growth (FCG) law was used in the evaluation using the through-wall stress distributions from the stress analyses in Attachment 2 of this RAI response letter. In addition, the through-wall weld residual stress profile from Figure 8-1 in EPRI Report 3002015906 and the 30 ksi clad residual stress equation discussed in Section 8.2.2.4 of EPRI Report 3002015905 were included in the evaluation. The fracture mechanics models identified in Section 8.2.2.4 of EPRI Report 3002015906 were used to determine the length of time for the postulated initial flaw to grow to a depth of 80% of the wall thickness (assumed to equate to leakage in this evaluation) or the depth at which the allowable toughness (upper shelf value of K_{IC} equal to 200 ksi√inch reduced by a structural factor of 2) was reached, whichever is less.

The results of the DFM evaluation for the MPS2 SG configuration are summarized in Table RAI-2(b)-1.

Table RAI-2(b)-1: Results of DFM Evaluation for the MPS2 SG Design

Item No.	Component Description	Case Identification (Note 1)	Years to Leak	Max. K at 80 Years (ksi√in)
B2.31	SG (primary side), stay cylinder-to-hemispherical head weld	**SGPSCH-P15A**	**694**	**68.9**
		SGPSCH-P15C	2,233	51.1
		SGPSCH-P16A	1,207	61.9
		SGPSCH-P16C	7,760	43.6
		SGPSCH-P17A	1,448	64.6
		SGPSCH-P17C	6,758	50.4
B2.31	SG (primary side), stay cylinder-to-tubesheet weld	SGPSCT-P18A	7,186	11.5
		SGPSCT-P18C	1,843	27.5
		SGPSCT-P19A	17,740	10.7
		SGPSCT-P19C	3,517	29.6
		SGPSCT-P20A	9,537	11.5
		SGPSCT-P20C	3,541	27.5

Note 1: The Case Identification terminology is as follows: SG for Steam generator; PSCH for primary stay cylinder-to-hemispherical head, PSCT for primary stay cylinder-to-tubesheet; P15 through P20 represent the crack paths (see Figure 6 of Attachment 2 of this RAI response letter); A for axial part-through-wall crack; and C for circumferential part-through-wall crack. The limiting case is displayed in **red bold** text.

Table RAI-2(b)-1 shows that for the DFM evaluation, a very long period is required for hypothetical postulated flaws to leak (in excess of 600 years), which indicates that the stay cylinder-to hemispherical head and stay cylinder-to-tubesheet welds are very flaw tolerant. These results are consistent with the DFM results for the SG design modeled in EPRI Report 3002015906 and summarized in Table 8-3 of the report. Because the DFM evaluation considered hypothetical postulated flaws, structural factors of 2.0 on primary loads and 1.0 on secondary loads consistent with ASME Code, Section XI, Appendix G were applied. Also, because the most dominant load is pressure, which results in a primary stress, the structural factor of 2.0 was conservatively applied to the

fracture toughness of 200 ksi√in. This results in an allowable fracture toughness of 100 ksi√in. Table RAI 2(b)-1 shows that the maximum K values for the applicable cases are below this allowable fracture toughness after an 80 year period, which is consistent with EPRI Report 3002015906.

The PFM evaluations were performed consistent with the approach described in Section 8.3 of EPRI Report 3002015906 using PROMISE, Version 2.0. The evaluations were performed for the base case identified in Section 8.3.4.1 of EPRI Report 3002015906 (pre-service inspection, PSI, only) followed by evaluation of the MPS2 plant-specific inspection scenario consisting of PSI followed by two 10-year inspections and one 30-year inspection (PSI+10+20+50). This inspection scenario is applicable to the primary side welds in the portions of the SGs that were replaced, and was addressed in Scenario No. 10 of Table 8-10 of EPRI Report 3002015906.

In addition, PFM sensitivity studies were performed for two cases: one case where the fracture toughness was reduced to 80 ksi√in (with a standard deviation of 5 ksi√inch), and another case where the stresses were increased by a factor of 1.25. Both of these cases evaluated the MPS2 inspection scenario (PSI+10+20+50).

The results of the PFM evaluation for the MPS2 SG design are presented in the Table RAI-2(b)-2 for the base case. This table shows that with PSI inspection alone, the probabilities of rupture and leakage at the applicable locations are below the acceptance criteria of 1.0×10^{-6} after 80 years of plant operation. These results are consistent with the results presented in Table 8-9 of EPRI Report 3002015906.

From the DFM results presented in Table RAI-2(b)-1, the critical Case ID is SGPSCH-P15A. This Case ID was used to perform the PFM evaluation for the MPS2 plant-specific inspection scenario. The results of the PFM evaluations are presented in Table RAI-2(b)-3. The two sensitivity studies using the MPS2 plant-specific inspection scenario are also presented in Table RAI-2(b)-3. The results indicate that the probabilities of rupture and leakage are below the acceptance criteria of 1.0×10^{-6} after 80 years of plant operation for these cases. These results are consistent with the results presented in Tables 8-9 and 8-10 (base case and inspection scenarios), Tables 8-13 through 8-16 (sensitivity for toughness), and Tables 8-17 through 8-20 (sensitivity for stress) of EPRI Report 3002015906.

The additional analyses specific to the MPS2 SG design demonstrate that Paths 15, 16, and 17, are appropriate for the stay cylinder base-to-head welds, and that Paths 18, 19, and 20 are appropriate for the stay cylinder-to-tubesheet welds. Furthermore, the DFM and PFM results for these selected paths are acceptable and consistent with the results in EPRI Report 3002015906.

Table RAI-2(b)-2: Results of PFM Evaluation for the MPS2 SG Design for the Base Case (PSI Only)

Case Identification	Probability of Leakage at 80 Years (per Year)	Probability of Rupture at 80 Years (per Year)
SGPSCH-P15A	1.25E-09	1.25E-09
SGPSCH-P15C	1.25E-09	1.25E-09
SGPSCH-P16A	1.25E-09	1.25E-09
SGPSCH-P16C	1.25E-09	1.25E-09
SGPSCH-P17A	1.25E-09	1.25E-09
SGPSCH-P17C	1.25E-09	1.25E-09
SGPSCT-P18A	1.25E-09	1.25E-09
SGPSCT-P18C	1.25E-09	1.25E-09
SGPSCT-P19A	1.25E-09	1.25E-09
SGPSCT-P19C	1.25E-09	1.25E-09
SGPSCT-P20A	1.25E-09	1.25E-09
SGPSCT-P20C	1.25E-09	1.25E-09

Table RAI-2(b)-3: PFM Results Comparing the Base Case to the MPS2 Plant-Specific Inspection Scenarios for Limiting Case ID SGPSCH-P15A

Case	Probability of Leakage at 80 Years (per Year)	Probability of Rupture at 80 Years (per Year)
Base Case	1.25E-09	1.25E-09
MPS2 Plant-Specific Inspection Case PSI + 10 + 20 + 50 (K_{IC} = 200 ksi√in)	1.25E-09	1.25E-09
Sensitivity Study #1: MPS2 Plant-Specific Inspection Case PSI + 10 + 20 + 50 (K_{IC} = 80 ksi√inch, STD = 5 ksi√inch)	1.25E-09	5.08E-07
Sensitivity Study #2: MPS2 Plant-Specific Inspection Case PSI + 10 + 20 + 50 (Stress x 1.25)	1.25E-09	1.25E-09

RAI-3

Regulatory Basis

The regulatory basis for the following requests has to do with demonstrating that the proposed alternative ISI requirements would ensure adequate structural integrity of the SG vessel welds of MPS2 for which EPRI report 3002015906 is referenced, and thereby providing an acceptable level of quality and safety per 10 CFR 50.55a(z)(1) for these welds.

Issue

Regarding the applied stresses determined in EPRI report 3002015906;

(a) The through-wall stress distribution plots in Figures 7-25 through 7-28 of EPRI report 3002015906 for the B2.31, B2.40, C1.10, C1.20, and C1.30 welds do not show the stress distributions for the unit pressure case. Thus, the NRC staff cannot verify if the pressure stress dominates in these welds. Sections 4.3.3 and 4.6 of EPRI report 3002015906 state the dominance of the radius-to-thickness (R/t) ratio, which reflect the dominance of the pressure stress.

(b) The through-wall stress distribution plots for the thermal transients in Figures 7-17 and 7-18 of EPRI report 3002015906 for the B3.130 welds show compressive stresses at the inner surface (except the Heatup/Cooldown transient). Tensile stresses at the inside surface are typically expected for transients that have temperature drops, such as Reactor Trip which has a temperature drop of 75°F in 10 seconds as described in Section 5.2.2 of EPRI report 3002015906.

(c) Figures A2 and A3 in Attachment 1 to the submittal show that the B2.31, B2.40, and B3.130 welds of MPS2 requested in the submittal are cladded. Figures 7-15 and 7-16 of EPRI report 3002015906 (B3.130 welds) and Figure 7-24 (B2.31 and B2.40) show that the stress paths for these welds include the clad. However, the NRC staff noted that clad residual stress is not included as a separate applied stress in EPRI report 3002015906. The NRC staff noted that in a similar EPRI report for cladded welds in the pressurizer vessel, non-proprietary EPRI Report No. 3002015905, "Technical Bases for Inspection Requirements for PWR Pressurizer Head, Shell-to-Head, and Nozzle-to-Vessel Welds", December 2019, clad residual stresses are included as a separate applied stress (Section 8.2.2.4 of EPRI report 3002015905).

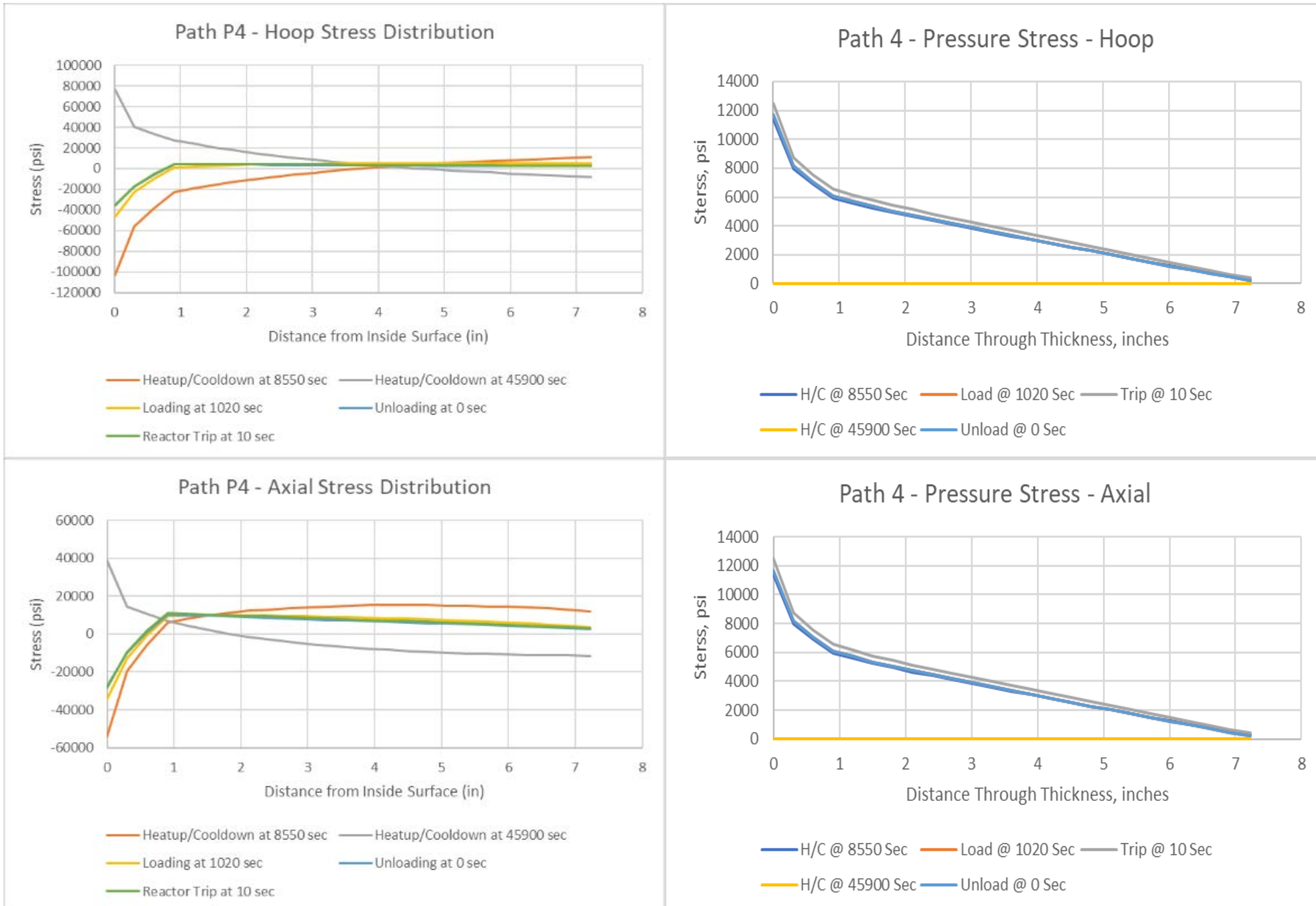
Request – RAI-3(a)

Show that the pressure stress dominates the thermal transient stresses for the B2.31, B2.40, C1.10, C1.20, and C1.30 welds requested in the submittal for MPS2.

DENC Response to RAI-3(a)

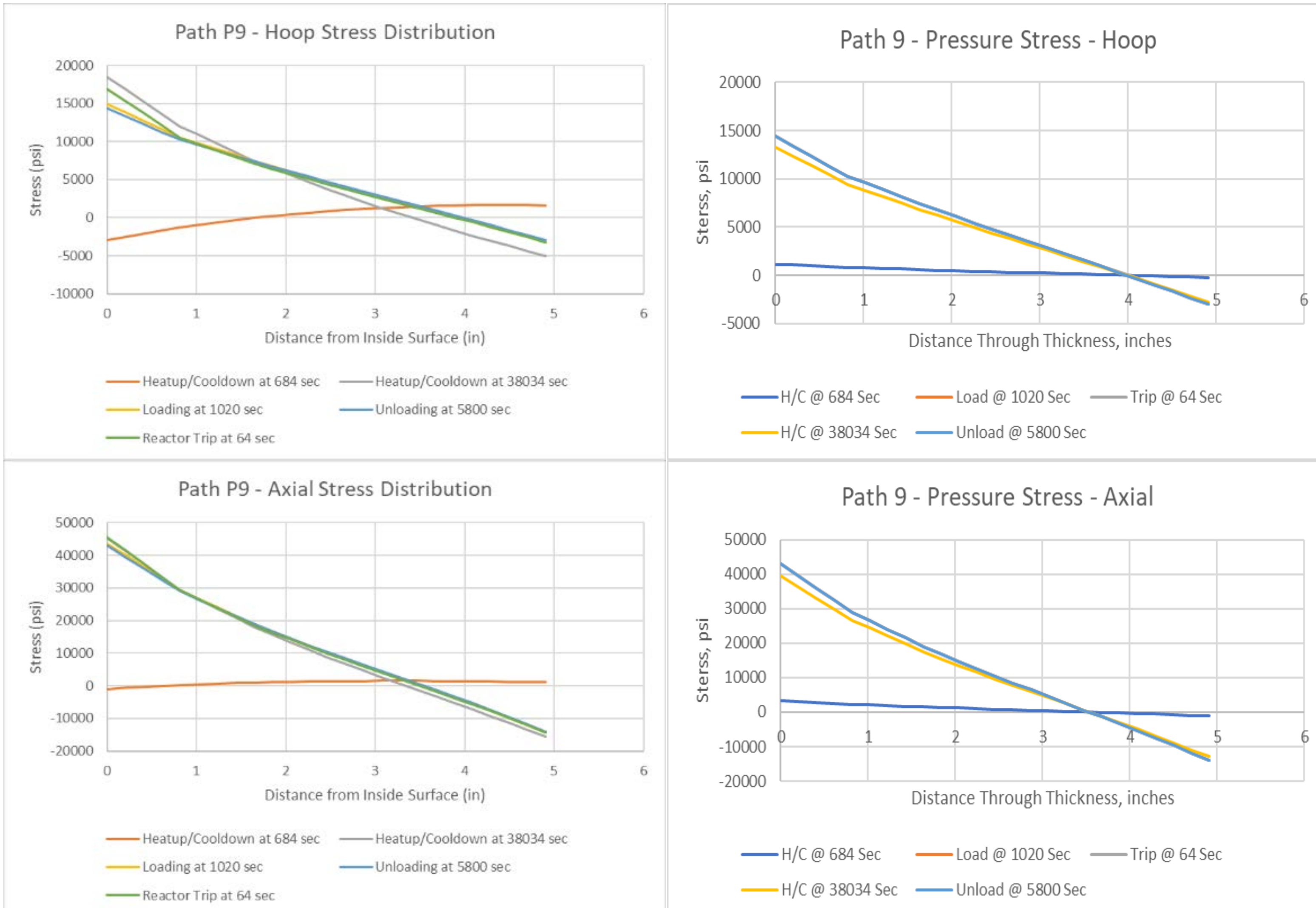
In Figures 7-25 through 7-28 in EPRI Report 3002015906 for ASME Section XI Items Nos. B2.31, B2.40, C1.10, C1.20, and C1.30, the thermal transients were all run with actual pressure loading included using the SG primary and secondary side pressures (which are different) during each transient. Extracting the effects of the pressure stresses alone were not possible from those runs. Therefore, the pressure stresses were not displayed separately for the SG welds as they were for the primary side inlet nozzles in Figures 7-17 and 7-18 in EPRI Report 3002015906. The primary side inlet nozzles are subjected only to primary side pressure so a unit pressure run was made independently of the thermal transient loadings.

To determine the effect of pressure stress alone, the analysis was reperformed with only the actual SG primary and secondary side pressures (not a unit pressure) applied for the associated thermal transients (thermal stresses not included). Figures 7-25 through 7-28 in EPRI Report 3002015906 are reproduced on the left side (a) of Figures RAI-3-1 through RAI-3-4. The pressure only stress distributions for the various transients are plotted separately on the right side (b) of each figure. Comparing the plots on the left side (a) of each figure (pressure plus thermal stresses) to the plots on the right side (b) of each figure (pressure stresses) indicates that the pressure stresses are dominant for all cases except for Path P4, which is the primary side bottom head weld. The stress distributions for Path P4 are significantly influenced by the different thermal coefficients of expansion for the stainless steel cladding and the low alloy steel base material using an assumed stress-free temperature of 70°F. However, starting at approximately two inches through the thickness from the inside surface, the thermal stresses caused by the differential thermal expansion of the two materials diminish, and the pressure stresses dominate. Therefore, for Path P4, pressure stresses dominate the total stress for the larger crack depths that contribute to the probabilities of leakage and rupture.



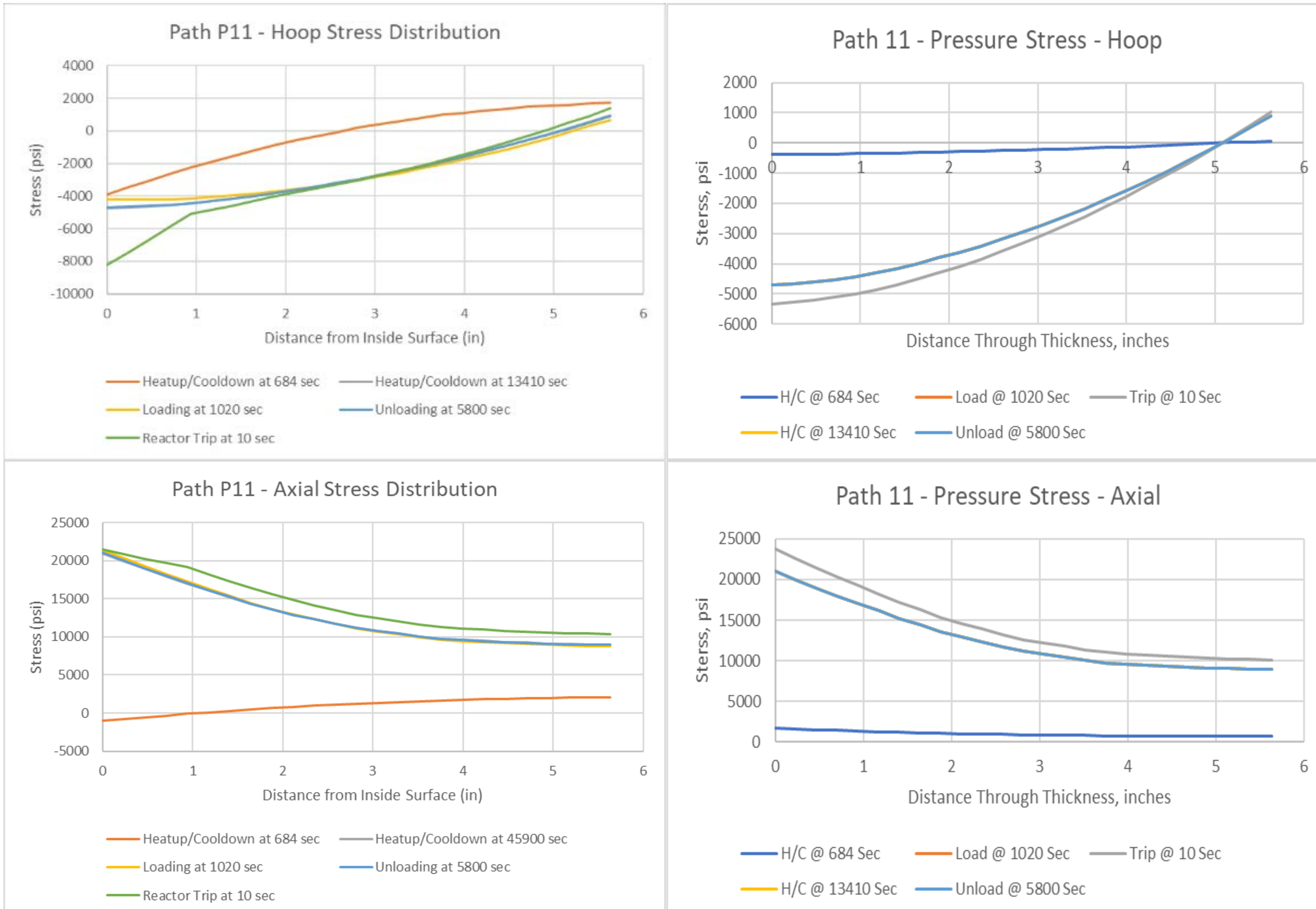
(a) (b)

Figure RAI-3-1
Through-Wall Stress Distributions at Path P4 for the PWR SG: (a) Pressure Plus Thermal Stresses (same as Figure 7-25 in EPRI Report 3002015906) and (b) Pressure Stresses



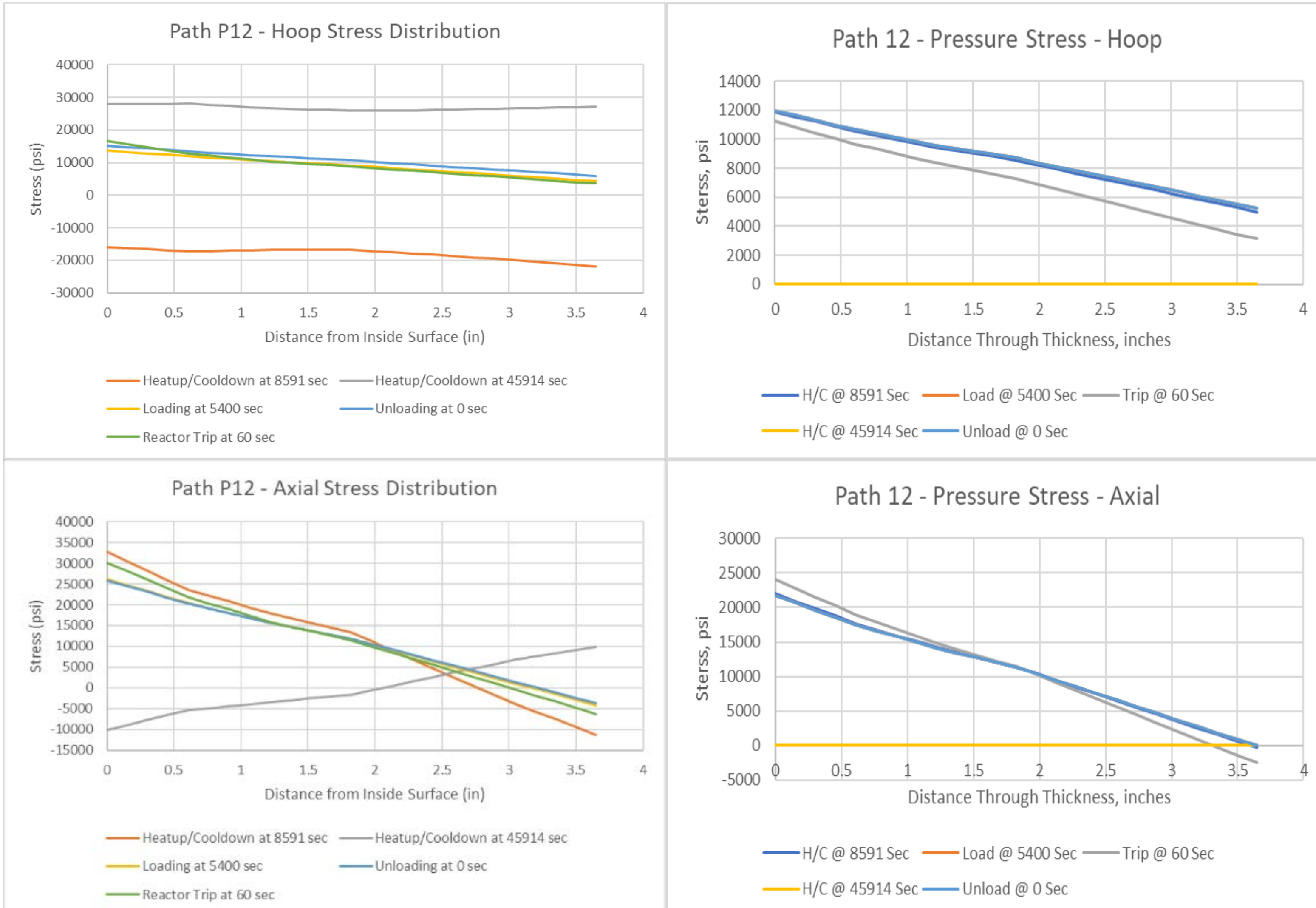
(a) (b)

Figure RAI-3-2
Through-Wall Stress Distributions at Path P9 for the PWR SG: (a) Pressure Plus Thermal Stresses (same as Figure 7-26 in EPRI Report 3002015906) and (b) Pressure Stresses



(a) (b)

Figure RAI-3-3
Through-Wall Stress Distributions at Path P11 for the PWR SG: (a) Pressure Plus Thermal Stresses (same as Figure 7-27 in EPRI Report 3002015906) and (b) Pressure Stresses



(a) (b)

Figure RAI-3-4
Through-Wall Stress Distributions at Path P12 for the PWR SG: (a) Pressure Plus Thermal Stresses (same as Figure 7-28 in EPRI Report 3002015906) and (b) Pressure Stresses

Request – RAI-3(b)

Explain if the thermal stresses on the inside surface shown in the plots in Figures 7-17 and 7-18 of EPRI report 002015906 become tensile at times other than those shown in the figures.

DENC Response to RAI-3(b)

The entire stress time histories on inside surfaces of the PWR SG inlet nozzle for the four transients reported in Figures 7-17 and 7-18 in EPRI Report 3002015906 for Path P1(N) (CE design) and Path P2N (B&W design) are shown in Figures RAI-3-5 and RAI-3-6, respectively.

The transients with temperature drops (i.e., the Reactor (or Plant) Trip and Plant Unloading transients) start with the SG hot and therefore, the inside surfaces at the start of these transients are in significant compression due to the differential thermal expansion between the stainless steel cladding and the low alloy steel base material using an assumed, stress-free temperature of 70°F. As shown in the two figures, the stresses on the inside surface at the onset of these two transients become less compressive. However, the temperature drop is not high enough during these transients for the stresses to become tensile. On the other hand, the cooldown portion of the Heatup/Cooldown transient generates through-thickness temperature variations that overcome the compressive stresses caused by the differential thermal expansion of the cladding and base material. This effect is large enough to produce tensile stresses, as indicated at a time of approximately 45,000 seconds, for the Heatup/Cooldown transient in Figures RAI-3-5 and RAI-3-6.

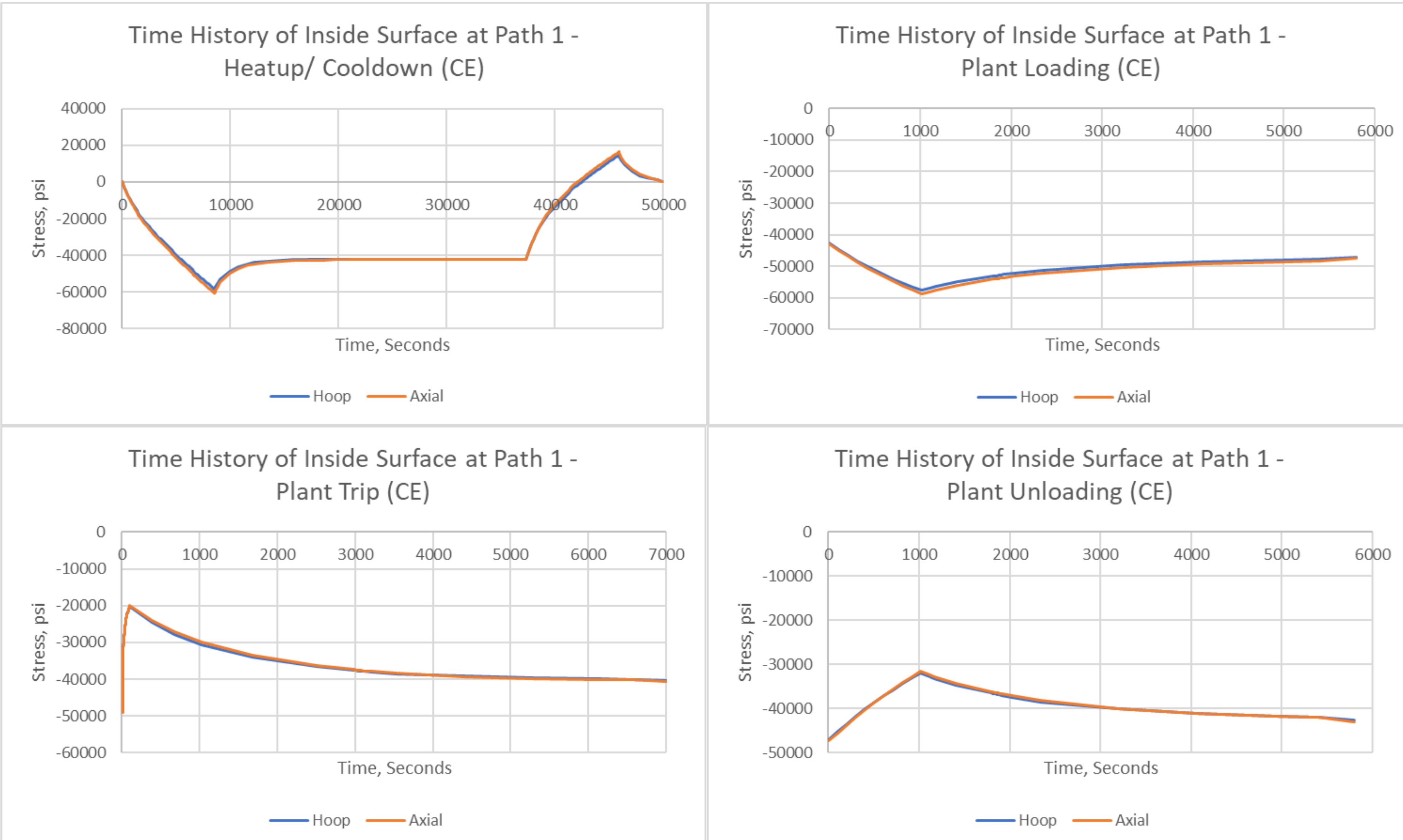


Figure RAI-3-5
Time History Stresses (Pressure Plus Thermal) at the Inside Surface for Path P1(N) for the CE Design PWR SG Inlet Nozzle (compares to Figure 7-17 in EPRI Report 3002015906)

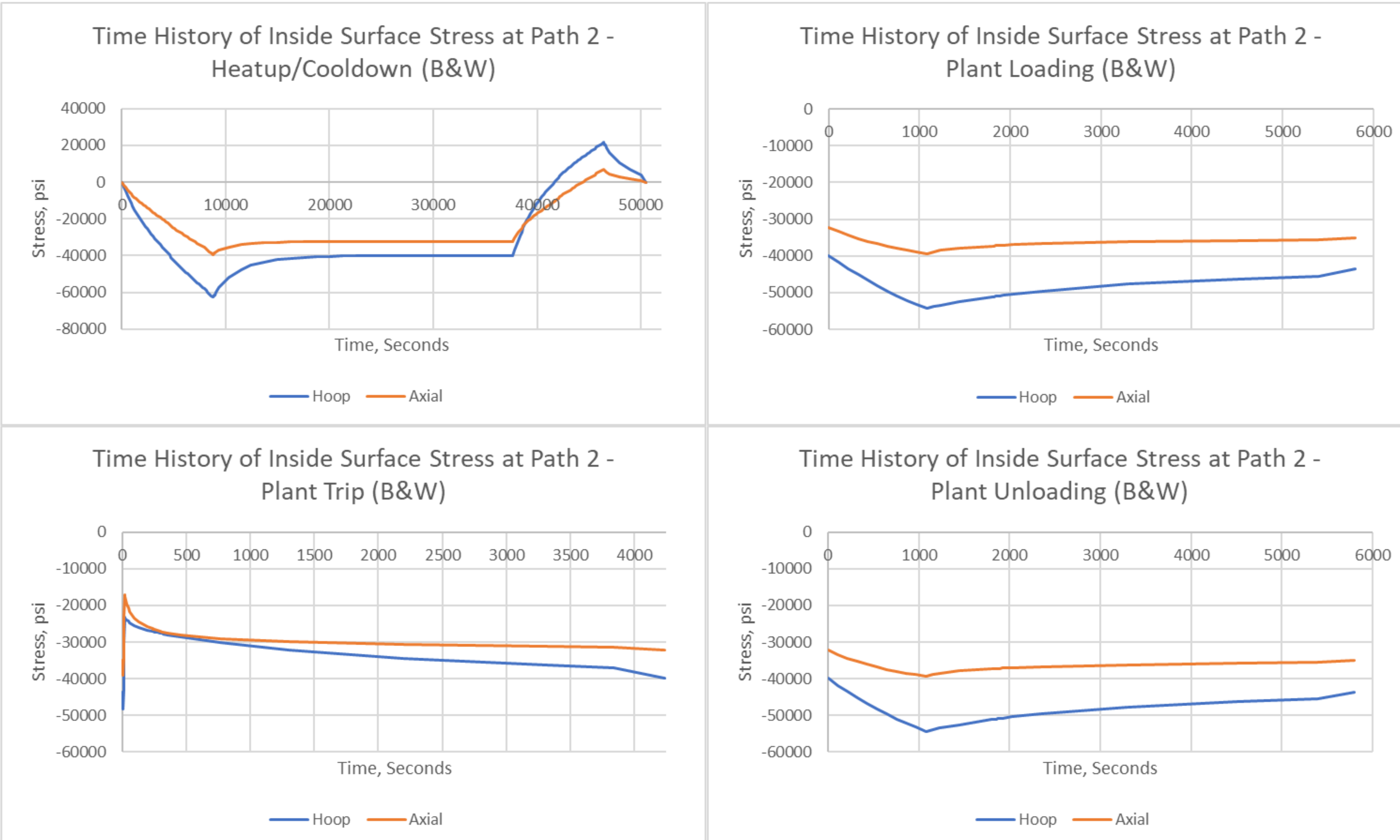


Figure RAI-3-6
Time History Stresses (Pressure Plus Thermal) at the Inside Surface for Path P2(N) for the B&W Design PWR SG Inlet Nozzle (compares to Figure 7-18 in EPRI Report 3002015906)

Request – RAI-3(c)

Justify how excluding the clad residual stress as a separate applied stress would adequately represent the total applied load in the B2.31, B2.40, and B3.130 welds of MPS2 requested in the submittal.

DENC Response to RAI-3(c)

In response to this RAI request, the analysis documented in EPRI Report 3002015906 has been supplemented to conservatively address the residual stress induced by differential thermal expansion of the cladding, using Equation (8-1) from Section 8.2.2.4 of EPRI Report 3002015905. The analysis was performed for the limiting Path SGPTH-P4A (as identified by the PFM results in Table 8-9 of EPRI Report 3002015906) and considered the MPS2 plant-specific inspection history (pre-service inspection (PSI) plus two 10-year inspection intervals, followed by one 30-year inspection interval, i.e. "PSI+10+20+50"). The results of the updated MPS2 specific PFM evaluation are shown in Table RAI-3-1. The probabilities for the base case from Table 8-9 of EPRI Report 3002015906 are also included for comparison.

Table RAI-3-1: Results of PFM Evaluation With and Without Cladding for Limiting Case ID SGPTH-P4A

Inspection Scenario	Stress-Free Temperature Used in Finite Element Analysis	Cladding Included in Finite Element Model?	Cladding Residual Stress Included	Probability of Rupture at 80 Years	Probability of Leakage at 80 Years
Base Case (PSI only, from Table 8-9 of EPRI Report 3002015906)	70°F	Yes	No	1.25E-09	5.00E-09
PSI, 10, 20, 50	70°F	Yes	No	1.25E-09	1.25E-09
PSI 10, 20, 50	70°F	Yes	Yes	1.25E-09	1.25E-09

As indicated in Table RAI-3-1, there is no difference in the probabilities of rupture and of leakage between the two cladding methods. Furthermore, the probabilities of rupture and leakage shown in Table RAI-3-1 are similar to the probabilities from Table 8-9 of EPRI Report 3002015906 for the Base Case (PSI only). Therefore, the method of modelling the cladding does not impact the results of the PFM analysis for the SG welds.

RAI-4

Regulatory Basis
The regulatory basis for the following RAI has to do with demonstrating that the proposed alternative ISI requirements would ensure adequate structural integrity of the MPS2 cladded welds identified as Examination Category B-B, Item Nos. B2.31 and B2.40, and Examination Category B-D, Item No. B3.130 in Section 1.0 of Attachment 1 to the submittal, and thereby providing an acceptable level of quality and safety per 10 CFR 50.55a(z)(1) for these welds.

Issue
For the B2.31, B2.40, and B3.130 welds of MPS2 requested in the submittal, Table A2 in Attachment 2 of the submittal shows the minimum and maximum temperatures (70°F and 535°F, respectively) for the Heatup/Cooldown transient. Section 8.2.2.5 of EPRI report 3002015906 states that the minimum temperature (200°F) during this transient corresponds to Figure 7-22 of EPRI report 3002015906, and therefore, a fracture toughness (KIC) set at the upper shelf value of the ASME Code KIC curve, 200 ksi√in, may be used. The NRC staff noted that Figure 7-22 of EPRI report 3002015906 is at the end of heatup (as noted in the figure). During the ramp periods in the beginning and ending of the Heatup/Cooldown transient, the temperatures at the subject locations of the B2.31, B2.40, and B3.130 welds of MPS2 could be lower than 200°F and thus, KIC could be lower than 200 ksi√in. For instance, during the audit summary report for PROMISE Version 1.0 (ADAMS Accession No. ML20258A002, Item No. 2.e.iii), the NRC staff observed that for the SG feedwater nozzle, KIC could be as low as 130 ksi√in during the Heatup/Cooldown transient and that this lower value of KIC is addressed by the sensitivity study results on fracture toughness in Tables 8-13 and 8-14 of EPRI report 3002014590. The NRC staff noted that the lowest KIC value, 80 ksi√in, used in the sensitivity study on fracture toughness in EPRI report 3002015906 is the same as the lowest KIC value used in the sensitivity study on fracture toughness in EPRI report 3002014590. However, the NRC staff noted that the pressure for the B2.31, B2.40, and B3.130 welds of MPS2 is more than twice the pressure analyzed for SG feedwater nozzle in the audit report since these welds are on the primary side of the reactor coolant system. The NRC staff also noted that the probability of rupture value of 1.50E-07 per year for the limiting case (SGPTH-P4A for a KIC value of 80 ksi√in) in Table 8-14 of EPRI report 3002015906 is higher than the probability of rupture value for the limiting case value of 3.75E-08 per year for the SG feedwater nozzle for a KIC value of 80 ksi√in. The NRC staff further noted in Attachment 3 to the submittal that the B2.31 welds requested for MPS2 had an examination coverage as low as 49.58%. With this examination coverage, the probability of rupture of value for the limiting case in Table 8-14 of EPRI report 3002015906 could exceed the criterion of 1E-06 per year since 1.50E-07 per year is only about one order magnitude lower than the criterion.

Request
Given the NRC staff's observations discussed above for the B2.31, B2.40, and B3.130 welds of MPS2, explain how the sensitivity study on fracture toughness in EPRI report

3002015906 addresses the lower KIC value during the ramp periods in the beginning and ending of the Heatup/Cooldown transient.

DENC Response to RAI-4

From the DFM results in Table 8-3 of EPRI Report 3002015906, the critical Case ID for Item No. B3.130 (for the CE SG design) is SGPNV-P1A-(N), and for Item Nos. B2.31 and B2.40 is SGPTH-P4A. These Case IDs also show the most sensitivity to fracture toughness in the PFM evaluation as presented in Tables 8-13 and 8-14 of EPRI Report 3002015906. The DFM results for Case ID SGPTH-P4A are bounding compared to the limiting Case ID SGPSCH-P15A for the MPS2 SG design. Specifically, the DFM result for Case ID SGPTH-P4A is 641 years from Table 8-3 of EPRI Report 3002015906, whereas the DFM result for Case ID SGPSCH-P15A is 694 years as identified in Table RAI-2-1 in the response to RAI-2. Therefore, Case IDs SGPNV-P1A-(N) and SGPTH-P4A were selected for an additional evaluation to respond to this RAI request.

The stress intensity factor (K) and temperature history at the inside surface for Case IDs SGPNV-P1A-(N) and SGPTH-P4A for Heatup/Cooldown transient (including RCS pressure) are provided in Figures RAI-4-1 and RAI-4-2 for three flaw depths that span the range of depths covered in the evaluation from EPRI Report 3002015906 (a/t = 0.2, 0.5 and 0.8). Based on a review of the supporting calculations, the maximum aspect ratio (crack half-length over depth, or *c*/*a*) from the DFM evaluation performed in Section 8.2 of EPRI Report 3002015906 for these two Case IDs is approximately 1 to 2. A conservative (longer flaw) aspect ratio of 3 was selected for use in this evaluation.

The stress intensity factor time histories were calculated for the three flaw depths for the Heatup/Cooldown transient, and the fracture toughness history for the transient was calculated using the equation for K_{IC} in Paragraph A-4200 of Nonmandatory Appendix A of ASME Code, Section XI. An upper bound value of RT_{NDT} of 0°F was used for the MPS2 SGs, based on a review of all available plant-specific SG materials specifications. As shown in Figures RAI-4-1 and RAI-4-2, the calculated applied MPS2-specific stress intensity factors are bounded with margin by the corresponding K_{IC} calculated as a function of temperature throughout the transient, for all three flaw depths.

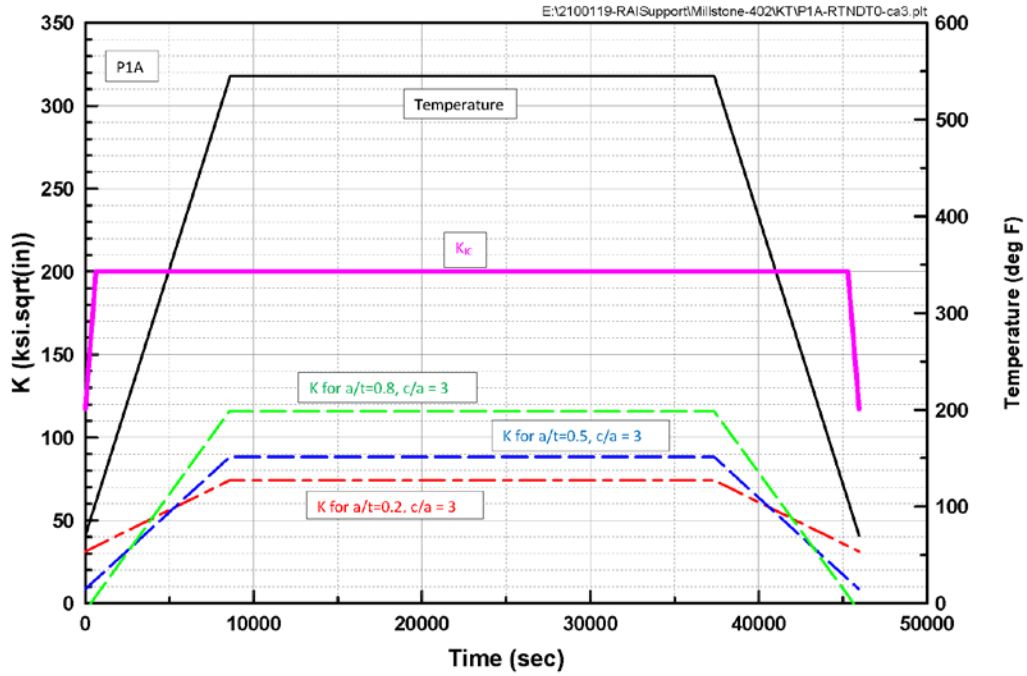


Figure RAI-4-1: Applied K vs. Fracture Toughness as a Function of Temperature for Case ID SGPNV-P1A-(N) (Using Plant-Specific Maximum SG Material RT_{NDT} Value of 0°F)

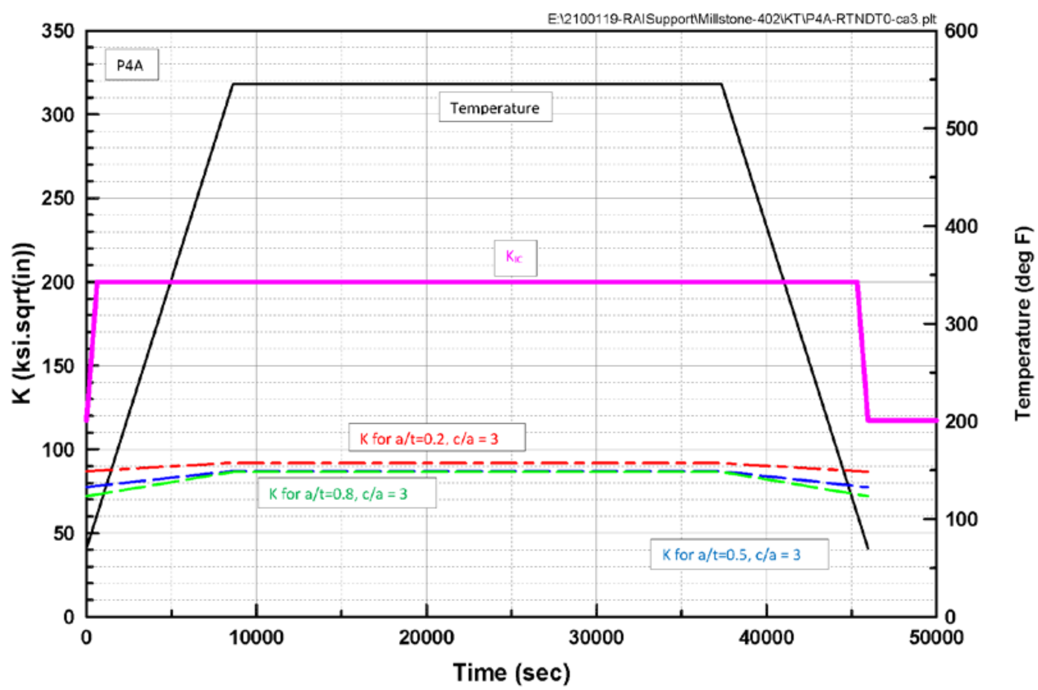


Figure RAI-4-2: Applied K vs. Fracture Toughness as a Function of Temperature for Case ID SGPTH-P4A (Using Plant-Specific Maximum SG Material RT_{NDT} Value of 0°F)

RAI-5

Regulatory Basis
The regulatory basis for the following RAI has to do with demonstrating that the proposed alternative ISI requirements would ensure adequate structural integrity of the SG vessel welds of MPS2 for which EPRI report 3002015906 is referenced, and thereby providing an acceptable level of quality and safety per 10 CFR 50.55a(z)(1) for these components.

Issue
The Reactor Trip transient (see Tables 5-7 through 5-9 of EPRI report 3002015906) was selected for the SG vessel welds of MPS2 for which EPRI report 3002015906 is referenced. A sample list of reactor coolant system design transients is given in Table 5-6 of EPRI report 3002015906. The NRC staff noted that Heatup, Cooldown, Plant loading, Plant unloading, and Reactor Trip in Table 5-6 are included in the transients selected for analysis. The NRC staff further noted that the rest of the transients in Table 5-6 are not included in the transients selected for analysis. It is not clear to the NRC staff whether the Reactor Trip transient selected for analysis in EPRI report 3002015906 accounted for transients not explicitly included in the analysis.

Request
Explain whether the Reactor Trip transient selected for the SG vessel welds of MPS2, for which the EPRI report 3002015906 is referenced, bounds and includes other transients and their cycles not explicitly included in the analysis in EPRI report 3002015906.

DENC Response to RAI-5

Table 5-6 of EPRI Report 3002015906 lists the number of cycles associated with typical design transients for a Combustion Engineering (CE) PWR (Arkansas Nuclear One, Unit 1) for 40 years and compares them with 60-year projections of actual operating cycles from Materials Reliability Program (MRP) Document 393 (EPRI Report 3002003085). The intent of that comparison was to provide insights as to the appropriate number of transients to consider in the evaluation that would reasonably bound the PWR fleet.

The Heatup, Cooldown, Plant Loading, Plant Unloading, and Reactor Trip transients were selected for evaluation in EPRI Report 3002015906. The reasoning behind these selections is described in Sections 5.2.1, 5.2.2, and 5.2.3 of EPRI Report 3002015906 for the SG weld Item Nos. in Examination Categories B-B, B-D, and C-A. For these five transients, the following number of cycles were assumed for 60 years (as indicated in Tables 5-7, 5-8, and 5-9 of the EPRI report):

- Heatup and Cooldown = 300 cycles
  (these transients were evaluated together as one single transient, as noted in Section 7.1.2.2 of 3002015906)
- Plant Loading = 5,000 cycles
- Plant Unloading = 5,000 cycles
- Reactor Trip = 360 cycles

The Large Step Load Decrease transient corresponds to a large step decrease (up to 50%) in turbine load and reactor power level. MPS2 can accommodate this large step decrease in power using the steam dump valves as a heat sink, without initiating a reactor trip or SG safety valve actuation. The Large Step Load Decrease transient is considered to be bounded by the Reactor Trip transient, based on expected pressure drop, and expected hot leg and cold leg temperature conditions. The Reactor Trip transient is explained in Sections 5.2.1, 5.2.2, and 5.2.3 of EPRI Report 3002015906. A total of 360 cycles was assumed for the Reactor Trip transient in EPRI Report 3002015906, which bounds the combined total of 220 cycles for these two transients (20 Large Step Load Decrease transients plus 200 Reactor Trip transients) that is anticipated over 60 years of operation, as identified in Table 5-6 of EPRI Report 3002015906 and based on MRP-393.

The remaining transients listed in Table 5-6 of EPRI Report 3002015906 were excluded in the evaluation because they are not typical or expected transients. In the case of the Pipe Break transient, it is considered “N/A” because it is a postulated faulted (accident) transient that is not expected to occur. All other transients were identified as “not typical” in MRP-393, as noted in Table 5-6. As described on page 5-8 of EPRI Report 3002015906, the transients identified as “not typical” in the last column of Table 5-6 are so noted because they are rare or have not occurred at most PWRs. As an example, this is validated in Table A4 in Attachment 2 of the MPS2 Request for Alternative submittal for the Loss of Power Transient, where zero cycles are projected over 60 years for MPS2. This same table also identifies in a footnote that Plant Loading and Plant Unloading transients are not counted at MPS2, whereas EPRI Report 3002015906 assumed 5,000 cycles for both of these transients.

Therefore, the selection of transients for Tables 5-7 through 5-9 of EPRI Report 3002015906 considers all typical transients and bounds a transient that is not explicitly included (Large Step Load Decrease). The transients selected for the evaluation and the associated number of cycles assumed in EPRI Report 3002015906 are considered conservative.

RAI-6

Regulatory Basis
The regulatory basis for the following RAI has to do with demonstrating that the proposed alternative ISI requirements would ensure adequate structural integrity of the MPS2 welds identified as Examination Category B-B, Item Nos. B2.31 and B2.40, and Examination Category B-D, Item No. B3.130 in Section 1.0 of Attachment 1 to the submittal, and thereby providing an acceptable level of quality and safety per 10 CFR 50.55a(z)(1) for these welds.

Issue
Table A1 in Attachment 2 to the submittal, the licensee states that MPS2 has not resulted in an event (unheated auxiliary feedwater being introduced into a hot SG) that can result in a thermal shock of the SG vessel. The NRC noted that the auxiliary feedwater event referred to in Table A1 affects the secondary side welds and NIR sections requested for MPS2 in the submittal (Item Nos. C1.10, C1.20, C1.30, C2.21, and C2.22) in Section 1.0 in Attachment 1 to the submittal. The NRC staff noted that a thermal shock event from the primary side can affect the primary side welds requested for MPS2 in the submittal (Item Nos. B2.31, B2.40, and B3.130).

Request
Similar to the thermal shock applicability discussion in Table A1 in Attachment 2 to the submittal, explain whether MPS2 has experienced a thermal shock event in the primary side of the reactor coolant system that can affect the KIC value assumed in the analysis in EPRI report 3002015906 referenced for the primary side welds requested for MPS2 in the submittal.

DENC Response to RAI-6

MPS2 has not experienced a thermal shock event in the primary side of the reactor coolant system that can affect the K_{IC} value assumed in the analysis in EPRI Report 3002015906 referenced for the primary side welds. The MPS2 design basis does not include a primary side thermal shock event analogous to the secondary side event of unheated auxiliary feedwater being introduced into a hot, dry SG. No beyond design basis event has occurred at MPS2 that introduced cold RCS primary coolant into a hot SG.

RAI-7

Regulatory Basis
The regulatory basis for the following RAI has to do with demonstrating that the proposed alternative ISI requirements would ensure adequate structural integrity of the SG vessel welds and NIR sections of MPS2 for which EPRI reports 3002015906 and

3002014590 are referenced, and thereby providing an acceptable level of quality and safety per 10 CFR 50.55a(z)(1) for these welds.

Issue
Regarding treatment of the pressure tests in EPRI reports 3002015906 and 3002014590;

(a) In Sections 5.2 of the EPRI reports, EPRI stated that it did not consider test conditions beyond a system leakage test in the analyses, and stated 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 pressure 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 primary and secondary side hydrostatic and leak tests.

(b) The transients selected for analysis in Tables 5-7, 5-8, and 5-9 in EPRI report 3002015906 and Table 5-5 in EPRI report 3002014590 do not include pressure tests (system leak and hydrostatic tests). Sections 4.2.1 and 4.3.2 of the MPS2 updated final safety analysis report indicates there are 10 cycles of primary and secondary side hydrostatic testing and 200 cycles of primary and secondary side leak testing. The NRC staff has determined in a safety evaluation for a similar submittal that the hydrostatic and leak testing are reasonably accounted for by the 300 cycles of the Heatup and Cooldown transient assumed in the EPRI reports. In Tables A1, A2, and A3 in Attachment 2 in the submittal, the licensee states 121 projected 60-year cycles of Heatup and Cooldown for MPS2. This leaves 179 cycles for pressure testing. The NRC staff noted a total of 210 cycles of pressure testing (as discussed above) for MPS2 applicable to both the primary side and secondary side. The NRC staff noted that 179 cycles of these 210 cycles of pressure testing are accounted for in the analysis in the EPRI reports. However, this still leaves 31 cycles of pressure testing of MPS2 unaccounted for. To determine whether the referenced analyses account for the MPS2 pressure tests, the staff is requesting the following RAIs.

Request RAI-7(a)

Confirm that at the maximum primary and secondary side pressures during the hydrostatic and leak tests of MPS2, the temperature of the primary side affecting the primary side welds requested for MPS2 in the submittal and the temperature of the secondary side affecting the secondary side welds requested for MPS2 in the submittal are high enough such that the upper shelf KIC of value of 200 ksi√in assumed in the EPRI reports for fracture toughness is appropriate, considering the value of the nil-ductility reference temperature (RTNDT) of 60°F assumed in calculating KIC in EPRI reports 3002015906 and 3002014590.

DENC Response to RAI-7(a)

As discussed in the response to RAI-7(b), leak tests conducted at normal plant operating pressures are performed in lieu of hydrostatic tests at MPS2. Furthermore, as discussed in the response to RAI-4, the applied stress intensity factors for the MPS2 SG primary side welds and secondary side welds throughout the Heatup/Cooldown transient (which includes leak tests) are bounded by the upper shelf fracture toughness (K_{IC}) value of 200 ksi√in assumed in EPRI Reports 3002015906 and 3002014590. The evaluation described in the response to RAI-4 considered the maximum plant-specific value for RT_{NDT} of 0°F to be appropriate for the MPS2 SGs.

Finally, the temperatures remain high enough to assure upper shelf behavior during other applicable transients. This is because the transients with temperature drops (i.e., the Reactor Trip and Plant Unloading transients) start with the SG hot, as described in the response to RAI-3(b).

Request RAI-7(b)

Discuss how the 31 cycles of pressure testing of MPS2 as discussed above are accounted for in the referenced analysis in the EPRI reports.

DENC Response to RAI-7(b)

The ten cycles of primary and secondary side hydrostatic testing and 200 cycles of primary and secondary side leak testing listed in Sections 4.2.1 and 4.3.2 of the MPS2 updated final safety analysis report reflect design basis transients. Although those transients are included in the MPS2 fatigue management program, they do not reflect current plant operating practices and the important pressurization cycles captured by the MPS2 fatigue management program.

With respect to system tests, leakage tests in accordance with ASME Code, Section XI, Examination Category C-H are conducted after each refueling outage for MPS2. Secondary side system leakage tests are performed in parallel with the Class 1 leakage tests which are performed at nominal operating temperature and pressure. The leakage tests are conducted while the plant is in Mode 3, as an integral part of the plant heatup process after refueling outages.

Hydrostatic tests refer to pressurization tests that are performed during construction prior to initial plant startup. Paragraph IWA-4540(a) of Section XI of the ASME Code requires an operating system leakage test or a secondary side system hydrostatic test following any major SG repairs. DENC has previously performed operating system leakage tests on the MPS2 SGs following repair and replacement activities, rather than secondary side system hydrostatic tests. DENC would also expect to perform operating leakage testing following any potential MPS2 SG repairs in the future.

The foregoing activities are typical practice for many PWRs. Therefore, hydrostatic tests were not considered in EPRI Report 3002015906, and only leakage tests were evaluated as part of the Heatup/Cooldown transient.

The Heatup and Cooldown transients were evaluated together as one single transient, as noted in Section 7.1.2.2 of EPRI Report 3002015906. As indicated in Tables 5-7, 5-8, and 5-9 of EPRI Report 3002015906, 300 cycles of heatup/cooldown were assumed for 60 years of operation. Each of these events include pressurization from atmospheric pressure to operating pressure, and depressurization to atmospheric pressure. Because the leakage tests are conducted as an integral part of the plant heatup process, no additional cycles were included solely for leakage testing. Therefore, 300 cycles of heatup/cooldown, which include integral leakage tests, were considered in the EPRI report. The projected number of heatup/cooldown (with integral leakage tests) events for MPS2 for 60 years of operation is 121 cycles, as identified in Tables A2, A3 and A4 of Attachment 2 of the MPS2 Request for Alternative.

Therefore, the projected 121 heatup/cooldown cycles (which also account for leak tests) for the MPS2 SGs over 60 years of operation is bounded by the corresponding 300 cycles evaluated in EPRI Report 3002015906.

RAI-8

Regulatory Basis
The regulatory basis for the following RAI has to do with demonstrating that the proposed alternative ISI requirements would ensure adequate structural integrity of the MPS2 welds identified as Examination Category B-B, Item Nos. B2.31 and B2.40 in Section 1.0 of Attachment 1 to the submittal, and thereby providing an acceptable level of quality and safety per 10 CFR 50.55a(z)(1) for these welds.

Issue
One set of the PFM results relevant to the B2.31 and B2.40 welds requested for MPS2 in the submittal is shown in Table 8-19 of EPRI report 3002015906. The probability of rupture stated for these welds is 7.33E-07 per year at 80 years, which is little margin from the criterion of 1E-06 per year such that an input parameter with high uncertainty or examination coverage can easily cause the probability of rupture value to exceed the criterion. This probability of rupture value, however, is with a stress multiplier of 1.9 that addresses the variation in radius-to-thickness (R/t) ratio from the geometric configuration modeled as discussed in Section 8.3.4.3.2 of EPRI report 3002015906. The R/t ratio applicable for Item Nos. B2.31 and B2.40 modeled in EPRI report 3002015906 is (169.75/2)/6.94 = 12.2 (dimensions from Table 4-2 of EPRI report 3002015906). The NRC staff noted that the corresponding R/t ratio for MPS2 can be calculated from the MPS2 SG lower head dimensions in Table A1 and Figure A2 in Attachment 2 to the submittal. However, the thickness of the SG lower head in Figure A2 in Attachment 2 to the submittal is not clearly shown in the figure. The NRC staff noted that the SG lower head diameter (165 inches), is given in Table A1 in Attachment

2 of the submittal. A calculation of the R/t ratio for the SG lower head of MPS2 could show that there is more margin in the probability of rupture of the B2.31 and B2.40 welds requested for MPS2 than is shown in Table 8-19 of EPRI report 3002015906.

Request
In order to verify that results for MPS2 are bounded by the results in Table 8-19 of EPRI report 3002015906, clarify the thickness of the SG lower head of MPS2 in Figure A2 in Attachment 2 to the submittal.

DENC Response to RAI-8

The thickness of the SG lower head of MPS2 is provided in Figure A3 of Attachment 2 of the MPS2 Request for Alternative submittal. The upper right portion of that figure indicates the lower head thickness is 7.0 inches, and the inside radius is 6 feet, 3 inches (or 75 inches). Therefore, the lower head outer diameter is 164 inches and the R/t ratio for the lower head is 11.7 (82 inches / 7 inches).

Table 4-2 of EPRI Report 3002015906 is reproduced in Table RAI-8-1. This table provides the relevant dimensions of the SG components selected for the evaluation and used for the finite element model of the SG in the EPRI report. As shown in this table, the thickness of the primary side of the SG used in the evaluation is 6.94 inches, which closely matches the 7-inch thickness for the MPS2 SG. The outside diameter of the primary side lower head used in the evaluation is 169.5 inches, which closely matches the 164-inches diameter of the MPS2 SG. The R/t ratio for the cylindrical portion of the primary side of the SG modeled in EPRI Report 3002015906 is 12.2 (169.75/2 inches / 6.94 inches), which is larger (and therefore more conservative) than the R/t ratio of 11.7 for the MPS2 SG.

Table RAI-8-1: Relevant Dimensions of SG Components Selected for Evaluation
(duplicates Table 4-2 in EPRI Report 3002015906)

SG Component	Item Nos.	Shell Diameter (in)	Shell/Clad Thickness (in)	Head Radius (in)	Head/Clad Thickness (in)
SG Primary Side	B2.31, B2.32, B2.40	169.75(1)	6.94 / 0.27	82.56	6.94 / 0.27
SG Secondary Side(2)	C1.10, C1.20, C1.30	169.75 170.07 240.69	3.65 / NA 4.31 / NA 4.91 / NA	Not Applicable	Not Applicable

Notes: 1. This is the SG tubesheet diameter dimension that attaches to the head.
2. For the SG secondary side, the shell diameters and thicknesses of the lower shell, intermediate shell, and upper shell, respectively, are listed.

However, it is noted that in response to RAI-2, the MPS2 SG design was specifically modeled and evaluated. Therefore, the revised analyses conducted in response to that RAI used the plant-specific dimensions for the MPS2 SG (including the lower head).

RAI-9

Regulatory Basis
The regulatory basis for the following RAI has to do with demonstrating that the proposed alternative ISI requirements would ensure adequate structural integrity of the SG vessel welds and NIR sections of MPS2 for which EPRI reports 3002015906 and 3002014590 are referenced, and thereby providing an acceptable level of quality and safety per 10 CFR 50.55a(z)(1) for these welds and sections.

Issue
Section 6.0 of Attachment 1 to the submittal states that MPS2 received preservice inspection followed by four 10-year interval inspections. However, the NRC staff noted that the MPS2 inspection history in Attachment 3 to the submittal shows the examination results and coverage achieved for only the last two inspection intervals (3rd and 4th 10-year ISI intervals). The licensee stated that the two MPS2 SGs were replaced during the 2nd 10-year ISI interval, but the top portion of the SGs were retained. The NRC staff noted that this explains why the MPS2 inspection history in Attachment 3 to the submittal shows examinations only for the 3rd and 4th 10-year ISI intervals. The NRC staff has determined that the examination results and coverage achieved for the 1st and 2nd 10-year ISI intervals for the SG vessel welds and NIR sections that were retained in the top portion of the SGs are needed to confirm that the

relevant PFM results in EPRI reports 3002015906 and 3002014590 are appropriate for the subject SG vessel welds and NIR sections requested for MPS2 in the submittal.

Request
(a) Provide the examination results and coverage achieved for the 1st and 2nd 10-year ISI intervals for the subject SG vessel welds and NIR sections of MPS2 that were retained.

OR

(b) Confirm that the examination results for the 1st and 2nd 10-year ISI intervals for the SG vessel welds and NIR sections listed in Attachment 3 to the submittal that were retained are the same as those in the attachment and that the coverages achieved for the 1st and 2nd 10-year ISI intervals for the SG vessel welds and NIR sections listed in Attachment 3 to the submittal that were retained are not less than the smallest coverage of 56.3% shown in the attachment.

DENC Response to RAI-9

The examination records for the 1st and 2nd 10-year ISI intervals for the MPS2 SG vessel welds and NIR sections listed in Attachment 3 to the Request for Alternative that were retained after the SG replacement were reviewed to assess the examination coverages obtained during those examinations. Table RAI-9-1 lists the applicable SG welds and NIR sections that were retained in the portions of both of the MPS2 SGs. DENC reviewed the examination records for these welds for all past examinations. The examinations performed during the 1st and 2nd intervals did not explicitly record the amount of examination coverage. However, the inspection records identified that there was limited coverage for some of the welds, the reasons for the limitations, and the examination techniques and scan angles that were used. This information for the 1st and 2nd interval examinations is the same as the information for the 3rd and 4th interval examinations where examination coverage was calculated. In addition, there have been no design configuration changes that would alter examination access for any of the welds or NIR sections.

Therefore, DENC's review concluded that the examination coverages for the 1st and 2nd intervals would have been no less than the coverages reported during the 3rd interval, per Request (b) of RAI-9. For the purposes of the coverage sensitivity study, the minimum examination coverage for all of the Items that were retained in the MPS2 SGs was 56.3%, which was recorded for the examination of Weld SG-2-MS-1 (Item No. C2.21) during the 3rd interval inspection (as identified in the last table of Attachment 3 of the MPS2 Request for Alternative). Also, as stated in Section 8.4 of both EPRI Report 3002015906 and 3002014590, and cited in the Attachment 1 conclusions of the MPS2 Request for Alternative, the PFM and DFM evaluations performed as part of the technical basis demonstrate that, after PSI, no other inspections are required until 80 years to meet the NRC safety goal of 10^{-6} failures per reactor year.

Table RAI-9-1: MPS2 SG Welds and NIR Sections That Were Retained After SG Replacement

MPS2 SG	Component ID	Component Description	ASME Item No.
1	1-SC-3	Cone to Upper Shell Weld	C1.10
	SG-1-THS-1	Secondary Head Circumferential Weld to Shell	C1.20
	SG-1-THS-2	Head Circumferential Weld	C1.20
	SG-1-FW-1	Feed Water Nozzle to Shell Weld	C2.21
	SG-1-FW-IR-1	Feed Water Nozzle Inside Radius Section	C2.22
2	SG-2-MS-1	Main Steam Nozzle to Head Weld	C2.21
	SG-2-MS-IR-1	Main Steam Nozzle Inside Radius Section	C2.22

ATTACHMENT 2

SUPPORTING MILLSTONE POWER STATION UNIT 2 STEAM GENERATOR DESIGN STRESS ANALYSIS FOR RAI-2(b)

MILLSTONE POWER STATION UNIT 2
DOMINION ENERGY NUCLEAR CONNECTICUT, INC.

1.0 INTRODUCTION

(similar to the content in Section 7, page 7-1 of EPRI Report 3002015906)

This attachment describes the stress analyses for the Millstone Power Station Unit 2 (MPS2) steam generator (SG) lower heads. The geometry of MPS2 SG lower heads was provided in Attachment 2, Figures A2 and A3 of the Dominion Energy Nuclear Connecticut, Inc. (DENC) Request for Alternative [1]. Due to the complex behavior of the stress distribution near the welds, finite element analyses (FEA) were performed for the applicable SG components. The FEA model, material properties, operating loads and transients are discussed in this section. The technical approach employed is similar to that used in Section 7 of EPRI Report 3001205906 [2]. Finite element models (FEMs) were developed for the components using the ANSYS finite element analysis software package [3].

Stress analyses were performed for thermal transients and internal pressure. For loads due to thermal transients, thermal analyses were performed to determine the temperature distribution histories for each transient. The temperature distribution histories were then used as inputs to perform stress analyses for the selected transients.

In performing the analyses, the following assumptions were made during development of the FEMs and thermal/mechanical stress evaluations, consistent with the assumptions made in EPRI Report 3002015906:

- The welds were not specifically modeled. The material properties between the base metals and the weld materials are similar enough that the effect of this assumption is assumed to be minimal.
- Representative heat transfer coefficients during thermal transients were conservatively assumed for each component.
- All thermal transients were assumed to start and end at a steady-state uniform temperature.
- The stress-free reference temperature for thermal stress calculations was assumed to be an ambient temperature of 70°F, which was also used for thermal strain calculations.
- All outside surfaces were assumed to be fully insulated and the insulation itself was treated as perfect, with zero heat transfer capability. This assumption is typical for stress analyses in similar components.
- Pressure stresses were separately calculated at a stress-free temperature of 70°F without any thermal stress effects.
- For thermal heat transfer analyses, 3,600 seconds was added to the end of each transient time to ensure that any lagging peak stresses were captured, followed by a steady state load step (at an arbitrary 400 seconds after the 3,600 seconds of additional time).

2.0 FINITE-ELEMENT MODEL

(similar to the content in Section 7.1.1 of EPRI Report 3002015906)

For the purposes of modeling the primary lower head and associated welds, an FEM was developed using the ANSYS finite element analysis software package [3]. The FEM is a 3-dimensional (3-D) half model of the SG primary head, stay cylinder, tubesheet, support skirt and the lower portion of the secondary shell. The model includes the SG primary head and cladding, tubesheet, stay cylinder and cladding (primary side), lower secondary shell, and primary divider plate with overall dimensions provided in Attachment 2, Figures A2 and A3 in DENC's Request for Alternative [1]. The MPS2 SG design information is included in Reference [4], however, a table of detailed dimensions for the MPS2 SG design (similar to Table 4-2 of EPRI Report 3002015906) is not provided in this attachment because the dimensions are proprietary to B&W.

Because of the geometric complexity of the lower head-to-stay cylinder and lower head-to-tubesheet welds and attached components, a 3-D half model was constructed using eight-node structural solid, SOLID45, elements. The thermal transient analyses used SOLID70 thermal elements. The FEM is shown in Figure 1.

The SG vessel penetrations (such as primary inlet/outlet nozzles, manways, snubber lugs, and support skirt) were not modeled, nor were any of the SG internal components (such as the U-tubes, tube support plates, and feed rings) other than the tubesheet and primary divider plate; this is because they do not significantly affect the temperature distributions in the SG welds under consideration.

The welds and the perforated region of the tubesheet were not specifically modeled. For the region of the perforated tubesheet, the equivalent material properties (that is, modulus of elasticity and Poisson's ratio) were applied. The equivalent modulus of elasticity, E, and Poisson's ratio values (v) were obtained from Reference [6] using the ligament efficiency for the tubesheet.

3.0 MATERIAL PROPERTIES

(similar to the content in Section 5.1 of EPRI Report 3002015906)

The SG tubesheet, stay cylinder and head are fabricated from SA-508, Class 3 material, while the secondary side shell plates and base support are fabricated from SA-533, Grade B, Class 1 material [5]. The cladding material is SA-240, Type 304L [5]. The primary side head divider plates are also fabricated from SA-204, Type 304L material [5].

The material properties were obtained from the relevant tables in the 2013 Edition of ASME Code, Section II, Part D [5]. Temperature-dependent material properties used in the FEA are listed in Table 1 through Table 4 of this attachment. Air was assumed for the fluid inside the stay cylinder.

4.0 Pressure / Thermal Stress Analysis

(similar to the content in Section 7.2.2 of EPRI Report 3002015906)

4.1 Internal Pressure Loading Analysis

(similar to the content in Section 7.2.2.1 of EPRI Report 3002015906)

For internal pressure, appropriate internal pressures for each transient were interpolated for each time step and applied in the stress analysis of each transient.

4.2 Thermal Heat Transfer Analyses

(similar to the content in Section 7.2.2.2 of EPRI Report 3002015906)

Heat transfer coefficients, fluid temperatures and pressures for the thermal transients listed in Table 5 of this attachment were applied to the interior surface nodes of the tubesheet, and primary and secondary shells. These transients are the same as described in Sections 5.2.1 and 5.2.3 of EPRI Report 3002015906. A heat transfer coefficient of 10,000 BTU/hr-ft^2-°F was applied to the inside surfaces of the primary side and the portions of the secondary side exposed to water. No heat transfer coefficients or temperatures were applied to the insulated outside surfaces of the model. As an example of the thermal boundary conditions, Figure 2 of this attachment shows representative plots of the thermal loads applied for Plant Loading transient.

4.3 Thermal Stress Analyses

(similar to the content in Section 7.2.2.3 of EPRI Report 3002015906)

Symmetric boundary conditions were applied on the symmetry plane of the FEM. The nodes at the base of the support skirt were restrained in the circumferential and vertical directions. The appropriate internal pressures for each transient were applied to the interior surfaces of the model. The bottom face of the support skirt and the air elements inside the stay tube were held at a constant temperature of 120°F. The reference temperature for the thermal stress evaluations was assumed to be 70°F. Figure 3 of this attachment shows an example plot of the pressure load and boundary conditions applied for the Plant Loading transient thermal stress analyses. The air elements inside the stay tube and the divider plate were removed from the stress analysis portion of the evaluations because they are non-structural.

5.0 STRESS ANALYSIS RESULTS

(similar to the content in Section 7.2.3 of EPRI Report 3002015906)

Representative temperature and stress contour plots for the Heatup/Cooldown transient are shown in Figure 4 and Figure 5 of this attachment, respectively. The time shown in Figure 5 of this attachment is when the maximum total stress intensity occurs for the entire model.

Figure 6 of this attachment shows the path locations where stresses were extracted. Paths P15 through P17 were selected for the stay cylinder-to-lower head circumferential

weld. Paths P18 through P20 were selected for the stay cylinder-to-tubesheet circumferential weld.

All stresses were extracted in a cylindrical coordinate system with “Y” corresponding to the hoop direction of the SG and “Z” corresponding to the axial (height) direction of the SG. Since Paths P15 through P17 are parallel to global “Y” (horizontal) direction, the global “X” direction stresses correspond to axial stresses for Paths P15 through P17. Representative through-wall stress distributions, at the time when the maximum total inside stress intensity for the entire model occurs during each transient, are shown in Figure 7 through Figure 12 of this attachment.

6.0 REFERENCES

1. Letter from M. D. Sartain (Dominion) to U.S. Nuclear Regulatory Commission, “Dominion Energy Nuclear Connecticut Inc., Millstone Power Station Unit 2, Alternative Request RR-05-06 – Inspection Interval Extension for Steam Generator Pressure-Retaining Welds and Full-Penetration Welded Nozzles,” dated July 15, 2020.

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

3. ANSYS Mechanical APDL (UP20170403) and Workbench (March 31, 2017), Release 18.1, SAS IP, Inc.

4. ETE-MP-2021-1029, “MPS2 Replacement Steam Generator Design Information,” Revision 0.

5. ASME Boiler and Pressure Vessel Code, Section II, Materials, Part D - Properties, 2013 Edition.

6. Thomas Slot, "Stress Analysis of Thick Perforated Plates," Technomics, 1972.

Table 1: Material Properties for Low Alloy Steel (SA-533, Grade B, Class 1)
(similar to Table 5-2 of EPRI Report 3002015906)

Temperature (°F)	Modulus of Elasticity (E) (10^6 psi)	Coefficient of Thermal Expansion (α) (10^{-6} in/in/°F)	Thermal Conductivity (K) (10^{-4} BTU/in-s-°F)	Specific Heat (C)(4) (BTU/lb-°F)
70	29.0	7.0	5.49	0.107
100	28.9(1)	7.1	5.46	0.108
150	28.7(1)	7.2	5.44	0.111
200	28.5	7.3	5.44	0.115
250	28.3(1)	7.3	5.42	0.117
300	28.0	7.4	5.42	0.121
350	27.8(1)	7.5	5.39	0.123
400	27.6	7.6	5.35	0.126
450	27.3(1)	7.6	5.32	0.129
500	27.0	7.7	5.25	0.131
550	26.7(1)	7.8	5.21	0.134
600	26.3	7.8	5.14	0.137
650	25.8(1)	7.9	5.07	0.139
700	25.3	7.9	5.00	0.142

Notes:

1. Linearly interpolated.
2. Density (ρ) = 0.280 lb/in^3 [5, Table PRD], assumed temperature independent.
3. Poisson's Ratio (v) = 0.3, assumed temperature independent.
4. Calculated per Note 1 of Table TCD [5].

Table 2: Material Properties for Low Alloy Steel (SA-508, Class 3)
(similar to Tables 5-3 or 5-4 of EPRI Report 3002015906)

Temperature (°F)	Modulus of Elasticity (E) (10^6 psi)	Coefficient of Thermal Expansion (α) (10^{-6} in/in/°F)	Thermal Conductivity (K) (10^{-4} BTU/in-s-°F)	Specific Heat (C)(4) (BTU/lb-°F)
70	27.8	6.4	5.49	0.107
100	27.6(1)	6.5	5.46	0.108
150	27.4(1)	6.6	5.44	0.111
200	27.1	6.7	5.44	0.115
250	26.9(1)	6.8	5.42	0.117
300	26.7	6.9	5.42	0.121
350	26.5(1)	7.0	5.39	0.123
400	26.2	7.1	5.35	0.126
450	26.0(1)	7.2	5.32	0.129
500	25.7	7.3	5.25	0.131
550	25.4(1)	7.3	5.21	0.134
600	25.1	7.4	5.14	0.137
650	24.9(1)	7.5	5.07	0.139
700	24.6	7.6	5.00	0.142

Notes:

1. Linearly interpolated.
2. Density (ρ) = 0.280 lb/in^3 [5, Table PRD], assumed temperature independent.
3. Poisson's Ratio (ν) = 0.3, assumed temperature independent.
4. Calculated per Note 1 of Table TCD [5].
5. SA-508, Class 3 has been reclassified as SA-508, Grade 3, Class 1 in Reference [5].

Table 3: Material Properties for Stainless Steel (SA-240, Type 304L)
(similar to Table 5-5 of EPRI Report 3002015906)

Temperature (°F)	Modulus of Elasticity (E) (10^6 psi)	Coefficient of Thermal Expansion (α) (10^{-6} in/in/°F)	Thermal Conductivity (K) (10^{-4} BTU/in-s-°F)	Specific Heat (C)(4) (BTU/lb-°F)
70	28.3	8.5	1.99	0.114
100	28.1(1)	8.6	2.01	0.114
150	27.8(1)	8.8	2.08	0.117
200	27.5	8.9	2.15	0.119
250	27.3(1)	9.1	2.22	0.121
300	27.0	9.2	2.27	0.122
350	26.7(1)	9.4	2.34	0.124
400	26.4	9.5	2.41	0.126
450	26.2(1)	9.6	2.45	0.127
500	25.9	9.7	2.52	0.129
550	25.6(1)	9.8	2.57	0.129
600	25.3	9.9	2.62	0.130
650	25.1(1)	9.9	2.69	0.131
700	24.8	10.0	2.73	0.132

Notes:

1. Linearly interpolated
2. Density (ρ) = 0.290 lb/in^3 [5, Table PRD], assumed temperature independent.
3. Poisson's Ratio (v) = 0.31, assumed temperature independent.
4. Calculated per Note 1 of Table TCD [5].

Table 4: Effective Material Properties for Thick Perforated Plates (SA-508, Class 3)

Temperature (°F)	Effective Modulus of Elasticity (E*) (10^6 psi)	Effective Modulus of Elasticity in Thickness Direction (E_z*) (10^6 psi)
70	6.56	14.18
100	6.52	14.09
150	6.46	13.96
200	6.40	13.82
250	6.35	13.72
300	6.30	13.62
350	6.24	13.49
400	6.18	13.36
450	6.13	13.23
500	6.07	13.11
550	6.00	12.95
600	5.93	12.80
650	5.87	12.67
700	5.81	12.54

Notes:

1. The effective modulus of elasticity (E*) will be applied to EX and EZ in ANSYS and the effective modulus of elasticity in thickness direction (E_z*) will be applied to EY in ANSYS.
2. The effective Poisson's ratio (v*) of 0.383, which will be applied to NUXY and NUYZ in ANSYS, while the standard value of 0.3 is applied to NUXZ.
3. The effective shear moduli (G*) is calculated as 1.99*10^6 psi at 600°F, which will be applied to GXZ in ANSYS.
4. The effective transverse shear modulus (G_z*) is calculated as 3.30*10^6 psi at 600°F, which will be applied to GXY and GYZ in ANSYS.
5. All other properties are per Table 2.

Table 5: Typical Thermal Transients for a Typical CE Steam Generator[1]
(similar to Tables 5-7 and 5-9 of EPRI Report 3002015906)

Description	Plant Condition	Time, sec	T_{PRI_HOT}[2], °F	T_{PRI_COLD}[2], °F	T_{SS}[2], °F	P_{PRI}[2], psig	P_{SEC}[2], psig	60-Year Cycles
Plant	Normal	0	70.0	70.0	70.0	0	0	300
Heatup/Cooldown		8550	545.0	545.0	545.0	2235	1000	
		37350[3]	545.0	545.0	545.0	2235	1000	
		45900	70.0	70.0	70.0	0	0	
Plant Loading,	Normal	0	550.0	545.0	545.0	2300	1000	5000
5%/min		1020	610.0	550.0	540.0	2300	1000	
(15% to 100%)		1800	610.0	550.0	540.0	2300	1000	
Plant Unloading,	Normal	0	610.0	550.0	540.0	2300	1000	5000
5%/min		1020	550.0	545.0	545.0	2300	1000	
(100% to 15%)		1800	550.0	545.0	545.0	2300	1000	
Reactor Trip	Upset	0	610.0	550.0	540.0	2235	1000	360
		8	615.0	565.0	552.0	2435	1104	
		10	615.0	565.0	555.0	2435	1130	
		15	580.0	565.0	555.0	2285	1130	
		20	560.0	550.0	545.0	2035	1030	
		60	540.0	535.0	530.0	1735	1000	
		100	530.0	530.0	530.0	1700	1000	
		3000	530.0	530.0	530.0	2235	1000	

Notes:

1. Above transients are defined based on the normal operating temperatures of 610°F (SG primary inlet), 550°F (SG primary outlet), and 540°F (SG secondary steam) and the normal operating pressures of 2,235 psi (SG primary) and 1,000 (SG secondary) and the transient descriptions in Reference [2].
2. T_{PRI_HOT} is primary side hot temperature, T_{PRI_COLD} is primary side cold temperature, and T_{SS} is secondary side temperature. Similarly, P_{PRI} is primary side pressure and P_{SEC} is secondary side pressure.
3. An arbitrary time of eight hours was assumed to reach steady state conditions after heatup.

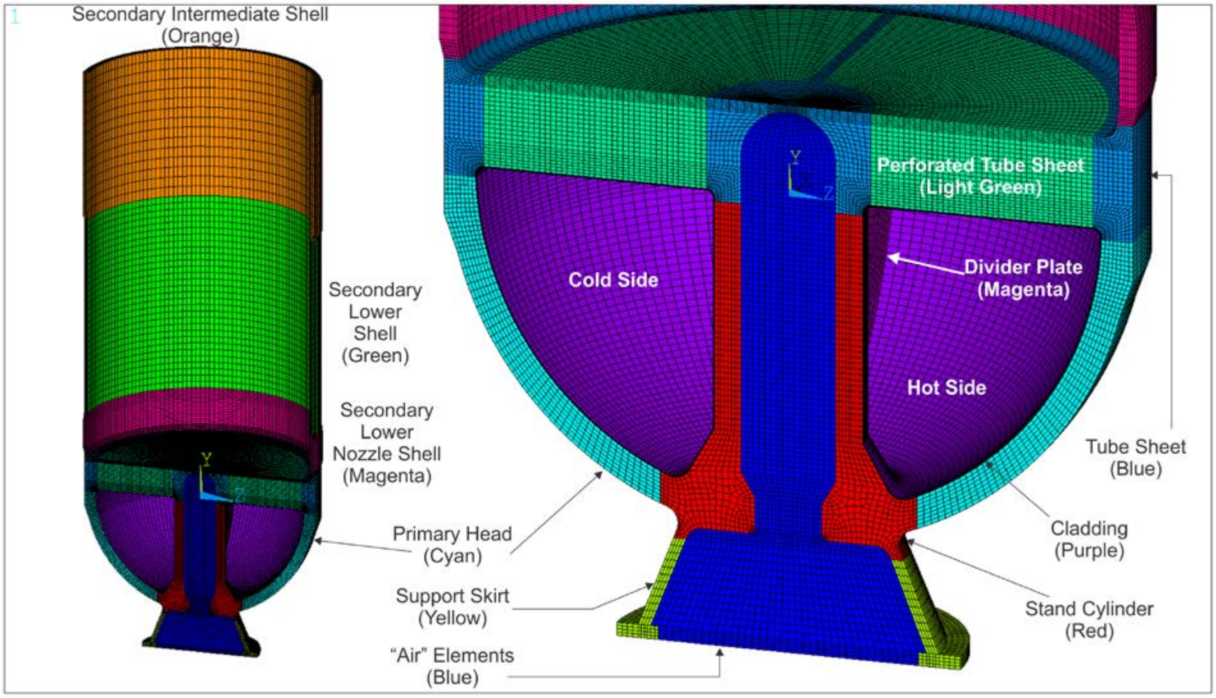
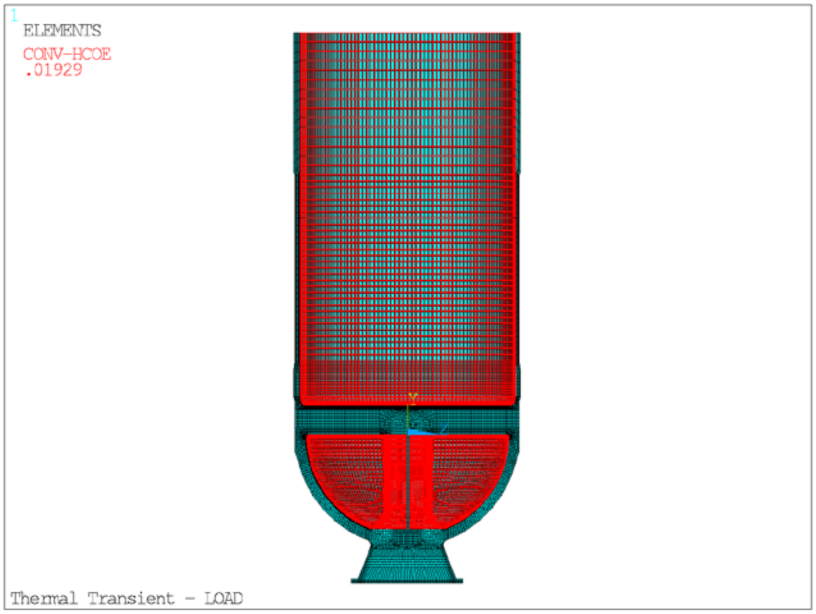
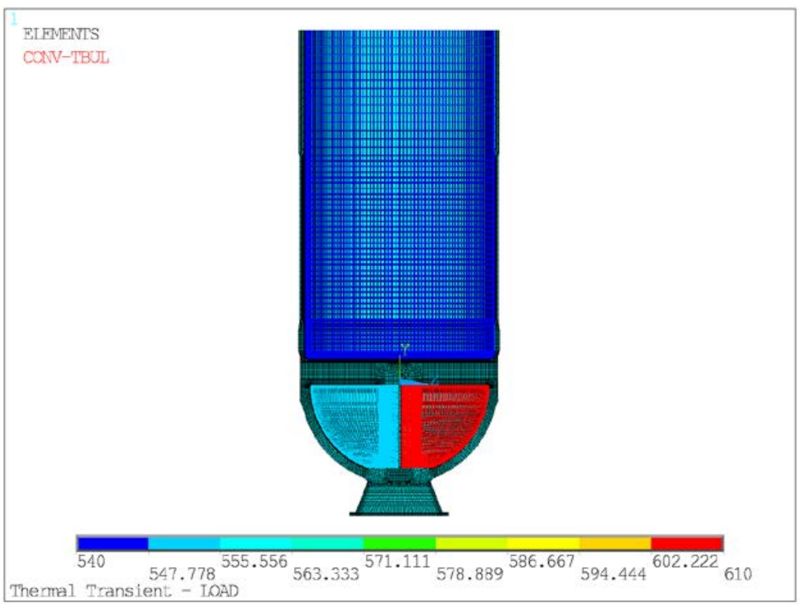


Figure 1. 3-D Finite Element Model and Mesh
(similar to Figure 7-19 of EPRI Report 3002015906)



Heat Transfer Coefficient



Bulk Temperature

Figure 2. Example of Applied Thermal Boundary Conditions for Thermal Transient Analyses

(similar to Figure 7-20 of EPRI Report 3002015906)

Plant Loading transient shown, loads applied at time = 1,800 seconds.

(Units for HTC are BTU/sec-in^2-°F, TBULK is °F)

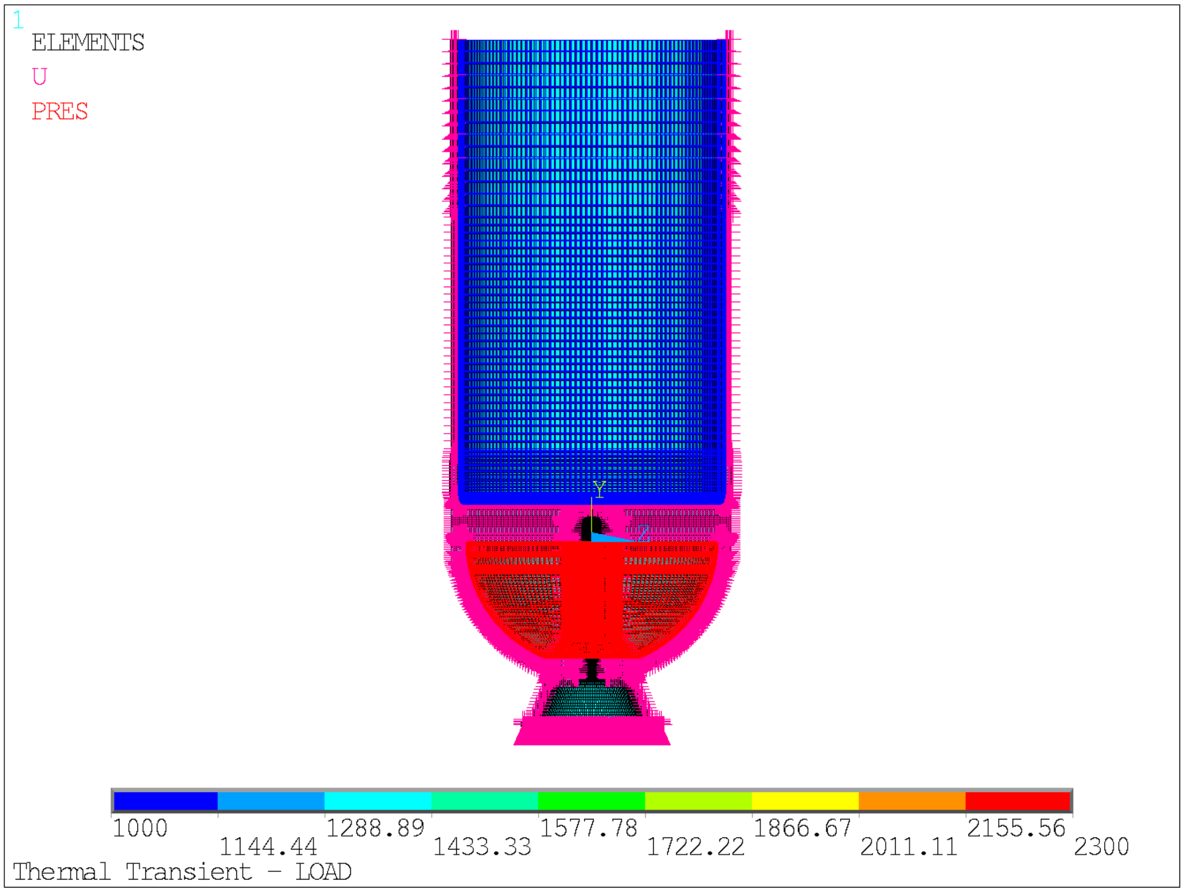


Figure 3. Example of Applied Mechanical Boundary Conditions and Pressure for Thermal Stress Analyses

(similar to Figure 7-21 of EPRI Report 3002015906)

Plant Loading transient shown, loads applied at time = 1,800 seconds.

(Units for pressure are psi)

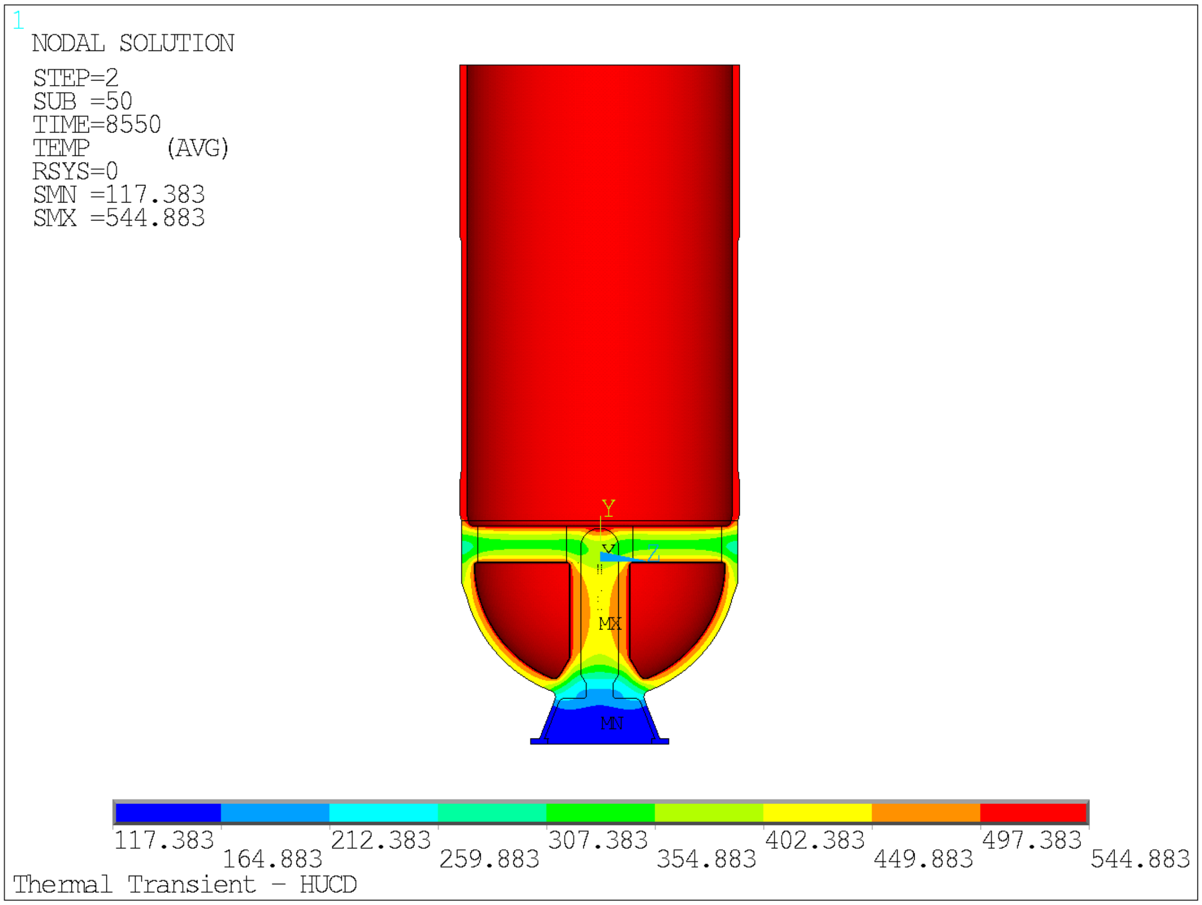
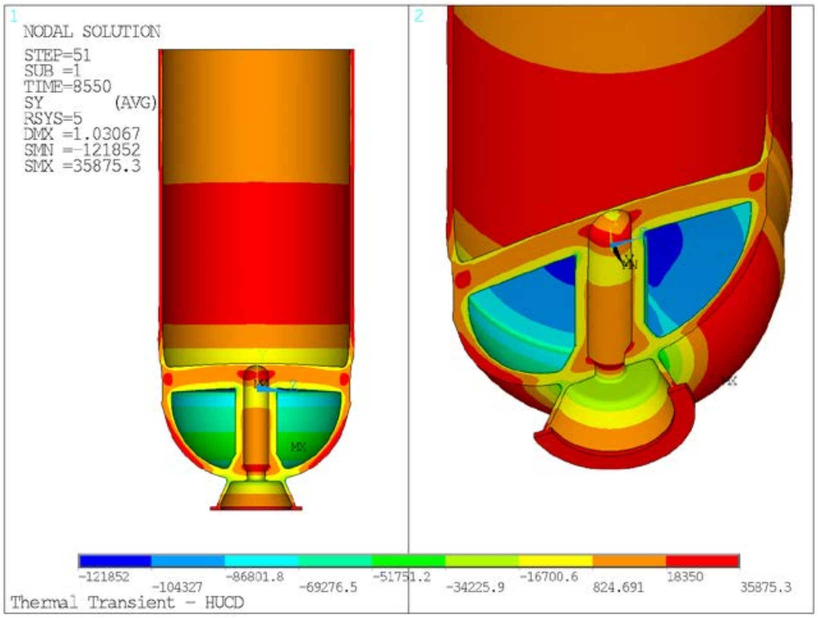


Figure 4. Example of Temperature Contour, Heatup/Cooldown Transient (Time = 8,550 seconds)

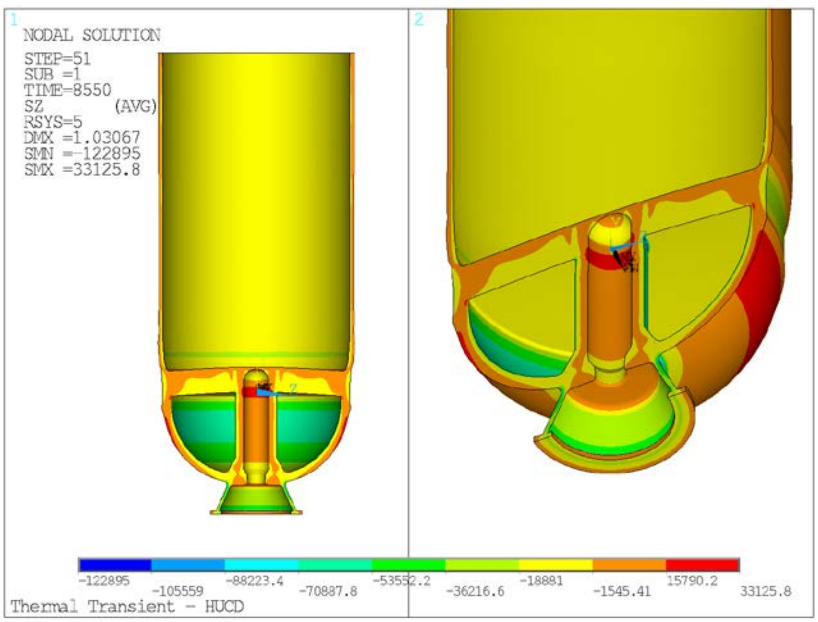
(similar to Figure 7-22 of EPRI Report 3002015906)

(Units for temperature are °F)

(Divider Plate and Air Elements included.)



Hoop Stress



Axial Stress

Figure 5. Example Stress Contours, Heatup/Cooldown Transient (Time = 8,550 seconds)

(similar to Figure 7-23 of EPRI Report 3002015906)

(Units for stress are psi)

(Divider Plate and Air Elements are removed.)

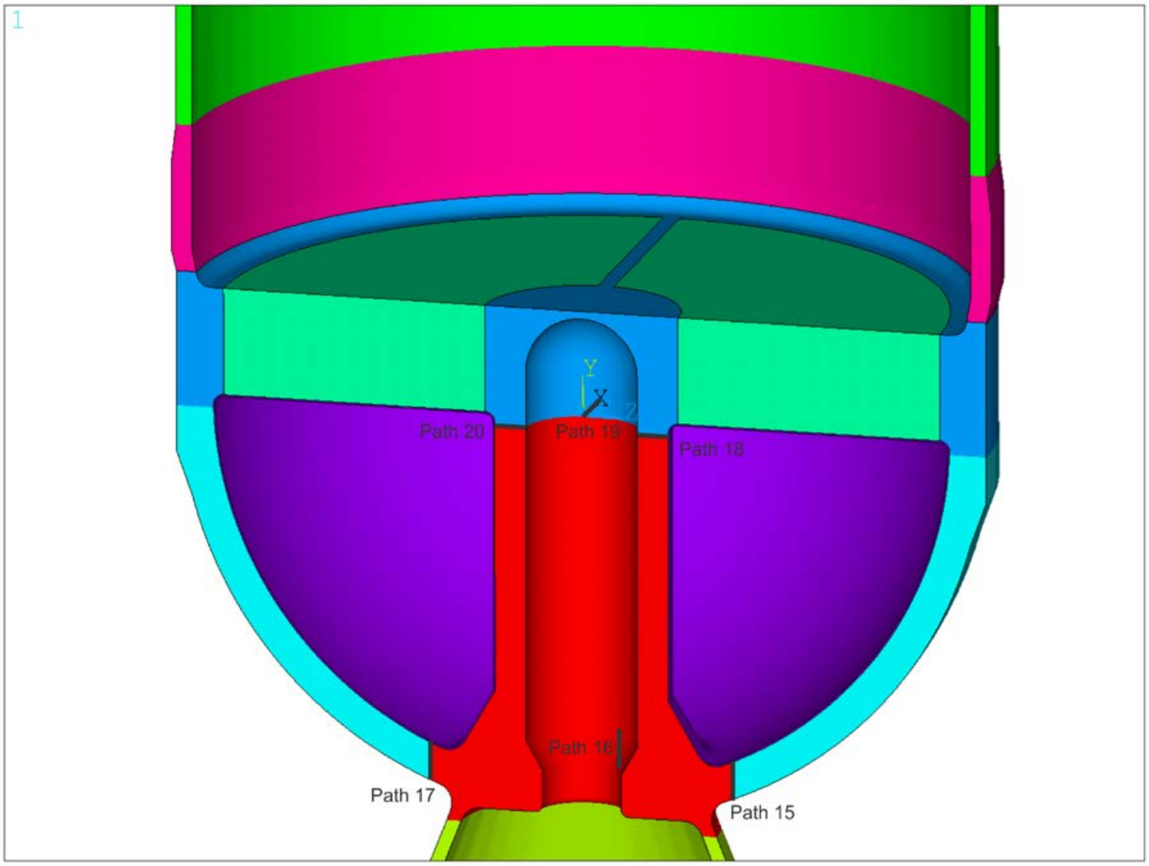


Figure 6. Path Locations
(similar to Figure 7-24 of EPRI Report 3002015906)

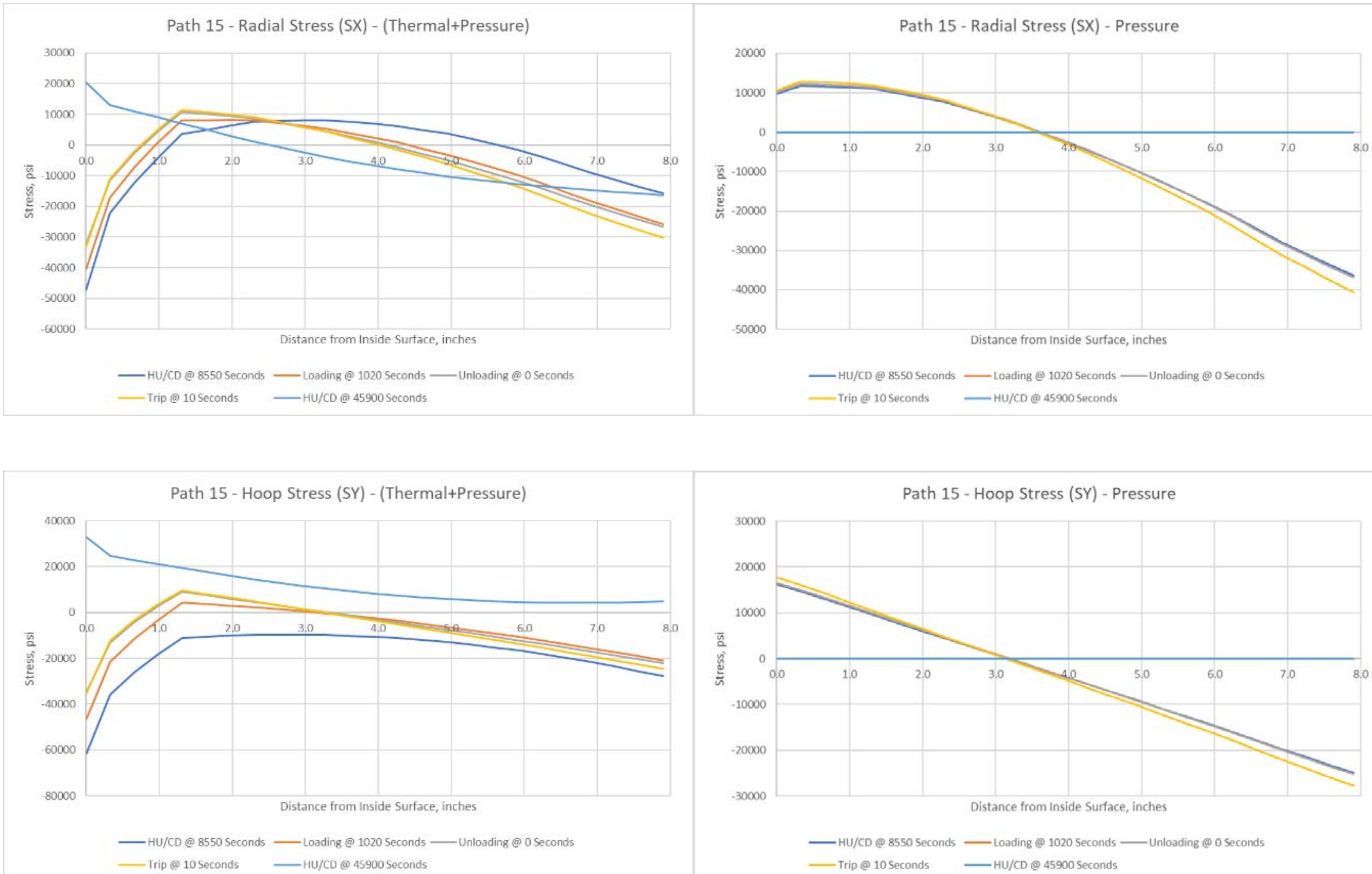


Figure 7. Through-Wall Stress Distribution for Path P15
(similar to Figures 7-25 through 7-28 of EPRI Report 3002015906)

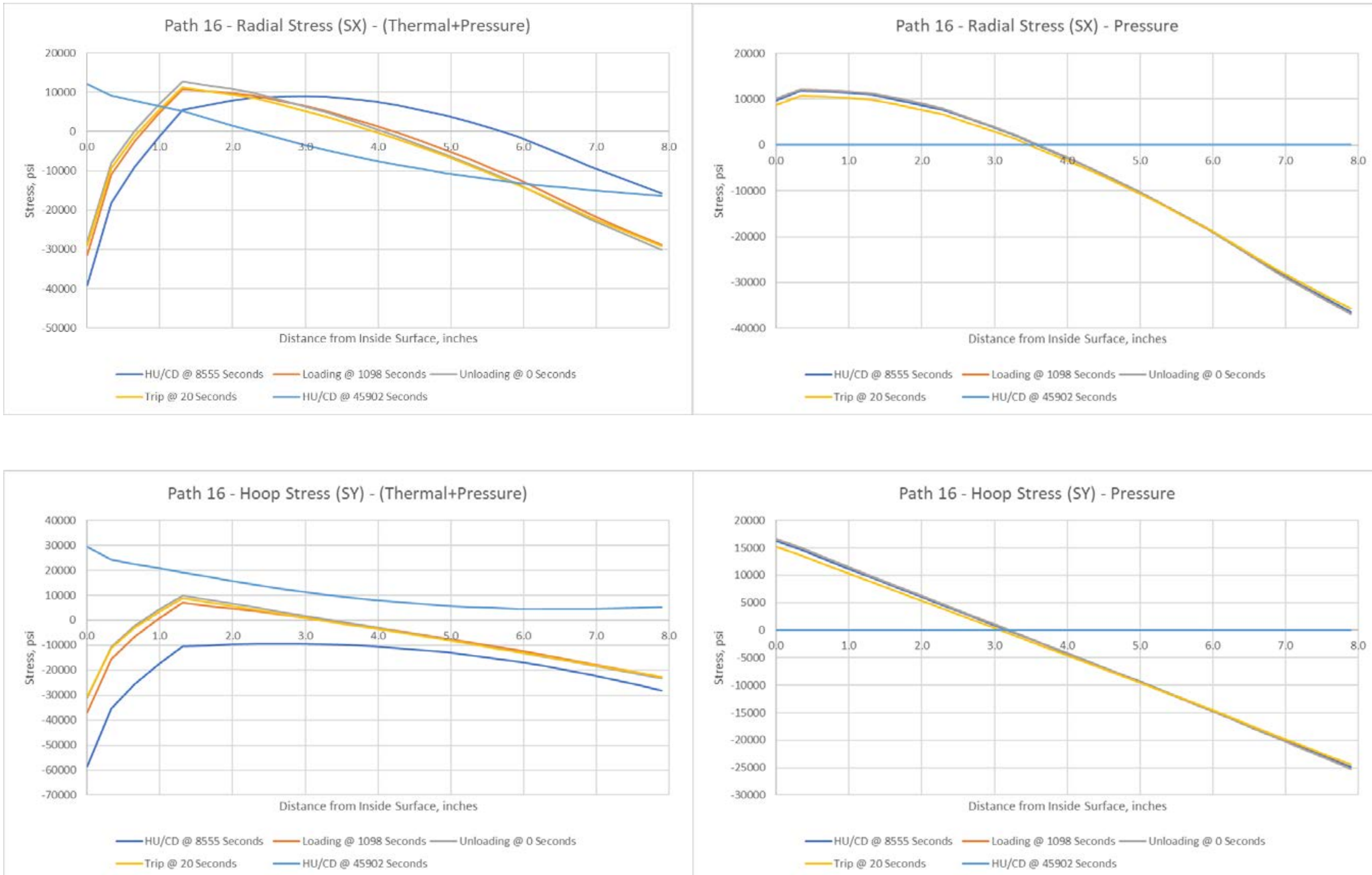


Figure 8. Through-Wall Stress Distribution for Path P16
(similar to Figures 7-25 through 7-28 of EPRI Report 3002015906)

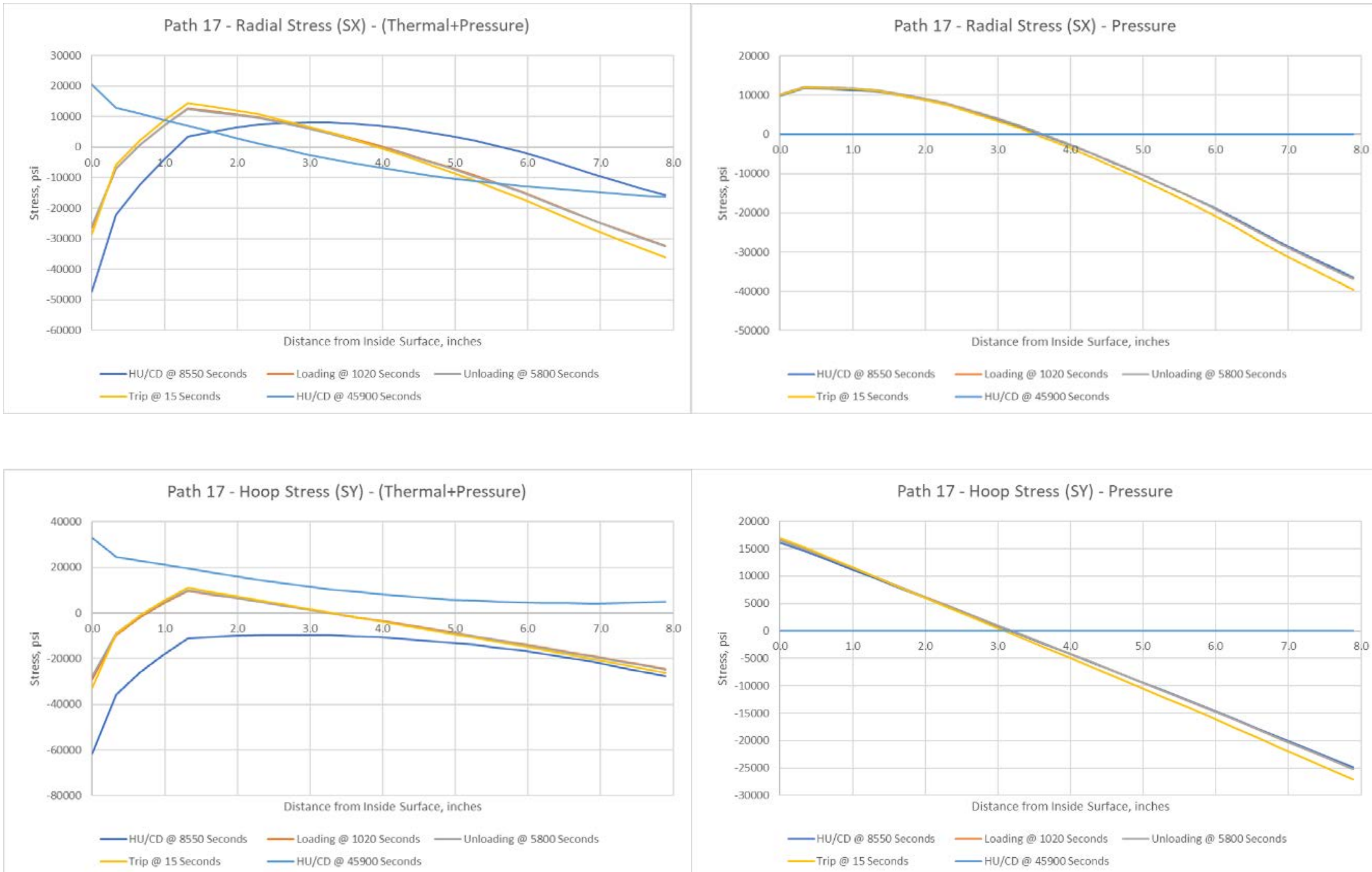


Figure 9. Through-Wall Stress Distribution for Path P17
(similar to Figures 7-25 through 7-28 of EPRI Report 3002015906)

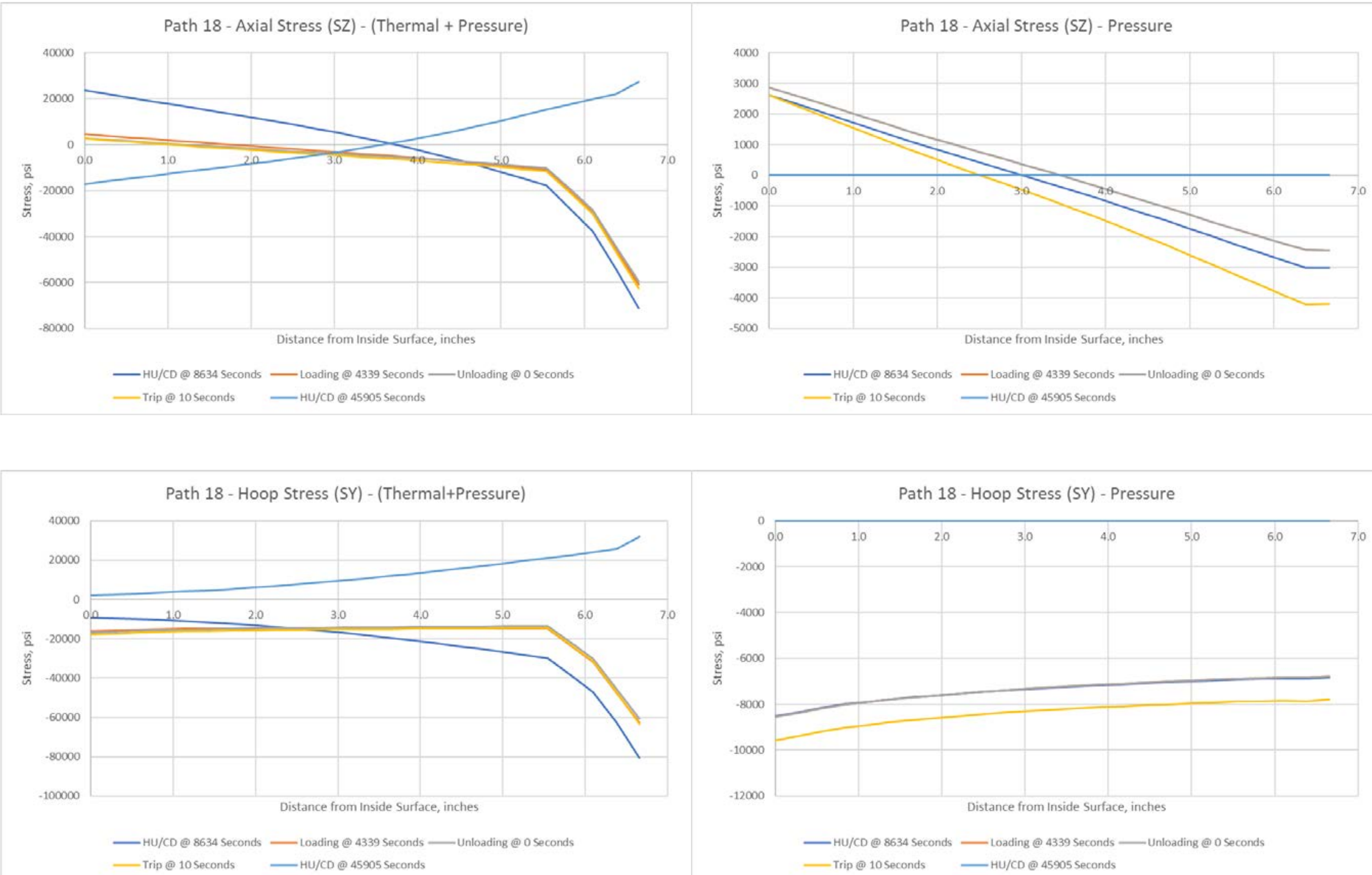


Figure 10. Through-Wall Stress Distribution for Path P18
(similar to Figures 7-25 through 7-28 of EPRI Report 3002015906)

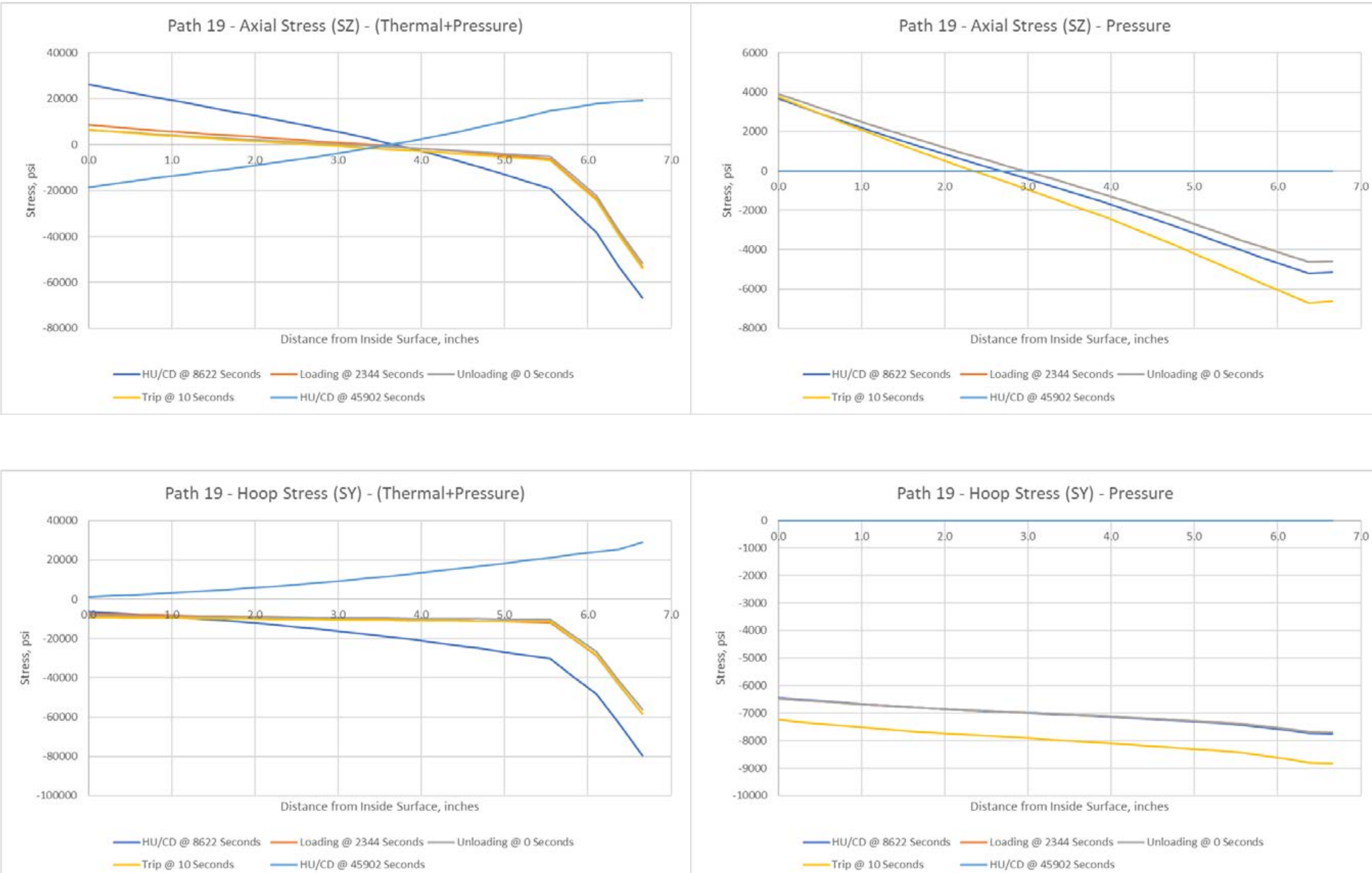


Figure 11. Through-Wall Stress Distribution for Path P19
(similar to Figures 7-25 through 7-28 of EPRI Report 3002015906)

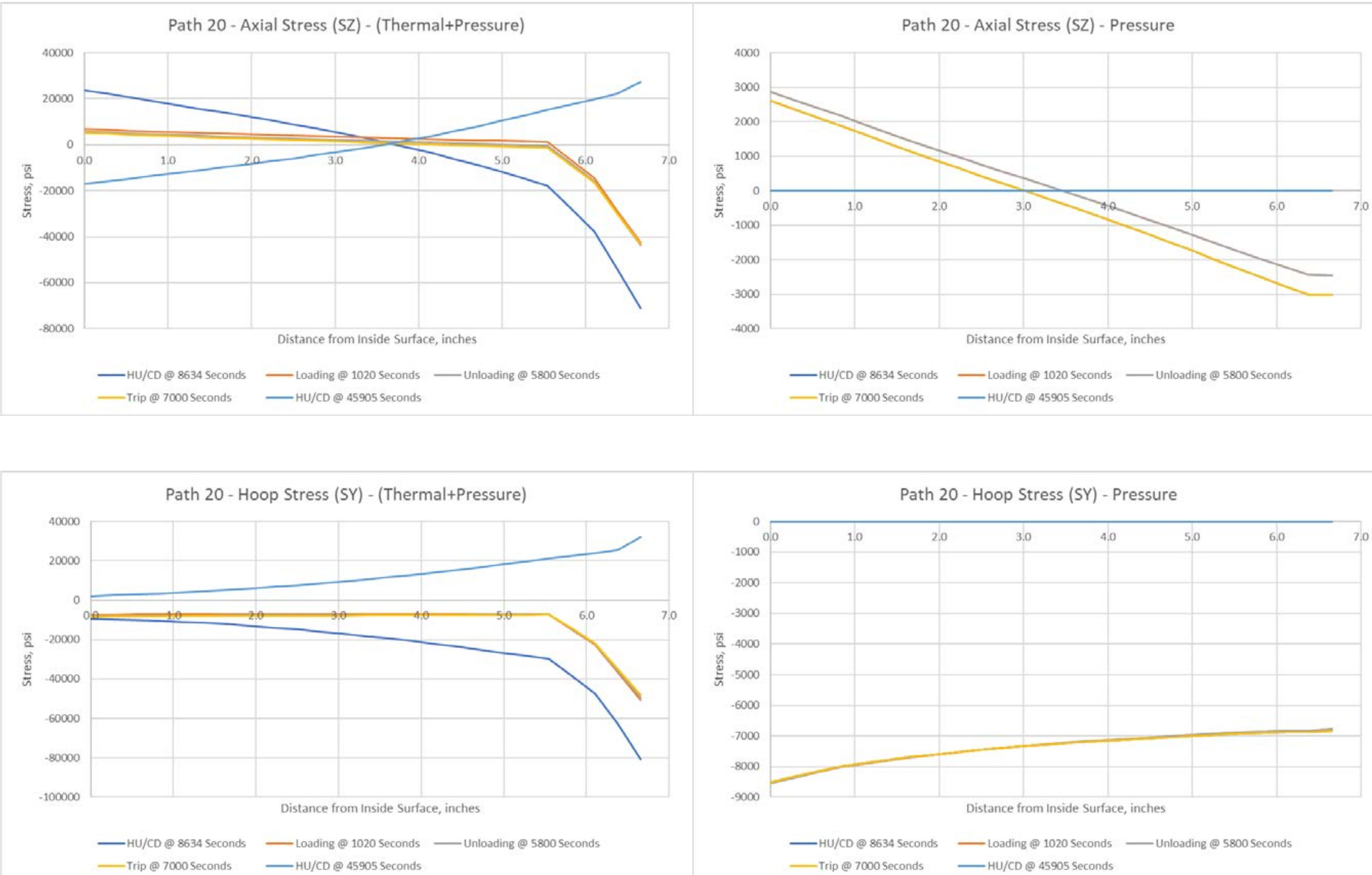


Figure 12. Through-Wall Stress Distribution for Path P20
(similar to Figures 7-25 through 7-28 of EPRI Report 3002015906)

ATTACHMENT 3

<u>SUPPORTING MILLSTONE POWER STATION UNIT 2 STEAM GENERATOR DESIGN DETERMINISTIC AND PROBABILISTIC FRACTURE MECHANICS EVALUATIONS FOR RAI-2(b)</u>

MILLSTONE POWER STATION UNIT 2
DOMINION ENERGY NUCLEAR CONNECTICUT, INC

1.0 INTRODUCTION

(similar to the content in Section 8.1 of EPRI Report 3002015906)

This section describes the deterministic fracture mechanics (DFM) and probabilistic fracture mechanics (PFM) analyses for the Millstone Power Station Unit 2 (MPS2) primary head-to-stay cylinder and stay cylinder-to-tubesheet welds using the results of the MPS2 SG design stress analysis in Attachment 2 and other relevant design inputs.

The objective of the DFM evaluation is to determine how long a postulated flaw will take to reach the allowable flaw size. The objective of the PFM evaluation is to determine the probabilities of failure for the two SG weld locations. The collective objective of the PFM and DFM analyses is to assess various inspection frequencies for the SG welds, including the MPS2-specific inspection scenario included in the Dominion Energy Nuclear Connecticut, Inc. (DENC) Request for Alternative [1]. The evaluations are performed consistent with Section 8 of EPRI Report 3002015906 [2].

2.0 DETERMINISTIC FRACTURE MECHANICS EVALUATION

(similar to the content in Section 8.2 of EPRI Report 3002015906)

The objective of the DFM evaluation is to determine the time it takes for a postulated flaw to grow to the ASME Code allowable flaw size. From the results of the DFM evaluation, critical path(s) are then selected for use in the PFM evaluation. The DFM evaluation is performed using average parameters to supplement the PFM evaluation. The DFM evaluation is performed consistent with Section 8.2 of EPRI Report 3002010596.

2.1 Technical Approach

(similar to the content in Section 8.2.1 of EPRI Report 3002015906)

The technical approach used in the DFM evaluation is to postulate an initial flaw size equivalent to the relevant ASME Code, Section XI acceptance standard. The ASME Code, Section XI fatigue crack growth (FCG) law, with the through-wall stress distributions from the stress analysis in Attachment 2 and appropriate fracture mechanics models, is then used to determine the length of time for the postulated initial flaw to grow to a depth of 80% of the wall thickness (assumed to equate to leakage in this evaluation) or the depth at which the allowable toughness (K_{IC} reduced by a structural factor of 2) is reached, whichever is less.

2.2 Design Inputs

(similar to the content in Section 8.2.2 of EPRI Report 3002015906)

The MPS2 specific Replacement SG design inputs used in the DFM evaluation are summarized in Table 1 of this attachment and are discussed in the following sections.

2.2.1 Geometry

(similar to the content in Section 8.2.2.1 of EPRI Report 3002015906)

The MPS2-specific geometries of the components considered in the evaluation are shown in Figures A2 and A3 of Reference [1]. The specific welds of interest in this evaluation are the stay cylinder-to-lower head weld and the stay cylinder-to-tubesheet welds. Figure 1 of this attachment is a duplicate of Figure 6 from the stress analyses in Attachment 2 that shows the stress paths for the two welds. Paths 15, 16, and 17

represent the stay cylinder-to-lower head weld, and Paths 18, 19 and 20 represent the stay cylinder-to-tubesheet weld.

2.2.2 Initial Crack Size and Shape

(similar to the content in Section 8.2.2.2 of EPRI Report 3002015906)

For the applicable SG welds, an initial crack size of 5.2% of the wall thickness (which corresponds to the most conservative ASME Code Section XI flaw acceptance standard for these components from Tables IWB-3510-1 and IWC-3510-1 of ASME Code Section XI [3]) was used in the DFM evaluation. This initial crack depth is the maximum depth from the two Section XI tables with an associated crack aspect ratio (half crack length-to-crack depth, c/a) of 1.0. This crack shape results in the most conservative initial stress intensity factor (K) at the deepest point of the crack. The aspect ratio was then subsequently allowed to vary during the crack growth process.

2.2.3 Applied Stresses

(similar to the content in Section 8.2.2.3 of EPRI Report 3002015906)

2.2.3.1 Operating Transient Stresses

(similar to the content in Section 8.2.2.3.1 of EPRI Report 3002015906)

The applied stresses consist of through-wall stresses due to pressure and the thermal transients described in the stress analyses in Attachment 2. Typical through-wall stress distributions for stress paths used in the evaluation from the stress analyses in Attachment 2 are reproduced in Figure 2 through Figure 7 of this attachment. Figure 1 of this attachment shows the stress paths where these stresses were extracted.

2.2.3.2 Weld Residual Stresses

(similar to the content in Section 8.2.2.3.2 of EPRI Report 3002015906)

Pressure vessel welds typically receive post-weld heat treatment (PWHT) to reduce the effects of weld residual stresses. In this evaluation, weld residual stresses remaining after PWHT were characterized in the form of a cosine distribution with a peak stress of 8 ksi [4]. This is shown in Figure 8, which is the same distribution used in Reference [2]. A clad tensile residual stress of 30 ksi was also assumed in the evaluation consistent with Section 8.2.2.4 of EPRI Report 3002015905 [5].

2.2.4 Fracture Mechanics Models

(similar to the content in Section 8.2.2.4 of EPRI Report 3002015906)

In this evaluation, pre-existing flaws were conservatively assumed to be surface flaws. Two different fracture mechanics models were used for axial and circumferential flaws. For the axial flaw, the stress intensity factor (K) solution for an internal, semi-elliptical crack from API-579/ASME-FFS-1 [6] was used. This model is shown in Figure 9 of this attachment. The aspect ratio (c/a) was allowed to vary during crack growth.

Similarly, for the circumferential flaw, the K solution for an internal, semi-elliptical crack from API-579/ASME-FFS-1 [6] was used. This model is shown in Figure 10 of this attachment. The aspect ratio (c/a) was allowed to vary during crack growth.

The flaw models are the same as those used in Section 8.2.2.4 of EPRI Report 3002015906 [2]. A large R/t ratio equal to 80 was used to approximate the stay cylinder-to-lower head weld since it approaches a flat plate rather than a cylinder.

These fracture mechanics models were incorporated into an SI-developed software program, TIFFANY [7], that determines the K distribution due to through-wall stress profiles for both circumferential and axial cracks. The outputs of TIFFANY are the maximum and minimum K distributions, as well as the ΔK distribution, for each transient.

2.2.5 Fracture Toughness

(similar to the content in Section 8.2.2.5 of EPRI Report 3002015906)

Since the materials under consideration are ferritic steels, the fracture toughness curve provided in ASME Code, Section XI, Appendix A (Figure A-4200-1) [3] was used for this evaluation. From the transients considered in the stress analyses in Attachment 2, the minimum temperature experienced across the applicable transients corresponds to the Heatup/Cooldown transient. The minimum fluid temperature across the applicable SG welds during the Heatup/Cooldown transient in the stress analyses in Attachment 2 is 70°F. However, the pressure at this temperature is zero. At full power where the maximum K occurs, the fluid temperature is 550°F. This full power temperature was therefore be used to determine the allowable fracture toughness. The maximum RT_{NDT} for the plates and nozzles of MPS2 lower head plates is 0°F [8]. The minimum possible temperature to use when entering ASME Code, Section XI, Figure A-4200-1 is therefore 550°F (550°F - 0°F). This temperature is greater than the temperature at the end of the K_{IC} curve shown in Figure A-4200-1, and as a result, an upper shelf fracture toughness of 200 ksi√in was used. Figure 11 of this attachment shows the fracture toughness of vessel steels as a function of temperature [9, 10] and indicates that the ASME Code, Section XI fracture toughness is a reasonable lower bound.

2.2.6 Fatigue Crack Growth Law

(similar to the content in Section 8.2.2.6 of EPRI Report 3002015906)

The FCG law for ferritic steels, as defined in ASME Code, Section XI, Appendix A, Paragraph A-4300 [3], was used in the evaluation. The FCG calculations were performed using the Structural Integrity Associates fracture mechanics program, pc-CRACK [12].

2.2.7 Summary of Design Inputs

(similar to the content in Section 8.2.2.7 of EPRI Report 3002015906)

The design inputs covered in Sections 2.2.1 through 2.2.6 of this attachment are summarized in Table 1 of this attachment and were used to perform the DFM evaluation.

2.3 Results of Deterministic Fracture Mechanics Evaluation

(similar to the content in Section 8.2.3 of EPRI Report 3002015906)

The results of the DFM evaluation are summarized in Table 2 of this attachment. This table shows that very long periods are required for hypothetical postulated flaws to leak (reach 80% through-wall), which indicates that the evaluated SG welds are very flaw

tolerant. Because the DFM evaluation considered hypothetical postulated flaws, structural factors of 2.0 on primary loads and 1.0 on secondary loads, consistent with ASME Code, Section XI, Appendix G, were applied. Also, because the most dominant load is pressure, which results in a primary stress, the structural factor of 2.0 was conservatively applied to the fracture toughness of 200 ksi√in. This results in an allowable fracture toughness of 100 ksi√in. Table 2 of this attachment also summarizes that the maximum applied K values for the applicable SG welds are below this allowable fracture toughness after 80 years.

One critical stress path was selected for further PFM evaluation based on the minimum number of years to leak from the DFM evaluation. These selected path (SGPSCH-P15A) is identified with red bold text in Table 2 of this attachment.

3.0 PFM EVALUATION

(similar to the content in Section 8.3 of EPRI Report 3002015906)

3.1 Technical Approach

(similar to the content in Section 8.3.1 of EPRI Report 3002015906)

The PFM evaluation was performed consistent with Section 8.3 of EPRI Report 3002015906 [3]. Monte Carlo probabilistic analysis techniques were used in the PFM analyses to determine the effect of various inspection scenarios on the probability of failure for the SG welds evaluated. The PFM overall technical approach is illustrated in Figure 8-5 of EPRI Report 3002015906 [2].

The acceptance criterion used for the PFM evaluation is that failure frequencies must be less than the U.S. NRC safety goal of 10^{-6} failures per year consistent with Section 8.3.2.9 of EPRI Report 3002015906. Failure is defined to occur either by rupture or by leakage. Rupture was assumed to occur in the probabilistic simulation when, for a given iteration, the applied K exceeded the material fracture toughness (K_{IC}), that is, $K > K_{IC}$. Leakage was assumed to occur when, for a given iteration, the crack depth exceeded 80% of the wall thickness ($a > 0.80t$). Due to the limitation in the fracture mechanics models used in this evaluation, it was conservatively assumed that leakage occurs when the crack depth reaches 80% of the component wall thickness.

Sensitivity studies performed on the various parameters listed in Section 8.3.4.3 of EPRI Report 3002015906 indicated that the most critical parameters that influence the probabilities of failure are the fracture toughness and stress. Therefore, sensitivity studies were limited to these two parameters for this PFM evaluation.

3.2 Design Inputs

(similar to the content in Section 8.3.2 of EPRI Report 3002015906)

The design inputs used for the PFM evaluation are shown in Table 3 of this attachment for the base case and Table 4 of this attachment for MPS2 plant-specific inspection history. The inputs are consistent with those used in Section 8.3.2 of EPRI Report 3002015906 [2]. The only difference between Table 3 and Table 4 of this attachment is that the MPS2 plant-specific inspection history from DENC's Request for Alternative [1] was used, as shown in Table 4 of this attachment. MPS2 has performed pre-service inspection (PSI) prior to the replacement SGs entering into service (i.e., Year 0)

followed by two successive 10-year in-service inspections (ISI) and one 30-year inspection. This plant-specific history was considered in this PFM evaluation and is denoted as (PSI+10+20+50).

3.3 PFM Evaluations

(similar to the content in Section 8.3.4 of EPRI Report 3002015906)

The PROMISE, Version 2.0 [11] software was used to perform the PFM evaluations. The evaluations were performed for the base case identified in Section 8.3.4.1 of EPRI Report 3002015906 (pre-service inspection, PSI, only) followed by evaluation of the MPS2 plant-specific inspection scenario consisting of PSI followed by two 10-year inspections and one 30-year inspection (PSI+10+20+50) described in DENC's Request for Alternative [1] applicable to the replacement SG welds.

In addition, PFM sensitivity studies were performed for two cases: one case where the fracture toughness was reduced to 80 ksi√in with a standard deviation of 5 ksi√inch (similar to the study performed in Section 8.3.4.3.1 of EPRI Report 3002015906), and another case where the stresses were increased by a factor of 1.25 (similar to the study performed in Section 8.3.4.3.2 of EPRI Report 3002015906). Both of these cases evaluated the MPS2 inspection scenario (PSI+10+20+50) for the critical Case ID (SGPSCH-P15A) determined in Section 2.3 of this attachment.

3.4 Results of PFM Evaluations

(similar to the content in Sections 8.3.4.1, 8.3.4.3.1 and 8.3.4.3.2 of EPRI Report 3002015906)

The results of the PFM evaluation for the MPS2 SG design are presented in Table 5 of this attachment for the base case. This table shows that with PSI inspection alone, the probabilities of rupture and leakage for the corresponding SG welds are below the acceptance criteria of $1.0x10^{-6}$ after 80 years of plant operation. These results are consistent with the results presented in Table 8-9 of EPRI Report 3002015906.

From the DFM results presented in Table 2 of this attachment, the critical Case ID is SGPSCH-P15A. This Case ID was used to perform the PFM evaluation for the MPS2 plant-specific inspection scenario. The results of the PFM evaluations are presented in Table 6 of this attachment. The two sensitivity studies using the MPS2 plant-specific inspection scenario are also presented in Table 6 of this attachment. The results indicate that the probabilities of rupture and leakage are below the acceptance criteria of $1.0x10^{-6}$ after 80 years of plant operation for these cases. These results are consistent with the results presented in Tables 8-9 and 8-10 (base case and inspection scenarios), Tables 8-13 through 8-16 (for toughness), and Tables 8-17 through 8-20 (for stresses) in EPRI Report 3002015906.

4.0 CONCLUSION

(similar to the content in Section 8.4 of EPRI Report 3002015906)

The following conclusions are made from the MPS2-specific PFM and DFM evaluations:

- The DFM evaluation demonstrated that a very long operating period (approximately 100 years) is necessary for a postulated initial flaw (with a depth equal to the ASME Code, Section XI acceptance standards) to propagate through 80% of the wall thickness (assumed as leakage in this study). After 80

years, the maximum K obtained from the analysis remains below the ASME Code, Section XI allowable fracture toughness, including the ASME Code, Section XI, Appendix G structural factor of 2.0 on primary stress, which indicates that the applicable ASME Code, Section XI structural margins have been satisfied. This indicates that the in-scope SG primary-side welds are very flaw tolerant.

- From the PFM evaluations, it was demonstrated that once PSI has been performed (base case), the failure probabilities (in terms of both rupture and leakage) are significantly below the acceptance criterion of 10^{-6} failures per year after 80 years of operation.
- For the MPS2 plant-specific inspection history (PSI+10+20+50), the PFM evaluation showed that the probabilities of rupture and leakage are also significantly below the acceptance criteria.
- Two separate sensitivity studies performed by reducing the fracture toughness from 200 ksi√in to 80 ksi√in (with a standard deviation of 5 ksi√in), and increasing the stresses by a factor of 1.25, resulted in probabilities of rupture and leakage below the acceptance criteria.

REFERENCES

1. Letter from M. D. Sartain (Dominion) to U.S. Nuclear Regulatory Commission, “Dominion Energy Nuclear Connecticut Inc., Millstone Power Station Unit 2, Alternative Request RR-05-06 – Inspection Interval Extension for Steam Generator Pressure-Retaining Welds and Full-Penetration Welded Nozzles,” dated July 15, 2020.

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

3. ASME Boiler and Pressure Vessel Code, Section XI, 2017 Edition.

4. Simonen, F. A. and Johnson, K. I., “Effects of Residual Stresses and Underclad Flaws on the Reliability of Reactor Pressure Vessels,” PVP-Vol. 251, *Reliability and Risk in Pressure Vessels and Piping*, ASME PVP Conference, 1993.

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

6. API Standard 579-1/ASME FFS-1, Fitness-for-Service, Second Edition, June 2016.

7. SI-TIFFANY 3.1, Structural Integrity Associates, September 2018.

8. Millstone Power Station Unit 2 FSAR, Table 4.6-3.

9. M. Kirk and M. Erickson, “Assessment of the Fracture Toughness of Ferritic Steel Fracture Toughness on or near the Lowe Shelf,” Paper No. PVP2015-45850, *Proceedings of the ASME Pressure Vessels and Piping Conference*, July 19–23, 2015, Boston, MA, USA.

10. *Application of Master Curve Fracture Toughness Methodology for Ferritic Steels*. EPRI, Palo Alto, CA: 1999. TR-108390, Revision 1.

11. Structural Integrity Associates Report DEV1806.402, *PROMISE 2.0 Theory and User’s Manual*, Revision 1.

12. pc-CRACK 5.0, Version Control No. 5.0.0.0, Structural Integrity Associates, January 2021.

Table 1: Summary of DFM Design Inputs
(similar to Table 8-2 of EPRI Report 3002015906)

Input	Value
Geometry	Based on Figures A2 and A3 of Reference [1]
Initial Crack Size	5.2% of the thickness, c/a = 1
Fracture toughness	200 ksi√in
Fatigue crack growth law	ASME Code, Section XI Appendix A, Paragraph A-4300
Operating Transient Stresses	Stress analyses in Attachment 2
Residual stresses	Cosine curve with 8 ksi peak (see Figure 8 of this attachment)
Clad residual stress	30 ksi tensile (Section 8.2.2.4 of Reference [5])

Table 2: Results of DFM Evaluation for the MPS2 SG Design
(similar to Table 8-3 of EPRI Report 3002015906)

Item No.	Component Description	Case Identification (Note 1)	Years to Leak	Max. K at 80 Years (ksi√in)
B2.31	SG (primary side), stay cylinder-to-hemispherical head weld	**SGPSCH-P15A**	**694**	**68.9**
		SGPSCH-P15C	2233	51.1
		SGPSCH-P16A	1207	61.9
		SGPSCH-P16C	7760	43.6
		SGPSCH-P17A	1448	64.6
		SGPSCH-P17C	6758	50.4
B2.31	SG (primary side), stay cylinder-to-tubesheet weld	SGPSCT-P18A	7186	11.5
		SGPSCT-P18C	1843	27.5
		SGPSCT-P19A	17740	10.7
		SGPSCT-P19C	3517	29.6
		SGPSCT-P20A	9537	11.5
		SGPSCT-P20C	3541	27.5

Note 1: The Case Identification terminology is as follows: SG for Steam generator; PSCH for primary stay cylinder-to-hemispherical head, PSCT for primary stay cylinder-to-tubesheet; P15 through P20 represent the crack paths (see Figure 6 of Attachment 2; A for axial part-through-wall crack; and C for circumferential part-through-wall crack. The limiting case is displayed in **red bold** text.

Table 3: Inputs for PFM Base Case
(similar to Table 8-8 of EPRI Report 3002015906)

No. of Realizations	Epistemic = 1, Aleatory = 10 million
No. of cracks per weld	1, constant
Crack depth distribution	PVRUF
Crack length distribution	NUREG/CR-6817-R1
Fracture toughness (ksi√in)	Normal (200,5)
Inspection coverage	100%
PSI	Yes
ISI	None
POD Curve	BWRVIP-108 (same as Figure 8-6 of EPRI Report 3002015906 [2])
Fatigue crack growth law and threshold	A-4300, log-normal, Second Parameter = 0.467
Uncertainties on transients	None
Weld residual stresses (ksi)	Cosine Curve (8, 8), constant (not random) (Figure 8 of this attachment)
Clad residual stress	30 ksi tensile (Section 8.2.2.4 of Reference [5])

Table 4: Inputs for MPS2 SG Design Inspection Scenario
(similar to Table 8-8 of EPRI Report 3002015906)

No. of Realizations	Epistemic = 1, Aleatory = 10 million
No. of cracks per weld	1, constant
Crack depth distribution	PVRUF
Crack length distribution	NUREG/CR-6817-R1
Fracture toughness (ksi√in)	Normal (200,5)
Inspection coverage	100%
PSI	Yes
ISI	10, 20, and 50 years (MPS2-specific)
POD Curve	BWRVIP-108 (same as Figure 8-6 of EPRI Report 3002010596 [2])
Fatigue crack growth law and threshold	A-4300, log-normal, Second Parameter = 0.467
Uncertainties on transients	None
Weld residual stresses (ksi)	Cosine Curve (8, 8), constant (not random) (Figure 8 of this attachment)
Clad Residual Stress	30 ksi tensile (Section 8.2.2.4 of Reference [5])

Table 5: Results of PFM Evaluation for the MPS2 SG Design for the Base Case (PSI Only)

(similar to Table 8-9 of EPRI Report 3002015906)

Case Identification	Probability of Leakage at 80 Years (per Year)	Probability of Rupture at 80 Years (per Year)
SGPSCH-P15A	1.25E-09	1.25E-09
SGPSCH-P15C	1.25E-09	1.25E-09
SGPSCH-P16A	1.25E-09	1.25E-09
SGPSCH-P16C	1.25E-09	1.25E-09
SGPSCH-P17A	1.25E-09	1.25E-09
SGPSCH-P17C	1.25E-09	1.25E-09
SGPSCT-P18A	1.25E-09	1.25E-09
SGPSCT-P18C	1.25E-09	1.25E-09
SGPSCT-P19A	1.25E-09	1.25E-09
SGPSCT-P19C	1.25E-09	1.25E-09
SGPSCT-P20A	1.25E-09	1.25E-09
SGPSCT-P20C	1.25E-09	1.25E-09

Table 6: PFM Results Comparing the Base Case to the MPS2 Plant-Specific Inspection Scenarios for Limiting Case ID SGPSCH-P15A

(comparable to Tables 8-9 and 8-10 (base case and inspection scenarios), Tables 8-13 through 8-16 (for toughness), and Tables 8-17 through 8-20 (for stresses) of EPRI Report 3002015906)

Case	Probability of Leakage at 80 Years (per Year)	Probability of Rupture at 80 Years (per Year)
Base Case	1.25E-09	1.25E-09
MPS2 Plant-Specific Inspection Case PSI + 10 + 20 + 50 (K_{IC} = 200 ksi√in)	1.25E-09	1.25E-09
Sensitivity Study #1: MPS2 Plant-Specific Inspection Case PSI + 10 + 20 + 50 (K_{IC} = 80 ksi√inch, STD = 5 ksi√inch)	1.25E-09	5.08E-07
Sensitivity Study #2: MPS2 Plant-Specific Inspection Case PSI + 10 + 20 + 50 (Stress x 1.25)	1.25E-09	1.25E-09

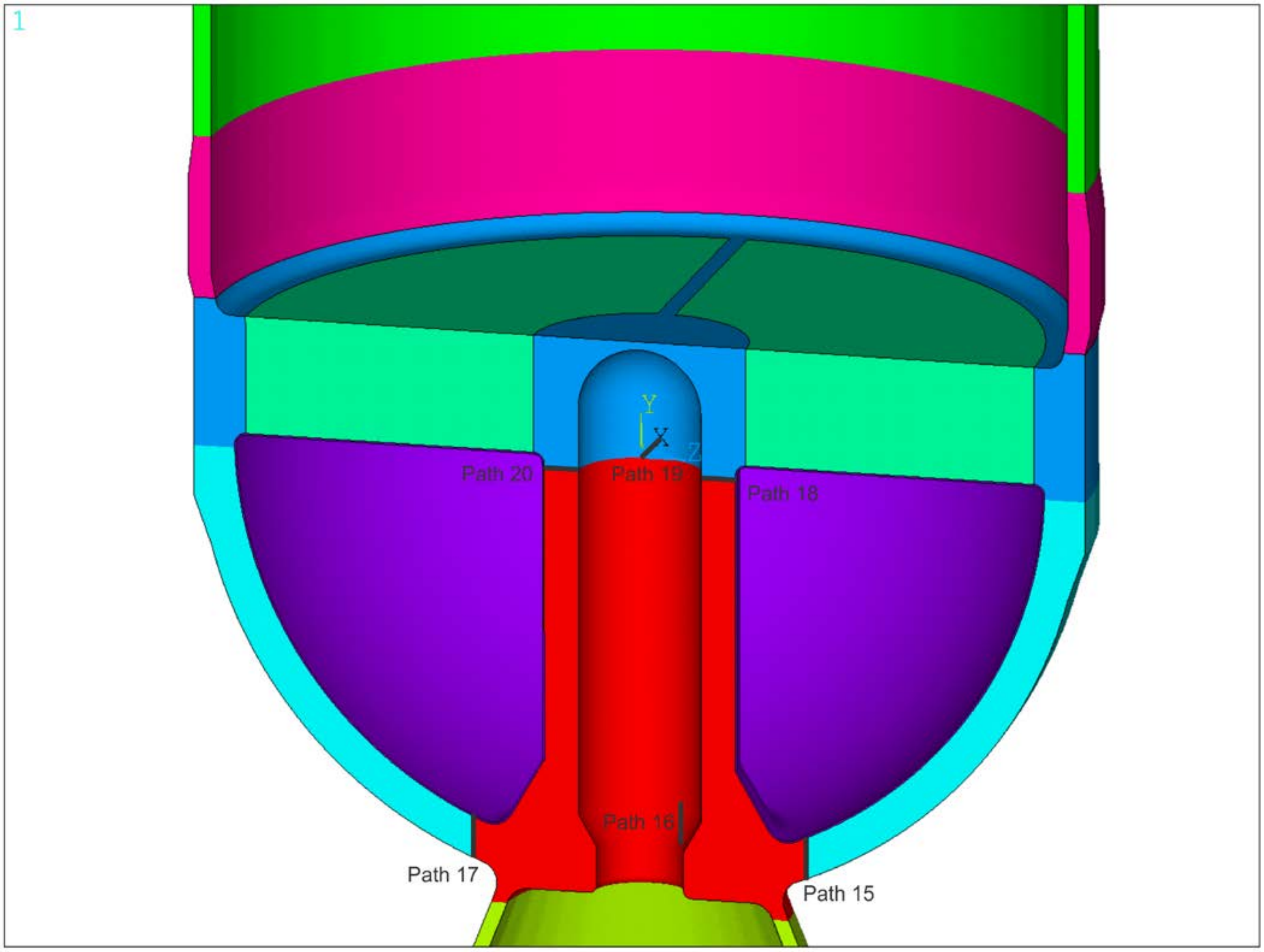
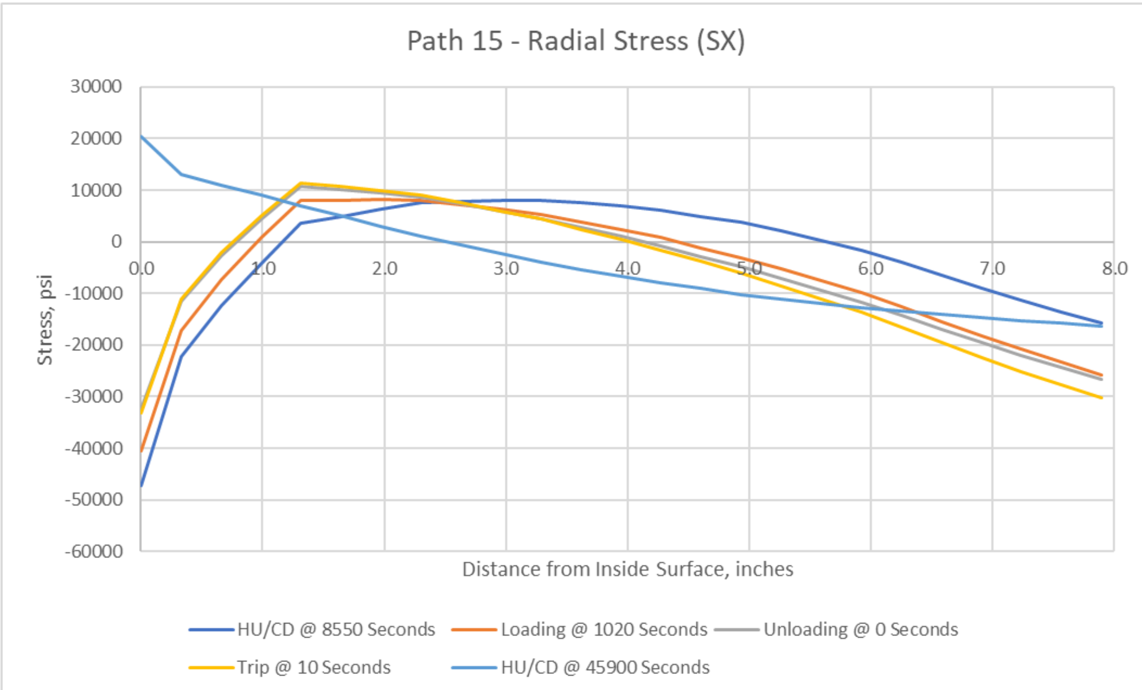


Figure 1. Path Locations
(Same as Figure 6 in Attachment 2, similar to Figure 7-24 of EPRI Report 3002015906)



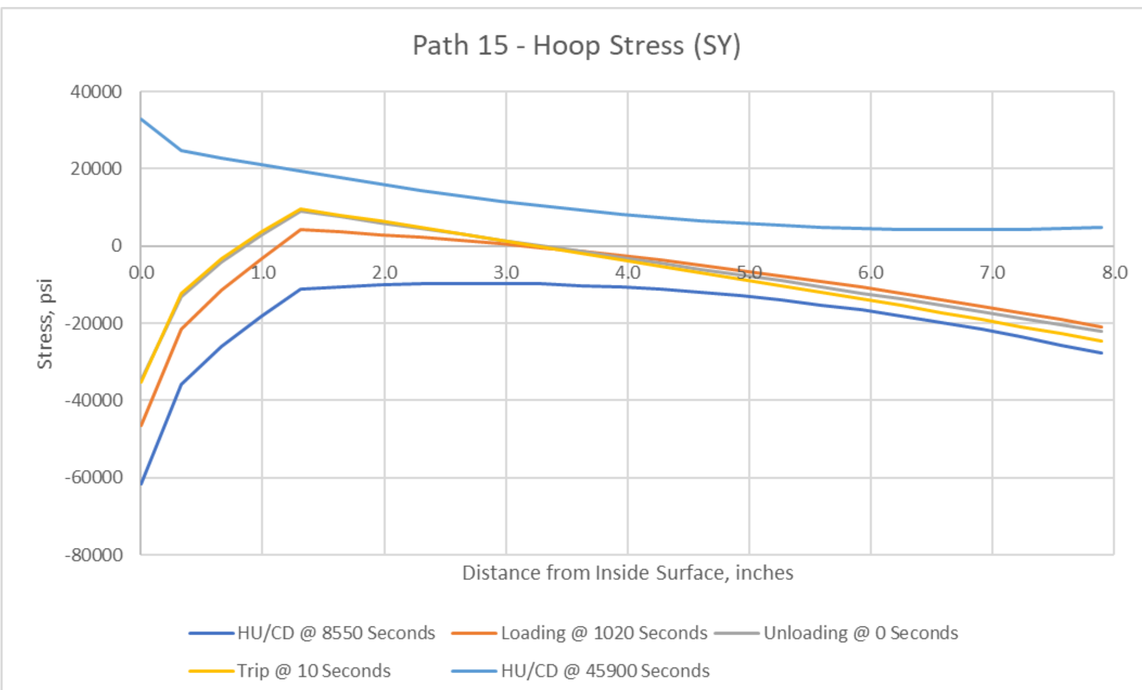
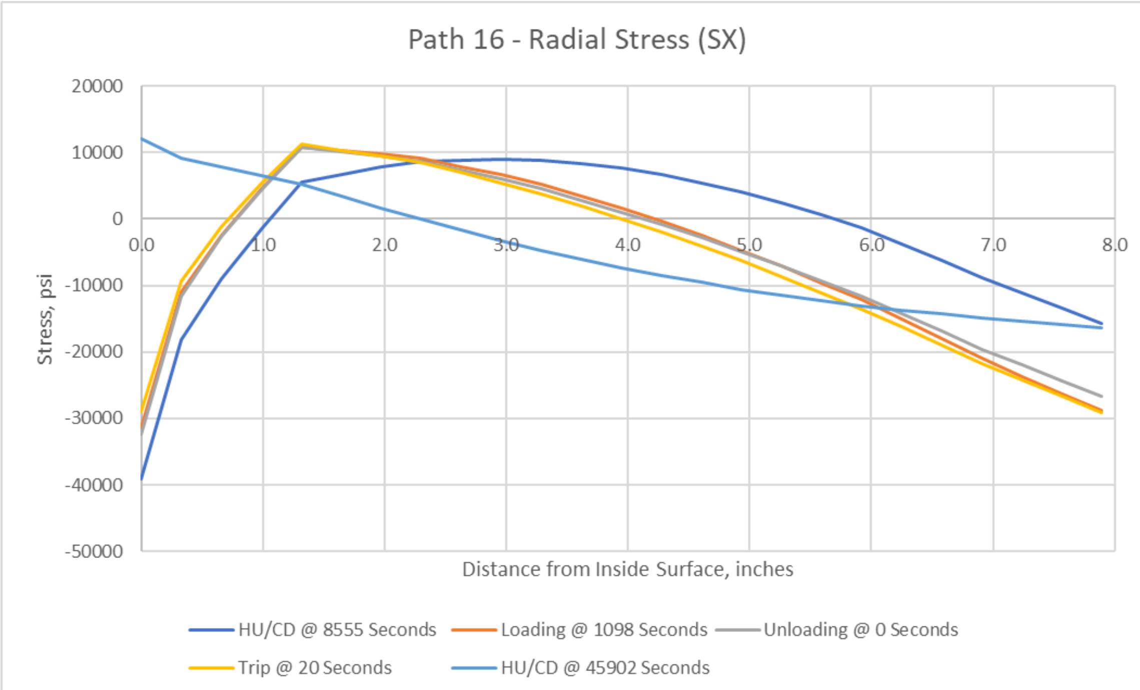


Figure 2. Through-Wall Stress Distribution for Path P15
(Same as the (Thermal+Pressure) portion of Figure 8 in Attachment 2)
(similar to Figures 7-25 through 7-28 of EPRI Report 3002015906)



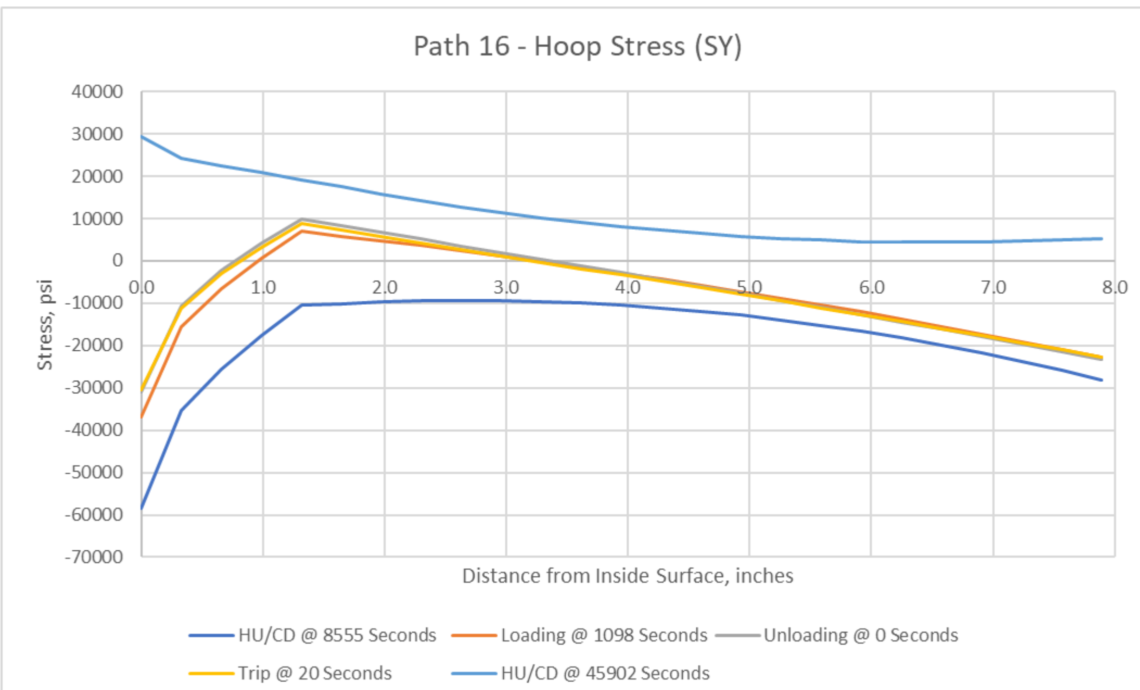
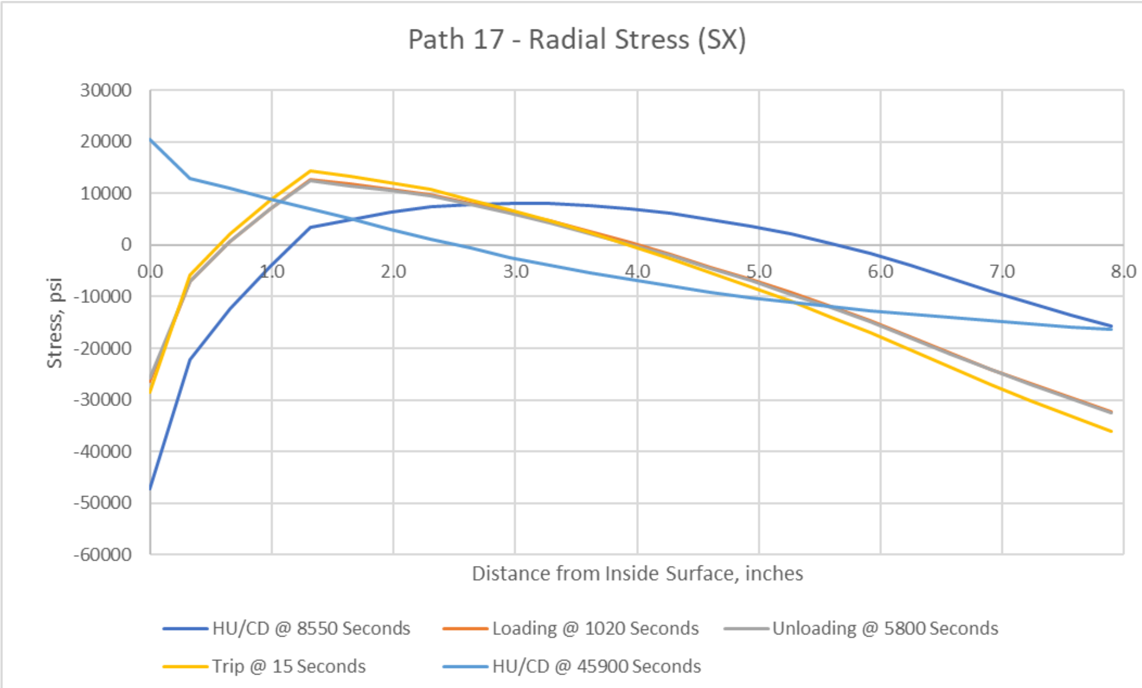


Figure 3. Through-Wall Stress Distribution for Path P16
(Same as the (Thermal+Pressure) portion of Figure 9 of Attachment 2)
(similar to Figures 7-25 through 7-28 of EPRI Report 3002015906)



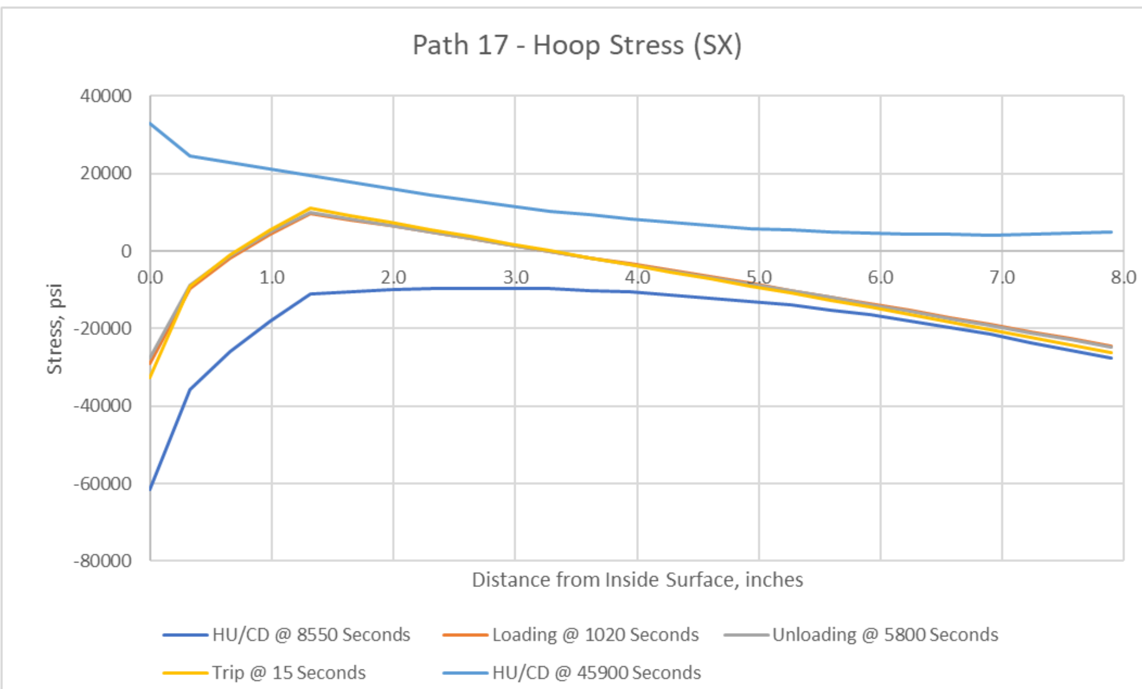
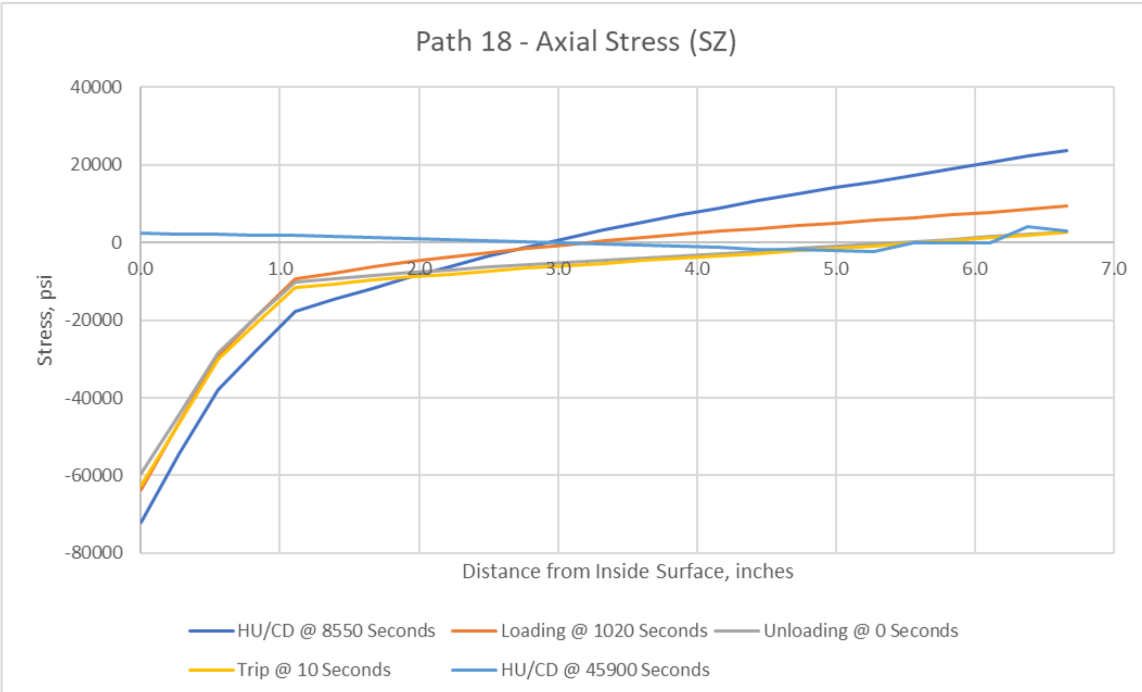


Figure 4. Through-Wall Stress Distribution for Path P17
(Same as the (Thermal+Pressure) portion of Figure 10 of Attachment 2)
(similar to Figures 7-25 through 7-28 of EPRI Report 3002015906)



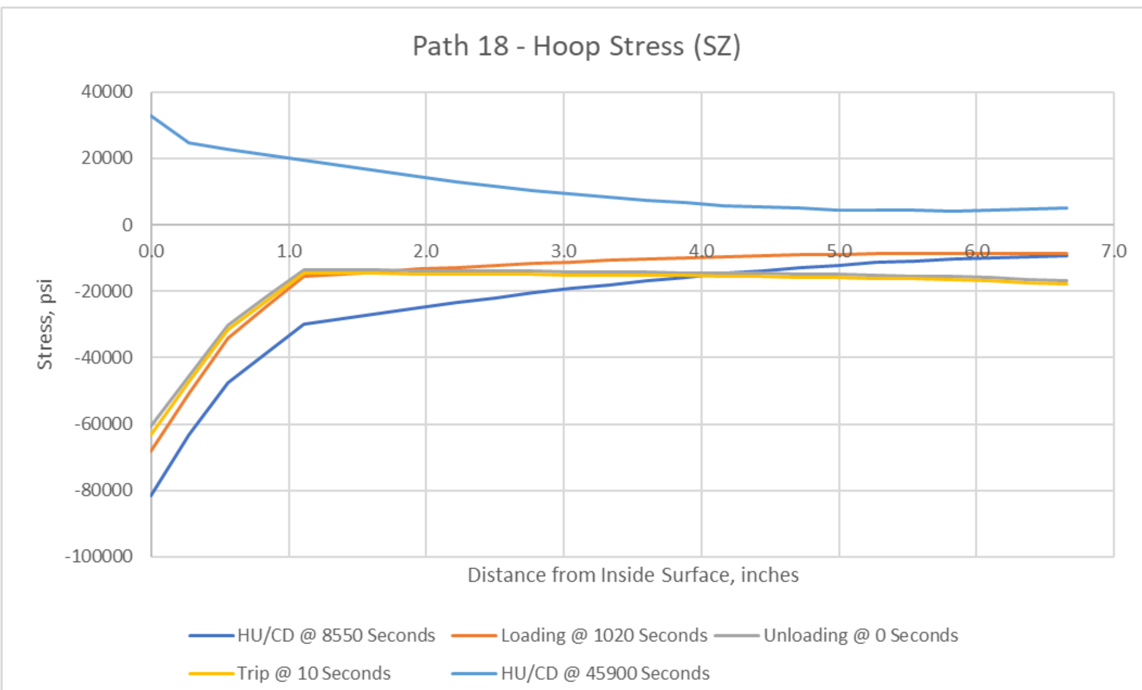
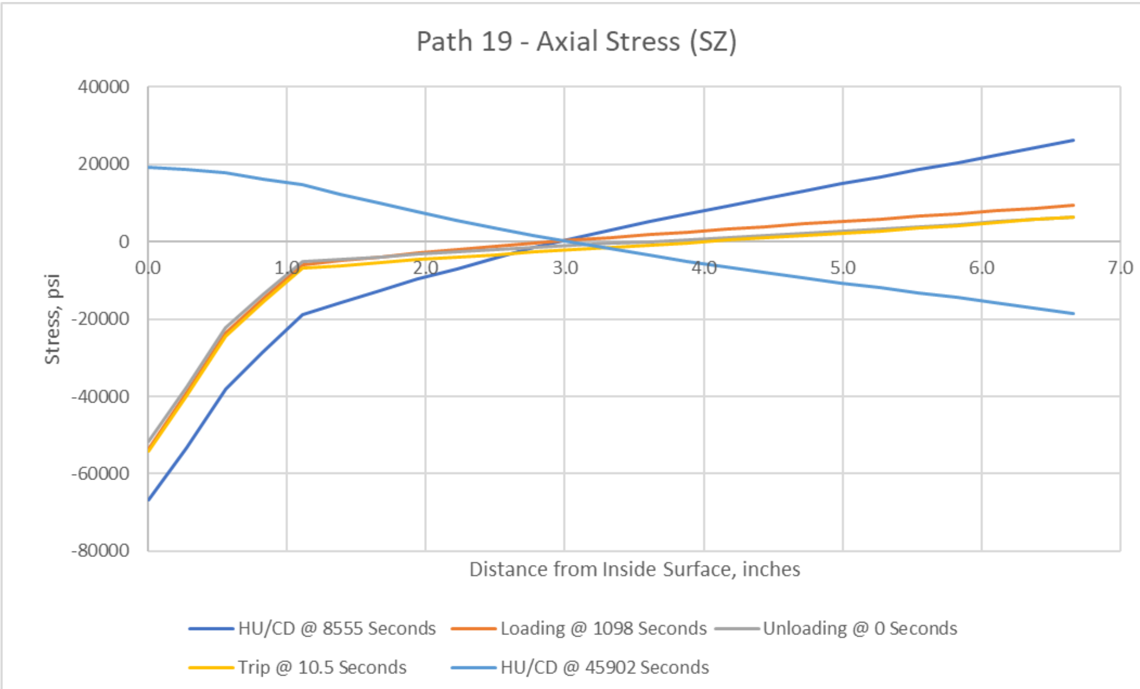


Figure 5. Through-Wall Stress Distribution for Path P18
(Same as the (Thermal+Pressure) portion of Figure 11 of Attachment 2)
(similar to Figures 7-25 through 7-28 of EPRI Report 3002015906)



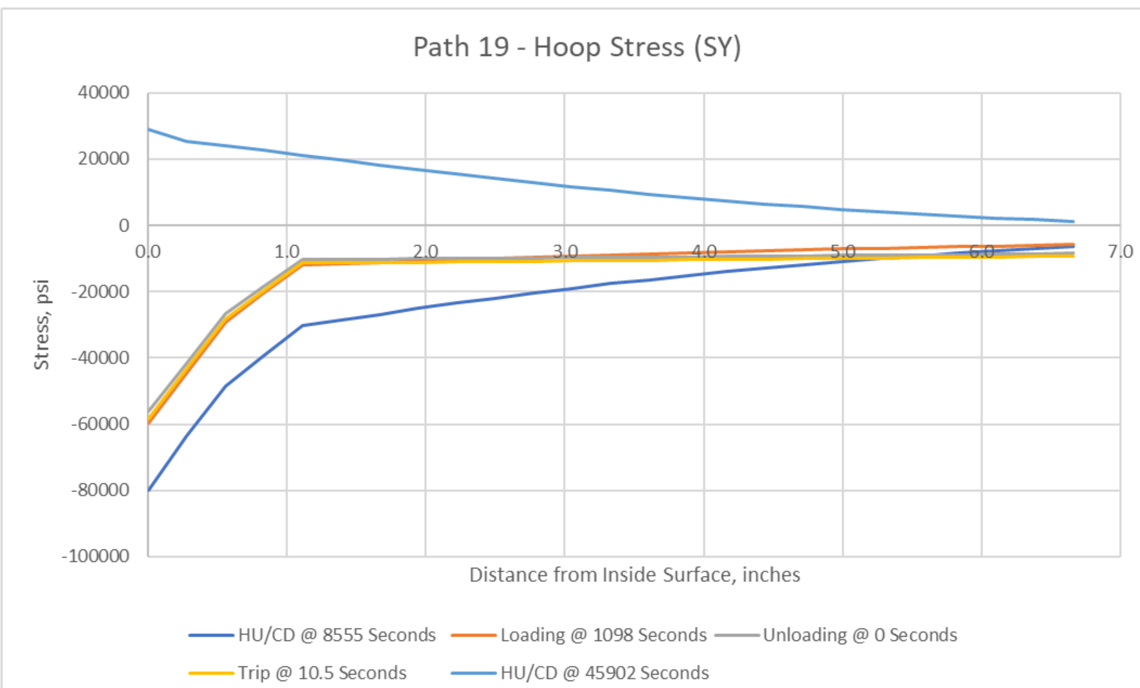
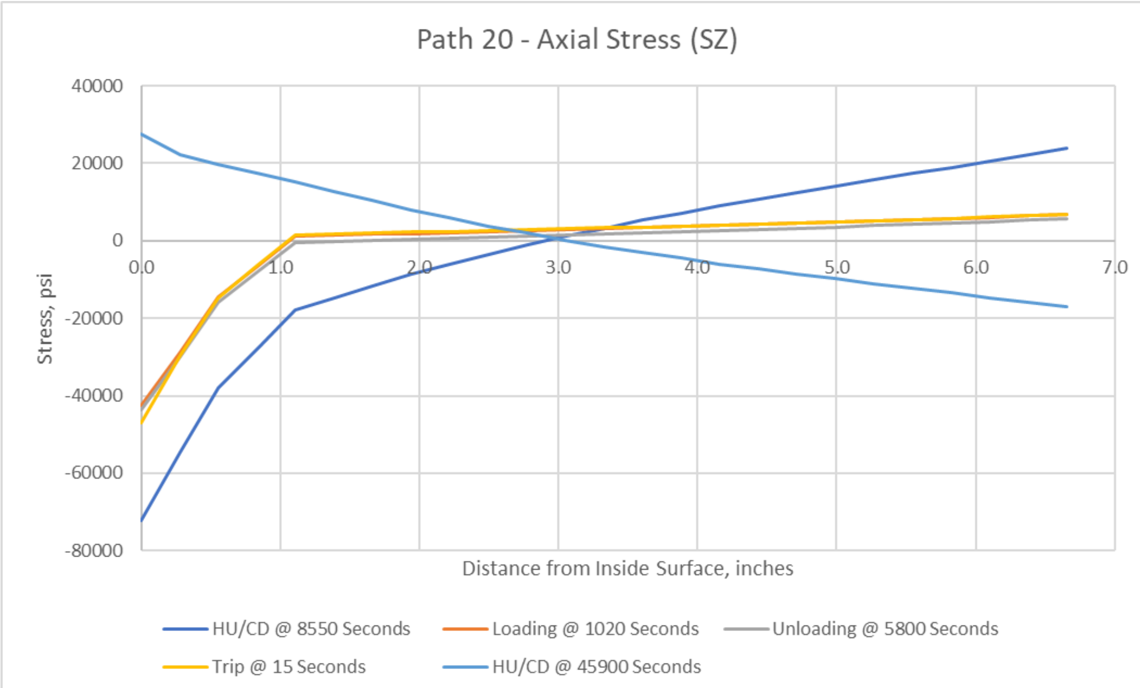


Figure 6. Through-Wall Stress Distribution for Path P19
(Same as the (Thermal+Pressure) portion of Figure 12 of Attachment 2)
(similar to Figures 7-25 through 7-28 of EPRI Report 3002015906)



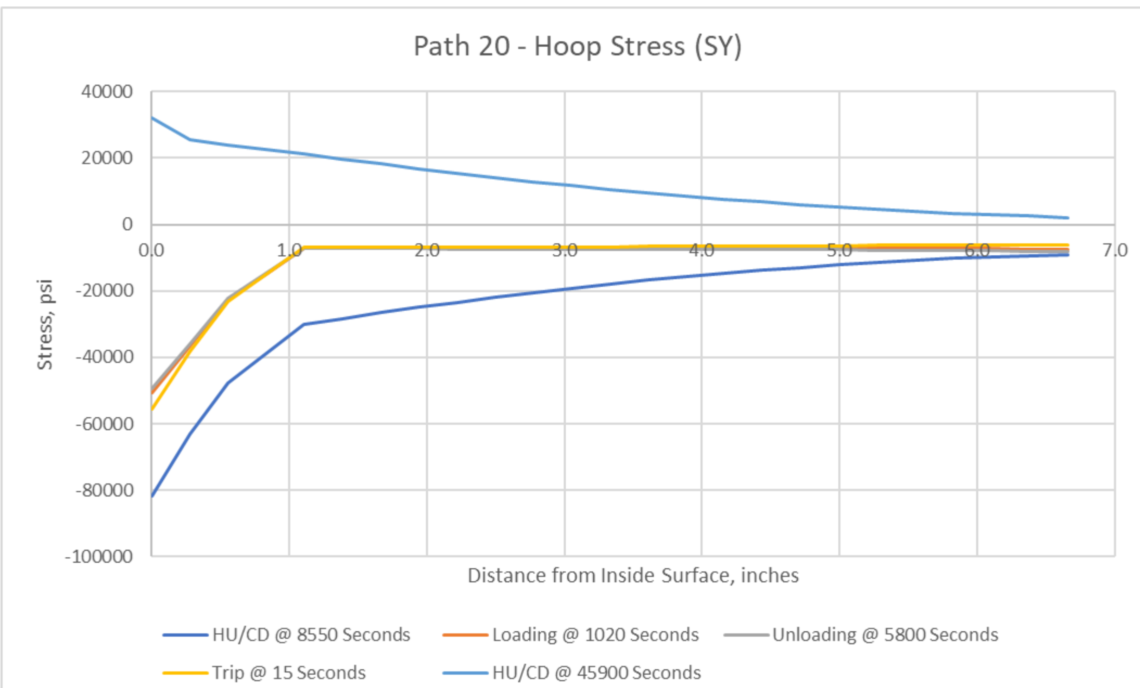


Figure 7. Through-Wall Stress Distribution for Path P20
(Same as the (Thermal+Pressure) portion of Figure 13 of Attachment 2)
(similar to Figures 7-25 through 7-28 of EPRI Report 3002015906)

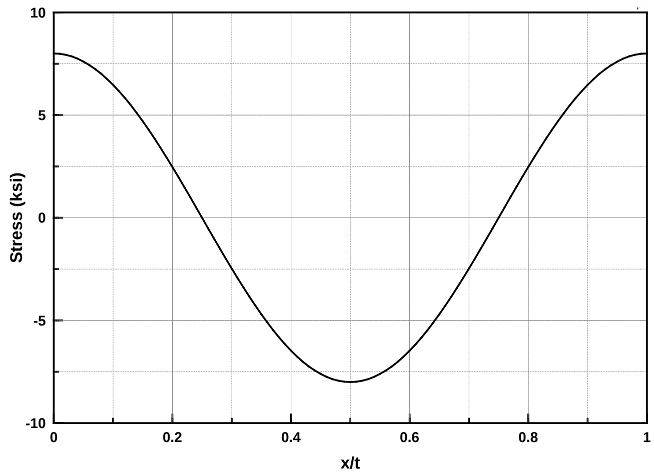


Figure 8. Weld Residual Stress Distribution
(same as Figure 8-1 of EPRI Report 3002015906)

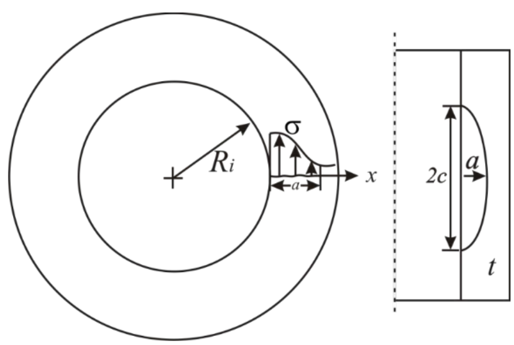


Figure 9. Semi-Elliptical Axial Crack in a Cylinder Model
(same as Figure 8-2 of EPRI Report 3002015906)

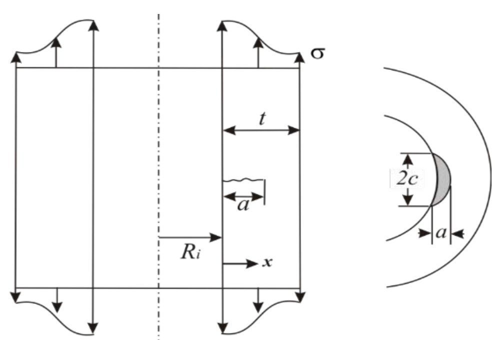


Figure 10. Semi-Elliptical Circumferential Crack in a Cylinder Model
(same as Figure 8-3 of EPRI Report 3002015906)

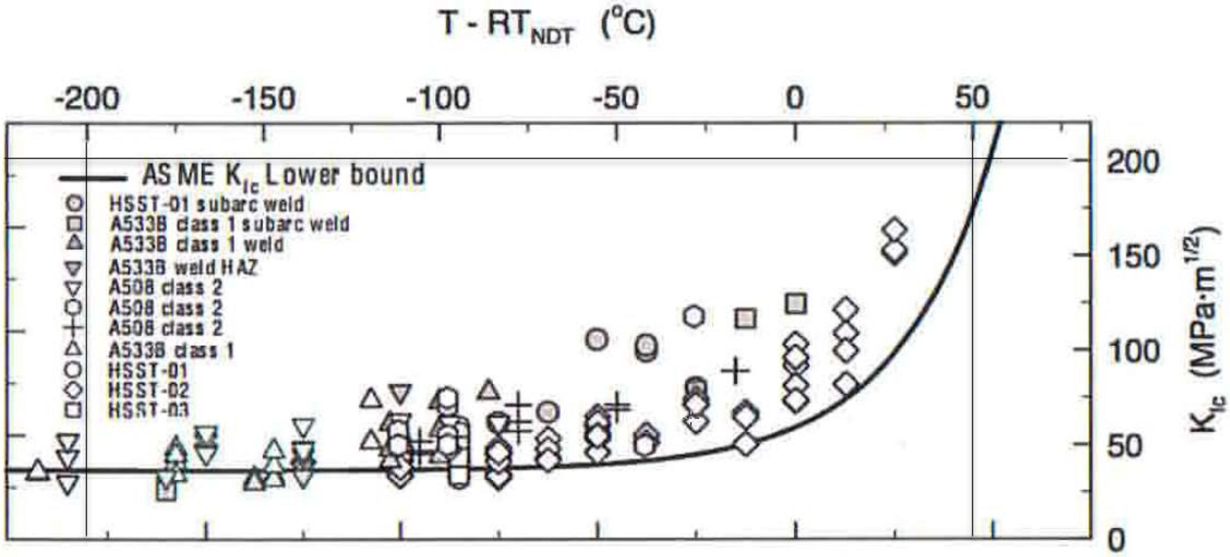


Figure 11. ASME Section XI Fracture Toughness Curve for Vessels vs. Experimental Data Points [9, 10]

(same as Figure 8-4 of EPRI Report 3002015906)